UK EPR	
	Title: PCSR – Sub-chapter 16.5 – Adequacy of the UK EPR design regarding functional diversity
	UKEPR-0002-167 Issue 01

Total number of pages: 328	Page No.: I / III
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REVISION HISTORY

Issue	Description	Date
00	First issue	30-03-2011
01	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 and addition of references - Clarification of text (§1, §2.4.1, §2.5.9.3, §2.7, § 3.3.2.5.6, §3.4.3.1.1, §3.5.2, §3.5.2.1.1) - New section 2.5.9.4 added regarding common cause failure analysis for sensors - New section 2.5.9.5 "Diversity analyses for loss of essential support systems" added - New section 2.5.10 "Loss of RCV [CVCS] faults" added - New section 2.5.11 "Emergency Operating Procedures" added covering diverse means available to reach a long-term safe shutdown state - New section 3.3.2.10 "ATWS excessive increase in Secondary Steam Flow due to TXS failure" added (with Tables 67 to 69, Figures 83 to 88) - New section 3.3.2.11 added covering rod drop/misalignment faults with ATWS due to failure of TXS" (with Tables 73 & 74, Figures 94 to 99) - New section 3.4.1.3 "Small break LOCA with failure of the VDAs [MSRT]s" added (with Tables 70 to 72, Figures 89 to 93). - New section 3.6.1.2 added "Diversity to safe shutdown state" (with Table 75 and Figure 100)	26-11-2012

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SUB-CHAPTER 16.5 – ADEQUACY OF UK EPR DESIGN REGARDING FUNCTIONAL DIVERSITY

1. INTRODUCTION

The purpose of this PCSR sub-chapter is to demonstrate the adequacy of the UK EPR design functional diversity.

The functional diversity is addressed for all the frequent Postulated Initiating Events (PIEs) as these have higher requirements for mitigation.

The role of the functional diversity analyses is to demonstrate that a functional second line of defence, diverse from the first line, is successful in mitigating frequent events with the loss of a safety function.

The safety functions are composed of plant level safety functions and the diversity is demonstrated within the plant level safety functions for all the frequent faults. Since some events are clearly more bounding than others for a given plant level safety function, a comprehensive review of the transients is performed to select the limiting events before their examination by calculations.

Therefore, the following PCSR sub-chapter is comprised of two parts.

- The methodology assessing the frequent events and the safety functions considered are presented and the selection of the most limiting events for each plant level safety function is demonstrated.
- Then, the limiting events are analysed by calculations or argumentation to demonstrate that the safety criteria are met. The analyses are performed using conservative assumptions.

2. METHODOLOGY FOR THE DEMONSTRATION OF FUNCTIONAL DIVERSITY

2.1. INTRODUCTION

The current section describes the functional diversity for the EPR design. Such diversity is assessed against a list of frequent PIEs deriving from the reconciliation of the deterministic and probabilistic lists of PIEs. The cut-off frequency is 10^{-3} per reactor per year and further investigations are provided to ensure there are no shortcomings in the demonstration.

The analysis of functional diversity presented in this sub-chapter demonstrates the completeness of the analysis, including consideration of frequent initiating events occurring from a range of possible plant states [Ref-1]. This list is consistent with the list of events presented in PCSR Sub-chapter 15.1 for the probabilistic safety analysis.

Section 2.2 describes the different levels of safety functions used to classify the events presented in the Fault Schedule.

Section 2.3 provides the list of design basis events which have a frequency greater than 10^{-3} per year. This list is consistent with the list of events presented in PCSR Sub-chapter 15.1 for the probabilistic safety analysis.

Section 2.4 presents the methodology used to carry out a comprehensive analysis of functional diversity based on all the frequent PIEs for each defined safety function.

Section 2.5 details each transient family (e.g. decrease in Reactor Coolant System (RCP [RCS]) water inventory) and safety function in order to identify the PIEs for which the loss of a safety function is the most onerous. These highlighted events are further analysed to identify cases which present the greatest challenge to the safety criteria.

Section 2.6 summarises the analysis of the bounding events selected.

2.2. INTRODUCTION TO THE EPR SAFETY FUNCTIONS

2.2.1. Definition of Safety Functions

The Fault Schedule presents several levels of safety functions derived from the following main safety functions:

- Control of fuel reactivity,
- Fuel heat removal,
- Confinement of radioactive material,
- Other.

In order to perform the fault analysis on a more precise level and to demonstrate that the main safety functions can be fulfilled, each main safety function is split into two sub–levels.

- Plant Level Safety Functions (PLSF),
- Lower Level Safety Functions (LLSF).

The definitions of the plant level safety functions are based on international standards for PWR (IAEA NS-R-1), international good practice as illustrated by Sizewell B, and analysis of EPR plant processes.

The lower level safety functions are a combination of the PLSF and the operating conditions of the EPR (normal, abnormal, accident). The LLSF are refined safety goals in specific conditions. The safety functional groups and safety features (Structures, Systems and Components, SSCs) used to perform the LLSF are derived from these safety functions[1].

[1] Further details regarding safety functional groups and safety features are provided in PCSR Sub-chapter 3.2.

The list of main safety functions, plant level safety functions and lower level safety functions is provided in Sub-chapter 16.5 – Table 1. The list of safety functions given are mainly those dedicated to the mitigation of the PIEs considered in this study. This list should not be considered as a comprehensive list of safety functions used to classify plant SSCs.

2.2.2. Application of Safety Functions to the Fault Schedule

For each event with a frequency higher than 10^{-3} per reactor per year, the main safety functions are listed and detailed into plant level safety functions and lower level safety functions.

For these functions, the safety functional groups and some safety features used to carry out the LLSF are detailed. A diverse line of protection, at the lower level safety function, is indicated showing diversity within the plant level safety functions. The safety functional groups used in the diverse lower level safety functions are also indicated to illustrate the independence of the diverse line of defence for the frequent events considered.

2.3. FREQUENT INITIATING EVENTS

2.3.1. Reconciliation with Probabilistic Safety Assessment events

The Probabilistic Safety Assessment presented in PCSR Chapter 15 provides the list of events to be considered as frequent faults. The following sections describe the list of initiating events to consider as frequent faults.

2.3.2. PCC-2 events

The list of PCC-2 events corresponds to events which have a frequency greater than 10^{-2} per year. This list is presented in PCSR Sub-chapter 14.3 and in Sub-chapter 16.5 – Table 2.

The spurious reactor trip (State A) is covered by the loss of condenser vacuum (State A), or any PCC-2 event leading to a reactor trip, and is thus not analysed further. Similarly, the turbine trip event is bounded by the loss of condenser event and is not analysed further.

2.3.3. Frequent PCC-3 events

The list of frequent PCC-3 events to be considered is extracted from PCSR Sub-chapter 14.4. The events with a frequency higher than 10^{-3} per reactor per year are listed in Sub-chapter 16.5 – Table 3 with the associated frequency.

Examples of PCC-3 events not considered due to their low frequency are:

- Leak in the gaseous or liquid waste processing systems (this event is only studied for the radiological consequences and large conservatism and no automatic actions are considered).
- Uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal (States B, C, D) (this event is also covered by the case at power).

The Small Break Loss of Coolant Accident (SB LOCA) in states A and B (break area smaller than 20 cm² - equivalent diameter less than 50 mm) is considered in the functional diversity analysis even though its frequency is about 6 $x10^{-4}$ per reactor per year. This frequency is lower than the range for frequent faults. However this transient is added to the analysis to demonstrate the absence of any potential cliff edge effect in the analysis. This event bounds the loss of primary coolant outside the containment due to the larger leak rate.

2.3.4. Fault Schedule

The fault schedule is presented in PCSR Sub-chapter 14.7. The fault schedule relevant to this analysis is limited to the frequent events given above.

The elements for the main protection line and the back-up line are listed. These elements are sorted using the different levels of safety functions. This classification is undertaken for all frequent events, grouped together by family.

2.4. FUNCTIONAL DIVERSITY ANALYSIS METHODOLOGY

2.4.1. Acceptance criteria

The first objective of the analysis is to meet the SAPs release target 4.

To ensure these criteria are met, the analysis must demonstrate that events analysed with an assumed failure of a lower level safety function at least meet the PCC-3/PCC-4 criteria (or lower if possible).

The following analyses rules are applied:

- Preventive maintenance is not considered in the analysis.
- The transients presented in this study are analysed with conservative assumptions for the values of key parameters. This includes pessimism of initial and boundary conditions (such as Protection System setpoints) and thresholds that are applied in the diversity study. Parameters to which the transient is not sensitive are not pessimised.

The values applied in the analyses are described in PCSR Sub-chapter 14.1. The main characteristics of the systems used to mitigate the transients are presented for each transient.

2.4.2. Methodology

The methodology for a comprehensive analysis of the functional diversity for frequent faults is presented below and summarised in Sub-chapter 16.5 – Figure 1.

Each event provided in section 2.3 is analysed at the level of the lower level safety function.

Each lower level safety function is assumed to be unavailable and is replaced by a diverse lower level safety function which provides the same plant level safety function. The methodology analyses the consequences of a potential failure of each LLSF for each transient. If the diversity assessment challenges the PCC-3/PCC-4 safety criteria, the most onerous event is assessed further.

In a family of events, the most onerous case(s) for meeting the relevant safety function is analysed further.

Finally, for each safety function, comparisons between events from different families are made to identify the most challenging case. Different transients may be analysed if required.

This analysis performed at the event level and at the plant safety level ensures a comprehensive assessment of all of the potential cases.

2.5. TRANSIENT SELECTION

The transients listed in Sub-chapter 16.5 – Tables 2 and 3 are analysed by event family in the following sections. This methodology is consistent with the presentation of the fault schedule.

2.5.1. Decrease in RCP [RCS] inventory

The “decrease in RCP [RCS] inventory” family of events includes:

- Chemical and Volume Control System (RCV [CVCS]) malfunction causing a decrease in RCP [RCS] inventory,
- Small break LOCA (< DN 50) including a break occurring on the extra boration system injection line (States A and B),
- Inadvertent opening of a pressuriser safety valve (State A),
- Uncontrolled RCP [RCS] level drop (States C, D),
- Steam Generator Tube Rupture (SGTR) (State A).

These events are analysed for each safety function in the following sections. As these events are in the same family of events, the same LLSF are usually used to mitigate the transients. Where differences exist, they will be clearly identified.

The inadvertent opening of a pressuriser safety valve is not analysed in detail as it is covered by the small break LOCA for RCP [RCS] inventory control and the spurious pressuriser spray actuation for the Departure from Nucleate Boiling Ratio (DNBR) analysis.

The ‘uncontrolled RCP [RCS] level drop (States C, D)’ is only analysed for the PLSF ‘H1 - maintain sufficient Reactor Coolant System water inventory for core cooling’ as this safety function is the only one challenged by this PIE.

2.5.1.1. R1 – Maintain core reactivity control

The lower level safety function is ‘control of boron concentration – slow variation’. This safety function is not challenged in the transients considered for the diversity analysis as it deals with normal operating conditions.

The operational method to provide post-trip boration in the long term is the RCV [CVCS]. The time scale to perform boration after reactor trip is large, thus leaving adequate margins for the operator to perform the action. Should the RCV [CVCS] fail, boration can be performed manually by the operator using the Extra Boration System (RBS [EBS]). Therefore, diversity is provided to fulfil this lower level safety function.

2.5.1.2. R2 - Shutdown and maintain core sub-criticality

The challenged lower level safety function is ‘negative reactivity fast insertion’ due to the need to shutdown the reactor trip in such transients.

The reactor trip occurs on low pressuriser pressure (Protection System). Following failure of the TXS I&C platform, the reactor trip would be actuated by the “low Hot Leg pressure” signal (by the Safety Automation System, SAS).

Diversity for the lower level safety function is provided by the ‘high concentration and high pressure boron injection’ Safety Functional Group (SFG). The extra boration system is actuated by the Anticipated Trip Without Scram (ATWS) signal from the TXS platform (Protection System).

Should the RCV [CVCS] malfunction, the leak rate causes a decrease in the RCP [RCS] inventory at a rate of 20 kg/s. Similarly, the leak rate following a steam generator tube rupture is about 25 kg/s. For the small break LOCA case, the leak rate considered is about 100 kg/s at the beginning of the transient. Therefore, the biggest challenge to PLSF R2 occurs following a small break LOCA as:

- the risk of core uncovery is higher,
- more boron is needed to maintain core sub-criticality because of the larger size of the break and the inability to isolate it.

The most onerous case for the consequences of the loss of the 'negative reactivity fast insertion' LLSF is the small break LOCA combined with the mechanical blockage of the rods (ATWS).

For completeness, the small break LOCA combined with the loss of the TXS platform is also considered as a bounding transient.

2.5.1.3. R3 - Prevention of uncontrolled positive reactivity insertion into the core

The lower level safety function is ‘RCP [RCS] overcooling protection’. This is provided by the safety functional groups ‘turbine trip’ and ‘full load main feedwater isolation’. These SFGs are actuated by the TXS platform (Protection System) following the reactor trip.

Should the TXS fail, the reactor trip signal from the SAS also initiates the turbine trip and the full load Main Feed Water (MFW) isolation.

From the PSLF R3 standpoint, the diverse steam isolation function is provided by the Main Steam Isolation Valve (VIV [MSIV]) closure, which is performed automatically by the reactor Protection System (RPR [PS]) or automatically by the SPPA-T2000 platform following a “low cold leg temperature” signal or manually. The RPR [PS] initiated VIV [MSIV] closure occurs from either a “SG pressure drop > MAX1” or a “SG pressure < MIN1” signal. Diversity from the above is also provided for the full load MFW isolation by the F2 classification and it can be performed by the SPPA-T2000 platform automatically or manually. This function acts on two redundant isolation valves. The low load isolation valve is closed following a “high SG level (> MAX0p)” signal and if the reactor trip has already occurred, the “SG level > MAX1p” signal leads to the closure of the full load, the low load and the main isolation valves in the MFW.

The assumed loss of the function is not an issue following either a SB LOCA or RCV [CVCS] malfunction causing a decrease in RCP [RCS] inventory as a cooldown by the secondary side has a beneficial impact on the transient.

Following a steam generator tube rupture (1 tube), the isolation of the main feedwater is required to prevent SG overfilling. The diversity of the function is demonstrated above.

2.5.1.4. R4 – Maintain sufficient sub-criticality of fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to a decrease in RCP [RCS] inventory event.

2.5.1.5. H1 - Maintain sufficient Reactor Coolant System water inventory for core cooling

The lower level safety functions are:

- Water injection into the RCP [RCS]
- Prevention of RCP [RCS] drainage through auxiliary lines

The LLSF 'water injection into the RCP [RCS]' is performed by the safety functional groups actuating the Medium Head Safety Injection (MHSI) pumps and the Main Steam Relief Trains (VDA [MSRT]). The partial cooldown is required to allow the RCP [RCS] to reach the MHSI injection pressure threshold. The Safety Injection System (RIS [SIS]) signal is generated following a "pressuriser pressure < MIN3" signal from the Protection System (RPR [PS]). Should the Protection System fail, an automatic RIS [SIS] actuation signal is received from the Safety Automation System (SAS) following a "hot leg pressure < MIN3" signal.

In cases where two safety functional groups are used to perform the LLSF, the analysis does not assume the failure of both groups within the same transient. Two analyses are performed. In the case of a RCV [CVCS] malfunction event or a SB LOCA event, diversity for this LLSF is provided by the 'water injection into the RCP [RCS]' function, which is performed by the LHSI pumps following a secondary cooldown. Should failure of the partial cooldown occur, and the RCP [RCS] pressure remains too high to allow MHSI injection, a secondary fast cooldown is performed by the manual opening the VDAs [MSRT]s.

Should complete failure of the four VDAs [MSRT]s occur, the operator would use the feed and bleed procedure. The detailed emergency operating procedures to start-up the feed and bleed actions are not part of the Generic Design Assessment, however, an analysis is performed to demonstrate the effectiveness of the feed and bleed actions based on assumptions for actuation by the operator.

In addition, the automatic reactor coolant pump trip is actuated by the Protection System following a SB LOCA by detection of a pressure difference over the reactor coolant pump lower than MIN1 and the presence of the RIS [SIS] signal. The reactor coolant pump can also be tripped manually via the SAS. The failure of this function is not a limiting case as all other safety injection means are available to maintain sufficient reactor coolant system inventory.

Manual actions to control the SG pressure via the VDAs [MSRT]s combined with LHSI injection provide the diversity for this function following a SGTR.

The ‘prevention of RCP [RCS] drainage through auxiliary lines’ function is provided by letdown line isolation and Residual Heat Removal (RHR) / RCV [CVCS] connection isolation following a RIS [SIS] signal generated via a “pressuriser pressure < MIN3” signal from the Protection System. Should the Protection System fail, the RIS [SIS] signal is triggered by the SAS as described above. The letdown line isolation is provided by the closure of two redundant isolation valves, the control valve on the RCV [CVCS] connecting line and the isolation valve from the RCV [CVCS]. The diversity in the actuation platforms and the redundancy in the isolation valves are sufficient to ensure the function will be achieved.

As discussed in section 2.5.1.2, a small break LOCA has a larger break flow which cannot be isolated and is therefore the most onerous PIE for this plant level safety function. Therefore, with respect to loss of RCP [RCS] water inventory, the small break LOCA is more onerous than the RCV [CVCS] malfunction causing a decrease in RCP [RCS] inventory and the SGTR leading to a larger decrease in RCP [RCS] inventory. Consequently, the three following cases will be further analysed to demonstrate diversity for the bounding events:

- SB LOCA without MHSI,
- SB LOCA without Partial Cooldown,
- SB LOCA without VDA [MSRT].

For the ‘uncontrolled RCP [RCS] level drop (states C, D)’ event, the limit values, various alarms and interlocks (PCSR Sub-chapter 14.3) ensure that this event is mitigated. The LHSI pumps and MHSI pumps provide the diverse inventory control following leakage. In addition, the RIS [SIS] actuation is also actuated via the SAS on low loop level in this case.

2.5.1.6. H2 – Remove heat from the core to the reactor coolant

This safety function is not challenged by these events as sufficient water inventory is maintained to ensure that the heat transfer capacity is sufficient and that the DNBR limit is not challenged. Emergency Core Cooling System diversity has been demonstrated for the H1 plant level safety function.

2.5.1.7. H3 - Transfer heat from the reactor coolant to the ultimate heat sink

The LLSF is ‘heat removal by steam generators - emergency shutdown mode’. It is performed by safety functional groups actuating the Emergency Feed Water System (ASG [EFWS]) and the VDA [MSRT]s. Emergency feedwater is actuated following a ”SG level < MIN2” signal for each steam generator from the Protection System.

For the RCV [CVCS] malfunction causing a decrease in RCP [RCS] inventory and the SB LOCA, the diverse lower level safety function is provided by the ‘heat removal by Low Head Emergency Core Cooling System (ECCS)’ function. This uses the safety injection (MHSI, LHSI and accumulators) and the severe accident discharge line. It is actuated manually using the SAS.

Following an SGTR, manual actions to control the SG pressure via the VDAs [MSRT]s and LHSI are used to provide the diverse function.

The RCV [CVCS] malfunction causing a decrease in RCP [RCS] inventory is more onerous than the small break LOCA or the SGTR event as it leads to a slower depressurisation and cooldown of the reactor coolant system. Thus, there is bigger requirement to transfer heat to the secondary side.

However, it should be noted that whilst the decrease in RCP [RCS] inventory events do not significantly challenge this safety function, a bigger challenge is presented by the decrease in heat removal events.

Note: The Main Steam Safety Valves (MSSVs), which are passive relief valves, provide the diverse function to the VDAs [MSRT]s. The two MSSVs together (per SG) have the same capacity as one VDA [MSRT]. The heat can therefore be removed by the steam generators via either route. It should also be noted that these events are not overpressure transients and do not, therefore, challenge the criteria.

2.5.1.8. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to the transients in the decreases in RCP [RCS] inventory family.

2.5.1.9. C1 - Maintain integrity of fuel cladding

This safety function is challenged but the diversity is addressed in the analysis of the R2 function.

2.5.1.10. C2 – Maintain integrity of the reactor coolant pressure boundary

The initiating events are failures of the reactor pressure boundary with the exception of the RCV [CVCS] malfunction. This safety function cannot therefore be met and is not analysed for this family of events.

2.5.1.11. C3 – Limit the release of radioactive material from the reactor containment

The lower level safety function performing this function is ‘containment isolation’ which occurs following a RIS [SIS] signal. Diversity for this actuation is not provided by another lower level safety function.

The RCV [CVCS] malfunction causing a decrease in RCP [RCS] inventory does not challenge this function as it does not involve a break of the reactor coolant system pressure boundary. In the decrease in reactor coolant inventory family, this function is challenged by a small break LOCA.

Therefore, the ‘containment building isolation’ function will be subject to an ALARP (As Low As Reasonably Practicable) justification.

For steam generator tube rupture, the plant level safety function is challenged if non isolation of the main feedwater or failure to open of the VDAs [MSRT]s occurs. These two functions and their diversity have been justified in sections 2.5.1.3 and 2.5.1.5.

2.5.1.12. C4 – Limit the release of radioactive waste and airborne material

This safety function is not challenged by the RCV [CVCS] malfunction causing a decrease in RCP [RCS] inventory and the SB LOCA events.

For the SGTR, the lower level safety function is 'prevention of radioactive release outside containment from radioactive steam generator'. This is achieved by the safety functional group performing the isolation of the affected steam generator (SGa) via the VDA [MSRT] setpoint increase, the VIV [MSIV] closure on the affected SG and the RCV [CVCS] charging line isolation.

The VDA [MSRT] setpoint increase is performed automatically by the Protection System following a "SG level > MAX2" signal if the partial cooldown has been completed. Should failure of the Protection System occur, a manual increase in the control valve setpoint can be performed from the SAS. The SGTR with failure of the VDA [MSRT] setpoint increase of the SGa is bounded by the case presented below covering the failure of the VIV [MSIV].

The VIV [MSIV] closure occurs following a "SG level > MAX2" signal if the partial cooldown has been completed. Should the Protection System fail, the VIV [MSIV] can be closed manually from the SAS by the operator 30 minutes after the first safety classified signal. To demonstrate sufficient safety margins, the transient 'SGTR + VIV [MSIV] failure to close of SGa (VIV [MSIV]a)' will be further assessed. An ALARP justification of the provision of diverse actions to the VIV [MSIV] closure will be carried out.

To prevent SG overfilling, the RCV [CVCS] charging line is automatically isolated following a "SG level > MAX2" signal. The isolation can also be performed manually by the SAS, which provides the necessary diversity for this function.

2.5.1.13. O1 – Prevent the failure or limit the consequences of failure of a structure, system or component whose failure would cause the impairment of a safety function

This plant level safety function is provided by the safety functional group actuating the VDAs [MSRT]s so as to prevent overpressure in the secondary side following turbine trip. The MSSVs, which are passive relief valves provide the capability diverse to the VDAs [MSRT]s. The two MSSVs together (per SG) have the same capacity as the VDA [MSRT]. The overpressure peak due to the temporary heat imbalance between the primary and secondary sides at turbine trip can therefore be withstood by the steam generators using either route.

2.5.1.14. Summary for decrease in RCP [RCS] inventory events

The summary of the plant level safety functions used for transient mitigation is provided in Sub-chapter 16.5 – Table 4. Function O1 is not presented in the summary table as sufficient diversity has been demonstrated.

2.5.2. Increase in RCP [RCS] inventory

The only event to be considered for the diversity analysis is:

- RCV [CVCS] malfunction causing an increase in RCP [RCS] inventory

2.5.2.1. R1 – Maintain core reactivity control

This safety function is not challenged by the transient.

2.5.2.2. R2 - Shutdown and maintain core sub-criticality

The lower level safety function is ‘negative reactivity fast insertion’. This is provided by the reactor trip actuation following either a “pressuriser pressure > MAX2” or a “pressuriser level > MAX1” signal from the Protection System (TXS platform).

Should the TXS I&C platform fail, a diverse reactor trip signal is not provided. The isolation of the charging line mitigates the event, without a reactor trip being required. The charging line can be isolated automatically (charging flow and auxiliary spray isolation) from the SPPA-T2000 platform following a “pressuriser level > MAX2” signal and from the TXS platform following a “pressuriser level > MAX2” (shutdown of RCV [CVCS] charging line) signal. This event leads to a higher pressuriser level and a pressurisation of the reactor coolant system, which may lead to the opening of the Pressuriser Safety Valves (PSVs), but is stopped by the isolation of the RCV [CVCS]. This event is covered by the SB LOCA event.

The diverse lower level safety function is provided by the ‘high concentration and high pressure boron injection’ function. The extra boration system is actuated by the ATWS signal from the TXS platform. The loss of the main safety function would lead to the ATWS sequence. This event is covered by the ATWS of the ‘decrease in heat removal’ family of events, as there are no changes to the heat removal rate following a RCV [CVCS] malfunction causing an increase in RCP [RCS] inventory.

2.5.2.3. R3 - Prevention of uncontrolled positive reactivity insertion into the core

The lower level safety function is ‘RCP [RCS] overcooling protection’. This is provided by the safety functional groups ‘turbine trip’ and ‘full load MFW isolation’. These SFGs are initiated by the Protection System following the reactor trip.

From the PSLF R3 standpoint, the diversity for the steam isolation function is provided by the VIV [MSIV] closure, which is performed automatically by the RPR [PS] or automatically by the SPPA-T2000 platform following a “low cold leg temperature” signal or manually. The VIV [MSIV] closure occurs via the RPR [PS] either on a “SG pressure drop > MAX1” or a “SG pressure < MIN1” signal. Diverse actuation of the full load MFW isolation is also, provided which differs from the above by its F2 classification and its action on the common isolation valve. The isolation can be performed automatically by the TXS platform or manually. This function operates via two redundant isolation valves. Following a “high SG level (> MAX0p)” signal, the low load isolation valve is closed. If a reactor trip has already occurred, a “SG level > MAX1p” signal will result in the closure of the full load, the low load and the main isolation valves in the MFW.

Therefore, the RCV [CVCS] malfunction causing an increase in RCP [RCS] inventory with an assumed failure in the R3 PLSF is covered by events in the ‘increase in heat removal family’.

2.5.2.4. R4 – Maintain sufficient sub-criticality of fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to this transient.

2.5.2.5. H1 - Maintain sufficient Reactor Coolant System water inventory for core cooling

This safety function is not challenged by this transient.

2.5.2.6. H2 – Remove heat from the core to the reactor coolant

This safety function is not challenged by this transient.

2.5.2.7. H3 - Transfer heat from the reactor coolant to the ultimate heat sink

The lower level safety function is 'heat removal by steam generators - emergency shutdown mode'. It is performed by the safety functional groups actuating the emergency feedwater and the VDAs [MSRT]s. Emergency feedwater is actuated following a "SG level < MIN2" signal on the associated steam generator by the Protection System.

The diverse capacity to the VDAs [MSRT]s is provided by the MSSVs, which are passive relief valves. The two MSSVs together (per SG) have the same capacity as the VDA [MSRT]. The heat can therefore be removed via the steam generators using either route.

Diversity for the lower level safety function is provided by the function 'heat removal by Low Head Emergency Core Cooling System (ECCS)', using safety injection (MHSI, LHSI and accumulators) and the severe accident discharge line. It is actuated manually from the SAS.

The 'RCV [CVCS] malfunction causing an increase in RCP [RCS] inventory' with an assumed failure in the H3 PLSF is covered by events in the 'decrease in heat removal family' as the heat removal rate is not affected by the RCV [CVCS] malfunction.

2.5.2.8. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to this transient.

2.5.2.9. C1 - Maintain integrity of fuel cladding

This safety function is not challenged by this transient.

2.5.2.10. C2 – Maintain integrity of the reactor coolant pressure boundary

The lower level safety function 'RCP [RCS] overpressure protection' is provided by the isolation of the charging flow and normal and auxiliary spray. As discussed in section 2.5.2.2, the isolation can be performed from the TXS I&C platform or the SPPA-T2000 platform.

Further diversity for this lower level safety function is provided by the Pressuriser Safety Valves (PSVs).

This event is covered by a decrease in heat removal event which leads to a higher overpressure.

2.5.2.11. C3 – Limit the release of radioactive material from the reactor containment

This safety function is not challenged by this transient.

2.5.2.12. C4 – Limit the release of radioactive waste and airborne material

This safety function is not challenged by this transient.

2.5.2.13. O1 – Prevent the failure or limit the consequences of failure of a structure, system or component whose failure would cause the impairment of a safety function

The lower level safety function 'essential component protection' is provided by the VDAs [MSRT]s. The diverse function to the VDAs [MSRT]s is provided by the MSSVs, as discussed above.

2.5.2.14. Summary

The safety functions challenged during this event are summarised in Sub-chapter 16.5 – Table 5.

2.5.3. Decrease in heat removal

The events considered for the decrease in heat removal events family are:

- Loss of condenser vacuum,
- Loss of main feedwater,
- Small feedwater system piping failure,
- Loss of off-site power,
- Inadvertent closure of one or four VIV [MSIV]s,
- Loss of one cooling train of the Safety Injection System/Residual Heat Removal System (RIS/RRA [SIS/RHRS]) in RHR mode (states C, D).

The event 'Loss of one cooling train of the RIS/RRA [SIS/RHRS] in RHR mode (states C, D)' is only analysed for the 'H3 - transfer heat from the reactor coolant to the ultimate heat sink' safety function as the other safety functions are not challenged. The only requirement is the removal of the residual heat in RHR mode.

2.5.3.1. R1 – Maintain core reactivity control

This safety function is not challenged by the transients.

2.5.3.2. R2 - Shutdown and maintain core sub-criticality

The lower level safety function is 'negative reactivity fast insertion'. This is provided by the reactor trip on either a "pressuriser pressure > MAX2", "SG level (Narrow Range, NR) < MIN1", "low reactor coolant pump speed" or "SG pressure > MAX1" signal from the Protection System. Which signal is generated first is dependent on the Postulated Initiating Event. Should failure of the TXS I&C platform occur, the signal would be actuated via a "SG level (Wide Range, WR) < MIN3" signal from the SPPA-T2000 platform for all of the PIE in the 'decrease in heat removal' family.

The diverse lower level safety function is provided by the 'high concentration and high pressure boron injection' function. The extra boration system is actuated by the ATWS signal from the TXS platform. The loss of the LLSF 'negative reactivity fast insertion' would lead to the ATWS sequence.

The most onerous transients are the loss of main feedwater and the Loss Of Off-site Power (LOOP) as they lead to the lowest SG water inventory at the time of the reactor trip signal. Consequently, the subsequent overheating is more severe than in the 'loss of condenser vacuum', 'small feedwater system piping failure' and 'inadvertent closure of VIVs [MSIV]s' events.

The assumed failure of the lower level safety function is covered by the loss of main feedwater combined with the mechanical blockage of the control rods (ATWS) and the LOOP ATWS.

For completeness, the loss of main feedwater and the loss of off-site power combined with the loss of the TXS platform are also considered as bounding transients.

2.5.3.3. R3 - Prevention of uncontrolled positive reactivity insertion into the core

The lower level safety function is 'RCP [RCS] overcooling protection'. This is provided by the safety functional groups 'turbine trip' and 'full load MFW isolation'. These SFGs are initiated by the Protection System following the reactor trip.

From the PSLF R3 standpoint, the diverse steam isolation function is provided by the VIV [MSIV] closure, which is automatically initiated by the RPR [PS] or by the SPPA-T2000 platform following a "low cold leg temperature" signal or manually. The VIV [MSIV] closure is initiated by the RPR [PS] following either a "SG pressure drop > MAX1" or a "SG pressure < MIN1" signal. Diversity for the full load MFW isolation is also provided by the F2 classification and can be performed automatically or manually by the SPPA-T2000 platform. This function acts on two redundant isolation valves. Following a "high SG level (> MAX0p)" signal, the low load isolation valve is closed and, if the reactor trip has already occurred, the "SG level > MAX1p" signal leads to the closure of the full load, the low load and the main isolation valves in the MFW.

Diversity to this function is provided by the VIV [MSIV] closure and the full load MFW isolation, which differs from the one above by its F2 classification and its action on the common isolation valve. The low load isolation valve is closed following a "high SG level (> MAX0p)" signal, and, if a reactor trip has already occurred, the "SG level > MAX1p" signal will initiate the closure of the full load, the low load and the main isolation valves in the MFW.

The assumed failure of the lower level safety function is beneficial for decrease in heat removal events as it would improve heat removal from the reactor coolant system to the secondary side. Therefore, failure of this function is not conservative.

2.5.3.4. R4 – Maintain sufficient sub-criticality of fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to the transients in the decrease in heat removal family.

2.5.3.5. H1 - Maintain sufficient Reactor Coolant System water inventory for core cooling

This safety function is not challenged by the transients in the decrease in heat removal family.

2.5.3.6. H2 – Remove heat from the core to the reactor coolant

The lower level safety function is ‘core heat removal by RCP [RCS] forced flow in power mode’. This LLSF is challenged if the reactor coolant pumps are assumed to be lost. The diverse function is provided by the LLSF ‘core heat removal by RCP [RCS] natural circulation in shutdown mode’. This lower level safety function occurs passively provided heat is removed via the steam generators. The difference in temperature between the reactor coolant system and the steam generators provides the driving function for the natural circulation.

The most onerous case is the loss of main feedwater with loss of the reactor coolant pumps. This transient is not limiting as it is also covered by functions H1 and H3.

2.5.3.7. H3 - Transfer heat from the reactor coolant to the ultimate heat sink

The lower level safety function is ‘heat removal by steam generators - emergency shutdown mode’. It is performed by the safety functional groups actuating the emergency feedwater and the VDAs [MSRT]s. Emergency feedwater is actuated following a “SG level < MIN2” signal independently for each steam generator by the Protection System.

The diverse function to the VDAs [MSRT]s is provided by the MSSVs, which are passive relief valves. The two MSSVs together (per SG) have the same capacity as the VDA [MSRT]. The heat can therefore be removed by the steam generators using either route.

Diversity for the lower level safety function is provided by the ‘heat removal by Low Head Emergency Core Cooling System (ECCS)’ function. This uses the safety injection (MHSI, LHSI and accumulators) and the severe accident discharge line. It is actuated manually from the SAS.

The assumed failure of the safety function combined with the initiating event is covered by the total loss of feedwater (State A) as the SG water inventory in this case is lower as a result of the early loss of the main feedwater. This sequence therefore leads to more significant overheating of the reactor coolant system and a larger accumulated heat load on the reactor coolant system.

Following the ‘Loss of one cooling train of the RIS/RRA [SIS/RHRS] in RHR mode (states C, D)’ event, the three RHR trains are in operation and one is on stand-by. In the PCSR Sub-chapter 14.3, it is demonstrated that two trains are sufficient to remove the residual heat in the most onerous case. The train on stand-by is not claimed. The design is therefore and a diverse capability is not needed.

2.5.3.8. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to the transients in this family of events.

2.5.3.9. C1 - Maintain integrity of fuel cladding

This safety function is not challenged by these transients as the water inventory is sufficient to remove heat and the pressure increase in the reactor coolant system resulting from the decrease in heat removal events prevents boiling in the RCP [RCS].

2.5.3.10. C2 – Maintain integrity of the reactor coolant pressure boundary

The lower level safety function 'RCP [RCS] overpressure protection' is provided by the pressuriser safety valves. These are passive relief valves with high opening reliability. The diverse function for this system is provided by the normal spray which limits the overpressure in the reactor coolant system when the reactor coolant pumps are in operation.

This event is covered by the inadvertent closure of all VIVs [MSIV]s as it leads to a faster pressure increase (the closure time of the VIV [MSIV] is shorter and the steam line is shorter as the main steam header is not included). The analysis of the transient 'inadvertent closure of the four VIVs [MSIV]s without PSVs' will be performed to show that the overpressure criterion of 130% of the design pressure is met, with or without spray actuation.

2.5.3.11. C3 – Limit the release of radioactive material from the reactor containment

This safety function is not relevant for this transient as these events do not lead to breaks of the reactor coolant system. If the pressuriser relief tank rupture disk were to fail, the event would be bounded by the small break LOCA.

2.5.3.12. C4 – Limit the release of radioactive waste and airborne material

This safety function is not applicable to these transients.

2.5.3.13. O1 – Prevent the failure or limit the consequences of failure of a structure, system or component whose failure would cause the impairment of a safety function

The lower level safety function 'essential component protection' is provided by the VDAs [MSRT]s. The diverse function to the VDAs [MSRT]s is provided by the MSSVs, as described above.

2.5.3.14. Summary of decrease in heat removal event

The summary of the plant level safety functions used in transient mitigation is provided in Sub-chapter 16.5 – Table 6.

2.5.4. Increase in heat removal

The events considered in the 'increase in heat removal' family of events are:

- Feedwater malfunction causing a reduction in feedwater temperature[2],
    - Feedwater malfunction causing an increase in feedwater flow rate[2],
- Excessive increase in steam flow,
- Small steam system piping failure (< DN 50).

The consequences of both feedwater malfunction events are covered by those of the excessive increase in steam flow or the small steam system piping break events.

[2] This event is not analysed in the PCSR and is not subject to an analysis in the current document

2.5.4.1. R1 – Maintain core reactivity control

This safety function is not challenged by these transients.

2.5.4.2. R2 - Shutdown and maintain core sub-criticality

The lower level safety function is ‘negative reactivity fast insertion’. It is provided by the safety functional group ‘reactor trip’ following a “SG pressure < MIN1” or “SG pressure drop > MAX1” signal from the Protection System for the excessive increase in steam flow fault. For the small steam line break fault, the reactor trip may not be actuated as the break under saturated steam conditions will only remove about 2% of the nominal power. The core is protected by dedicated signals ”low DNBR” or “high core power level” if the break flow is sufficient to reach the associated setpoints. Should failure of the TXS I&C platform occur, the reactor trip signal is generated following either “cold leg temperature (WR) < MIN1”, “SG level < MIN1”, or “High neutron flux” signal in the SPPA-T2000 platform, if needed.

The diverse lower level safety function is provided by the ‘high concentration and high pressure boron injection’ function. The extra boration system is actuated by the ATWS signal from the TXS platform. The loss of the LLSF ‘negative reactivity fast insertion’ would lead to the ATWS sequence.

Should the reactor trip fail, the SG pressure decreases until the low SG pressure setpoint is reached. The main steam isolation valves are closed following this signal, which terminates the cooldown caused by the initiating event.

The bounding case is the excessive increase in steam flow. In this case, the power removed by the opening of the main steam bypass valve is ~10% nominal power, which is larger than that associated with the small steam line break. Consequently, the resultant cooldown is larger following an excessive increase in steam flow.

For completeness, the excessive increase in steam flow combined with the loss of the TXS platform is also assessed as a bounding transient.

2.5.4.3. R3 - Prevention of uncontrolled positive reactivity insertion into the core

The lower level safety function is ‘RCP [RCS] overcooling protection’. It is provided by the safety functional groups ‘turbine trip’ and ‘full load MFW isolation’. These SFGs are initiated by the Protection System following the reactor trip.

Should the TXS fail, the reactor trip signal via the SPPA-T2000 also actuates the turbine trip and the full load MFW isolation. The reactor trip signal generated following a “cold leg temperature < MIN1” signal also actuates the automatic closure of the main steam isolation valves.

From the PSLF R3 standpoint, the diverse steam isolation function is provided by the VIV [MSIV] closure, which is automatically actuated by the RPR [PS] or by the SPPA-T2000 platform on a “low cold leg temperature” signal or manually. The VIV [MSIV] closure is actuated via the RPR [PS] on either a “SG pressure drop > MAX1” or a “SG pressure < MIN1” signal. The SPPA-T2000 platform provides the diverse full load MFW isolation to the one above by the F2 classification and can be performed either automatically or manually. This function actuates two redundant isolation valves. Following a “high SG level (> MAX0p)” signal, the low load isolation valve is closed and, if the reactor trip has already occurred, the “SG level > MAX1p” signal leads to the closure of the full load, the low load and the main isolation valves in the MFW.

The diverse function is provided by the VIV [MSIV] closure and the full load MFW isolation, which differs from the one above as it is F2 classified and it action on the common isolation valve. Following a “high SG level (> MAX0p)” signal, the low load isolation valve is closed and, if a reactor trip has already occurred, the “SG level > MAX1p” signal will lead to the closure of the full load, the low load and the main isolation valves in the MFW.

Following the inadvertent opening of the VDA [MSRT], the isolation of the main steam relief train when the SG pressure is lower than MIN3 terminates the event. This prevents further positive reactivity insertion and does not challenge the minimum shutdown margin. The isolation of the VDA [MSRT] leads to the closure of the control and of the isolation valve.

The inadvertent opening of the main steam bypass, which is one of the excessive increase in steam flow scenarios, is more onerous than the small steam line break due to the larger break size. Consequently, the failure of the VIV [MSIV] closure combined with the excessive increase in steam flow will be subject to further analysis. This event is also relevant for the ‘C1 – Maintain integrity of fuel cladding‘ as it may result in damage to the fuel cladding by Departure from Nucleate Boiling (DNB) as a consequence of the RCP [RCS] pressure decrease.

2.5.4.4. R4 –‘Maintain sufficient sub-criticality of fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to these transients.

2.5.4.5. H1 - Maintain sufficient Reactor Coolant System water inventory for core cooling

This safety function is not challenged by these transients as the RCP [RCS] water inventory does not change.

2.5.4.6. H2 – Remove heat from the core to the reactor coolant

This safety function is not challenged by these transients.

2.5.4.7. H3 - Transfer heat from the reactor coolant to the ultimate heat sink

The lower level safety function ‘heat removal by steam generators – emergency shutdown mode’ is performed by the safety functional groups actuating the emergency feedwater and the main steam relief trains. This safety function is continually used during the transfer to a stable state. However, it is not required as the excessive increase in steam flow event leads to over cooling of the reactor coolant system.

Following closure of the turbine inlet valve and of the main steam bypass, heat transfer to the ultimate heat sink is performed by the VDAs [MSRT]s. The diverse function is provided by the MSSVs, which are passive relief valves. The two MSSVs together (per SG) have the same capacity as the VDA [MSRT]. The heat can therefore be removed by the steam generators.

The diverse lower level safety function is provided by the ‘heat removal by Low Head Emergency Core Cooling System (ECCS)’ function, which uses the safety injection (MHSI, LHSI and accumulators) and the severe accident discharge line. It is actuated manually using the SAS.

This safety function is not significantly challenged following the events resulting in an increase in heat removal as the heat transfer from the reactor coolant system to the secondary side is increased by the event.

2.5.4.8. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

This safety function is not challenged by these transients.

2.5.4.9. C1 - Maintain integrity of fuel cladding

The lower level safety function 'R3 – prevention of uncontrolled positive reactivity insertion into the core' demonstrates that the excessive increase in steam flow with failure of the VIV [MSIV] might challenge this safety function. Functional diversity to ensure C1 is then covered by R3 diversity analyses.

2.5.4.10. C2 – Maintain integrity of the reactor coolant pressure boundary

This safety function is not challenged by these transients as the reactor coolant system depressurises.

2.5.4.11. C3 – Limit the release of radioactive material from the reactor containment

This safety function is not applicable to these transients.

2.5.4.12. C4 – Limit the release of radioactive waste and airborne material

This safety function is not applicable to these transients.

2.5.4.13. O1 – Prevent the failure or limit the consequences of failure of a structure, system or component whose failure would cause the impairment of a safety function

The lower level safety function 'essential component protection' is provided by the VDAs [MSRT]s. The MSSVs provide the diverse function to the VDA [MSRT]s, as described above.

2.5.4.14. Summary of increase in heat removal event

The summary of the plant level safety functions used for transient mitigation is provided in Sub-chapter 16.5 – Table 7.

2.5.5. Reactivity insertion faults

The events considered in the 'reactivity insertion faults' family of events are:

- Uncontrolled RCCA bank withdrawal at power,
- Forced decrease in reactor coolant flow,
- Uncontrolled RCCA bank withdrawal from hot zero power conditions,
- Start-up of an inactive reactor coolant pump at an incorrect temperature,

- RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant,
- Uncontrolled single control rod withdrawal.

These events are analysed for each safety function in the following sections. As these events are in the same family of event, broadly the same LLSF mitigate the transients. Where there are differences, these are clearly identified.

2.5.5.1. R1 – Maintain core reactivity control

The lower level safety function is 'Control of Boron Concentration – Slow Variation'. It is not challenged in the transients considered for the diversity analysis as it deals with normal operating conditions.

The operational control of post-trip boration in the long term is provided by the RCV [CVCS]. The time scale to perform boration after reactor trip is large, providing sufficient time for the operator to perform the action. If failure of the RCV [CVCS] occurs, boration can be performed manually by the operator using the RBS [EBS].

2.5.5.2. R2 - Shutdown and maintain core sub-criticality

The lower level safety function is 'Negative Reactivity Fast Insertion' following a Reactor Trip (RT) on either a "Low DNBR" or a "High Neutron Flux Rate of Change" or a "High Linear Power Density" (Protection System) signal. Should the TXS I&C platform fail, no diverse signals are provided on the SPPA-T2000 platform.

However, Limiting Condition of Operation (LCO) functions actuated by the Reactor Control Surveillance and Limitation system (RCSL) are triggered via a "Low DNBR" or "High Linear Power Density" signal. These functions inhibit further RCCA withdrawal and mitigate the consequences of the RCCA withdrawal faults.

The lower level safety function 'Anti Dilution Protection' is claimed for the Boron Dilution fault. This safety function is actuated by the Protection System following a "Low Boron Concentration" signal and isolates the RCV [CVCS].

The diverse function to these LLSF is provided by the 'High Concentration and High Pressure Boron Injection' function. The Extra Boration System (EBS) system is actuated by the ATWS signal or on a combination of a reactor trip signal and a "High Rod Position (or High Flux) after an appropriate delay" signal from the TXS platform. The loss of the LLSF would lead to the ATWS sequence. Following the ATWS signal, the RCV [CVCS] is isolated downstream of the RCV [CVCS] volume control tank. This stops the dilution, and the RBS [EBS] starts automatically to provide boron injection. In addition, the ATWS signal causes the primary coolant pumps to trip when the low SG level setpoint is reached.

The most onerous ATWS sequence is the 'Uncontrolled RCCA Bank Withdrawal at Power (URBWP) with RCCA mechanical blockage' sequence.

For completeness, the uncontrolled RCCA bank withdrawal at power combined with the loss of the TXS platform is also assessed as a bounding transient.

Similarly, the forced decrease in reactor coolant flow combined with the RCCA mechanical blockage sequence and the loss of the TXS platform is also assessed, since it also challenges the safety function.

2.5.5.3. R3 - Prevention of uncontrolled positive reactivity insertion into the core

This plant level safety function is bounded by other family of events such as ‘increase in heat removal’.

For the Boron Dilution fault, the lower level safety function ’Anti Dilution Protection’ is claimed. This safety function is actuated by the Protection System following a “Low Boron Concentration” signal and isolates the RCV [CVCS].

In addition, an LCO function is actuated by the RCSL following an “Insertion Limit” signal. This LCO blocks the generator power increase, prevents dilution instructions and injects boric acid. During standard shutdown states, a limitation function is actuated by the RCSL following a “Low Boron concentration” signal. This function blocks further dilution and injects boric acid that mitigates any dilution fault.

The most onerous ATWS sequence is ‘RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant with loss of Protection System’ sequence.

2.5.5.4. R4 – Maintain sufficient sub-criticality of fuel stored outside the reactor coolant system but within the site

This safety function is not challenged by these transients.

2.5.5.5. H1 - Maintain sufficient Reactor Coolant System water inventory for core cooling

This safety function is not challenged by these transients.

2.5.5.6. H2 – Remove heat from the core to the reactor coolant

This safety function is not challenged by these transients.

2.5.5.7. H3 - Transfer heat from the reactor coolant to the ultimate heat sink

This plant level safety function is bounded by another family of events such as ‘decrease in heat removal’.

2.5.5.8. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

This safety function is not challenged by these transients.

2.5.5.9. C1 - Maintain integrity of fuel cladding

This safety function is challenged but the provision of diverse protection is addressed in the analysis of the R2 function.

2.5.5.10. C2 – Maintain integrity of the reactor coolant pressure boundary

The lower level safety function ‘RCP [RCS] overpressure protection’ is provided by the safety functional group covering the pressuriser safety valves. These are passive relief valves with a high opening reliability. Diversity to this system is not provided.

An ALARP justification of the reliability of the system will be provided.

2.5.5.11. C3 – Limit the release of radioactive material from the reactor containment

This safety function is not applicable to these transients.

2.5.5.12. C4 – Limit the release of radioactive waste and airborne material

This safety function is not applicable to these transients.

2.5.5.13. O1 – Prevent the failure or limit the consequences of failure of a structure, system or component whose failure would cause the impairment of a safety function

This safety function is not applicable to these transients.

2.5.6. Other reactivity and power distribution faults

The events considered in this family of events are:

- Partial loss of core coolant flow (Loss of one reactor coolant pump),
- RCCA misalignment up to rod drop.

2.5.6.1. Partial loss of core coolant flow (Loss of one reactor coolant pump)

2.5.6.1.1. R1 – Maintain core reactivity control

This safety function is not applicable to this transient.

2.5.6.1.2. R2 - Shutdown and maintain core sub-criticality

The lower level safety function is ‘Negative Reactivity Fast Insertion’. This is provided by the safety functional group ‘reactor trip’ actuated on a “Low-low loop flow rate (in one loop)” signal from the Protection System. If the TXS I&C platform fails, a diverse signal is not provided in the SPPA-T2000 platform.

However, a limitation channel actuated by the RCSL is actuated following the loss of one reactor coolant pump. This function initiates a partial trip (fast insertion of a certain number of RCCAs) and a coincident generator power reduction.

Diversification of the lower level safety function is provided by the ‘high concentration and high pressure boron injection’ function. The extra boration system is actuated by the ATWS signal from the TXS platform. The loss of the main safety function would lead to the ATWS sequence.

2.5.6.1.3. R3 - Prevention of uncontrolled positive reactivity insertion into the core

This plant level safety function is bounded by another family of events such as ‘increase in heat removal’.

2.5.6.1.4. R4 – Maintain sufficient sub-criticality of fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to this transient.

2.5.6.1.5. H1 - Maintain sufficient Reactor Coolant System water inventory for core cooling

This safety function is not challenged by this transient.

2.5.6.1.6. H2 – Remove heat from the core to the reactor coolant

This safety function is not challenged by this transient.

2.5.6.1.7. H3 - Transfer heat from the reactor coolant to the ultimate heat sink

This plant level safety function is bounded by another family of events such as ‘decrease in heat removal’.

2.5.6.1.8. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to this transient.

2.5.6.1.9. C1 - Maintain integrity of fuel cladding

The analysis of this function is the same as for R2 function.

2.5.6.1.10. C2 – Maintain integrity of the reactor coolant pressure boundary

The lower level safety function ‘RCP [RCS] overpressure protection’ is provided by the pressuriser safety valves. They are passive relief valves with high opening reliability. A diverse system is not provided.

An ALARP justification of the reliability of the system will be provided.

2.5.6.1.11. C3 – Limit the release of radioactive material from the reactor containment

This safety function is not applicable to this transient.

2.5.6.1.12. C4 – Limit the release of radioactive waste and airborne material

This safety function is not applicable to this transient.

2.5.6.1.13. O1 – Prevent the failure or limit the consequences of failure of a structure, system or component whose failure would cause the impairment of a safety function

This safety function is not challenged by this transient.

2.5.6.2. RCCA misalignment up to rod drop

2.5.6.2.1. R1 – Maintain core reactivity control

This safety function is not applicable to this transient.

2.5.6.2.2. R2 - Shutdown and maintain core sub-criticality

The lower level safety function is ‘negative reactivity fast insertion’. This is initiated by the reactor trip following a “low DNBR” signal from the Protection System. Should failure of the TXS I&C platform occur, a diverse signal is not provided in the SPPA-T2000 platform.

However, several LCO and limitation channels actuated by the RCSL may be triggered following a rod drop.

Diversity for the lower level safety function is provided by the ‘high concentration and high pressure boron injection’ function. The extra boration system is actuated by the ATWS signal from the TXS platform. The loss of the LLSF ‘high concentration and high pressure boron injection’ would lead to the ATWS sequence.

2.5.6.2.3. R3 - Prevention of uncontrolled positive reactivity insertion into the core

This plant level safety function is bounded by another family of events such as ‘increase in heat removal’.

2.5.6.2.4. R4 – Maintain sufficient sub-criticality of fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to this transient.

2.5.6.2.5. H1 - Maintain sufficient Reactor Coolant System water inventory for core cooling

This safety function is not challenged by this transient.

2.5.6.2.6. H2 – Remove heat from the core to the reactor coolant

This safety function is not challenged by this transient.

2.5.6.2.7. H3 - Transfer heat from the reactor coolant to the ultimate heat sink

This plant level safety function is bounded by another family of events such as ‘decrease in heat removal’.

2.5.6.2.8. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to this transient.

2.5.6.2.9. C1 - Maintain integrity of fuel cladding

This safety function is not applicable to this transient.

2.5.6.2.10. C2 – Maintain integrity of the reactor coolant pressure boundary

The lower level safety function 'RCP [RCS] overpressure protection' is ensured by the pressuriser safety valves. They are passive relief valves with high opening reliability. This system is not diversified.

An ALARP justification of the reliability of the system will be provided.

2.5.6.2.11. C3 – Limit the release of radioactive material from the reactor containment

This safety function is not challenged by this transient.

2.5.6.2.12. C4 – Limit the release of radioactive waste and airborne material

This safety function is not challenged by this transient.

2.5.6.2.13. O1 – Prevent the failure or limit the consequences of failure of a structure, system or component whose failure would cause the impairment of a safety function

This safety function is not challenged by this transient.

2.5.6.3. Summary for reactivity events

The summary of the plant level safety functions used for transient mitigation is provided in Sub-chapter 16.5 – Table 8. Function O1 is not presented in the summary tables as the diversity is sufficient.

2.5.7. Fuel pool transients

The fuel rack designed to maintain the stored fuel assemblies sub-critical provides the R4 plant level safety function. This function is consequently not analysed further.

2.5.7.1. Loss of one train of the Fuel Pool Cooling System (PTR [FPCS]) or of a supporting system (state A)

2.5.7.1.1. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

The lower level safety function is fuel pool heat removal. It is provided by the safety functional group 'manual start of the other PTR [FPCS] main train' following detection of a fuel pool water temperature increase using an analogue measurement. If the second train fails, the diverse PTR [FPCS] third train can be manually started and is cooled by a diverse heat sink.

Water make-up to the fuel pool using the Classified Fire Fighting Water Supply System (JAC/JPI [NIFPS]) is also available to maintain the water level in the fuel pool should boiling occur.

2.5.7.2. Long-term LOOP, fuel pool cooling aspects (state A)

2.5.7.2.1. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

The lower level safety function is fuel pool heat removal. It is provided by the safety functional group ‘manual start of a PTR [FPCS] main train’ once the corresponding Emergency Diesel Generator (EDG) has automatically started. If the second train or the associated EDG fail, the diverse PTR [FPCS] third train can be manually started and is cooled by a diverse heat sink.

Water make-up to the fuel pool by Classified Fire Fighting Water Supply System (JAC/JPI [NIFPS]) is also available to maintain the water level in the fuel pool should boiling occur.

2.5.7.3. Isolatable piping failure on a system connected to the spent fuel pool - Draining via the RCV [CVCS] draining line (state E)

2.5.7.3.1. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

The lower level safety function is fuel pool heat removal. It is provided by the safety functional group ‘manual isolation of the RCV [CVCS] drain line’ that was inadvertently opened. The time delay of more than 3 hours occurs before the water level falls to {CCI} [a] and the operating PTR [FPCS] pumps is automatically switched off .Sufficient time is available for the operator to carry out diagnosis and to locally close the RCV [CVCS] drain line.

Should the level continue to fall, automatic detection of a water level in the reactor building transfer compartment below {CCI} [a] results in the automatic isolation of the RIS/RRA [SIS/RHR] suction line. This stops the discharge via the RCV [CVCS] drain line (isolation of the line by one valve closing). Detection of the draining using the analogue spent fuel pool water level measurement and manual isolation of the RIS/RCV [SIS/CVCS] drain line is also possible.

Spent Fuel Pool cooling was lost when the water level reached {CCI} [a]. Consequently, water make-up to the fuel pool by Classified Fire Fighting Water Supply System (JAC/JPI [NIFPS]) must be activated to increase the water level in the fuel pool to the required level {CCI Removed} [a]. This allows the manual start-up of a PTR [FPCS] main train.

2.5.7.4. Isolatable piping failure on a system connected to the spent fuel pool - Voluntary draining of the reactor building pool, spent fuel pool not isolated (state D or state F)

2.5.7.4.1. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

The lower level safety function is ‘fuel pool heat removal’. It is provided by the safety functional group ‘manual closing of the reactor building pool drain lines isolation valves’. This action follows the identification of the inadvertent draining via the automatic detection of a low water level in the spent fuel pool {CCI Removed} [a]. this terminates the draining phase before the loss of fuel pool heat removal system.

Should this action fail, automatic detection of a low water level in the spent fuel pool {CCI Removed} [a] results in the automatic trip of the PTR [FPCS] purification pumps and stops the draining before the loss of the fuel pool heat removal system. Detection of the draining using the analogue spent fuel pool water level measurement and manual PTR [FPCS] purification pumps trip or manual closing of the transfer tube isolation valve are other diverse lines of protection.

2.5.7.5. Isolatable piping failure on a system connected to the spent fuel pool - Inadequately prepared transfer between the loading pit and the fuel building transfer compartment (state A to D)

2.5.7.5.1. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

The lower level safety function is fuel pool heat removal. It is provided by the automatic detection of a low water level in the spent fuel pool {CCI Removed} [a]. This leads to the automatic closure of the fuel building pool drain lines isolation valves which halts the draining before the loss of the fuel pool heat removal system.

Should this protection fail, an automatic trip of the PTR [FPCS] purification pumps occurs following a “low spent fuel pool water level” signal {CCI Removed} [a]. This halts the draining of the fuel pool before the loss of the fuel pool heat removal system.

Diverse protection following the draining of the fuel pool is provided by the spent fuel pool water level analogue measurement followed by:

- manual PTR [FPCS] purification pump trip,
- or water make-up to the fuel pool using the Classified Fire Fighting Water Supply System (JAC/JPI [NIFPS]) with the manual start of a PTR [FPCS] main train.

2.5.7.6. Summary of fuel pool transients

The summary of the plant level safety functions used for transient mitigation is provided in Sub-chapter 16.5 – Table 9.

The ‘Spent fuel pool draining via the RCV [CVCS] unloading line (state E)’ case is the most onerous of the fuel pool frequent events. This event is used to demonstrate the diversity available for the H4 safety function.

Justification is based on the following analysis:

- Failure of the main line does not lead to any fuel pool cooling loss following a deliberate draining of the reactor building pool, spent fuel pool not isolated (state D) or an inadequately prepared transfer between the loading pit and the fuel building transfer compartment (state A to D).
- Loss of one train of the fuel pool cooling system (PTR [FPCS]) or of a supporting system (state A) and the long-term LOOP for the fuel pool cooling aspects (state A) leading to a loss a fuel pool heat removal in state A, where the decay heat in the fuel pool is about four times lower than in state E.

- Failure of the main line isolation following draining via the RCV [CVCS] unloading line (state E) leads to a loss of the fuel pool heat removal in state E and requires make up before the start of the PTR [FPCS] main train can occur. This is the bounding case because of the shortest grace time and the time required for the diverse line to be initiated.

2.5.8. Miscellaneous

The frequent events considered in this family of events are:

- Spurious actuation of pressuriser spray leading to a decrease in RCP [RCS] pressure,

- Spurious actuation of pressuriser heaters leading to an increase in RCP [RCS] pressure.

These transients are events affecting the reactor coolant system pressure.

2.5.8.1. R1 – Maintain core reactivity control

This safety function is not challenged by this transient.

2.5.8.2. R2 - Shutdown and maintain core sub-criticality

The lower level safety function is ‘negative reactivity fast insertion’. It is provided by the safety functional group ‘reactor trip’ following a “pressuriser pressure < MIN2” signal, a “low DNBR” signal or a “pressuriser pressure > MAX2” signal from the Protection System. Should the TXS I&C platform fail, the reactor trip occurs following a ”Hot leg pressure (WR) < MIN2” signal in the SPPA-T2000 platform. The diverse parameter for the signal on high pressuriser pressure is not provided. A manual reactor trip from the SAS can be actuated in this case.

Diversity for the lower level safety function is provided by the ‘high concentration and high pressure boron injection’ function. The extra boration system is actuated by the ATWS signal from the TXS platform. The loss of the LLSF ‘high concentration and high pressure boron injection’ would lead to the ATWS sequence.

The more onerous case is the ‘Spurious actuation of pressuriser spray’ as the pressure in the RCP [RCS] decreases and this results in a challenge to the DNBR criteria. This case is more onerous for the 'shutdown and maintain sub-criticality' function and the plant level safety function C1 – Maintain integrity of fuel cladding.

For completeness, the spurious actuation of the pressuriser spray combined with the loss of the TXS platform is also considered as a bounding transient.

2.5.8.3. R3 - Prevention of uncontrolled positive reactivity insertion into the core

The lower level safety function is ‘RCP [RCS] overcooling protection’. It is provided by the safety functional groups ‘turbine trip’ and ‘full load MFW isolation’. These actions are actuated by the Protection System following the reactor trip.

Should failure of the TXS occur, the reactor trip signal from the SPPA-T2000 also initiates the turbine trip and the full load MFW isolation. The reactor trip signal following a “Cold leg temperature < MIN1” signal also automatically closes the main steam isolation valves.

From the PSLF R3 standpoint, the diversity for this function is provided by the VIV [MSIV] closure, which is performed automatically by the RPR [PS], or automatically by the SPPA-T2000 platform following a “low cold leg temperature” signal, or manually. The VIV [MSIV] closure occurs via the RPR [PS] either on a “SG pressure drop > MAX1” or a “SG pressure < MIN1” signal. Diverse actuation of the full load MFW isolation, which differs from the one above by its F2 classification, can be performed automatically by the TXS platform or manually. This function operates via two redundant isolation valves. Following a “high SG level (> MAX0p)” signal, the low load isolation valve is closed and, if a reactor trip has already occurred, the “SG level > MAX1p” signal results in the closure of the full load, the low load and the main isolation valves in the MFW.

This safety function is not significantly challenged by these transients.

2.5.8.4. R4 – Maintain sufficient sub-criticality of fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to these transients.

2.5.8.5. H1 - Maintain sufficient Reactor Coolant System water inventory for core cooling

This safety function is not challenged by these transients as the RCP [RCS] water inventory does not vary.

2.5.8.6. H2 – Remove heat from the core to the reactor coolant

This safety function is not challenged by these transients.

2.5.8.7. H3 - Transfer heat from the reactor coolant to the ultimate heat sink

The lower level safety function ‘heat removal by steam generators – emergency shutdown mode’ is performed by the emergency feedwater and the main steam relief trains. This safety function is used throughout the transfer to a stable state. However, there is no challenge to the heat transfer capacity as the pressure transients presented have no impact on this function.

Following closure of the turbine inlet valve and of the main steam bypass, heat transfer to the ultimate heat sink is performed by the VDAs [MSRT]s. The diverse function is provided by the MSSVs, which are passive relief valves. The two MSSVs together (per SG) have the same capacity as the VDA [MSRT]. The heat can therefore be removed by the steam generators using either route.

The diverse lower level safety function is provided by the ‘Heat removal by Low Head Emergency Core Cooling System (ECCS)’ function, which uses the safety injection (MHSI, LHSI and accumulators) and the severe accident discharge line. It is actuated manually from the SAS.

2.5.8.8. H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site

This safety function is not applicable to these transients.

2.5.8.9. C1 - Maintain integrity of fuel cladding

The lower level safety function 'R2 – Shutdown and maintain core sub-criticality' demonstrates that the spurious pressuriser spray event with failure of the reactor trip may affect this safety function. Functional diversity is provided within this R2 PLSF.

2.5.8.10. C2 – Maintain integrity of the reactor coolant pressure boundary

This safety function is challenged by the spurious pressuriser heaters actuation event as it results in the opening of the pressuriser safety valves. Auxiliary and normal spray are sufficient to compensate for the overpressure due to the actuation of the heaters following to a short pressure increase.

2.5.8.11. C3 – Limit the release of radioactive material from the reactor containment

This safety function is not challenged by these transients.

2.5.8.12. C4 – Limit the release of radioactive waste and airborne material

This safety function is not challenged by these transients.

2.5.8.13. O1 – Prevent the failure or limit the consequences of failure of a structure, system or component whose failure would cause the impairment of a safety function

The lower level safety function 'Essential component protection' is provided by the VDAs [MSRT]s. The MSSVs provide the diverse function to the VDA [MSRT] as described above.

2.5.8.14. Summary for miscellaneous events

The summary of the plant level safety functions used in transient mitigation is provided in Sub-chapter 16.5 – Table 10.

2.5.9. Support systems

2.5.9.1. Electrical supply

The systems needing electrical supply are supplied by the grid as a normal source of power. A diverse power supply is provided by the emergency diesel generators. They supply the following safety systems (list not exhaustive):

- MHSI pumps,
- LHSI pumps,
- RCV [CVCS] charging pumps,
- ASG [EFWS] pumps,
- Component Cooling Water System (RRI [CCWS]) pumps,
- Essential Service Water System (SEC [ESWS]) pumps,

- RBS [EBS] pumps.

A third diverse line is provided by the Ultimate Diesel Generators (UDG), also called the Station Black-Out (SBO) diesel generators.

Therefore, the loss of off-site power is the limiting event for the electrical supply with the first line of defence being the emergency diesel generators. The adequacy of the SBO diesel generators is demonstrated by the Station Black-Out analysis. The analysis is described below.

Note: The reactor trip is independent of electrical power supply as the loss of electrical power leads to the release of the breakers.

2.5.9.2. Cooling chain

Diversity within the cooling chain is provided by:

- Component Cooling Water System/Essential Service Water System
- Diverse cooling chain using the safety chilled water system (PCSR Sub-chapter 9.2)

Therefore, the bounding scenario following the loss of the RRI/SEC [CCWS/ESWS] is the total loss of cooling chain, which is analysed below.

2.5.9.3. Instrumentation and Control

The diversity at the I&C level is provided by the diversity of platforms. The TXS platform performs action from the Protection System (RPR [PS]). The SPPA-T2000 performs actions from the Safety Automation system (SAS). It ensures functional diversity regarding the I&C.

In addition to this, a further level of diversity is provided by the Non-Computerised Safety System (NCSS) for a total loss of digital I&C. In the event of a total loss of computerised I&C, the NCSS supports functional diversity [Ref-1]; details are provided in PCSR Sub-chapter 7.4. However, such loss is not considered in the present analysis as the total loss of digital I&C is not a frequent fault.

The relevant diversity will be demonstrated during the accident analysis.

2.5.9.4. Diversity of sensors

The requirements for the instrumentation are listed in PCSR Sub-chapter 7.6. The diversification of sensors and independence of information is presented in the functional analysis for common cause failure of sensors [Ref-1].

2.5.9.5. Diversity analyses for loss of essential support systems

This sub-chapter demonstrates the provision of diverse protection for frequent faults involving a loss of essential support systems (e.g. loss of cooling chain, electrical, HVAC)", and demonstrates that any diverse systems claimed are appropriately categorised.

The analysis of failure of the essential support systems shows that five initiating events are identified as frequent faults:

- Mechanical failure of an RRI/SEC [CCWS/ESWS] train,
- Leak/break in an RRI [CCWS] train,
- Leak/break in a RRI [CCWS] common auxiliary,
- Loss of a Safety Chilled Water System (DEL [SCWS]) or a Safeguard Building Uncontrolled Area Ventilation System (DVL [SBVSE]) train,
- Loss of an electrical switchboard.

The consequences of these different PIEs lead to a range of reactor transients, and some bounding plant transients are identified.

Two additional Design Basis Events are studied within the diversity studies:

- two reactor coolant pumps trip event with a lost safeguard division.
- four reactor coolant pumps leak event with a lost safeguard division.

These two reactor plant transients are studied in a similar manner to existing Design Basis Events for the diversity demonstration. Firstly, the two initiating events are analysed until reaching a controlled state. Then, from the controlled state to the final state, a generic demonstration is provided (section 2.5.9.5.3).

2.5.9.5.1. Two reactor coolant pumps trip event with a lost safeguard division

The different Lower Level Safety Functions (and associated Safety Functional Groups) identified as main and diverse lines to manage the PIE “Partial loss of core coolant flow (Loss of one reactor coolant pump)” and “Forced decrease of reactor coolant flow (four pumps)” are identical to the LLSF used to manage a 2 reactor coolant pumps trip event.

Every diverse LLSF is ensured by Safety Functional Groups which have been checked and confirmed to still be operable with a safeguard division lost.

2.5.9.5.2. Seal LOCA events

The different Lower Level Safety Functions (and associated Safety Functional Groups) identified as main and diverse lines to manage the PIE “Small break LOCA (< DN 50) including a break occurring on the extra boration system injection line (states A and B)” are identical to the LLSF used to manage a leak at 4 reactor coolant pumps event.

Every diverse LLSF is ensured by Safety Functional Groups which have been checked and confirmed to still be operable with loss of a safeguard division.

2.5.9.5.3. From controlled state to final state

In the event of the loss of an electrical and I&C division, one train is lost but the final state can be reached by using feed and bleed with three RIS [SIS] trains and the Primary Depressurisation System (PDS) valves.

Should the Low Head Safety Injection (LHSI) be unavailable, the Containment Heat Removal System (EVU [CHRS]) is needed to remove RCP [RCS] heat in addition to the secondary side.

The adequacy of the functional diversity in the EPR design is consequently demonstrated for frequent faults involving the loss of essential support systems.

2.5.10. Loss of RCV [CVCS] faults

There are two automatic systems available on the EPR to provide protection following loss of RCV [CVCS] after reactor trip, in addition to operator action. The first is a high neutron flux signal on the source range detector, which is claimed to automatically actuate the RBS [EBS]. The second is a limitation function on the RCSL which also automatically actuates the RBS [EBS] when it determines that the boron concentration is about to fall below the critical boron concentration.

An ALARP analysis [Ref-1] has been carried out to review the reasonably practicable options for providing diverse protection against an RCV [CVCS] malfunction resulting in a decrease in boron concentration in the reactor coolant in a shutdown state. A UK EPR design change has been raised in order to carry forward the investigation of the design options presented in the ALARP analysis into the site licensing phase, to ensure that a diverse protection function is implemented in the design of the UK EPR.

2.5.10.1. Automatic actuation of RBS [EBS]

The function “high neutron flux (source range)” initiating boration on detection of the source range detectors high neutron flux is described in PCSR Sub-chapter 16.1. It is an RRC-A feature (actuated via the RPR [PS]) to mitigate non RCV [CVCS] homogeneous dilution with operator failure to isolate the dilution source in states C and D.

The second means of boration called “Anti-dilution in shutdown states” does not occur through the EBS [RBS]. The signal is transferred to the Reactor Boron and Water Make up System (REA [RBWMS]), which orders both REA [RBWMS] boration lines to start the pump and to fully open the valve. The second RCV [CVCS] charging pump start-up command is sent by the REA [RBWMS] to the RCV [CVCS]. The “block demineralised water injection” demand is also activated, which isolates the main source of dilution from the REA [RBWMS].

This signal is not considered here, as it is a part of the reactor control and surveillance system, which is not included in the GDA scope.

In summary, two diverse protection channels exist to mitigate a fault in the RCV [CVCS] after reactor trip.

- The “high neutron flux (source range)” channel actuates the EBS [RBS] on a high neutron flux signal via the RPR [PS],

- The “Anti-dilution in shutdown states” from the REA [RBWMS] prevents dilution from the RCV [CVCS] and borates the reactor coolant system using the REA [RBWMS]. The REA [RBWMS] uses the RCV [CVCS] lines and pumps. Thus, this action prevents a boron dilution originating in the RCV [CVCS]. It is routed via the RCSL.

The plant is protected in the event of a loss of RCV [CVCS] boration by the “high neutron flux (source range)” protection, which leads to the automatic RBS [EBS] actuation.

2.5.10.2. Source range detectors

The Source Range Detectors (SRDs) are designed to monitor the neutron flux in sub-critical conditions representative of shutdown states, and for use during the approach to criticality for reactor start-up.

The permissive interlock to restore SRD functionality upon return to the sub-critical state is ensured by the Intermediate Range Detector (IRD) based on a permissive value of neutron flux corresponding to a power level defined between 10-^{6} and 10-^{7} P/Pr. Shortly after reactor trip, the core becomes sub-critical and the flux decreases dramatically. Once the value of the neutron flux is below the permissive value, the protective actions linked to the SRD are automatically restored and the SRD remains in operation until manual action is taken. After this time, the neutron population consists only of intrinsic sources (e.g. spontaneous decay of Curium, etc.) and any primary or secondary neutron sources (used for the first few cycles to ensure adequate neutron source for the SRDs). The resulting neutron flux corresponds to a core power level in the range of 10^{-10} P/Pr or 10^{-8} P/Pr. Therefore, the neutron flux levels around 16 to 20 hours after reactor trip will be monitored by the SRDs.

2.5.11. Emergency Operating Procedures

This section assesses the diverse means available to reach a stable state from the controlled state for all the frequent faults. The controlled state is characterised by short-term heat removal capacity, core sub-criticality and stable core coolant inventory. Hence, the starting point of the analysis is that the controlled state has been reached following the postulated initiating event (PIE) and that no failure of any system has occurred up to that point, other than the system leading to the PIE. Consequently, the three main safety functions, i.e. reactivity control, heat removal and containment, are already ensured – notably the confinement of radioactive materials.

The following analysis first explains the necessary steps to connect to the RRA [RHRS] and then reviews all the frequent events, by event family, to perform the demonstration of diversity. As the emergency operating procedures described in the PCC fault analyses correspond to the safe path, the analysis intends to demonstrate that the feed and bleed procedure is adequate to provide the diversity of the safe path. Subsequently, the objective is not necessarily to demonstrate a safe shutdown state on RRA [RHRS] but it is to achieve a long-term non-hazardous stable state, in which the core is sub-critical, residual heat is removed by primary or secondary systems and off-site radioactive discharges remain acceptable.

2.5.11.1. RRA [RHRS] Connection

The RRA [RHRS] connecting conditions are reached when the reactor coolant system hot leg temperature is lower than 180°C and the hot leg pressure is lower than 32 bar abs.

Four actions must be performed to reach the safe shutdown state from the controlled state:

- RCP [RCS] boration to ensure the core is sub-critical as the temperature and the pressure in the reactor coolant system decrease to reach RRA [RHRS] connecting conditions
- RCP [RCS] depressurisation
- RCP [RCS] cooldown
- Connection to the RRA [RHRS].

The analysis is performed for the frequent postulated initiating events (PIEs).

The operational system used to perform the boration is the RCV [CVCS]. It is F2 and therefore cannot be credited in the safety analyses. The RBS [EBS] is the safety classified means of performing the boration to ensure long-term sub-criticality. Should the RBS [EBS] fail, the boration can be performed by a combination of F1 and F2 systems. For instance, boration can be performed by the feed and bleed procedure, which uses the safety injection system RIS [SIS] and the severe accident discharge lines.

The operational system used to perform the depressurisation of the reactor coolant system is the normal or the auxiliary spray. The depressurisation of the reactor coolant system can also be performed by the safety classified Pressuriser Safety Valves (PSV). Should the pressuriser safety valves fail, the depressurisation of the reactor coolant system can be performed by the PDS. However, the PDS actuation requires the use of the RIS [SIS] to compensate for the flow lost through the PDS and to maintain the RCP [RCS] inventory.

The cooldown of the reactor coolant system is performed during normal operation by the main steam bypass. The safety classified means to cooldown the reactor coolant system are the VDAs [MSRT]s. Should they fail, the feed and bleed procedure is actuated if the plant situation cannot be stabilised without the VDAs [MSRT]s.

The connection to RRA [RHRS] can be performed only if the LHSI pumps are available, as one LHSI train is necessary to ensure the residual heat removal. Should the LHSI pumps not be available, the residual heat can be removed via the steam generators fed by the ASG [EFWS] and using the VDAs [MSRT]s. MHSI may be required to maintain the reactor coolant system inventory.

The use of a diverse line may not always be necessary as the plant can be maintained in a controlled state. In particular, this is true in cases where:

- the automatic actions mitigate the PIE, or
- the integrity of the reactor coolant system and the secondary side are not impaired so that their inventories can be maintained after a controlled state is reached.

In these cases, the residual heat is removed by the secondary side in the long term because the steam generator water inventory is high and the reactor coolant system inventory is stable. Typically, the case of a spurious reactor trip falls into this category and is not analysed here any further.

The water volume in the ASG [EFWS] tanks provide sufficient inventory to ensure heat removal for 24 hours at hot shutdown. The ASG [EFWS] tanks can be refilled with water during that time, via the JAC (fire fighting water supply) system.

2.5.11.2. Frequent Postulated Initiating Events

Frequent PIEs are analysed in the following sections in the light of the arguments provided above to demonstrate that a suitably classified diverse line exists to bring the plant to a long-term non-hazardous stable state.

2.5.11.2.1. Increase in RCP [RCS] inventory

The postulated initiating event considered is:

- RCV [CVCS] malfunction causing an increase in reactor coolant inventory

For this event, the diverse line is not necessary as the plant is in a stable state after mitigation of the initiating event by automatic actions, such as isolation of the RCV [CVCS] charging line on high pressuriser level.

In a similar manner to the situation after a reactor trip, the plant is stabilised and can subsequently be maintained in a long-term stable state.

2.5.11.2.2. Decrease in RCP [RCS] inventory

The postulated initiating events considered are:

- RCV [CVCS] malfunction causing decrease in reactor coolant inventory (state A)
- Inadvertent opening of a pressuriser safety valve
- Small break (not greater than DN 50), including a break occurring on the Extra Boration System injection line (State A)

In the case of the RCV [CVCS] malfunction, the initiating event is mitigated by automatic actions, such as letdown isolation. The plant can remain in the hot shutdown condition in the long term, even in the event of failure of a system needed to reach the safe shutdown state. Moreover, reaching the RRA [RHRS] connecting conditions under these conditions is bounded by the SB LOCA case.

The case of the inadvertent opening of a pressuriser safety valve is similar to that of the SB LOCA in the phase from the controlled state to the safe shutdown state since, at that stage, the PSV opening discharges steam in a similar manner to the SB LOCA case.

Therefore, the analysis is carried out for the case of the SB LOCA. The controlled state is reached when the MHSI flow compensates for the break flow rate and the RCP [RCS] inventory is stable. This ensures sufficient boration in the reactor coolant system. The table below presents the main and diverse lines of the different stages necessary to reach the safe shutdown state.

	Main line	**Diverse line**	**Comments**
RCP [RCS] boration	Emergency boron injection into the core – Manual	RIS [SIS] + PDS	The efficiency of the feed and bleed procedure in the case of SB LOCA has been demonstrated
RCP [RCS] cooldown	SG Pressure Control – Cooling (VDA [MSRT])	Not necessary	Heat removal is ensured by the break flow rate. The efficiency of the feed and bleed procedure in the case of SB LOCA without VDAs [MSRT]s has been demonstrated.

	Main line	Diverse line	Comments
RRA [RHRS] connection	Stop MHSI (1 train) - manual	No diversity	In the event of failure of the MHSI to stop, the RRA [RHRS] cannot be connected, but the MHSI and EVU [CHRS] can be used to ensure the core remains covered and to provide heat removal from the In-containment Refuelling Water Storage Tank (IRWST).
RRA [RHRS] connection	LHSI switch to RHR mode (1 train)	MHSI + EVU [CHRS] +VDA [MSRT] + ASG [EFWS]	In the event of failure of LHSI, the MHSI and the EVU [CHRS] are used to ensure the core remains covered and to provide heat removal from the IRWST.

Main and Diverse Lines to Reach Safe Shutdown State

2.5.11.2.3. Increase in heat removal

The postulated initiating events considered are:

- Feedwater malfunction - causing a reduction in feedwater temperature
- Feedwater malfunction – causing an increase in feedwater flow rate
- Excessive increase in steam flow
- Inadvertent opening of a SG relief train (state A)

For the feedwater malfunction events, as soon as the initiating event is mitigated, by reactor trip and full load main feedwater isolation, the plant is stabilised. The transfer to the safe shutdown state is bounded by the excessive increase in steam flow event.

The inadvertent opening of a SG relief train is mitigated by the closure of the main steam relief train control valve. The plant is then stabilised and heat can be removed by the remaining VDAs [MSRT]s. The transfer to the safe shutdown state is performed in the same way as other events in which the integrity of the reactor coolant system or the steam lines is not impaired.

The table below presents the main and diverse lines used to reach a safe shutdown state.

	Main line	Diverse line	Comments
RCP [RCS] boration	Emergency boron injection into the core - Manual	RIS [SIS] + PDS	
-	Low Load ARE [MFWS] isolation (4 SGs) – manual	Not necessary	Main feedwater isolation (main feedwater isolation valve closure) can be used if necessary. This system is not critical since the plant could operate with the Low Load ARE [MFWS] feeding the SG to remove heat.
RCP [RCS] cooldown	SG Pressure Control – Cooling (VDA [MSRT])	RIS [SIS] + PDS	
-	Steam line isolation (1 SG) – auto	No diversity	Steam line isolation may be necessary, depending on the initiating event. In the case of FW malfunction, ARE [MFWS] isolation is sufficient, the transient has been stopped and no further actions are required. In the event of non isolation of the excessive steam line flow, the other SGs must be isolated. No Common Cause Failure (CCF) is postulated on the VIVs [MSIV]s [Ref-1], therefore the secondary side can be used to cool down the RCP [RCS].
RCP [RCS] depressurisation	RCP [RCS] depressurisation by Pressuriser safety valves	RIS [SIS] + PDS	The depressurisation can be performed by the PDS, leading to the feed and bleed.
RRA [RHRS] connection	RRA [RHR] connection and start-up (no SI signal)	VDA [MSRT] + ASG [EFWS]	If LHSI is not available, ASG [EFWS] + VDA [MSRT] are used to remove heat, eventually leading to the need to refill the ASG [EFWS] tanks for long-term mitigation. However, the problem may come from the steam line isolation, depending on the initiating event. In case of non isolation of the excessive steam line flow, the other SGs must be isolated. No CCF is postulated on the VIVs [MSIV]s [Ref-1], therefore the secondary side can be used to cool the RCP [RCS] down.

Main and Diverse Lines to Reach a Safe Shutdown State

2.5.11.2.4. Decrease in heat removal

The postulated initiating events considered are:

- Turbine trip
- Loss of condenser vacuum
- Loss of normal feedwater flow
- Small feedwater system piping failure
- Inadvertent closure of one or all main steam isolation valves

The turbine trip is bounded by the loss of condenser vacuum. The latter event leads to a reactor trip. Under these conditions, the plant can remain in this hot shutdown state as long as boration is performed and water is provided to the steam generators.

The inadvertent closure of one or all VIVs [MSIV]s event leads to a similar scenario with the additional isolation of the steam generators. After reactor trip, the plant can remain in the hot shutdown state as long as water is provided to the steam generators.

The bounding event is the loss of main feedwater, due to the lower steam generator inventory. Regarding the loss of main feedwater, if the RBS [EBS] is unavailable, the plant is stabilised by the use of the ASG [EFWS] and VDA [MSRT]. The transfer to the safe shutdown state is not necessary as the ASG [EFWS] tank can be supplied with additional water to remove the heat from the reactor coolant system.

Should the PSVs fail, the increase in RCP [RCS] heat is removed by the secondary side as demonstrated by the analysis of closure of the 4 VIVs [MSIV]s without PSV in the short term. In the long term, as the failure is postulated on the PSVs, the RCP [RCS] heat can be removed by the ASG [EFWS] and the VDAs [MSRT]s. The plant is stabilised in this configuration. The same is true if there is a failure on the LHSI (RHR). The plant is stabilised in a non-hazardous stable state and maintenance can be performed on the equipment.

The table below presents the main and diverse lines used to reach a safe shutdown state.

	Main line	**Diverse line**	**Comments**
RCP [RCS] boration	Emergency boron injection into the core – Manual	RIS [SIS] + PDS	Cooldown is necessary to perform RIS [SIS] injection and reach adequate boron concentration.
RCP [RCS] cooldown	SG Pressure Control – Cooling (VDA [MSRT])	RIS [SIS] + PDS	Demonstration of the feed and bleed procedure in case of Total Loss of Feedwater (TLOFW)
RCP [RCS] depressurisation	RCP [RCS] depressurisation by pressuriser safety valves	RIS [SIS] + PDS	The depressurisation can be performed by the PDS, leading to the feed and bleed procedure.

	Main line	**Diverse line**	**Comments**
RRA connection	RHR connection and start-up (no SI signal)	VDA [MSRT] + ASG [EFWS]	If LHSI are not available, ASG [EFWS] +VDA [MSRT] are used to remove heat, eventually leading to the need to refill the ASG [EFWS] tanks for long-term mitigation.

Main and Diverse Lines to reach a Safe Shutdown State

2.5.11.2.5. Electrical power supply fault

The postulated initiating event considered is:

- Short-term loss of off-site power (LOOP)

This case is similar to those identified above, as the short-term LOOP leads to a decrease in heat removal. Therefore, section 2.5.11.2.4 presents the main and diverse lines used to reach a safe shutdown state.

Moreover, the additional failure that can be combined with the LOOP is the loss of the EDGs. It is demonstrated in PCSR Sub-chapter 16.1 that, in the event of Station Black Out, the safe shutdown state can be reached. Additional details are provided below.

To reach the safe shutdown state, the systems used must be supplied by power in the long term. This is true for:

- RBS [EBS]
- ASG [EFWS]
- RIS [SIS]

The VDAs [MSRT]s and PSVs (solenoid pilots) are supplied by two-hour batteries.

Moreover, the two pilots of the third PSV are supplied by electrical divisions 1 and 4, which are supplied by SBO diesel generators. The two pilots must open to actuate the safety valve.

Similarly, the main steam relief control valves of steam lines 1 and 4 are supplied by SBO diesel generators.

The LOOP leads to the loss of the reactor coolant pumps, Main Feedwater System (ARE [MFWS] and turbine trip. Therefore, the ASG [EFWS] and VDAs [MSRT]s are necessary to mitigate the event. The heat exchange in the steam generators ensures that the reactor coolant flows by natural circulation due to the temperature difference between the core and the steam generators. Therefore, heat removal is ensured as long as the water inventory in the steam generator is sufficient. The plant can be stabilised in the hot shutdown state.

2.5.11.2.6. Steam generator tube rupture

In the case of steam generator tube rupture, the controlled state is reached when the leak is compensated by RCP [RCS] water make-up. In the fault studies, SGTR scenarios are demonstrated up to the end of the short-term phase where the SGTR leak flow rate is terminated by establishing a pressure balance between the RCP [RCS] and the affected SG.

The plant can remain in this condition in the long term, without further actions. Boration has been performed by the RIS [SIS], or the RCV [CVCS] if it is operational, which is more onerous because it prevents RIS [SIS] actuation.

The reactor coolant system inventory is stable due to the RCV [CVCS] or the RIS [SIS], and three steam generators contribute to the residual heat removal.

Radioactive releases are stabilised by isolation of the affected SG. If operational systems are available, they can be used to perform their functions. Therefore, the plant can remain in a long-term stable state. Cooldown occurs without intervention in the steam generators due to heat losses and causes the temperature of the reactor coolant system to reduce naturally, allowing the possibility of repairing the impaired systems.

2.5.11.2.7. Reactivity transients

The postulated initiating events considered are:

- Uncontrolled RCCA bank withdrawal at power
- Uncontrolled RCCA bank withdrawal from Hot Zero Power (HZP)
- RCCA misalignment up to control rod drop
- Start-up of an inactive reactor coolant loop at an incorrect temperature
- RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant
- Uncontrolled single RCCA withdrawal

These events lead to a reactor trip and do not impair the integrity of the reactor coolant system and the secondary side so that their inventories can be maintained after the controlled state is reached. Therefore, the non-hazardous stable state can be maintained in the long term.

2.5.11.3. Conclusions

The use of diverse means to reach the long-term non-hazardous stable state has been demonstrated, when necessary, for all the frequent initiating events. The feed and bleed procedure is used in most cases to ensure depressurisation, cooldown and boration. In the other cases, the plant can remain in a long-term non-hazardous stable state.

2.6. SUMMARY

2.6.1. Transient selection

Sub-chapter 16.5 – Table 11 presents for each plant level safety function which event (per family) is bounding or indicates that an ALARP justification must be performed.

Therefore, the list of transients to be studied is the following:

- Reactivity control
  - R1: normal operation, hence no associated transient is identified
  - R2:
    - ATWS by mechanical blockage of the rods and by loss of RPR [PS] - SB LOCA,
    - ATWS by mechanical blockage of the rods and by loss of RPR [PS] - loss of main feedwater,
    - ATWS by mechanical blockage of the rods and by loss of RPR [PS] - excessive increase in steam flow,
    - ATWS by mechanical blockage of the rods and by loss of RPR [PS] - LOOP,
    - ATWS by loss of TXS - excessive increase in steam flow.
  - R3:
    - Excessive increase in steam flow without Main Steam Isolation Valves,
    - ATWS by failure of the Protection System - RCV [CVCS] malfunction which results in a decrease in boron concentration in the reactor coolant. This event is bounding as the SGTR without main feedwater isolation triggers a lower overcooling than the excessive increase in steam flow. The ALARP justification concerning the VIV [MSIV] is performed for function C4.
  - R4: this safety function is not challenged by fuel pool events
- Heat removal
  - H1:
    - SB LOCA without MHSI,
    - SB LOCA without partial cooldown.
    - SB LOCA without MSRT [VDA]
  - H2:
    - loss of feedwater with loss of reactor coolant pumps
  - H3:
    - Total loss of feedwater

- o H4:
  - Draining via the RCV [CVCS] unloading line (state E)

- Containment
  - o C1:
    - ATWS by mechanical blockage of the rods and by loss of TXS platform - uncontrolled RCCA bank withdrawal at power
    - ATWS by mechanical blockage of the rods and by loss of TXS platform - forced decrease in reactor coolant flow
    - ATWS by mechanical blockage of the rods - RCV [CVCS] malfunction which results in a decrease in boron concentration in the reactor coolant
    - ATWS by mechanical blockage of the rods and by loss of PS - spurious pressuriser spray
  - o C2:
    - Inadvertent closure of four VIVs [MSIV]s without PSVs (with and without pressuriser spray)

The closure of four VIVs [MSIV]s is a more onerous transient for overpressure than the spurious pressuriser heaters as it leads to a total loss of the secondary side heat removal whilst the secondary relief trains remain closed.

- o C3:
  - ALARP justification concerning containment isolation
- o C4:
  - ALARP justification concerning VIV [MSIV] failure to close (in particular, the SGTR with failure of the VIV [MSIV]a closure is considered for the demonstration)

- Other
  - o O1: Diversity for VDA [MSRT] in the protection against overpressure role is ensured by the MSSVs.

This list summarises the analyses to be performed to demonstrate the diversity of the systems at the plant level safety functions.

2.7. CONCLUSIONS

The diversity analysis is performed on the frequent events list determined from the reconciliation between the PCSR Chapter 14 and the probabilistic safety analysis.

The analysis demonstrates the diversity for the plant level safety functions for all the frequent events considered in the PCSR. The events challenging the Plant Level Safety Functions have been highlighted and the most onerous ones will be further analysed.

The analysis of functional diversity presented in this sub-chapter demonstrates the completeness of the analysis, including consideration of initiating events occurring from a range of possible plant states [Ref-5].

Some potential shortfalls are addressed through justification of the current design. ALARP justification is provided for the VIV [MSIV] design [Ref-2], the containment isolation valve design [Ref-3] and operation of the Station Black-Out diesel (SBO) [Ref-4] regarding adequacy for diversity studies.

Diverse protection for the PSVs is not provided. However, other means of limiting the RCP [RCS] pressure can be claimed, such as the normal spray function and, indirectly, the secondary side overpressure protection (MSSV and MSRT). The current PSV design also gives intrinsic diverse overpressure protection means and improvements compared to previous designs [Ref-1].

3. FUNCTIONAL DIVERSITY TRANSIENT ANALYSES

3.1. INTRODUCTION

Section 3 presents the quantified diversity analysis in the Plant Level Safety Function (PLSF) for all identified representative frequent Postulated Initiating Events (PIE) discussed in section 2.

It follows the functional diversity analysis for the EPR design presented in section 2. This functional analysis has highlighted a list of limiting frequent PIEs which demonstrates a suitable degree of diversity within the design for each PLSF.

Transient analyses of the selected PIEs combined with the loss of a safety function are presented to prove the safety criteria are met while considering these additional failures.

Section 3.2 introduces the methodology used to study the transients by focusing on the assumptions and codes used.

Sections 3.3 to 3.6 provide the analyses for each PLSF.

It should be noted that the following diverse reactor trip signals are used in the analysis.

- Low SG level,
- Low hot leg pressure,
- Low cold leg temperature.

The following additional reactor trip signals have been claimed:

- High hot leg pressure,
- High axial offset,

- Low reactor coolant pump speed,
- High neutron flux.

The last four signals will be allocated to a sufficiently classified non TXS platform.

The Non-Computerised Safety System is not claimed in the analyses as it is only designed to mitigate the total loss of the computerised I&C.

3.2. METHODOLOGY

3.2.1. Review of the selection of the transients

Each frequent event (initiating frequency higher than 10^{-3} per reactor per year) is analysed at the level of the Lower Level Safety Function (LLSF).

Each LLSF is assumed unavailable and is replaced by a diverse LLSF which provides the Plant Level Safety Function (PLSF). The study is based on the analysis of the consequences of a potential failure for each LLSF for each transient. If the diverse LLSF challenges the safety criteria, the most onerous event is further analysed.

In a family of events, the most onerous case(s) when assessed against the considered safety function is (are) further analysed.

Eventually, for each safety function, comparisons with events from other families are made to identify which case is the most limiting. Different transients may be analysed if required.

This analysis performed at the event level and at the plant safety level provides a comprehensive assessment of all cases. Sub-chapter 16.5 – Table 12 summarises the cases to be analysed for each PLSF.

3.2.2. Assumptions

3.2.2.1. Initial and boundary conditions

The transients presented in the present analysis are analysed with conservative assumptions. The initial and boundary conditions considered for the transient analyses for the diversity study are conservative.

The key parameters, including initial conditions, setpoints and thresholds, are pessimised.

Parameters to which the transient is not sensitive are not pessimised as they do not affect the transient.

The values considered for the analyses are described in PCSR Sub-chapter 14.1. The main characteristics of the systems used to mitigate the transients are presented for each transient. {CCI Removed} b

3.2.2.2. Rules for operator actions

A distinction is made between two phases of a transient, the automatic phase and the manual phase:

- the automatic phase, which lasts from event occurrence up to the first manual action,
- the manual phase, which lasts from the first manual action, until the safe shutdown state or stable state is reached.

During the manual phase, as described in PCSR Sub-chapter 3.1, manual actions are taken into account in the accident analysis, in addition to automatic actions. Operator action times are defined as follows:

- a manual action from the Main Control Room (MCR) is assumed to take place, at the earliest, 30 minutes after the first significant information is provided to the operator,
- a local manual action, i.e. a manual action that must be performed outside the MCR, is assumed to take place, at the earliest, 1 hour after the first significant information is provided to the operator.

In the majority of cases, the controlled state is reached using automatic actions only. However this is not mandatory. Reliance on manual actions to reach the controlled state is allowed.

The most significant operator actions are taken into account in the analysis. These actions are modelled to demonstrate the relevance of the back-up line to reach a stable state, but shall not be considered as detailed Emergency Operating Procedures.

3.2.2.3. Application of the single failure criterion

The analyses covered here deterministically assume the loss of a safety function, which is consequently not combined with any additional single failure.

Therefore, a single failure is not included in the transient analyses.

3.2.2.4. Application of preventive maintenance

Preventive maintenance is not considered in the analysis.

3.2.2.5. Loss of off-site power

Loss of off-site power is not combined to the PIEs considered in the analysis as this would combine two initiating events with the loss of the LLSF.

3.2.3. Codes

The codes and methods used for each transient analysis are chosen consistent with the requirements for the case studied.

3.2.3.1. CATHARE

CATHARE 2 V2.5 is an advanced, two-fluid, thermal-hydraulic code designed for use in realistic studies of accident thermal-hydraulics in a Pressurised Water Reactor (PWRs). The transients of interest are those in which core degradation is limited to fuel cladding deformation and bursting. While this excludes the severe accident domain, it does cover all Loss Of Coolant Accidents (LOCA), all degraded operating conditions in the steam generators (SG) secondary systems following ruptures or system malfunctions, and insofar as all PWR systems can be simulated, all of the incident or accident transients in which they are involved as initiators or participants.

Therefore, the code is used for the different types of SB LOCA analyses (including ATWS), the total loss of feedwater event and the loss of feedwater with additional failures.

3.2.3.2. SCIENCE nuclear code package

MANTA is a computer code for the simulation of all non LOCA PWR plant transients, including variations in core reactivity, SG heat removal capability, primary flow, primary pressure, and primary mass inventory. Its objectives address:

- the simulation of any complex fluid system, such as the primary and secondary circuits of the PWR, but also any other fluid system such as the Reactor Heat Removal System (RIS/RRA [SIS/RHRS]) or Chemical and Volumetric Control System (RCV [CVCS]) if required,

- flexible coupling of thermal-hydraulics with different neutronic modelling (0D, 1D, 3D): the code fully describes the neutronic core behaviour and its feedback on the thermal-hydraulic behaviour, as required for the analysed transient and dominant phenomena,

- user friendly detailed modelling of Instrumentation and Control (I&C) systems, in order to save time and effort for description of these systems.

Therefore, MANTA is used for the overpressure transients (inadvertent closure of four VIVs [MSIV]s) with 0D (point) neutronic data.

MANTA may be coupled to SMART and FLICA for the ATWS computations to increase the accuracy of the prediction of the neutronic effects on the transient.

FLICA calculates the core thermal hydraulics.

The SMART neutronic model provides the 3D nuclear power distribution. The nuclear power is split into a part (f) deposited in the fuel pins and a part (1-f) directly deposited in the water. The SMART fuel pin model provides the Doppler temperature (Tceff) to the neutron model and the heat flux across the clad (Ptherm) to the MANTA thermal-hydraulic model. MANTA provides in turn water specific volume (vs) and boron concentration (Cb) to the neutron model, and the wall temperature at the clad internal surface (Tpig) to the SMART fuel thermal model.

COMBAT calculates the transient temperature distribution in a cross-section of a fuel rod (cladding, pellet-cladding gap, UO_2 pellet) and the transient heat flux at the surface of the cladding, using as input the nuclear flux, the fuel neutronic and mechanical characteristics with or without burnable poisons in the core, and the time dependant coolant parameters.

3.2.4. Acceptance criteria

The acceptance criterion for the analyses is to meet release target 4 of the HSE Safety Assessment Principles (SAPs).

To ensure this safety criterion is met, the aim of the analyses is to demonstrate that events analysed with an assumed failure of a LLSF meet at least the PCC-3/PCC-4 criteria, as defined in PCSR Sub-chapter 14.0. The decoupled acceptance criteria are provided for each event analysed below.

For all loss of coolant accidents, the acceptance criteria are:

- the peak cladding temperature remains below the 1200°C acceptance criteria,
- the maximum oxidation of the cladding does not exceed 17% of the total thickness of the cladding at the hot spot,
- there is no cladding failure,
- the integrity of the core geometry is maintained,
- long-term cooling is ensured.

For all other accidents, the acceptance criteria are:

- The number of fuel rods experiencing DNB remains below 10%,
- Peak clad temperature must remain below 1482°C,
- Melted fuel at the hot spot must not exceed 10% by volume,
- the RCP [RCS] integrity is not challenged (as an acceptance criterion, the pressure at the worst point of the RCP [RCS] does not exceed 130% of the design pressure, i.e. 228.5 bar abs (Sub-chapter 3.4).

3.3. REACTIVITY CONTROL SAFETY FUNCTION

3.3.1. R1 – Maintain core reactivity

The event 'Loss of RCV [CVCS] after Reactor Trip' challenges the R1 PLSF as it prevents the operation of the normal boration route to provide long-term sub-criticality after a reactor trip.

In such a case, the boration occurs via the RBS [EBS], which is the F2 classified system used to increase the boron concentration in the reactor coolant system before cooling down the plant. The consequences of this event are bounded by those of ATWS events, which are used to demonstrate diversity for the PLSF 'R2 – Shutdown and maintain sub-criticality' (see section 3.3.2). These ATWS events are more onerous as the rods do not drop due to mechanical blockage of the RCCAs and the RBS [EBS] is the F2 means used to ensure sub-criticality in the long term.

3.3.2. R2 – Shutdown and maintain core sub-criticality

The R2 PLSF is challenged by all the PIEs leading to a reactor trip. The loss of the LLSF 'Negative reactivity fast insertion' leads to ATWS events. The following five ATWS events are bounding when compared to the other sequences:

- Small Break Loss Of Coolant Accident (SB LOCA) – see sections 3.3.2.2 and 3.3.2.3.
- Loss of main feedwater – see sections 3.3.2.4 and 3.3.2.5.
- Loss Of Off-site Power (LOOP) – see sections 3.3.2.6 and 3.3.2.7.
- Excessive Increase in Steam Flow (EISF) – see sections 3.3.2.8, 3.3.2.9 and 3.3.2.10.
- Rod drop faults (and rod misalignment faults) with ATWS due to failure of TXS – see section 3.3.2.11.

These five events are studied with the mechanical blockage of the rods. In addition, in order to demonstrate the performance of the I&C back-up platform, the above mentioned PIEs are studied assuming the loss of the TXS I&C platform.

3.3.2.1. Mechanical blockage of the rods

The ATWS events are studied with the total loss of a F1 function following an initiating event as described in PCSR Sub-chapter 16.1 (section 2.1.1.1). The total loss of the "negative reactivity fast insertion" function must therefore be assumed.

In the current framework, the objective of the ATWS events analysis is to demonstrate the functional diversity of the "Shutdown and maintain sub-criticality" PLSF. The "negative reactivity fast insertion" function must therefore be assumed to be totally lost. All rods are thus assumed to be blocked during ATWS events.

The case of LOOP with a partial mechanical blockage is different from the PCSR assumption and thus not included in the PCSR. A description of this particular case is given in the following paragraph.

The ATWS with the successful drop of some rods has some similarities with the first phase of the "spurious drop of several rods at power" transient. During the first phase of these two transients, the drop of some rods induces a core power decrease and thus a DNBR margin increase. In the second phase, the LOOP-ATWS transient with partial mechanical blockage differs from the rod drop transient. In the case of a rod drop transient, the power increases due to the Average Coolant Temperature (ACT) control. The minimum DNBR is reached during this power increase. In the case of a LOOP event, a reactor trip signal is generated following a "low reactor coolant pump speed" signal and the ACT control is switched to manual mode, thus preventing any insertion/withdrawal signals to the control rods. The control rods are therefore not withdrawn and the ACT control has no effect on the core power. The ATWS postulated in the case of a LOOP event, with the successful drop of some rods, is therefore not limiting in terms of DNBR.

Therefore, there is no need to consider a LOOP and ATWS event with the successful drop of several rods.

More generally and for the reason explained above, ATWS events postulating the mechanical blockage of the rods, and presented in this PCSR sub-chapter, do not assume the successful drop of any rods.

3.3.2.2. ATWS by mechanical blockage of the rods - Small Break LOCA (< DN 50) including a break occurring on the extra boration system injection line (states A and B)

3.3.2.2.1. Introduction

This section presents analysis of the small break loss of coolant accident (up to 20 cm²) combined with a failure in the R2 PLSF leading to the ATWS.

In case of ATWS, the PLSF R2 'Shutdown and maintain core sub-criticality' is challenged as the lower level safety function ‘Negative reactivity fast insertion’ cannot be fulfilled. The PLSF is then provided by the LLSF ‘Highly concentrated boron injection’.

3.3.2.2.2. Typical sequence of events

This accident is initiated by a 20 cm² break located in the cold leg (this is conservative for the boron concentration within the RCP [RCS] as borated water is lost via the break. A break on the hot leg would release mainly saturated steam, increasing the boron concentration in the core.

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

A reactor trip (RT) occurs following a “low pressuriser pressure < MIN2” signal. The RT signal automatically trips the turbine and closes the ARE [MFWS] full-load lines. The reactor continues to operate at full power as no scram is triggered.

As the secondary side pressure increases, the Main Steam Bypass (GCT [MSB]) valves open allowing steam dump to the condenser. If the condenser is unavailable for steam dump, the VDA [MSRT] opens allowing the steam to be dumped to the atmosphere. The steam generators (SGs) are fed by the ARE [MFWS] through the low-load lines. If the ARE [MFWS] is unavailable, the Start-up and Shutdown Feedwater System (AAD [SSS]) pump starts and feeds the SG through the low-load lines. If the AAD [SSS] is unavailable, the ASG [EFWS] is actuated following a “low SG level < MIN2” signal.

The secondary system saturation temperature increases with the pressure. Because of the isolation of the ARE [MFWS] full-load lines, the feedwater flow rate is not sufficient to remove all the energy produced by the core at full power. Therefore, the secondary system water mass decreases, reducing the heat transferred between the RCP [RCS] and the SG. The RCP [RCS] average temperature and pressure sharply increase. This effect, combined with the loss of primary coolant, leads to boiling within the core.

The Pressuriser Safety Valves (PSV) may then open to limit the primary side overpressure.

A dedicated ATWS signal is implemented to prevent this high pressure peak. The ATWS signal is triggered in the RPR [PS], from information showing "RT signal and rods out (or flux high) after a time delay". This ATWS signal (and associated actions) is a RRC-A feature and is specifically implemented to cope with the "ATWS by rods failure” sequences. It actuates all RCP [RCS] pumps trip on a “low SG-water level” signal (SG level WR < MIN2) before the SG depletion occurs.

The ATWS signal also automatically initiates RBS [EBS] injection using 7000 ppm enriched boron (corresponding to 11200 ppm natural boron) thus automatically providing core sub-criticality in the long term. The RCV [CVCS] is also able to perform this boration function.

When boiling occurs in the core, the void effect becomes dominant in the reactivity balance. Therefore, the core becomes significantly sub-critical and the power produced decreases, eventually reaching the value of the decay heat power.

A Safety Injection (SI) signal is actuated following a “pressuriser pressure < MIN3” signal. The safety injection signal automatically starts the Medium Head Safety Injection (MHSI) and Low Head Safety Injection (LHSI) pumps and initiates a partial cooldown of the secondary system. During the partial cooldown, the RCP [RCS] pressure decreases sufficiently to allow MHSI injection into the cold legs. The partial cooldown is performed by all SGs using the steam dump to the condenser. This is performed by automatically decreasing the respective relief valve setpoints for a constant cooling rate of -250°C/h. The cooling continues to a specific pressure value which is low enough to allow the necessary MHSI injection but high enough to prevent core re-criticality.

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

The RCP [RCS] inventory depletion stops once sufficient MHSI flow is available to compensate for the break flow. The RCP [RCS] boron concentration keeps increasing due to the RBS [EBS] and the RIS [SIS] (MHSI, LHSI, accumulators) injection. The increase in the boron concentration is sufficient to compensate for the core cool down and to maintain core sub-criticality. The controlled state is reached with significant margins on core reactivity.

3.3.2.2.3. Safety criteria

For this demonstration, the following PCC-3/PCC-4 acceptance criteria are considered:

- the maximum cladding temperature shall remain below 1200°C,
- the core geometry remains coolable,
- the RCP [RCS] integrity is not challenged (as an acceptance criterion, the pressure at the worst point in the RCP [RCS] shall not exceed 130% of the design pressure, i.e. 228.5 bar abs).

3.3.2.2.4. Assumptions for the analysis

The study is performed with conservative assumptions.

3.3.2.2.4.1. Initial conditions

Conservative initial conditions are considered.

They are summarised in Sub-chapter 16.5 – Table 13.

Initial core power is 102% FP, i.e. 4590 MW.

Thermal-hydraulic flow conditions are considered within the primary circuit.

3.3.2.2.4.2. General neutronic data

Conservative neutronic data are considered for the study of this case. In particular, a minimum moderator density coefficient of 0.09 Δk/k per g/cm^3 is assumed, which corresponds to the beginning of life value. The associated initial boron concentration is at its maximum of 625 ppm, which corresponds to the UO_2, Beginning of Life with Xenon (BLX) conditions. In addition, the Doppler temperature coefficient is at a minimum value of -4.03 pcm/°C. All these data are presented in PCSR Sub-chapter 14.1 - Tables 4 and 5.

3.3.2.2.4.3. Assumptions related to systems

The assumptions related to systems and controls are provided in Sub-chapter 16.5 – Table 14.

The relevant systems modelled in the transient analysis and relevant for the mitigation of the event are: RIS [SIS], RBS [EBS], PSV, VDA [MSRT], MSSV and ASG [EFWS].

All the characteristics of the systems are pessimised to minimise their mitigation of the event.

3.3.2.2.4.4. Assumptions related to controls

Conservative I&C setpoints are used assuming the uncertainties for degraded conditions.

In particular, the ATWS signal is triggered 20 seconds after reactor trip.

3.3.2.2.4.5. Single failure and preventive maintenance

Single failure and preventive maintenance are not considered in the analysis.

3.3.2.2.5. Method

Calculations for the overall plant, system and core behaviour are performed using the CATHARE 2 V2.5 computer code with the core kinetic model. For this transient, neutronic data is used to model the core response.

3.3.2.2.6. Results

This study is performed with the same assumptions as a typical LOCA. The non-safety systems are not claimed except the pressuriser heaters as their action is conservative for the RCP [RCS] pressure.

The core long-term sub-criticality is demonstrated using boron injection by the RIS [SIS] and the RBS [EBS].

The typical sequence of events is given in Sub-chapter 16.5 – Table 15.

The key parameters are presented in the following figures:

- Sub-chapter 16.5 – Figure 2

- Sub-chapter 16.5 – Figure 3
- Sub-chapter 16.5 – Figure 4
- Sub-chapter 16.5 – Figure 5
- Sub-chapter 16.5 – Figure 6

3.3.2.2.6.1. Sequence of events

At time 0 seconds, a small break (20 cm²) in a cold leg in state A results in a loss of RCP [RCS] coolant inventory. This causes a decrease of the RCP [RCS] pressure and the pressuriser level. The pressuriser heaters operate at full power (2596 kW) and attempt to compensate for the depressurisation. This is conservative for the break flow rate and delays the reactor trip and ATWS signals.

At 79 seconds, the pressuriser pressure reaches the MIN2 setpoint (135 bar – 3 bar uncertainty). The reactor trip is actuated, followed by the turbine trip and the ARE [MFWS] full load isolation. In the absence of Rod Cluster Control Assemblies drop, the core power remains high. The secondary pressure rises to the VDA [MSRT] setpoint (95.5 bar + 1.5 bar uncertainty) and then to the MSSV opening pressure (105 bar) as the GCT [MSB] is not considered for this study.

At 99 seconds, the ATWS signal is actuated.

The RCP [RCS] pressure and temperature increase sharply. Boiling occurs in the core at 120 seconds and the first PSV opens at 211 seconds releasing steam and preventing RCP [RCS] overpressure. The start of boiling in the core stops the neutronic reactions. The moderator effect is dominant in the reactivity balance and the core becomes significantly sub-critical. Subsequently, only decay-heat has to be removed.

At 177 seconds, all RCP [RCS] pumps are tripped as the ATWS signal has been generated. Natural circulation occurs within the RCP [RCS], maintaining the core cooling.

At 194 seconds, the steam generator level reaches the MIN2 setpoint (40% WR - 5% uncertainty). The ASG [EFWS] is actuated to maintain the SG cooling capability.

At 700 seconds, the boron injected by the RBS [EBS] reaches the core inlet.

At 825 seconds, the pressuriser level falls below 12% of the measured range. The pressuriser heaters are switched off.

At 1055 seconds, the pressuriser pressure reaches the MIN3 setpoint (115 bar – 3 bar uncertainty). The safety injection signal is actuated and initiates the partial cooldown signal. The secondary side pressure starts to decrease at the rate of -250°C/h. The RCP [RCS] pressure follows the secondary side pressure.

At 1508 seconds, the secondary side pressure has reached 61.5 bar (60 bar + 1.5 bar uncertainty). As long as the primary and secondary sides are not decoupled, the RCP [RCS] pressure remains at the secondary side pressure. This pressure is not low enough to allow the accumulators or the LHSI to inject. The MHSI cannot compensate for the break flow, therefore the RCP [RCS] coolant inventory keeps on decreasing.

At 2588 seconds, the RCP [RCS] pressure has reached the accumulator injection pressure (45 bar). From 2000 seconds the RIS [SIS] injection totally compensates for the break flow. The RCP [RCS] inventory is stabilised and the decay heat removal is provided by the secondary side. The RIS [SIS] provides the boron injection and the core remains sub-critical.

3.3.2.2.6.2. Final state

At the end of calculation, the RCP [RCS] inventory is stabilised at 103 tons and starts to increase.

The core remains covered.

The decay-heat is removed by the SGs and by the coolant loss via the break.

The long-term core sub-criticality is maintained by the boron injection through the RIS [SIS] and the RBS [EBS].

The final state is reached with a core sub critical by greater than 10,000 pcm.

3.3.2.2.7. Conclusions

The core stays covered throughout the transient and the cladding temperature reaches a maximum of 378°C. Therefore:

- the peak cladding temperature remains below the 1200°C acceptance criteria,
- the maximum oxidation of the cladding does not exceed 17% of the total thickness of the cladding at the hot spot,
- there is no cladding failure,
- the integrity of the core geometry is maintained,
- long-term cooling is guaranteed.

All the loss of coolant accident acceptance criteria are met with significant margins and without the need for any non-safety systems.

Boron injection through the RIS [SIS] maintains the long-term sub-criticality of the core.

This demonstration shows that, for the ATWS sequence Small-Break LOCA combined with the mechanical blockage of the rods, the PCC-3/PCC-4 acceptance criteria are met and the required controlled state can be reached by means of only F1A features even making conservative assumptions.

As a consequence, the lower level safety function ‘Highly concentrated boron injection’ provides an efficient diverse mean to mitigate the event following the assumed loss of the lower level safety function ‘Negative reactivity fast insertion’.

3.3.2.3. ATWS by loss of RPR [PS] - Small break LOCA (< DN 50) including a break occurring on the extra boration system injection line (states A and B)

The ATWS SB LOCA fault assuming the loss of the Protection System is bounded by the fault assuming the mechanical blockage of the rods as the diverse reactor trip from the non-TXS I&C platform would occur automatically following a “Hot leg pressure (WR) < MIN2” signal.

The safety injection signal with the associated partial cooldown occurs shortly after following a “Hot leg pressure (WR) < MIN3” signal. This would lead to a less onerous transient when compared to the ATWS with mechanical blockage.

This leads to the conclusion that, in such a case, the lower level safety function ‘Highly concentrated boron injection’ provides an efficient diverse means to mitigate the event assuming the loss of the lower level safety function ‘Negative reactivity fast insertion’.

3.3.2.4. ATWS by mechanical blockage of the rods - Loss of normal feedwater flow (loss of all ARE [MFWS] pumps and of the start-up and shutdown pump)

3.3.2.4.1. Typical sequence of events

The sequence considered is initiated by the total loss of all ARE [MFWS] pumps, which implies the loss of three operating pumps and the stand-by pump.

Following the loss of main feedwater supply, reactor trip/turbine trip signals are actuated. As the control/shutdown rods have failed to enter the core due to mechanical blockage, the reduction of the reactor power can only result in the short-term from the inherent reduction of reactivity following the decrease ion the moderator density. This is due to RCP [RCS] temperature increase. This phenomenon is due to:

- the secondary side pressure increase (turbine valves are closed, secondary side heat removal is made via VDA [MSRT] or MSSVs), the primary temperature increases due to the thermal coupling via the SG tubes, or
- the heat transfer capability of the steam generators decrease (depletion of steam generator inventory), or
- the primary coolant flow reduction (trip of RCP [RCS] pumps).

To compensate for the pressuriser pressure increase, the primary pressure control function increases the pressuriser sprays flow rate. This insufficient to halt the pressure rise and the Pressuriser Safety Valves (PSV) open to limit the primary side overpressure.

Without any additional actions, this state would be stable provided the steam generators have sufficient water inventories to remove the primary power (core plus RCP [RCS] pumps power). Nevertheless as the SG water inventory is reduced, the ASG [EFWS] will be actuated and heat removal will be partially recovered. However, the heat transfer capability is significantly reduced, and pressure as well as temperature will increase.

To limit the magnitude of the pressure peak, a dedicated ATWS signal is implemented. The ATWS signal is actuated in the RPR [PS], from detecting a "RT signal and rods out (or flux high) after a time delay". This ATWS signal (and associated actions) is a RRC-A feature and it is specifically implemented to cope with the "ATWS by rods failure" sequences, being classified F2. It trips all RCP [RCS] pumps following a “low SG-water level” signal (SG level < MIN2). By this action, the reactor power is reduced more smoothly with the decreasing coolant flow rate, which leads to a reduced pressure increase on the primary side.

The ATWS signal also automatically initiates RBS [EBS] injection with 7000 ppm enriched boron (corresponding to 11200 ppm natural boron) which automatically maintains long-term core sub-criticality. The RCV [CVCS] could also perform this boration function, but is not considered in this study.

3.3.2.4.2. Assumptions for the analysis

3.3.2.4.2.1. Definition of studied case

In order to minimise the core power decrease during the transient, the initial power state is considered at BLX conditions, when the moderator temperature coefficient is at its minimum absolute value.

PCSR studies at a lower core power level of 4250 MWth have shown that the transient at 100% NP initial state covers the 60% power level. It demonstrates that both RCP [RCS] temperature and power are among the most important parameters for Departure from Nucleate Boiling (DNB) calculations.

In the present study, in order to pessimise the DNB assessment, the calculation is performed at full power. It covers all power levels in state A.

3.3.2.4.2.2. Single failure and preventive maintenance

Single failure and preventive maintenance are not taken into account in this analysis.

3.3.2.4.2.3. Initial and boundary conditions

Conservative initial conditions are considered, including a core power of 102% NP.

RCP [RCS] thermal hydraulic design flow rate is assumed.

For this study, the systems which are not claimed (most of the non-classified systems, except those which pessimise the transient) and classified systems which do not intervene in the study (e.g. RIS [SIS]) are not modelled. Other systems are available.

Boundary conditions defining system performance are based on conservative characteristics (minimum or maximum data claimed for the limiting purposes, e.g. PSV for RCP [RCS] overpressure limitation, definition of ATWS signal, RBS [EBS] for core boration).

The initial conditions are presented in Sub-chapter 16.5 – Table 16.

3.3.2.4.2.4. Neutronic data

The neutronic data refer to BLX conditions for the 18 month cycle, which is representative of the different fuel cycles. The same approach is considered for the all the transient analyses. The safety margins shown for the different cases are sufficient to ensure that a change in the fuel cycle considered would not jeopardise meeting the acceptance criteria.

The moderator coefficient is pessimised, and its value is set at -13.2 pcm/°C at the beginning of the transient.

The bounding initial RCP [RCS] boron concentration is 1594 ppm.

3.3.2.4.2.5. Protection and mitigation actions

The following I&C functions provide protection and mitigation following a loss of main feedwater combined with a reactor trip failure due to the mechanical failure of the rods:

- Reactor trip / turbine trip signal on "SG level < MIN1" (F1A),
- Reactor trip / turbine trip signal on “pressuriser pressure > MAX1 (F1A)”,
- ATWS signal on “Reactor trip signal and high rods position (or high flux) after a time delay (F2)”,
- RBS [EBS] actuation on ATWS signal (F2),
- RCP [RCS] pumps trip on "SG level (WR) < MIN2" if the ATWS signal has been obtained (F2),
- VDA [MSRT] opening on "SG pressure > MAX1" (F1A),
- ASG [EFWS] actuation on "SG level (wide range) < MIN2" (F1A).

In addition, the following systems are also available

- Three pressuriser safety valves (F1A)
- Two main steam safety valves per SG (F1A).

3.3.2.4.2.6. Assumptions related to controls

The following assumptions related to control systems are made in this accident:

- SG level and RCP [RCS] temperature control are not considered as the ARE [MFWS] and control rods are unavailable,
- Pressuriser pressure control via normal spray and heaters is taken into account, limiting the pressure increase which is conservative for the DNB criteria (beginning of the transient),
- SG pressure control via GCT [MSB] is not taken into account,
- Pressuriser level control is not modelled in the present analysis.

3.3.2.4.2.7. Assumptions related to protections

The F1A systems assumed to operate are: ASG [EFWS], PSV, VDA [MSRT], MSSV.

The F2 functions assumed to operate are: RBS [EBS] boration and RCP [RCS] pumps cut-off on ATWS signal.

The setpoints, delays and flow capacities are all pessimised and listed in Sub-chapter 16.5 – Table 17.

In addition, automatic boration via the RCV [CVCS] pumps is not claimed in the analysis.

3.3.2.4.2.8. Special assumptions not included in the above

- The ATWS signal is actuated 20 seconds after the reactor trip signal, starting the RBS [EBS] with 7000 ppm boron acid injection (corresponding to 11200 ppm natural boron). When the wide range SG level falls below the MIN2 setpoint, all the main coolant pumps are tripped,
- Automatic boration via RCV [CVCS] pumps is not claimed in the analysis.

3.3.2.4.3. Safety criteria

For this demonstration, the following decoupled acceptance criteria are considered:

- The number of fuel rods experiencing DNB remains below 10%,
- The RCP [RCS] integrity is not challenged (as an acceptance criterion, the peak pressure in the RCP [RCS] does not exceed 130% of the design pressure.

3.3.2.4.4. Methods of analysis

The analysis is carried out using the internal coupling of:

- the MANTA V3.7 code for the overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), and modelling F2/F1 systems operations,
- the SMART V4.8.1/FLICA-IIIF V3 codes for the neutronic and thermal-hydraulic behaviour of the core.

The DNBR calculation is performed by considering the axial distribution in the hot channel and the FΔH provided by 3D core calculations from the SMART code at the time of minimum DNBR.

For these ATWS analyses, no conservatism on the local thermal power is considered for DNBR calculation.

3.3.2.4.5. Results

Sub-chapter 16.5 - Table 18 gives the sequence of events.

The change in parameters versus time is presented in:

- Sub-chapter 16.5 – Figure 7
- Sub-chapter 16.5 – Figure 8
- Sub-chapter 16.5 – Figure 9
- Sub-chapter 16.5 – Figure 10
- Sub-chapter 16.5 – Figure 11
- Sub-chapter 16.5 – Figure 12
- Sub-chapter 16.5 – Figure 13
- Sub-chapter 16.5 – Figure 14

At 5 seconds all the ARE [MFWS] pumps are tripped as the initiating event. The SG level decreases and the heat transfer capability from the primary to the secondary side is reduced. This leads to a primary temperature and pressure increase.

The change in the key parameters (steam generator level, pressure and temperature) eventually leads to the actuation of a reactor trip signal. The first one reached is the RT on “SG level < MIN1 (20% NR)”.

However, as the rods are assumed to be mechanically blocked, the core power does not decrease following the reactor trip. However, the turbine is tripped by the reactor trip check-back signal which leads to a secondary pressure increase and to the VDA [MSRT] opening. At the same time, the temperature in the primary side increases, and due to the moderator density effect the core power is reduced. Relief via the PSVs is demanded.

At 20 seconds after the reactor trip check back, the ATWS signal is triggered. This results in the actuation of the RBS [EBS] and to the tripping of the reactor coolant pumps (on ATWS signal combined with SG level < MIN1).

Despite the core power reduction, the steam generators are unable to support the required heat transfer. The steam generator level continues to decrease until the MIN2p threshold is reached, giving ASG [EFWS] actuation. This actuation reduces the rate at which the SG water inventory decreases.

At about 600 seconds, the plant is in a stable state:

- the SG pressure is controlled by VDA [MSRT],
- the SG water inventory is stabilised,
- the RCP [RCS] temperature and pressure are also under control,
- and even with reactor coolant pumps shutdown, the boron reaches the core.

The plant is stabilised with the primary heat removed by the VDA [MSRT] and ASG [EFWS]. The reactivity will progressively decrease as the boron injected by RBS [EBS] to the cold leg, reaches the core.

3.3.2.4.6. Conclusions

The initial DNBR at the beginning of the transient for the 102% FP case is the minimum DNBR of the entire transient. This is due to the increase of the primary pressure and the power reduction that occur as the temperature increases. As a result, the DNBR remains higher than the acceptance criterion.

The decay heat is safely removed via the VDA [MSRT] and the ASG [EFWS] via the SG.

The activity release during the accident is under control as none of the barriers (fuel and RCP [RCS]) is challenged.

The calculation results show that for the sequence "ATWS by rods failure – Loss of Main Feedwater", the acceptance criteria are met and the required final state is reached.

As a conclusion, in such case, the lower level safety function ‘Highly concentrated boron injection’ provides an efficient diverse mean to mitigate the event following the assumed loss of the lower level safety function ‘Negative reactivity fast insertion’.

3.3.2.5. ATWS by loss of RPR [PS] - Loss of normal feedwater flow (loss of all ARE [MFWS] pumps and of the start-up and shutdown pump)

This section presents the assessment of the ATWS resulting from a RPR [PS] failure with Loss Of main Feed Water (LOFW) in State A at power. In this case, the lower level safety function “Fast reactor shutdown” is assumed to be lost.

The LOFW is an overheating event due to a reduction of the capability of the secondary side to remove the primary heat load. This event causes an overheating on the primary and secondary side.

3.3.2.5.1. Typical sequence of events

The sequence considered is initiated by the total loss of all ARE [MFWS] pumps, the loss of three operating pumps and the stand-by pump.

Following the loss of main feedwater supply, the main plant parameters, e.g. SG level, pressuriser pressure, deviate from their steady state values. As the Protection System is assumed to have failed, neither RPR [PS] reactor trip nor ASG [EFWS] start-up are actuated. The primary heat is removed by the steam generators. However, because there is no feedwater feeding the steam generators, the SG level decreases. Eventually conditions deteriorate sufficiently for a reactor trip actuation via the non TXS platform. This additional feature (F2 classified) is provided to actuate the fast power reduction safety function following RPR [PS] failure.

When the turbine is tripped, the valves are closed and the main steam pressure increases. The VDA [MSRT] is not actuated as the opening function is implemented in RPR [PS]. However, the MSSV are opened, and the secondary side pressure is stabilised. On the primary side, the temperature increases resulting in a primary pressure increase, leading to PSV actuation.

The SG level continues to decrease, until the reactor coolant pumps are tripped once the setpoint is reached on at least three steam generators. This operation helps to maintain SG inventory by halting the heat addition to the primary side by the pumps. After a certain period of time, the steam generators are empty and the secondary side can no longer remove the power. In these conditions, the primary system heat removal is through the PSVs.

After these events the operator intervenes and manually starts the ASG [EFWS]. This action maintains primary side heat removal to the secondary side without emptying of the SGs. If necessary the operator will, as for a total loss of feedwater event, initiate feed and bleed operations.

3.3.2.5.2. Safety criteria

For this demonstration, the following acceptance criteria are considered:

- The number of fuel rods experiencing DNB remains below 10%,
- the RCP [RCS] integrity is not impaired (as an acceptance criterion, the pressure at the worst point of the RCP [RCS] does not exceed 130% of the design pressure, i.e. 228.5 bar abs (Sub-chapter 3.4).

The analysis presented below addresses the DNBR criterion.

The maximum primary pressure reached is given for information.

3.3.2.5.3. Specific assumptions

3.3.2.5.3.1. Definition of studied case

In the present study, to increase the likelihood of core boiling, the calculation is performed at 102% FP (with NP 4500 MWth).

It covers all power levels in state A.

The calculation is performed in two steps. The first step uses a 3D core model with MANTA SMART FLICA coupling. In this model the initial power state is assumed at BLX when the moderator temperature coefficient is at its smallest absolute value. The second step is a middle of transient performed with MANTA 0D, based on the first calculation and conservative assumptions for decay heat.

3.3.2.5.3.2. Single failure and preventive maintenance

Single failure and preventive maintenance are not taken into account.

3.3.2.5.3.3. Initial and boundary conditions

Conservative initial conditions are considered, including the core power at 102% FP.

RCP [RCS] thermal hydraulic design flow rate is assumed.

For this study, the systems which are not claimed are not modelled. In particular, classified systems which do not contribute to the transient (e.g. RIS [SIS]) are not modelled. All other relevant and classified systems are available.

Boundary conditions defining system performance are based on conservative characteristics (minimum or maximum data assumed for the limiting purposes (e.g. PSV for RCP [RCS] overpressure limitation).

Initial conditions are presented in Sub-chapter 16.5 - Table 19.

3.3.2.5.3.4. Neutronic data

At the beginning of the transient, the neutronic data refer to BLX conditions used for 3D modelling.

The moderator coefficient is pessimised, with a value of -13.2 pcm/°C, and the bounding initial RCP [RCS] boron concentration is 1594 ppm.

3.3.2.5.3.5. Protection and mitigation actions

The following I&C functions provide protection and mitigation in a non TXS platform following LOFW combined with RPR [PS] failure causing a reactor trip failure:

- Diverse reactor trip / turbine trip signal on high hot leg pressure,
- Reactor coolant pump trips on low low SG level.

In addition, the following systems are also available:

- Three PSVs (F1A),
- Two MSSVs per SG (F1A).

The operator mitigation actions related to emergency operating procedures are not modelled in this calculation.

3.3.2.5.3.6. Assumptions related to controls

The controls are not claimed in this transient since either they have no effect on the transient, or their effect is beneficial.

3.3.2.5.3.7. Assumptions related to protections

In this study only the automatic actions are modelled. The non-TXS platform is designed to initiate some automatic actions, such as the diverse reactor trip, RIS [SIS] actuation, etc. However in this transient only the reactor trip setpoint on high hot leg pressure is reached.

The manual operations and associated systems are assumed to be used by the operator, but this phase is not presented in the study:

- The F1A systems assumed to operate are: PSV, MSSV.
- The F2 functions assumed to operate are: diverse reactor trip and reactor coolant pumps trips.

Conservative values for the setpoints, delays and flow capacities are assumed and are listed in Sub-chapter 16.5 - Table 20.

In addition, automatic boration via the RCV [CVCS] pumps is not claimed in the analysis as this non-classified action is a benefit to the transient.

3.3.2.5.4. Methods of analysis

In order to model this transient, the calculations are performed with two steps:

- the first step uses the MANTA SMART FLICA coupling until the reactor trip (short-term transient). This step models a 3D core.
- the second step is a MANTA calculation based on a conservative power transient from the first step calculations and a conservative decay heat curve {CCI Removed}[b]. This step models a 0D core.

The analysis is carried out using:

- the MANTA V3.7 code for the overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), taking account of F1/F2 systems operations,
- the SMART V4.8.1/FLICA-IIIF V3 codes for the neutronic and thermal-hydraulic behaviour of the core.

The DNBR calculation is performed with the FLICA code. The axial distribution in the hot channel and the $F\Delta H$ are provided by a 3D core calculation using the SMART code at time of minimum DNBR. For this calculation, no additional conservatism on the local thermal power is considered for the DNBR calculation. This conservatism would decrease the initial DNBR value to the LCO value in a manner decoupled from the core physics.

3.3.2.5.5. Results

Sub-chapter 16.5 - Table 21 gives the sequence of events.

The changes in parameters versus time are presented in:

- Sub-chapter 16.5 – Figure 15
- Sub-chapter 16.5 – Figure 16
- Sub-chapter 16.5 – Figure 17
- Sub-chapter 16.5 – Figure 18
- Sub-chapter 16.5 – Figure 19
- Sub-chapter 16.5 – Figure 20

At 5 seconds all the ARE [MFWS] pumps are tripped, to represent the initiating event. The steam generator levels fall, and the heat transfer rate from the primary to the secondary side is reduced. This leads to an increase in the primary temperature and pressure. The maximum pressuriser pressure reached is 173.8 bar abs.

The change in conditions (steam generator level, pressure and temperature) leads to the generation of the non-TXS reactor trip signal on “HL pressure > MAX (173 bar abs)” as the RPR [PS] has failed.

The power is rapidly reduced and the steam generators continue to provide the decay heat removal until about 1200 seconds. The SG level continues to fall and the reactor coolant pumps are tripped on “very low SG level” signal. Subsequently, the primary system heats up and the primary pressure increases. The PSVs are demanded several times, which provides part of the primary heat removal. In these conditions the RCP [RCS] inventory decreases. However, by the time the operator initiates the emergency operating procedures, the RCP [RCS] mass has decreased by about 50 tons.

When the operator begins emergency operating procedures, the plant is stable.

3.3.2.5.6. Conclusions

The maximum pressure downstream of the RCP [RCS] pumps occurs for the loss of main feedwater event from the 102% FP initial state. The peak value of 173.8 bar is significantly below the acceptance criterion of 130% design pressure.

The initial DNBR for the 102% NP case does not fall below the initial value during the transient. This is caused by the increase of the primary pressure and the power reduction that that are a bigger benefit than the penalty from the temperature increase. As a result, the DNBR remains higher than 1.0, with no fuel rods experiencing DNB.

The calculation results show that for the sequence "ATWS by RPR [PS] failure – Loss of Main Feedwater", the acceptance criteria are met and the final state can be reached using the emergency operating procedures defined for a total loss of feedwater.

Consequently, the diverse means to provide the lower level safety function ‘fast negative reactivity insertion’ is sufficient to mitigate the event.

NB: The NCSS is not specifically designed to provide functional diversity, but the NCSS automatic ASG [EFWS] actuation initiated following loss of main feedwater and total loss of TXS provides additional margins.

3.3.2.6. ATWS by mechanical blockage of the rods – Short-term loss of off-site power (≤ 2 hours)

3.3.2.6.1. Typical sequence of events

The sequence considered is initiated by the loss of off-site power, which causes a turbine trip, and trips all the reactor coolant pumps and ARE [MFWS] pumps.

The loss of ARE [MFWS] pumps leads to a decrease in the secondary side heat removal and the primary flow coast-down reduces the capacity of the primary coolant to remove heat from the core. Consequently, primary and secondary pressures and temperatures increase.

Reactor scram, or any other power reduction by rod insertion, does not occur following a reactor trip signal due to assumed mechanical blockage of the rods. The heating of the core causes the reactivity to decrease through the moderator temperature feedback effect.

The continued secondary side heat removal is provided by the VDA [MSRT] and main steam safety valves (MSSV), with feed only provided by the ASG [EFWS].

The Pressuriser Safety Valves (PSV) open to limit the primary side overpressure.

The ATWS signal is then generated by the reactor Protection System, following the detection of "RT signal and high rod position (or high flux) after a time delay". This ATWS signal (and the associated actions) is, specifically provided to protect against the "ATWS by rods failure" sequences, being F2 classified. It actuates boration using 7000 ppm enriched boron (corresponding to 11200 ppm natural boron) via the RBS [EBS] pumps. The boration by the RCV [CVCS] pumps is not considered.

3.3.2.6.2. Assumptions for the analysis

3.3.2.6.2.1. Definition of studied case

In order to minimise the core power decrease during the transient, the initial power state is considered to be BLX, when the moderator coefficient is at its minimum absolute value.

PCSR studies at a lower core power level of 4250 MWth have shown that the transient at 100% NP initial state bounds the 60% power level. It demonstrates that both RCP [RCS] temperature and power are the most important parameters for DNB calculations.

In the present study, in order to pessimise the DNB assessment, the calculation is performed at full power. It bounds all the power levels in state A.

3.3.2.6.2.2. Single failure and preventive maintenance

Single failure and preventive maintenance are not taken into account in this analysis.

3.3.2.6.2.3. Initial and boundary conditions

Typical initial conditions are at 102% FP (see Sub-chapter 16.5 - Table 22).

The RCP [RCS] thermal hydraulic design flow rate is assumed.

For this study, the systems, which are not claimed, are not modelled:

- Most of the non classified systems (except those which pessimise the transient),
- Safety classified systems which do not contribute to the transient mitigation (e.g. RIS [SIS]).

All other relevant systems are available.

Boundary conditions defining system performance are generally based on conservative characteristics. Sensitivity studies are used to identify the conservative direction to pessimise the DNB.

For systems directly contributing to meeting the safety/decoupling criteria, which includes the ATWS dedicated features, boundary conditions are based on conservative characteristics– minimum or maximum data is conservatively assumed (e.g. PSV for RCP [RCS] overpressure limitation, definition of ATWS signal, RBS [EBS] for core boration, ASG [EFWS] temperature).Some additional assumptions are taken. Signals related to reactor trip are delayed to delay RBS [EBS] actuation and the effective boron injection into the core.

3.3.2.6.2.4. Neutronic data

The neutronic data refer to BLX conditions

The moderator coefficient is pessimised, and a value of -13.2 pcm/°C applies at the start of the transient.

The bounding initial RCP [RCS] boron concentration is 1594 ppm.

3.3.2.6.2.5. Protection and mitigation actions

The following I&C functions provide protection and mitigation following a loss of off-site power followed by a reactor trip failure due to a mechanical failure of the rods:

- Reactor trip signal on "RCP [RCS] pump speed < MIN1" (F1A),
- Reactor trip signal on "SG pressure > MAX1" (F1A),
- Reactor trip signal on "SG level (narrow range) < MIN1" (F1A),
- Reactor trip signal on "pressuriser pressure > MAX2" (F1A),
- ATWS signal on "reactor trip signal, and high rods position (or high flux) after a time delay" (F2),
- RBS [EBS] actuation on ATWS signal (F2),
- VDA [MSRT] opening on "SG pressure > MAX1" (F1A),
- ASG [EFWS] actuation on "SG level (wide range) < MIN2" (F1A).

In addition, three PSV and two SG safety valves (MSSV) per SG are available (F1A).

The protection setpoints are pessimised in order to reduce the minimum DNB ratio.

3.3.2.6.2.6. Assumptions related to controls

SG level and RCP [RCS] temperature controls are not relevant (ARE [MFWS]/AAD [SSS] and control rods unavailable).

Pressuriser pressure control via normal spray and emergency power heaters is not taken into account, because, the conservative initial pressure (152.5 bar) would cause the pressure control to actuate the heaters at the beginning of the transient. This would result in a higher pressuriser pressure at the time of minimum DNB which is non-conservative.

Steam generator pressure control via GCT [MSB] is not accounted for as the system is non classified.

Pressuriser level control is not modelled in the present analysis.

3.3.2.6.2.7. Assumptions related to systems

The F1A systems used are: ASG [EFWS], PSV, VDA [MSRT], SG safety valves (MSSV).

The F2 functions used are: RBS [EBS] boration. RBS [EBS] is an F1A system, automatically actuated by the F2 classified ATWS signal. The automatic RBS [EBS] boration function is therefore F2 classified.

The setpoints, delays and flow capacities are shown in Sub-chapter 16.5 - Table 23.

3.3.2.6.2.8. Special assumptions not included in the above

The ATWS signal is actuated 20 seconds after the reactor trip signal, and starts the RBS [EBS] which provides 7000 ppm boric acid injection (corresponding to 11200 ppm natural boron). When the SG level wide range falls below the MIN2 value, all the main reactor coolant pumps are tripped. Automatic boration via the RCV [CVCS] pumps is not claimed in the analysis.

3.3.2.6.3. Safety criteria

For this demonstration, the following decoupled acceptance criteria are considered:

- The number of fuel rods experiencing DNB remains below 10%,
- The RCP [RCS] integrity is not challenged as the pressure at the worst point of the RCP [RCS] does not exceed 130% of the design pressure, i.e. 228.5 bar abs (Sub-chapter 3.4).

3.3.2.6.4. Methods of analysis

The analysis is carried out using the internal coupling of:

- the MANTA V3.7 code for the overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), including the effect of F2/F1 systems operations,
- the SMART V4.8.1/FLICA-IIIF V3 codes for the neutronic and thermal-hydraulic behaviour of the core.

The DNBR calculation is performed using the FLICA code. The DNBR calculation takes account of the axial power distribution in the hot channel and the FΔH provided by a 3D core calculation from the SMART code at the time of minimum DNBR. For the current analyses, no conservatism on the local thermal power is assumed for the DNBR calculation.

3.3.2.6.5. Results

Sub-chapter 16.5 - Table 24 gives the sequence of events.

The change in parameters versus time is presented in:

- Sub-chapter 16.5 – Figure 21
- Sub-chapter 16.5 – Figure 22
- Sub-chapter 16.5 – Figure 23
- Sub-chapter 16.5 – Figure 24

- Sub-chapter 16.5 – Figure 25
- Sub-chapter 16.5 – Figure 26
- Sub-chapter 16.5 – Figure 27
- Sub-chapter 16.5 – Figure 28

At 5 seconds, the LOOP trips the turbine, the reactor coolant pumps and all the ARE [MFWS] pumps. The SG level decreases and the heat transfer rate from the primary to the secondary side falls. This leads to an increase in the primary temperature and pressure. The deterioration in the reactor conditions (SG level, pressure, temperature, reactor coolant pump speed) leads to the generation of a reactor trip signal. The first trip signal reached in this transient is the RT on "reactor coolant pumps speed < MIN1 (91% nominal speed)".

However, as the rods are assumed to fail to enter the core, the core power does not decrease following reactor trip. As the turbine has been lost due to the LOOP initiating event, the secondary pressure increases until opening the VDA [MSRT] occurs. At the same time, the temperature in the primary side is increasing, and the moderator density effect causes the core power to fall. This is not sufficient to stop a demand on the PSVs to limit the primary overpressure.

At 20 seconds after reactor trip, detection of the control rods still at a high position results in the generation of the ATWS signal. This leads to the automatic actuation of the RBS [EBS].

Even with the core power reduction, the steam generators are unable to provide the required heat transfer. The steam generator level continues to decrease until the MIN2 setpoint is reached, leading to actuation of the ASG [EFWS]. This reduces the rate at which SG water inventory is decreasing.

At about 800 seconds, when boron reaches the core, the plant is at a controlled state:

- the SG pressure is controlled by VDA [MSRT],
- the SG water inventory has stabilised,
- the RCP [RCS] temperature and pressure are also stable,
- the core is sub-critical.

The plant is stable with the primary heat removed by the VDA [MSRT] and the ASG [EFWS]. The reactivity will continue to fall as the boron, injected by RBS [EBS] in the cold leg, reaches the core.

3.3.2.6.6. Conclusions

The minimum DNBR is reached 7 seconds after the RCP [RCS] pumps are tripped at a value of 2.12 compared to its initial value of 2.17. Consequently, the no fuel rods experience DNB.

The decay heat is safely removed via the VDA [MSRT] and the ASG [EFWS] using the SG. Once the boron reaches the core, even with the reactor coolant pumps tripped, the reactivity will decrease. The activity release during the accident is limited as none of the barriers (fuel and RCP [RCS]) are breached.

In addition the maximum pressure reached in the pressuriser is 176.6 bar abs, which is substantially below the decoupling criterion of 130% of the design pressure.

The calculation results show that for the sequence "ATWS by rods failure – Loss of off-site power", the acceptance criteria are met and the required final state is reached.

Therefore it is concluded that the lower level safety function ‘Highly concentrated boron injection’ provides an efficient diverse mean to mitigate the event assuming the loss of the lower level safety function ‘Negative reactivity fast insertion’.

3.3.2.7. ATWS by loss of RPR [PS] – Short-term loss of off-site power (≤ 2 hours)

This section presents the ATWS transient resulting from a RPR [PS] failure with loss of off-site power (LOOP) in State A at power. In this case, the LLSF “Fast reactor shutdown” is temporarily lost.

The loss of off-site power is an overheating event due to a reduction in primary circuit heat removal via the secondary side. The event causes an overheating on both the primary and secondary sides.

3.3.2.7.1. Typical sequence of events

The sequence considered is initiated by the loss of off-site power. It causes a turbine trip, and loss of all the reactor coolant pumps and the ARE [MFWS] pumps.

Following the loss of off-site power, the plant parameters, such as SG level or pressuriser pressure change from their normal values. As the Protection System is assumed to fail, neither F1A actions such as RPR [PS] reactor trip nor ASG [EFWS] start-up are actuated. The primary heat continues to be is removed by the steam generators. However as there is no feedwater addition, the SG levels decrease. The deterioration in the conditions leads to the non TXS platform setpoints being reached. This additional feature (F2 classified) is provides a diverse fast power reduction capability following RPR [PS] failure.

When the turbine is tripped, the valves are closed and main steam pressure increases. The VDAs [MSRT]s are not demanded the opening function is implemented in the RPR [PS]. However, the MSSVs are finally opened, and stabilise the secondary side pressure. On the primary side, the temperature and pressure increase leading to PSV actuation.

After a certain period of time, as long as the steam generators have not emptied, the secondary and primary parameters stabilise.

3.3.2.7.2. Specific assumptions

3.3.2.7.2.1. Definition of studied case

In the present study, to increase the likelihood of core boiling, the calculation is performed at 102% FP (with NP 4500 MWth). It bounds all power levels in state A.

The calculation is performed in two steps. The first step uses a 3D core model with MANTA SMART FLICA coupling. In this model the initial power state of BLX is modelled when the moderator temperature coefficient is at its minimum absolute value. The second step is a mid transient calculation performed with MANTA 0D, based on the first calculation and conservative assumptions for decay heat.

3.3.2.7.2.2. Single failure and preventive maintenance

Single failure and preventive maintenance are not taken into account.

3.3.2.7.2.3. Initial and boundary conditions

Conservative initial conditions are assumed, including a core power of 102% NP.

The RCP [RCS] thermal hydraulic design flow rate is chosen although not required by accident analyses rules.

For this study, the systems which are not claimed (most of the non- classified systems, except those which pessimise the transient) and classified system which does do not intervene in the study (e.g. RIS [SIS]) are not modelled. Other systems are available.

Boundary conditions defining system performance are based on conservative characteristics (minimum or maximum data assumed for the limiting purposes, (e.g. PSV for RCP [RCS] overpressure limitation).

Initial conditions considered for this analysis are presented in Sub-chapter 16.5 - Table 25.

3.3.2.7.2.4. Neutronic data

At the beginning of the transient, the neutronic data for BLX conditions are modelled.

The moderator coefficient is pessimised with a value of -13.2 pcm/°C.

The bounding initial RCP [RCS] boron concentration is 1594 ppm.

3.3.2.7.2.5. Protection and mitigation actions

The following I&C functions provide protection and mitigation following the loss of off-site power with ATWS due to RPR [PS] failure:

- diverse reactor trip / turbine trip signal on high hot leg pressure.

In addition, the following systems are also available:

- three PSV (F1A),
- two MSSVs per SG (F1A).

Consistent with the safety analysis rules, no operator mitigation actions occur before 30 minutes after the RT signal.

This later phase dealing with Emergency Operating Procedures is not modelled in this calculation.

3.3.2.7.2.6. Assumptions related to controls

The controls are not modelled in this transient because of the assumed RPR [PS] failure.

3.3.2.7.2.7. Assumptions related to protections

In this study only the automatic actions are considered.

The non TXS platform is designed to initiate some automatic actions, such as the diverse reactor trip, etc. For this transient the reactor trip setpoint on high hot leg pressure is reached.

The manual operations and associated systems are assumed to be used by the operator after 30 minutes, but this phase is not assessed in the study. Finally:

- The F1A systems assumed to operate are: PSV, MSSV.
- The F2 functions assumed to operate are: diverse reactor trip.

The setpoints, delays and flow capacities assumed are listed in Sub-chapter 16.5 - Table 26.

In addition, automatic boration via RCV [CVCS] pumps is not claimed in the analysis.

3.3.2.7.3. Methods of analysis

In order to model this transient, the calculations are performed with two steps:

- the first step uses the MANTA SMART FLICA coupling until the reactor trip (short-term transient). This step models a 3D core.
- the second step is a MANTA calculation based on the conservative power transient from the first step calculations with a conservative decay heat curve {CCI Removed} [b]. This step models a 0D core.

The analysis is carried out using:

- the MANTA V3.7 code for the overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), modelling the operation of the F1/F2,
- the SMART V4.8.1/FLICA-IIIF V3 codes for the neutronic and thermal-hydraulic behaviour of the core.

The DNBR calculation is performed using the FLICA code. The axial power distribution in the hot channel and the FΔH at the time of minimum DNBR are provided by a 3D core calculation using the SMART code. For this calculation, an additional conservatism on the local thermal power is not considered for the DNBR calculation. This conservatism would decrease the initial DNBR value to the LCO value independently of the core physics.

3.3.2.7.4. Results

Sub-chapter 16.5 – Table 27 gives the sequence of events.

The change in parameters versus time is presented in:

- Sub-chapter 16.5 – Figure 29
- Sub-chapter 16.5 – Figure 30

- Sub-chapter 16.5 – Figure 31
- Sub-chapter 16.5 – Figure 32

At 5 seconds, the LOOP trips the turbine, the reactor coolant pumps and all the ARE [MFWS] pumps. The SG level decreases, and the rate of heat transfer from the primary to the secondary side is reduced. As a consequence of the turbine trip the primary temperature and pressure increase. The deterioration in the conditions (SG level, pressure, temperature) leads to the actuation of the diverse reactor trip as failure of the RPR [PS] is assumed. The reactor trip is actuated following a “hot leg pressure > MAX (173 bar)” signal.

The power is rapidly reduced and the SG is capable of removing the decay heat until 1800 seconds, as the reactor trip occurs within the first 20 seconds of the event. At 20 seconds the steam generators are filled with almost 75 tons of water. During the transient the SG levels decrease steadily and support continued residual heat removal. When the operator intervenes, each SG still contains approximately 23 tons of water.

When the operator begins the emergency operating procedures 30 minutes after the RT signal (around 1815 seconds), the plant is under acceptable conditions. The manual operations assumed to manage the transient are ASG [EFWS] operation and VDA [MSRT] operation.

3.3.2.7.5. Conclusions

The minimum DNBR is reached 7 seconds after the reactor coolant pumps are tripped; its value is 2.12 compared to its initial value of 2.17. As a result, the DNBR remains significantly above the acceptance criterion. The maximum pressuriser pressure reached is 175.8 bar abs, which is lower than the acceptance criterion of 130% of the design pressure.

The calculation results show that for the sequence "ATWS by RPR [PS] failure – Loss of off-site power", the acceptance criteria are met and the required final state is reached.

As a conclusion, in such a case, the lower level safety function ‘Highly concentrated boron injection’ provides an efficient diverse mean to mitigate the event assuming the loss of the lower level safety function ‘Negative reactivity fast insertion’.

3.3.2.8. ATWS by mechanical blockage of the rods - Excessive increase in secondary steam flow

3.3.2.8.1. Typical sequence of events

The sequence considered is initiated by an increase in steam flow to the turbine.

In the secondary side, an increase of turbine steam flow leads to a reduction in secondary pressure and steam generator level. In the reactor coolant system, an increase of turbine steam flow leads to a decrease of primary temperature, pressure and pressuriser level and an increase in reactivity. A reactor trip signal and turbine trip are assumed to be actuated following a low SG level in the modelled sequence.

As the control/shutdown rods have failed to insert into the core, the reduction of the reactor power only results from the reduction in reactivity caused by the decrease in moderator density in the short term.

The Pressuriser Safety Valves (PSV) open to limit the primary side overpressure.

Without any additional actions, this state would be stable whilst the steam generators have sufficient water inventories to remove the primary power (core power plus RCP [RCS] pump heat).

To limit the primary pressure transient, a dedicated ATWS signal is provided. The ATWS signal is generated in the RPR [PS], from detection of "RT signal and rods out (or flux high) after a time delay". This ATWS signal (and the associated actions) is a RRC-A feature and is specifically implemented to protect against "ATWS by rods failure" sequences and is F2 classified. It trip the RCP [RCS] pumps following a “very low SG-water level” signal (SG level WR < MIN2), which occurs before the complete SG depletion occurs. By this action, the reactor power is reduced consistently with the decreasing coolant flow rate, which results in a lower pressure increase on the primary side.

The ATWS signal also automatically initiates the RBS [EBS] injection of 7000 ppm enriched boron (corresponding to 11200 ppm natural boron) thus automatically providing core sub-criticality in the medium term. The non-safety classified REA [RBWMS] via the RCV [CVCS] could also perform this boration function, but is not considered in this study.

3.3.2.8.2. Assumptions for the analysis

3.3.2.8.2.1. Definition of studied case

To minimise the core power decrease during the transient the initial power state is considered at beginning of life (BLX) when the moderator temperature coefficient has its minimum absolute value.

Studies at a lower core power level of 4300 MWth have shown that the transient initiated from the full power initial state bounds all power levels between 100% NP and 0% NP.

Therefore the ATWS excessive increase in secondary steam flow at full power is analysed in the present study.

3.3.2.8.2.2. Single failure and preventive maintenance

Single failure and preventive maintenance are not taken into account in this analysis.

3.3.2.8.2.3. Initial and boundary conditions

Typical initial core power is at 102% NP.

The initial conditions are summarised in Sub-chapter 16.5 – Table 28.

Systems lost by definition of the sequence (e.g. no rod drop), non-classified systems and systems with no impact on the transient are not considered. The steam generator level control is modelled as this results in a more conservative transient.

Boundary conditions defining system performance are generally based on conservative assumptions.

For systems directly supporting the achievement of a safety/acceptance criteria (which includes the RRC-A dedicated features), the boundary conditions are based on conservative characteristics (minimum or maximum data assumed for the relevant criterion, e.g. PSV for RCP [RCS] overpressure limitation, definition of ATWS signal, RBS [EBS] for core boration).

3.3.2.8.2.4. Neutronic data

At the beginning of the transient, the neutronic data are the BLX conditions used for 3D modelling.

The moderator coefficient is pessimised, and has a value of -13.2 pcm/°C. The bounding initial RCP [RCS] boron concentration is 1594 ppm.

3.3.2.8.2.5. Protection and mitigation actions

The following I&C functions provide protection and mitigation following an excessive increase in secondary steam flow followed by a reactor trip failure due to a mechanical failure of the rods:

- Reactor trip / turbine trip signal on "SG level (narrow range) < MIN1" (F1A),
- ATWS signal on reactor trip signal and high rods position (or high flux) after a time delay (F2),
- RBS [EBS] actuation on ATWS signal (F2),
- RCP [RCS] pumps trip on "SG level (wide range) < MIN2" if the ATWS signal has been generated (F2),
- VDA [MSRT] opening on "SG pressure > MAX1" (F1A),
- ASG [EFWS] actuation on "SG level (wide range) < MIN2" (F1A).

In addition, three PSVs and two MSSVs per steam generator are available (F1A).

3.3.2.8.2.6. Assumptions related to controls

Control rods are unavailable due to the postulated mechanical blockage of the rods.

Pressuriser pressure control via normal spray is modelled to give a higher depressurisation during primary pressure peaks, thus pessimising the DNBR calculation.

Pressuriser pressure control via the pressuriser heaters is not taken into account.

SG pressure control via GCT [MSB] is not taken into account.

SG level regulation is modelled.

Pressuriser level control is not modelled in the present analysis.

3.3.2.8.2.7. Assumptions related to systems

The F1A systems assumed to operate are: ASG [EFWS], PSV, VDA [MSRT] and MSSV.

The F2 functions assumed to operate are: RBS [EBS] boration.

The setpoints, delays and flow capacities are listed in Sub-chapter 16.5 – Table 29.

3.3.2.8.2.8. Special assumptions not included in the above

The ATWS signal is actuated 20 seconds after the reactor trip signal, starting the RBS [EBS] giving 7000 ppm boric acid injection (corresponding to 11200 ppm natural boron). When the SG level wide range falls below the MIN2 setpoint, all main coolant pumps are tripped.

Automatic boration via the RCV [CVCS] pumps is not claimed in the analysis.

3.3.2.8.3. Safety criteria

For this demonstration, the following decoupled acceptance criteria are considered:

- The number of fuel rods experiencing DNB remains below 10%,
- the RCP [RCS] integrity is not challenged (as an acceptance criterion, the pressure at the worst point of the RCP [RCS] does not exceed 130% of the design pressure, i.e. 228.5 bar abs (Sub-chapter 3.4).

3.3.2.8.4. Methods of analysis

The analysis is carried out using the internal coupling of:

- the MANTA V3.7 code for overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), modelling the necessary F2/F1 systems operation,
- the SMART V4.8.1/FLICA-IIIF V3 codes for the neutronic and thermal-hydraulic behaviour of the core.

The DNBR calculation is performed with the FLICA code. The DNBR calculation takes account of the axial power distribution in the hot channel and the $F\Delta H$ provided by a 3D core calculation at the time of minimum DNBR using the SMART code. For the current analyses, no conservatism on the local thermal power is considered for DNBR calculation.

3.3.2.8.5. Results

Sub-chapter 16.5 – Table 30 gives the sequence of events.

The main parameters versus time are presented in:

- Sub-chapter 16.5 – Figure 33
- Sub-chapter 16.5 – Figure 34
- Sub-chapter 16.5 – Figure 35
- Sub-chapter 16.5 – Figure 36
- Sub-chapter 16.5 – Figure 37
- Sub-chapter 16.5 – Figure 38
- Sub-chapter 16.5 – Figure 39

- Sub-chapter 16.5 – Figure 40

The maximum pressure downstream of the RCP [RCS] pumps is reached just after the opening of the first PSV. The VDA [MSRT] are actuated as a result of turbine trip.

The MSSVs are actuated as a result of turbine trip. The second and third PSVs are not actuated during the transient. The reactor trip signal is reached 313 seconds after the initiating event.

The ASG [EFWS] is actuated 407.0 seconds after the initiating event.

3.3.2.8.6. Conclusions

The peak pressure downstream of the RCP [RCS] pumps is smaller than the acceptance criterion of 130% Design Pressure (DP), by a large margin (175.03 bar abs is 99.4% of DP).

The DNBR calculated before turbine trip for the 102% NP case is the minimum DNBR during the transient. The reactor coolant pump trip occurs after the power reduction caused by turbine trip and PSV opening. The primary temperature is high, the primary pressure and power low, and consequently the DNBR is high. As a result, the DNBR remains higher than 1.0. Therefore no fuel rod experiences DNB.

The decay heat is safely removed via the VDA [MSRT], low-load ARE [MFWS] and ASG [EFWS] via the SG.

The activity release during the accident is limited as none of the barriers (fuel and RCP [RCS]) is breached.

The calculation results show that for the sequence "ATWS by rods failure – Excessive increase in secondary steam flow", the acceptance criteria are met and the required final state is reached.

Consequently, in such a case, the lower level safety function ‘Highly concentrated boron injection’ provides an efficient diverse means to mitigate the event assuming the loss of the lower level safety function ‘Negative reactivity fast insertion’.

3.3.2.9. ATWS by loss of RPR [PS] - Excessive increase in secondary steam flow

3.3.2.9.1. Typical sequence of events

The sequence considered is initiated by an increase of turbine steam flow.

On the secondary side, an increase in turbine steam flow results in a decrease of the secondary pressure and steam generator (SG) level. In the reactor coolant system, an increase of turbine steam leads to a withdrawal of control rods, an increase of reactivity and primary pressure. The reactor trip signal and turbine trip are actuated following a low SG level via the diversified reactor trip.

After reactor trip and turbine trip, residual power leads to an increase and stabilisation of the primary temperatures, pressuriser level and primary pressure. On the secondary side, after reactor trip and turbine trip, the low ARE [MFWS] flow rate injects water into the steam generator. SG levels increase slowly and residual power is successfully removed. As a consequence of the complete failure of the reactor trip system on demand from the RPR [PS], the VDA [MSRT] and ASG [EFWS] are not available. The secondary pressure increases until the MSSVs open. The plant is stable once the valves have opened.

This state is stable until manual operator actions are taken, 30 minutes after the reactor trip at the end of the simulation.

The RCV [CVCS] could perform the boration function, but is not considered in this study as it is not safety classified.

3.3.2.9.2. Assumptions for the analysis

3.3.2.9.2.1. Definition of studied case

To minimise the core power decrease during the transient the initial power state is at beginning of life (BLX) when the moderator temperature coefficient at its minimum absolute value.

Studies at a lower core power level of 4300 MWth have shown that the transient at 100% NP initial state bounds all power levels between 100% NP and 0% NP.

Therefore the ATWS excessive increase in secondary steam flow at full power is analysed in the present study.

3.3.2.9.2.2. Single failure and maintenance

Single failure and maintenance are not taken into account.

3.3.2.9.2.3. Initial and boundary conditions

The typical initial conditions are at 102% FP, see Sub-chapter 16.5 – Table 31.

The RCP [RCS] thermal hydraulic design flow rate is chosen.

Systems lost by the definition of the sequence, non classified systems and systems which do not impact the transient are not considered.

In order to pessimise the transient, the steam generator level control is modelled.

Boundary conditions defining system performance are generally based on conservative assumptions. For systems directly contributing to the achievement of a safety/acceptance criterion (which includes the RRC-A dedicated feature), the boundary conditions are based on conservative characteristics (minimum or maximum data assumed for the relevant criterion, e.g. PSV for RCP [RCS] overpressure limitation).

3.3.2.9.2.4. Neutronic data

At the beginning of the transient, the neutronic data used for 3D modelling are at BLX conditions.

The moderator coefficient is pessimised at a value of -13.2 pcm/°C with a bounding initial RCP [RCS] boron concentration of 1594 ppm.

3.3.2.9.2.5. Assumptions related to controls

Pressuriser pressure control via the normal spray is assumed to increase the depressurisation during primary pressure increases thus reducing the calculated DNBR.

Average coolant temperature control via the rods is assumed to increase the power at the beginning of the transient, thus reducing the calculated DNBR.

Pressuriser pressure control via the pressuriser heaters is not taken into account.

Steam generator pressure control via GCT [MSB] and pressuriser level control are not taken into account in the present analysis.

3.3.2.9.2.6. Assumptions related to systems

The systems controlled by the Protection System (RPR [PS]) are unavailable: RT system, VDA [MSRT], ASG [EFWS].

SG level control is modelled.

The setpoints, delays and flow capacities are pessimised and listed in Sub-chapter 16.5 – Table 32.

3.3.2.9.3. Safety criteria

For this demonstration, the following PCC-3/PCC-4 decoupling acceptance criteria are considered:

- the number of fuel rods experiencing DNB is not higher than 10% of the total core,
- the RCP [RCS] integrity is not challenged (as an acceptance criterion, the pressure at the worst point of the RCP [RCS] does not exceed 130% of the design pressure).

The analysis presented in this document demonstrates that the DNBR criterion is met. The maximum primary pressure reached will be given for information.

3.3.2.9.4. Methods of analysis

The analysis is carried out using the MANTA V3.7 code to model the overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), accounting for F2/F1 systems operation. The DNBR calculation is performed using the FLICA code.

The ATWS event ‘excessive increase in steam removal’ due to the failure of the RPR [PS] case is studied with 0D MANTA modelling to model the average coolant temperature control by the RCCAs, as this control is conservative for this transient. The core modelling is less precise than in the other ATWS events, and therefore an additional conservatism is included in the initial DNBR. The initial DNBR is fixed at the LCO threshold (DNBR = 1.50) due to the bounding 0D modelling of the core.

3.3.2.9.5. Results

Sub-chapter 16.5 - Table 33 gives the sequence of events.

The most representative parameters are presented in:

- Sub-chapter 16.5 – Figure 41

- Sub-chapter 16.5 – Figure 42
- Sub-chapter 16.5 – Figure 43
- Sub-chapter 16.5 – Figure 44
- Sub-chapter 16.5 – Figure 45
- Sub-chapter 16.5 – Figure 46

The increase in steam flow results in a decrease in the secondary side pressure and a decrease in the reactor coolant temperature. The average coolant temperature control acts to limit this temperature variation. The reactor power is increased and stabilises at around 115% NP. Following the core power increase, the DNBR decreases and stabilises at its minimum value of 1.10. The plant conditions remain stable except for the SG inventory which decreases despite the actions of the SG level control.

The diverse reactor trip setpoint is reached 923.5 seconds after the initiating event.

The peak pressure downstream of the RCP [RCS] pumps is reached after reactor trip, whilst secondary pressure is stabilised by operation of the MSSVs.

The MSSVs are opened as a result of turbine trip and VDA [MSRT] unavailability.

Pressuriser safety valves are not actuated during the transient.

The lowest DNBR is reached in the first part of the transient ($DNBR_{min}$ = 1.10).

3.3.2.9.6. Conclusions

The decoupling safety criteria dealing with core damage are met as the number of fuel rods experiencing DNB is not higher than 10% of the total core, peak clad temperature remains below 1482°C and melted fuel at the hot spot does not exceed 10% by volume. The lowest DNBR is reached before turbine trip ($DNBR_{min}$ = 1.10).

There is no activity release during the accident as none of the barriers (fuel and RCP [RCS]) is breached.

The value of the peak pressure downstream of the RCP [RCS] pumps is lower than the acceptance criterion of 130% design pressure, with a high significant margin, as the PSV are not actuated.

The calculation results show that for the sequence "ATWS by a complete failure of the Protection System RPR [PS] – Excessive increase in secondary steam flow", the acceptance criteria are met and the required final state is reached.

Consequently, in this case, the lower level safety function ‘fast negative reactivity insertion’ is performed via efficient diverse means to mitigate the event.

3.3.2.10. ATWS excessive increase in Secondary Steam Flow due to TXS failure

3.3.2.10.1. Typical sequence of events

The sequence considered is initiated by an increase in turbine steam flow. On the secondary side, an increase in turbine steam flow leads to a decrease in secondary side pressure and steam generator (SG) level. In the reactor coolant system, an increase in turbine steam flow leads to withdrawal of the control rods, and to an increase in reactivity and primary pressure. The reactor trip signal and turbine trip are actuated on low SG level or high Neutron Flux (diversified reactor trip), if needed. After reactor trip and turbine trip, the residual power leads to an increase and stabilisation of primary circuit temperatures, pressuriser level and primary pressure. On the secondary side, after reactor trip and turbine trip, the low ARE [MFWS] flow rate injects water into the steam generator. SG levels increase slowly and the residual power is removed. Due to the complete failure of the reactor trip system on demand from the RPR [PS], the VDA [MSRT] and ASG [EFWS] are not available. Secondary pressure increases until the MSSVs open. The state is stable after their opening.

This state is stable until manual operator actions, which are assumed to occur 30 minutes after reactor trip (end of simulation).

The RCV [CVCS] could perform the boration function, but it is not considered in this analysis because it is not safety classified.

3.3.2.10.2. Assumptions for the analysis

3.3.2.10.2.1. Definition of case analysed

In order to minimise the core power decrease during the transient, the initial power state is considered at beginning of life (BLX) when the absolute value of moderator temperature coefficient is minimised. Studies at a lower core power level of 4300 MWth have shown that the transient from 100% NP initial state covers all power levels between 100% NP and 0% NP. Therefore, the ATWS excessive increase in secondary steam flow at full power is analysed in the present study.

3.3.2.10.2.2. Single failure and maintenance

Single failure and maintenance are not taken into account.

3.3.2.10.2.3. Initial and boundary conditions

The typical initial condition is at 102% FP, see Sub-chapter 16.5 - Table 67.

RCP [RCS] thermal hydraulic design flow rate is assumed. Systems lost by definition of the sequence, non-classified systems, and systems without impacts are not considered.

In order to pessimise the transient, steam generator level control is credited. Boundary conditions defining system efficiency are generally based on conservative assumptions. In particular, for systems directly participating in the achievement of a safety/acceptance criteria (which includes the RRC-A dedicated feature), the boundary conditions are based on conservative characteristics (minimum or maximum data assumed for limiting purposes, e.g. PSV for RCP [RCS] overpressure limitation).

3.3.2.10.2.4. Neutronic data

At the beginning of the transient, the neutronic data are those for BLX conditions used for 3D modelling.

The moderator coefficient is pessimised, with a value of -13.2 pcm/°C. The bounding initial RCP [RCS] boron concentration is 1594 ppm.

3.3.2.10.2.5. Assumptions related to controls

Pressuriser pressure control via normal spray is assumed, in order to have a higher depressurisation rate in response to primary pressure peaks, which is conservative for DNBR calculations. Average coolant temperature control by the rods is assumed, in order to increase power at the beginning of the transient, which is also conservative for DNBR. Pressuriser pressure control via the pressuriser heaters is not represented. Steam generator pressure control via the GCT [MSB] and pressuriser level control are not represented in the present analysis.

3.3.2.10.2.6. Assumptions related to systems

The systems controlled by the Protection System RPR [PS] are unavailable: RT system, VDA [MSRT], and ASG [EFWS]. SG level control is modelled.

The setpoints, delays and flow capacities are pessimised and are listed in Sub-chapter 16.5 – Table 68.

3.3.2.10.3. Safety criteria

For this demonstration, the following PCC-3/PCC-4 decoupling acceptance criteria are considered:

- the number of fuel rods experiencing DNB remains below 10% of the total number in the core,
- the RCP [RCS] integrity is not impaired (as an acceptance criterion, the pressure at the worst point in the RCP [RCS] does not exceed 130% of the design pressure, i.e. 228.5 bar abs (Sub-chapter 3.4).

The analysis presented below demonstrates that the DNB criterion is met. The maximum primary pressure reached is given for information.

3.3.2.10.4. Methods of analysis

The analysis is carried out using the internal coupling of:

- the MANTA V3.7 computer code (PCSR Appendix 14A) for analysis of the overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), accounting for F2/F1 system operation,
- the SMART V4.8 /FLICA-IIIF V3 computer codes for analysis of neutronic and thermal-hydraulic behaviour of the core.

The ATWS event ‘excessive increase in steam removal’ due to the failure of the RPR [PS] case is analysed taking into account the average coolant temperature control by the RCCAs, as this control is conservative for this transient. The initial DNBR value is fixed at the LCO threshold (DNBR = 1.32).

3.3.2.10.5. Results

Sub-chapter 16.5 – Table 69 gives the sequence of events.

The most representative parameters are presented in:

- Sub-chapter 16.5 – Figure 83: Core power and reactor coolant pump speed,
- Sub-chapter 16.5 – Figure 84: RCP [RCS] temperatures and pressuriser pressure,
- Sub-chapter 16.5 – Figure 85: SG pressure and PSV flow rate,
- Sub-chapter 16.5 – Figure 86: Pressuriser and SG levels,
- Sub-chapter 16.5 – Figure 87: Main feedwater flow rate and steam flow rate,
- Sub-chapter 16.5 – Figure 88: DNBR transient.

The increase in steam flow results in a decrease in the secondary side pressure and a decrease in the reactor coolant temperature. The average coolant temperature control limits this temperature variation. The reactor power is increased and stabilises at around 115% NP. Following the core power increase, the DNBR decreases and stabilises at its minimum value of 0.97. The number of fuel rods experiencing DNB is 0.39%, thus the criterion of 10% is fulfilled.

The plant conditions remain stable except for the SG inventory, which decreases despite the SG level control.

The diverse reactor trip setpoint is reached 860.0 seconds after the initiating event. The peak pressure downstream of the RCP [RCS] pumps is reached after reactor trip, whilst secondary pressure is stabilised by operation of the MSSVs. The MSSVs are opened as a result of turbine trip and VDA [MSRT] unavailability. Pressuriser safety valves are not actuated during the transient. The lowest DNBR in reached in the first part of the transient ($DNBR_{min}$ = 0.97).

Additional analyses of a steam line break, which leads to a similar increase in steam flow to the event described above, have been performed with different assumptions regarding the main feedwater to cover remaining uncertainties in its design. By considering a feedwater flow rate equal to the steam flow rate, the SG level does not decrease.

Specific causes of the increase in steam flow could result in the following break sizes:

- VDA [MSRT] opening ⇔253 cm²
- MSSV opening ⇔ 158 cm²

The largest excessive increase in steam flow is due to the spurious VDA [MSRT] opening. According to the calculations of the DNB for the spectrum of break sizes, the plant would not enter conditions leading to DNB. Thus additional protection in the RPR [PS] or SAS is not required, since the PCC-2 decoupling safety criterion is met (no DNB) without triggering any RT signal.

The operator would have time to identify that the plant is not in its normal conditions, e.g. by following the power transient, and to perform a manual reactor trip from the RPR [PS] or from a diversified platform in the case of the RPR [PS] being unavailable.

The diversified High Neutron Flux signal implemented in the SAS protects the plant from the risk of excessive power increase due to overcooling. It is designed to be actuated at around the same level of power as the RPR [PS] function RT on high core power level.

3.3.2.10.6. Conclusions

The decoupling safety criteria dealing with core damage are met with high margins, as the calculated number of fuel rods experiencing DNB is 0.39%. The lowest calculated DNBR is obtained before turbine trip ($DNBR_{min}$ = 0.97). There is no activity release during the accident as none of the barriers (fuel and RCP [RCS]) are impaired.

Additionally, should the increase in steam flow rate be higher, the SAS reactor trip signal on “high neutron flux” would act. It limits the percentage of rods entering DNB to less than 0.25%.

A non-coupled calculation has demonstrated that the value of the peak pressure downstream of the RCP [RCS] pumps is lower than the acceptance criterion of 130% DP, with a high margin, as the PSVs are not actuated.

The calculation results show that for the sequence "ATWS by a complete failure of the Protection System RPR [PS] – Excessive increase in secondary steam flow", the acceptance criteria are met and the required final state is reached. Consequently, in this case, the lower level safety function ‘fast negative reactivity insertion’ is performed via efficient diverse means to mitigate the event.

3.3.2.11. Rod drop faults (and rod misalignment faults) with ATWS due to failure of TXS

This section presents analysis of RCCA misalignment up to rod drop combined with a complete failure of the Teleperm XS (TXS) platform leading to an ATWS. A complete failure of the TXS includes a failure of both the PS and RCSL systems.

A complete failure of the reactor trip system on demand from the reactor RPR [PS] can result from either:

- a failure of the automatic F1A reactor shutdown signals (i.e. none of the signals sent by the RPR [PS] de-energises the control rod drive coils),or
- failure of the control and shutdown rods to insert into the core after de-energisation of their drive coils. In this case, actuation of the rods due to control or limitation signals also fails.

This section deals with the first case, due to a failure of the TXS platform.

3.3.2.11.1. Typical sequence of events

The accidental drop of one or more RCCAs into the core is characterised by a fast negative reactivity insertion and a temporary reduction of the core power whereas the secondary power remains constant. Without automatic reactor protection, the primary/secondary-side power imbalance results in a decrease in primary coolant temperature, as well as primary and main steam pressures.

The initial core power decrease is mainly terminated by the reactivity feedback effects of the negative Doppler reactivity coefficient. Subsequently, in the absence of temperature regulation the core power increases due to moderator reactivity feedback effects. A new primary/ secondary-side equilibrium state is reached. This equilibrium is determined by the secondary-side power level and the reduced primary coolant temperature with elevated power peaking factors.

As no automatic shutdown signals are generated by the TXS platform, neither on low DNBR nor on low primary pressure, the combination of the power increase, the decrease in primary pressure and a distorted power distribution, caused by the presence of dropped rods, may result in DNB. The mid-term phase does not present any risk of damage to the fuel and cladding, since the core power and primary pressure of the new equilibrium state are roughly the same as the initial values, the core temperature is lower, and the increase in axial power distribution is not significant enough to result in DNB.

Sub-chapter 16.5 – Figures 96 and 97 represent the typical variation in the relevant parameters for 102% FP, i.e. 4590 MW (core power, inlet temperature, axial offset, primary pressure and rod bank positions).

3.3.2.11.2. Safety criteria

For the safety demonstration, the safety criterion of no degradation of the fuel cladding used is defined in PCSR Sub-chapter 14.0:

- the number of fuel rods experiencing DNB must not be higher than 10% of the total core.

3.3.2.11.3. Assumptions for the analysis

The analysis methodology is based on the following approach:

- identification of the dominant phenomena
- application of conservative PCC analysis rules.

The dominant phenomena are split into two different categories:

- System transient dominant phenomena:
    - Asymmetric Reactor Coolant System (RCP [RCS]),
    - RCP [RCS] depressurisation.
- Core transient dominant phenomena:
    - A top skewed axial power distribution (Pz),
    - The shape of the fuel census curve (representation of the enthalpy increase factor in the core).

The application of conservative PCC analysis rules leads to pessimism of the plant initial conditions.

The assumption applied in the analysis provide conservative results; they are presented in the following sub-sections.

3.3.2.11.3.1. Initial conditions

The initial conditions that are assumed in order to provide conservative results correspond to:

- Thermal hydraulic flow rate,
- Hot Full Power (HFP) at both BLX and EOL,
- Upper threshold (rightmost) for Axial Offset (AO),
- RCCAs at their insertion limits,
- Initial DNBR.

The assumed initial conditions are detailed in Sub-chapter 16.5 – Table 73.

3.3.2.11.3.2. Assumptions on dominant phenomena

This sub-section presents the assumptions made for the dominant phenomena identified in this section.

a) System transient phenomena

Assuming no temperature regulation due to the loss of TXS, the relevant neutronic parameters for this transient analysis are:

- Enthalpy peaking factor (FΔH): a high enthalpy rise peaking factor variation is conservative regarding DNBR. As the initial FΔH is set to the DNBR LCO, the relevant parameter is the enthalpy peaking factor variation (ΔFΔH)
- Dropped rod worth (Δρ): assuming a constant load demand, a low rod worth reduces the contribution of the neutronic feedback in relation to the power increase and leads to a higher value of primary power, which is conservative regarding DNBR. At a fixed value of ΔFΔH, a lower rod worth is more conservative.

Therefore, for failure of temperature regulation in the case of a rod drop event, the dominant parameter for DNBR is the enthalpy rise peaking factor variation (ΔFΔH) during the transient.

b) Selected cases

The selected configurations have been chosen from conservative cases used for Pellet Clad Interaction (PCI) studies.

In order to maximise the RCP [RCS] asymmetry, cases for three dropped RCCAs have been considered.

Without a functional TXS platform, the cases selected for analysis are those which maximise the FΔH during the transient. Among these cases, those having a low rod worth must be considered.

The case with the maximum FΔH variation is then selected. Sub-chapter 16.5 – Figures 94 and 95 show that a 5% reduction in the rod worth results in at least a 7% reduction in FΔH variation and justifies that there is no requirement to analyse other cases with lower rod worth. Neutronic data for the retained cases are given in Sub-chapter 16.5 – Table 74.

c) Core transient phenomena

The core power axial distribution is conservatively set at 30%; this corresponds to the preliminary value of RT setpoint on a non TXS platform. Then, the axial power shape (Pz) is taken from the fuel assembly with the maximum power.

3.3.2.11.4. Results and conclusion

Sub-chapter 16.5 – Figure 98 represents the variation in rods experiencing DNB following rod drop for the most onerous case studied (BLX with maximum FΔH increase, see Sub-chapter 16.5 – Table 74). The most onerous configuration, for comparison with the safety criteria, is reached before the equilibrium state, and the number of fuel rods experiencing DNB remains below 1% throughout the transient. As the maximum clad temperature reached during the transient remains below 1190°C (see Sub-chapter 16.5 – Figure 99), the structural integrity of the fuel rods is ensured.

An analysis of RCCA misalignment up to rod drop with ATWS due to loss of TXS platform has been performed. The study uses conservative assumptions, and concludes that PCC-4 criteria are met; therefore there is no requirement to add an extra trip signal on a non TXS platform.

3.3.3. R3 – Prevention of uncontrolled positive reactivity insertion in the core

3.3.3.1. Prevention of overcooling via the secondary side

Prevention of an uncontrolled positive reactivity insertion is performed by the closure of the main feedwater supply. Several means provide the closure of the feedwater supply. Operator Aid Functions, in the Process Automation System (PAS), lead to a closing signal to the full load and low load control valves. The Protection System (RPR [PS]) demands closure of the full load and low load isolation valves if the SG level > MAX1 (NR). In addition, a diverse isolation instruction to the full load and low load isolation valves are provided in the non TXS platform if the SG level > MAX (WR).

The excessive increase in steam flow without main steam isolation valve (VIV [MSIV]) closure is not analysed in this section. The ATWS excessive increase in steam flow does not demand the VIV [MSIV] closure because the depressurisation rate is too slow to reach the closure setpoint (see section 3.3.2.8). This event is also bounded by the ATWS event for the DNBR criteria.

3.3.3.2. Reactivity transients

The uncontrolled RCCA bank withdrawal at power and the RCV [CVCS] malfunction causing a decrease in boron concentration challenge this PLSF. However, they are bounded by the C1 PLSF ‘maintain integrity of the fuel cladding’ and are presented in section 3.5.1.

3.3.4. R4 – Maintain sufficient sub-criticality of fuel stored outside the reactor coolant system but within the site

The plant level safety function R4 is not applicable to these events.

3.4. HEAT REMOVAL SAFETY FUNCTION

3.4.1. H1 – Maintain sufficient reactor coolant system water inventory for core cooling

The plant level safety function H1 is challenged by events leading to a loss of reactor coolant system water. The bounding transient among the frequent PIEs is the Small break LOCA (< DN 50) including a break occurring on the extra boration system injection line (states A and B). The lower level safety function 'water injection into the RCP [RCS]' is provides this plant level safety function via the safety functional groups 'Steam Generator Pressure Control - Cooling - Auto (VDA [MSRT])' and 'MHSI – Auto'.

Therefore, three cases are analysed here to demonstrate the diversity to deliver the plant level safety function.

- Small break LOCA (< DN 50) including a break occurring on the extra boration system injection line (states A and B) without MHSI
- Small break LOCA (< DN 50) including a break occurring on the extra boration system injection line (states A and B) without a partial cooldown signal
- Small break LOCA with failure of the VDA [MSRT]

3.4.1.1. Small break LOCA (< DN 50) including a break occurring on the extra boration system injection line (states A and B) without MHSI

Following a Small Break LOCA, the PLSF H1 "Maintain sufficient Reactor Coolant system water inventory for core cooling" is challenged as the RCP [RCS] coolant is gradually lost via the break. If the lower level safety function 'Medium head injection into the RCP [RCS]' cannot be achieved, it is replaced by 'Fast water injection into the RCP [RCS]' and 'Low head injection into the RCP [RCS]' for mitigation of the accident.

3.4.1.1.1. Typical sequence of events

The first part of the sequence is identical to a SB LOCA.

The second part of the sequence is specific to this event, involving the operator actions required to mitigate the accident.

The first part of the sequence, identical to a typical SB LOCA, is described below:

- The break results in a loss of reactor coolant inventory. The RCV [CVCS] cannot compensate for the break. The loss of primary coolant results in a decrease in primary system pressure and pressuriser level.
- A reactor trip occurs following a low pressuriser pressure (< MIN2) signal. The reactor trip signal automatically trips the turbine and closes the ARE [MFWS] full load control valves.
- As the secondary side pressure increases, the Main Steam Bypass (GCT [MSB]) valves open, allowing steam dump to the condenser.

- The steam generators are fed by the ARE [MFWS] through the low load control valves. If the ARE [MFWS] is unavailable, the start-up and shutdown pump starts and feeds the SG through the low load control valves.

- A Safety Injection signal is actuated following a very low pressuriser pressure (< MIN3) signal. The RIS [SIS] signal automatically starts the RIS [SIS] pumps and initiates a partial cooldown of the secondary system. This action cools the primary system and lowers the RCP [RCS] pressure. MHSI is assumed to be unavailable.

- RCP [RCS] pumps trip is actuated via the RIS [SIS] signal combined with a low ΔP (< 80%) across three pumps. Subsequent natural circulation maintains the heat transfer via the secondary side and provides the decay heat removal.

- The pressuriser heaters are shut down following a low pressuriser level (< 12% Maximum Range, MR) signal.

The second part of the sequence is specific to the loss of MHSI:

- At the end of the partial cooldown, the RCP [RCS] pressure is about equal to the secondary side at around 60 bar with the RCP [RCS] saturated. This is above the pressure required for injection from the accumulator or LHSI (MHSI is assumed to be unavailable). Therefore the mitigation of the accident requires operator action.

- In this case, the operator strategy consists in decreasing the secondary side pressure to below the accumulator injection pressure and subsequently the LHSI shut-off head using the VDA [MSRT]. The GCT [MSB] is not used because it would cause main steam header isolation on a SG pressure drop signal or “SG pressure < MIN1”.

- The criterion for initiating fast cooldown of the secondary side is defined conservatively as:
    - All MHSI unavailable and a 30 minutes delay after the SI signal.

- As the RCP [RCS] water inventory can be low when the criterion is reached and the ΔTsat < ε at the core outlet, the cooldown is required to be fast to allow the rapid injection by the accumulators. This operation is called "fast cooldown".

- The fast cooldown is actuated by adjusting the setpoint of the VDA [MSRT] to a value lower than the LHSI delivery pressure, resulting in the full opening of the four VDA [MSRT].

- As soon as the fast cooldown is actuated by the operator, the RCP [RCS] pressure decreases rapidly to allow injection by the accumulators and then by LHSI. The accumulator injection rapidly restores the coolant inventory in the reactor pressure vessel. The long-term core cooling is provided by the LHSI.

- During the period that the break cannot remove the decay heat, the RCP [RCS] pressure is governed by the secondary side at a level compatible with LHSI and core cooling.

- The final state is then maintained by control of RCP [RCS] inventory using at least one LHSI and control of the RCP [RCS] temperature via heat transfer to the SG (VDA [MSRT] or GCT [MSB] if available, and ASG [EFWS]) or via at least one other LHSI. The transfer to the LHSI/RHR operating mode is possible once the conditions inside the RCP [RCS] loops allow the operation of the LHSI pumps in RHR mode.
  - o RCP [RCS] hot leg pressure < 32 bar.
  - o RCP [RCS] hot leg temperature < 180°C.
  - o ΔTsat and Reactor Pressure Vessel (RPV) level consistent with LHSI/RHR suction from the hot leg.

3.4.1.1.2. Assumptions for the analysis

The study is performed with conservative assumptions.

Operator actions are claimed 30 minutes after reactor trip.

3.4.1.1.2.1. Initial conditions

The conservative initial conditions considered are presented in Sub-chapter 16.5 - Table 34.

Initial core power is 102% FP, i.e. 4590 MWth.

Thermal-hydraulic flow conditions are considered within the primary circuit.

3.4.1.1.2.2. Decay Heat

A conservative decay heat curve is considered
{CCI Removed} [b].

3.4.1.1.2.3. Assumptions related to systems

The non-safety systems are not considered for the safety analysis unless their effect is pessimistic for the study.

The F1A systems taken into account are: RIS [SIS], VDA [MSRT] and ASG [EFWS] (see Sub-chapter 16.5 – Table 35).

3.4.1.1.2.4. Assumptions related to controls

Conservative I&C setpoints are used with uncertainties appropriate to degraded conditions (see Sub-chapter 16.5 – Table 35).

3.4.1.1.2.5. Single failure and preventive maintenance

Neither single failure nor preventive maintenance are taken into account.

3.4.1.1.2.6. Fast cooldown assumptions

The cooldown is actuated 30 minutes after the RIS [SIS] signal with no MHSI. If $\Delta T_{sat} < \varepsilon$ at the core outlet, a fast cooldown is performed. The operator fully opens the VDA [MSRT] to decrease the pressure inside the SGs until atmospheric conditions are reached.

3.4.1.1.2.7. Reactivity balance

A reactivity calculation is not performed in the CATHARE calculation. The core is assumed to remain sub-critical for the entire transient once reactor trip occurs. This assumption is based on the following arguments:

- At the actuation of the partial cooldown, the void fraction of the coolant increases due to the fast RCP [RCS] depressurisation rate.
- When the “fast cooldown” is actuated, the void fraction in the core is high.
- The accumulators inject borated water once the RCP [RCS] pressure reaches 45 bar, corresponding to a saturation temperature of 257°C. At this temperature, the core remains sub-critical with a high shutdown margin as all rods are inserted.
- The accumulator borated water enters the core at a fast rate, as the fast RCP [RCS] depressurisation rate causes a high accumulator injection flow rate, with a low RPV water inventory at the time of accumulator injection.
- The LHSI provides additional borated water once the RCP [RCS] pressure falls below 20 bar, corresponding to a saturation temperature of 210°C. In this temperature range, core re-criticality could occur in the absence of additional boron injection. However at the time of LHSI injection, the core boron concentration has already been significantly increased, by the injection of approximately 40 tons of borated water into the RPV by the accumulators.

3.4.1.1.3. Safety criteria

For this sequence, it must be demonstrated that the acceptance criteria for LOCA in PCC-3/ PCC-4 are met:

- The peak cladding temperature shall remain lower than 1200°C.
- The maximum cladding oxidation shall remain lower than 17% of the total cladding thickness.
- the maximum hydrogen generation, produced by the chemical reaction of the cladding with water or steam, must not exceed 1% of the hydrogen that would be produced if all the cladding materials had reacted (with exclusion of the expansion volume cladding),
- The core geometry shall remain coolable, any calculated changes in core geometry shall be such that the core remains coolable.
- The long-term cooling shall be ensured: the RCP [RCS] coolant inventory shall be stabilised or increased and the decay heat shall be removed.

3.4.1.1.4. Method

Calculations for the overall plant, system and core behaviour are performed using the CATHARE 2 V2.5 computer code.

3.4.1.1.5. Results

This study is performed with the same assumptions as a typical LOCA. The non-safety systems are not claimed except the pressuriser heaters as their action is pessimistic for the calculation of the RCP [RCS] pressure.

Typical sequence of events is given in Sub-chapter 16.5 – Table 36.

The most representative parameters are presented in the following figures:

- Sub-chapter 16.5 – Figure 48
- Sub-chapter 16.5 – Figure 49
- Sub-chapter 16.5 – Figure 50
- Sub-chapter 16.5 – Figure 51

3.4.1.1.5.1. Sequence of events

At time 0 seconds, a small break (20 cm²) in a cold leg in state A causes the loss of RCP [RCS] coolant. It results in a decrease in the RCP [RCS] pressure and pressuriser level. The pressuriser heaters operate at full power (2596 kW), trying to alleviate the depressurisation. This is conservative for the break flow rate, and delays the generation of the reactor trip and RIS [SIS] signals.

At 79 seconds, the pressuriser pressure reaches the MIN2 setpoint (135 bar -3 bar uncertainty). The reactor trip is actuated, followed by the turbine trip and the ARE [MFWS] full-load isolation. The RCP [RCS] temperature decreases, increasing the depressurisation. As the MSB is not considered for this study, the secondary pressure rises to the VDA [MSRT] setpoint (95.5 bar + 1.5 bar uncertainty).

At 110 seconds, the pressuriser level falls below 12% measured range. The pressuriser heaters are switched off.

At 172 seconds, the pressuriser pressure reaches the MIN3 setpoint (115 bar -3 bar uncertainty). The RIS [SIS] signal is actuated, followed by the partial cooldown signal. The secondary side pressure starts to decrease at -250°C/h. The RCP [RCS] pressure follows the secondary side pressure.

At 534 seconds, the ΔP across three RCP [RCS] pumps is lower than 75% (80% - 5% uncertainty). This, combined with the existing RIS [SIS] signal leads to the generation of a RCP [RCS] pump trip signal. Natural circulation takes place inside the primary side, to provide the core cooling.

At 625 seconds, the secondary side pressure has reached 61.5 bar (60 bar + 1.5 bar uncertainty). As long as the primary and secondary sides remain coupled, the RCP [RCS] pressure remains close in value to the secondary side pressure. This pressure is too high to allow injection from the accumulators and the LHSI. The RIS [SIS] cannot compensate for the break flow and therefore the RCP [RCS] coolant inventory continues to decrease.

At 1013 seconds, the SG level reaches the MIN2 setpoint (40% WR - 5% uncertainty). The ASG [EFWS] is actuated to maintain the SG cooling capability.

The operator actions start 30 minutes after the RIS [SIS] signal. As $\Delta T_{sat} < \varepsilon$ at the core outlet, the operator performs a “fast cooldown” on the secondary side using the VDA [MSRT] to cool down the RCP [RCS] and decrease its pressure.

At 2056 seconds, the RCP [RCS] pressure has reached the accumulators injection pressure of 45 bar. The LHSI at 2239 seconds matches the break flow. The RCP [RCS] pressure and coolant inventory subsequently increase until reaching equilibrium conditions with the injection flow matching the break flow. The decay heat removal is provided by the secondary side.

3.4.1.1.5.2. Final state

At 50 minutes after the beginning of the accident, the RCP [RCS] coolant inventory is restored. The loops are totally flooded, allowing the RRA [RHR] pumps to be initiated.

The primary pressure is stabilised at 16 bar. This value is defined by the difference between the break flow rate and the LHSI flow rate. The RCP [RCS] coolant temperature, which has fallen below 180°C, continues to decrease.

The RRA [RHR] connecting conditions are met.

3.4.1.1.6. Conclusions

The core stays completely covered and the RCP [RCS] coolant inventory is stabilised. Therefore, the following conclusions can be drawn concerning the SB LOCA with loss of MHSI:

- The peak cladding temperature remains below the 1200°C acceptance criteria,
- The maximum oxidation of the cladding does not exceed 17% of the total thickness of the cladding at the hot spot,
- There is no cladding failure,
- The integrity of the core geometry is maintained,
- Long-term cooling is ensured.

All the LOCA acceptance criteria are met with significant margins.

The residual power is efficiently removed. The final state is reached. The water inventory is maintained using at least one LHSI pump. The SG, or at least one other LHSI pump operating in residual heat removal mode, is sufficient to cooldown the RCP [RCS].

Following accumulator injection and injection from the LHSI pump into the RCP [RCS], core sub-criticality is maintained. In this case, no additional boration is required.

The calculation results show that, for the transient "Small break LOCA without MHSI", the PCC-3/PCC-4 acceptance criteria are met and the required final state can be reached using the RRC-A feature “manual activation of fast secondary cooldown”. This has been demonstrated assuming conservative initial conditions and only claiming classified safety systems.

The diversity of the PLSF ‘Maintain sufficient Reactor Coolant System water inventory for core cooling’ is demonstrated. The LHSI are an adequate diverse means to inject water in the reactor coolant system.

Consequently, in such a case, the plant level safety function ‘Maintain sufficient Reactor Coolant System water inventory for core cooling’ is achieved.

The lower level safety functions ‘Fast water injection into the RCP [RCS]’ and ‘Low head injection into the RCP [RCS]’.are efficient diverse means to fulfil this plant level safety function via the safety functional groups ‘Steam Generator Pressure Control - Cooling - Auto (VDA [MSRT])’ and ’MHSI – Auto’.

3.4.1.2. Small break LOCA (< DN 50) including a break occurring on the extra boration system injection line (states A and B) without partial cooldown signal

Following a small break LOCA, the PLSF H1 ”Maintain sufficient Reactor Coolant System water inventory for core cooling” is challenged since the RCP [RCS] coolant is gradually lost via the break. The LLSFs ‘Medium head injection into the RCP [RCS]’, ‘Fast water injection into the RCP [RCS]’ and ‘Low head injection into the RCP [RCS]’ cannot be achieved as the RCP [RCS] pressure is too high to allow injection.

A manual cooldown shall be performed by the operator to restore the above safety functions and mitigate the accident.

3.4.1.2.1. Typical sequence of events

The first part of the sequence is identical to a typical SB LOCA. The second part of the sequence is specific to this event, involving the operator actions required to mitigate the accident.

The first part of the sequence, identical to a typical SB LOCA, is described below:

- The break results in a loss of reactor coolant inventory. The RCV [CVCS] cannot compensate for the break. The loss of primary coolant results in a decrease in primary system pressure and pressuriser level.
- A reactor trip occurs following a “low pressuriser pressure (< MIN2)” signal. The reactor trip signal automatically trips the turbine and closes the ARE [MFWS] full load control valves.
- As the secondary side pressure increases, the Main Steam Bypass (GCT [MSB]) valves open, allowing steam dump to the condenser.
- The pressuriser heaters are shut down following a “low pressuriser level (< 12% MR)” signal.
- The SG are fed by the ARE [MFWS] through the low load control valves. With the unavailability of the ARE [MFWS], the start-up and shutdown pump starts and feeds the SG through the low load control valves.

- A Safety Injection signal is actuated following a “very low pressuriser pressure (< MIN3)” signal. The RIS [SIS] signal automatically starts the MHSI and LHSI pumps and actuates the signal for the partial cooldown of the secondary system. This signal fails and the depressurisation of primary and secondary systems does not occur.

- RCP [RCS] pump trip is actuated via a combination of the RIS [SIS] signal and a “low ΔP (< 80%) across three pumps” signal. Natural circulation maintains the heat transfer to the secondary side and maintains the decay heat removal.

The second part of the sequence is specific to the failure of the partial cooldown signal:

- Following failure of the partial cooldown signal, the SG pressure remains at the GCT [MSB] setpoint of 90 bar. The RCP [RCS] pressure is approximately equal to the secondary side at about 90 bar as the RCP [RCS] reaches saturation. This is too high to allow injection by the RIS [SIS] (MHSI, LHSI or accumulators). The RIS [SIS] cannot compensate for the break and therefore the mitigation of the accident relies on operator action.

- In that case, the operator strategy consists of decreasing the secondary side pressure to reach the MHSI injection pressure using the VDA [MSRT]. The GCT [MSB] is not used because main steam header isolation would occur following either a “SG pressure drop” signal or “SG pressure < MIN1” signal.

- The criterion for initiating the manual partial cooldown of the secondary side is conservatively defined as:

  - No secondary system automatic partial cooldown after the SI signal combined with a 30 minute delay after the SI signal.

- The manual partial cooldown is achieved by decreasing the setpoint of the VDA [MSRT] to 60 bar at a rate of -250°C/hour.

- As soon as the manual partial cooldown is actuated by the operator, the RCP [RCS] pressure decreases, allowing the MHSI. The MHSI rapidly restores the RCP [RCS] coolant inventory.

- During the period when the break cannot remove the decay heat, the RCP [RCS] pressure is governed by the secondary side at a level compatible with MHSI and core cooling.

- The final state is then achieved by control of RCP [RCS] inventory using the MHSI and control of the RCP [RCS] temperature through the heat transfer to the SG (VDA [MSRT] or GCT [MSB] if available, and ASG [EFWS]).

3.4.1.2.2. Assumptions for the analysis

The study is performed with conservative assumptions.

Operator actions can be claimed 30 minutes after reactor trip.

3.4.1.2.2.1. Initial conditions

The conservative initial conditions considered are presented in Sub-chapter 16.5 – Table 37.

The initial core power is 102% FP, i.e. 4590 MWth.

Thermal-hydraulic flow conditions are assumed within the primary circuit.

3.4.1.2.2.2. Decay Heat

A conservative decay heat curve is modelled
{CCI Removed} [b].

3.4.1.2.2.3. Assumptions related to systems

The non-safety systems are not considered for the safety analysis unless they are pessimistic for the study.

The systems modelled are: RIS [SIS], VDA [MSRT] and ASG [EFWS] (see Sub-chapter 16.5 – Table 38).

3.4.1.2.2.4. Assumptions related to controls

Conservative I&C setpoints are used with uncertainties for degraded conditions (see Sub-chapter 16.5 –Table 38).

3.4.1.2.2.5. Single failure and preventive maintenance

Neither single failure nor preventive maintenance are taken into account.

3.4.1.2.2.6. Partial cooldown assumptions

The operator initiates a manual partial cooldown of the secondary system with a depressurisation rate of -250°C/hour, and stabilises the SG pressure at 61.5 bar (60 bar +1.5 bar uncertainty). This action is performed 30 minutes after the RIS [SIS] signal assuming no partial cooldown of the secondary system.

3.4.1.2.2.7. Reactivity balance

A reactivity calculation is not performed in the CATHARE calculation. The core is assumed to remain sub-critical for the entire transient once reactor trip has occurred. This assumption is based on the following arguments:

- When the manual partial cooldown is actuated, the void fraction of the coolant increases due to the rapid RCP [RCS] depressurisation rate.

- The MHSI injects borated water once the RCP [RCS] pressure falls below 85 bar, corresponding to a saturation temperature of 299 C. At this temperature, the core remains sub-critical with a significant shutdown margin as all the rods are inserted.

- At the end of the partial cooldown, the RCP [RCS] pressure is about 60 bar, corresponding to a saturation temperature of 275°C. At this temperature level, the core remains sub-critical with a significant shutdown margin as all the rods are inserted.

3.4.1.2.3. Safety criteria

For this sequence, it must be demonstrated that the acceptance criteria for the LOCA in PCC-4 are met:

- The peak cladding temperature shall remain lower than 1200°C.
- The maximum cladding oxidation shall remain lower than 17% of the total cladding thickness.
- the maximum hydrogen generation, produced by the chemical reaction of the cladding with water or steam, must not exceed 1% of the hydrogen that would be produced if all the cladding materials had reacted (with exclusion of the expansion volume cladding),
- The core geometry shall remain coolable: calculated changes in core geometry shall be such that the core remains coolable.
- The long-term cooling shall be achieved: the RCP [RCS] coolant inventory shall be stabilised or increasing and the decay heat shall be removed.

3.4.1.2.4. Method

Calculations for the overall plant, system and core behaviour are performed using the CATHARE 2 V2.5 computer code.

3.4.1.2.5. Results

This study is performed with the same assumptions as a typical LOCA. The non-safety systems are not claimed except the pressuriser heaters (Their action is conservative for the RCP [RCS] pressure and break flow).

Typical sequence of events is given in Sub-chapter 16.5 – Table 39.

The most representative parameters are presented in the following figures:

- Sub-chapter 16.5 – Figure 52
- Sub-chapter 16.5 – Figure 53
- Sub-chapter 16.5 – Figure 54
- Sub-chapter 16.5 – Figure 55

3.4.1.2.5.1. Sequence of events

At time 0 seconds, a small break (20 cm²) in a cold leg in state A results in the loss of RCP [RCS] coolant. It results in a decrease in the RCP [RCS] pressure and pressuriser (PZR) level. The pressuriser heaters are at full power (2596 kW), attempting to offset the depressurisation (This assumption is conservative for the break flow rate).

At 79 seconds, the pressuriser pressure reaches the MIN2 setpoint (135 bar -3 bar uncertainty). The reactor trip is actuated, followed by the turbine trip and the ARE [MFWS] full load isolation. The RCP [RCS] temperature decreases, increasing the depressurisation. As the MSB is not considered for this study, the secondary pressure rises to the VDA [MSRT] setpoint (95.5 bar + 1.5 bar uncertainty).

At 110 seconds, the pressuriser level falls below 12% measured range. The pressuriser heaters are switched off.

At 172 seconds, the pressuriser pressure reaches the MIN3 setpoint (115 bar – 3 bar uncertainty). The RIS [SIS] signal is actuated but failure of the partial cooldown signal occurs. The secondary side pressure remains at the VDA [MSRT] setpoint of 97 bar. Therefore, the RCP [RCS] pressure remains close to the SG pressure and higher than the maximum MHSI injection pressure.

At 684 seconds, the ΔP across three of the RCP [RCS] pumps falls below 75% (80% - 5% uncertainty). This, combined with the RIS [SIS] signal causes the tripping of the RCP [RCS] pumps. Natural circulation takes place inside the primary side, maintaining core cooling.

The operator actions start 30 minutes after the RIS [SIS] signal. The operator performs a manual partial cooldown on the secondary side using the VDA [MSRT] to cool down the RCP [RCS] and decrease its pressure to allow MHSI.

At 2058 seconds, the SG level reaches the MIN2 setpoint (40% WR - 5% uncertainty). The ASG [EFWS] is actuated to maintain the SG cooling capability.

At 2145 seconds, the RCP [RCS] pressure has reached the MHSI injection pressure (85 bar). The primary and secondary pressures continue to decrease.

The manual partial cooldown ends at 2424 seconds. The SG pressure has reached 61.5 bar (60 bar + 1.5 bar uncertainty). The RCP [RCS] pressure is close to the SG pressure, allowing RIS [SIS] flow rate of approximately 70 kg/s.

The RCP [RCS] temperature continues to fall as the power removed from the RCP [RCS] by the break and the SG is greater than the residual power. Therefore the RCP [RCS] pressure follows the temperature decrease, increasing the rate of RIS [SIS] injection.

At 3510 seconds, the RCP [RCS] pressure has reached the accumulators injection pressure of 45 bar. At this point the MHSI totally compensates for the break. The RCP [RCS] coolant inventory starts to increase. The decay heat removal is maintained by the secondary side.

3.4.1.2.5.2. Final state

By 50 minutes after the beginning of the accident, the RCP [RCS] coolant inventory is stabilised at above 100 ton. The MHSI fully compensates for the break. The core remains flooded. The RCP [RCS] pressure and temperature continue to fall. The core cooling is maintained by the secondary systems and the break. The controlled state is reached, the long-term operator actions can start.

3.4.1.2.6. Conclusions

The core stays completely covered and the RCP [RCS] coolant inventory is stabilised. Therefore, the following conclusions can be drawn concerning a SB LOCA without partial cooldown:

- The peak cladding temperature remains below the 1200°C acceptance criteria,
- The maximum oxidation of the cladding does not exceed 17% of the total thickness of the cladding at the hot spot,
- There is no cladding failure,
- The integrity of the core geometry is maintained,
- Long-term cooling is ensured.

All the LOCA acceptance criteria are met with significant margins and without the requirement for the operation of non-safety systems.

The residual power is efficiently removed. The controlled state is reached. The water inventory is maintained by the MHSI. The core cooling is provided by the SG and the break.

The calculation results show that, for the "Small break LOCA with failure of partial cooldown signal" transient, the PCC-3/PCC-4 acceptance criteria are met and the required final state can be reached by means of the RRC-A feature “manual partial cooldown”. This can be achieved with conservative initial conditions and assuming the operation of classified safety systems only.

Should complete failure of the VDAs [MSRT]s occur, the manual actuation of the VDAs [MSRT]s contributing to the LLSF ‘heat removal by SG – emergency shutdown mode’ is not achieved and is backed up by the LLSF ‘Heat removal by ECCS’ through the feed and bleed procedure. The performance of the feed and bleed procedure is demonstrated for the bounding case loss of feedwater combined with the loss of the LLSF ‘heat removal by SGs – emergency shutdown mode’.

Consequently, in such a case, the lower level safety function ‘Heat removal by ECCS’ provides an efficient diverse means to mitigate the event assuming the loss of the lower level safety function ‘removal by SG – emergency shutdown mode’.

3.4.1.3. Small break LOCA with failure of the VDAs [MSRT]s

In the case of a small break LOCA, the plant level safety function H1 ”Maintain sufficient Reactor Coolant system water inventory for core cooling” is challenged since the RCP [RCS] coolant is gradually lost from the break. The lower level safety functions ‘Medium head injection into the RCP [RCS]’, ‘Fast water injection into the RCP [RCS]’ and ‘Low head injection into the RCP [RCS]’ cannot be fulfilled because the RCP [RCS] pressure is too high to allow the injection. In the case of inoperability of the GCT [MSB], the RCP [RCS] pressure cannot be lowered by the secondary side and a Feed and Bleed (opening of the Primary Depressurisation System (PDS) with RIS [SIS] injection) will be actuated by the operator.

The purpose of this analysis is to demonstrate the efficiency of such a back-up line with regard to core consequences. For this demonstration, the criterion for the operator to actuate Feed and Bleed is low loop level (RPV level lower than the bottom of the hot leg). This enables the operator to restore the previous safety functions and mitigate the accident.

Note that Feed and Bleed could also be actuated on an RCP [RCS] overheating criterion. The final decision regarding the feed and bleed actuation criterion (and its optimisation) should be considered taking account of the detailed Emergency Operating Procedures, and will be part of the site licensing phase. Nevertheless, the current transient analysis demonstrates that Feed and Bleed, when actuated in due time, provides an efficient back-up line.

3.4.1.3.1. Typical sequence of events

The first part of the sequence is identical to a SB LOCA. The second part of the sequence is specific to this event, involving the operator actions required to mitigate the event. The first part of the sequence, identical to a typical SB LOCA, is described below:

- The break results in a loss of reactor coolant inventory. The RCV [CVCS] cannot compensate for the break. The loss of primary coolant results in a decrease in the primary system pressure and the pressuriser level.

- A reactor trip occurs following a low pressuriser pressure (< MIN2) signal. The Reactor Trip signal automatically trips the turbine and closes the ARE [MFWS] full-load control valves.

- As the secondary side pressure increases, the Main Steam Bypass (GCT [MSB]) valves open, allowing steam dump to the condenser.

- The pressuriser heaters are shut down following a low pressuriser level (< 12% MR) signal.

- The steam generators (SGs) are fed by the ARE [MFWS] through the low-load control valves. If the ARE [MFWS] is unavailable, the start-up and shutdown pump starts and feeds the SG through the low-load control valves.

- A Safety Injection signal is actuated following a very low pressuriser pressure (< MIN3) signal. The RIS [SIS] signal automatically starts Medium Head Safety Injection (MHSI) and Low Head Safety Injection (LHSI) pumps and initiates a partial cooldown of the secondary system. The Main Steam Relief Train (VDA [MSRT]) fails to open and the depressurisation of primary and secondary systems is not initiated.

- RCP [RCS] pumps trip is actuated on the RIS [SIS] signal combined with a low ΔP (< 80%) over three pumps. Subsequent natural circulation maintains the heat transfer to the secondary side and provides the decay heat removal.

The second part of the sequence is specific to the failure of the VDAs [MSRT]s:

- Due to the failure of the partial cooldown signal and of the VDAs [MSRT]s to open, the SG pressure remains at the GCT [MSB] setpoint (90 bar), if available, or at the pressure setpoint of the MSSVs. The RCP [RCS] pressure is approximately equal to the secondary side pressure, which is too high to enable injection by the RIS [SIS] (MHSI, LHSI or accumulators). The RIS [SIS] cannot compensate for the break; therefore the mitigation of the accident relies on operator actions.

- In this case, the strategy consists of decreasing the primary side pressure to reach the MHSI injection pressure by means of one pressuriser discharge valve (PDS). The GCT [MSB] is not used in the analysis because it is not safety classified.

- The criterion for initiating the Feed and Bleed is defined conservatively as:
  - Reactor Pressure Vessel (RPV) level lower than the bottom of the hot leg
- The PDS is fully opened
- As soon as the PDS is opened, the RCP [RCS] pressure decreases, allowing RIS [SIS] injection. The RIS [SIS] injection quickly restores the RCP [RCS] coolant inventory.
- The final state is then maintained by the equilibrium between the break/PDS flow rates and RIS [SIS] flow rates.

3.4.1.3.2. Assumptions for the analysis

The analysis is performed with conservative assumptions. Operator actions are claimed conservatively at 30 minutes after the reactor trip. Neither LOOP nor earthquake is added to the initiating event.

3.4.1.3.2.1. Initial conditions

The conservative initial conditions considered are presented in Sub-chapter 16.5 – Table 70.

Initial core power is 102% FP, i.e. 4590 MWth.

Thermal-hydraulic flow conditions are considered within the primary circuit (see Sub-chapter 16.5 – Table 70).

3.4.1.3.2.2. Decay Heat

A conservative decay heat curve is considered {CCI Removed} [b].

3.4.1.3.2.3. Assumptions related to systems

The non-safety systems are not represented for the safety analysis unless their inclusion is pessimistic for the study.

The F1A systems taken into account are RIS [SIS] and ASG [EFWS].

The F2 system taken into account is PDS.

3.4.1.3.2.4. Assumptions related to controls

Conservative I&C setpoints are used with uncertainties appropriate to degraded conditions (see Sub-chapter 16.5 – Table 71).

3.4.1.3.2.5. Single failure and preventive maintenance

Neither single failure nor preventive maintenance are taken into account.

3.4.1.3.2.6. Feed and Bleed assumptions

The Feed and Bleed is actuated on RPV level lower than the bottom of the hot leg.

3.4.1.3.2.7. Reactivity balance

A reactivity calculation is not performed in the CATHARE computer code calculation. The core is assumed to remain sub-critical for the entire transient after RT occurs. This assumption is based on the following arguments:

- At the actuation of the Feed and Bleed, the void fraction of the coolant increases due to the fast RCP [RCS] depressurisation rate.
- The MHSI injects borated water once the RCP [RCS] pressure reaches 85 bar, corresponding to a saturation temperature of 299°C.
- The accumulator borated water enters the core at a fast rate, due to the fast RCP [RCS] depressurisation rate, and due to the low RPV water inventory at the time of accumulator injection.
- The LHSI provides additional borated water once the RCP [RCS] pressure falls below 20 bar, corresponding to a saturation temperature of 210°C. In this temperature range, core re-criticality could occur in the absence of additional boron injection. However, at the time of LHSI injection, the core boron concentration is already high, since about 40 tons of borated water from the accumulators has already been injected into the RPV.

3.4.1.3.3. Safety criteria

For this sequence, it must be demonstrated that the acceptance criteria for the LOCA in PCC-4 are met:

- The peak cladding temperature shall remain lower than 1200°C.
- The maximum local cladding oxidation shall remain lower than 17% of the total cladding thickness.
- the maximum hydrogen generation, produced by the chemical reaction of the cladding with water or steam, must not exceed 1% of the hydrogen that would be produced if all the cladding materials had reacted (with exclusion of the expansion volume cladding),
- The core geometry shall remain coolable: calculated changes in core geometry shall be such that the core remains coolable.
- The long-term cooling shall be ensured: the RCP [RCS] coolant inventory shall be stabilised or increasing and the decay heat shall be removed.

3.4.1.3.4. Method

Calculations for the overall plant, system and core behaviour are performed using the CATHARE 2 V2.5 computer code.

3.4.1.3.5. Results

This study is performed with the same assumptions as a typical LOCA. The non-safety systems are not claimed except for the pressuriser heaters, as their action is conservative for the calculation of RCP [RCS] pressure.

The typical sequence of events is given in Sub-chapter 16.5 – Table 72. The most representative parameters are presented in the following figures:

- Sub-chapter 16.5 – Figure 90: Primary and secondary exchanged power, Primary and secondary pressure
- Sub-chapter 16.5 – Figure 91: SG wide range levels, Core temperatures.
- Sub-chapter 16.5 – Figure 92: Upper plenum liquid level, Core liquid level
- Sub-chapter 16.5 – Figure 93: RIS [SIS], PDS, and break flow rates, Primary and Secondary masses.

3.4.1.3.5.1. Sequence of events

At time 0 seconds, a small break (20 cm²) in a cold leg in state A causes a loss of RCP [RCS] coolant. It results in a decrease of the RCP [RCS] pressure and the pressuriser level. The pressuriser heaters operate at full power (2596 kW), attempting to compensate for the depressurisation. (This assumption is conservative for the calculation of break flow rate).

At 80 seconds, the pressuriser pressure reaches the MIN2 setpoint (135 bar -3 bar uncertainty). The reactor trip is actuated, followed by the turbine trip and the ARE [MFWS] full-load isolation. The RCP [RCS] temperature decreases, increasing the depressurisation. Since the GCT [MSB] is not considered for this study, the secondary pressure rises to the MSSV setpoint (105 bar +1.5 bar uncertainty).

At 125 seconds, the pressuriser level falls below 12% of the measured range. The pressuriser heaters are switched off.

At 215 seconds, the pressuriser pressure reaches the MIN3 setpoint (115 bar – 3 bar uncertainty). The RIS [SIS] signal is actuated, with failure of the VDAs [MSRT]s. The secondary side pressure remains at the SG safety valve setpoint (102 bar). Therefore, the RCP [RCS] pressure is approximately the same as the SG pressure and stays higher than the MHSI injection pressure

At 680 seconds, the ΔP over three RCP [RCS] pumps is less than 75% (80%- 5% uncertainty). The RIS [SIS] signal has already been actuated; therefore, the RCP [RCS] pumps trip. Subsequent natural circulation within the primary circuit maintains the core cooling.

At 2500 seconds the low loop level is reached. The operators start the Feed and Bleed procedure by opening the PDS and allowing the primary side pressure to decrease and the RIS [SIS] to inject into the RCP [RCS].

At 2650 seconds, the RCP [RCS] pressure has fallen to the MHSI injection pressure (85 bar).

At 2930 seconds, the accumulator injection restores the coolant inventory in the reactor pressure vessel.

At 4500 seconds, equilibrium between the break/PDS and RIS [SIS] flow rates is reached, allowing the RCP [RCS] pressure to stabilise at 20 bar, with a stable average temperature of 149°C.

3.4.1.3.5.2. Final state

80 minutes after the beginning of the accident, the RCP [RCS] coolant inventory is stabilised above 175 ton. The MHSI flow rate fully compensates for the break. The core remains flooded. The RCP [RCS] pressure and temperature are stable. The core cooling is provided by the RIS [SIS] and the flow rate at the break and at the PDS. The controlled state is reached, allowing the long-term operator actions to start.

3.4.1.3.6. Conclusions

The core stays mainly covered throughout the transient, the cladding temperature remains below 380°C, and the RCP [RCS] coolant inventory is stabilised. Therefore:

- the peak cladding temperature remains below the 1200°C acceptance criteria,
- the maximum local oxidation of the cladding does not exceed 17% of the total thickness of the cladding at the hot spot,
- there is no cladding failure,
- the integrity of the core geometry is maintained,
- long-term cooling is ensured.

All the LOCA acceptance criteria are met with significant margins and without the need for any non-safety systems.

The residual power is efficiently removed. The controlled state is reached. The water inventory is maintained by the MHSI. The core cooling is provided by the Feed and Bleed and the break flow.

The calculation results show that, for the RRC-A transient "Small break LOCA with failure of 4 Main Steam Relief Trains", the PCC-4 acceptance criteria are met and the required final state can be reached by means of the RRC-A feature “Feed and Bleed” even with initial conservative conditions and consideration of only classified safety systems.

3.4.2. H2 – Remove heat from the core to the reactor coolant

The loss of main feedwater fault challenges the PLSF ‘H2 – Remove heat from the core to the reactor coolant’. The LLSF ‘Core heat removal by RCP [RCS] forced flow in power mode’ is assumed to be unavailable, which leads to the total loss of feedwater event.

The heat transfer from the RCP [RCS] to the steam generators gradually deteriorates and the plant level safety function H2 is achieved by the LLSF ‘Core heat removal by RCP [RCS] natural circulation in shutdown mode’, which mitigates the accident.

3.4.2.1. Loss of normal feedwater flow (loss of all ARE [MFWS] pumps and of the start-up and shutdown pump) without reactor coolant pumps

3.4.2.1.1. Typical sequence of events

Following the loss of normal feedwater flow, the inventory of the steam generators decreases as the core power remains the same and the reactor coolant system pressure and temperature increase. The signal “loss of feedwater”, which generates the drop of all control and shutdown rods and a turbine trip, is normally actuated.

Reactor trip (RT) is automatically actuated following either a high primary pressure, low steam generator level, low DNBR, or “low RCP [RCS] pumps speed” signal. Turbine trip is actuated on reactor trip check-back and the secondary side pressure is limited by steam generators relief devices. Either the main steam bypass or the main steam relief trains, if the main steam bypass is unavailable, can remove the steam produced in the steam generators.

The steam generator water level will continue to decrease and eventually, once the low-low steam generator water level setpoint is reached, the dedicated Emergency Feed Water System (ASG [EFWS]) pump is actuated, to provide the residual heat removal. Subsequently, the controlled state is reached with residual heat being removed via the main steam relief trains on all steam generators. The feedwater supply is provided by the ASG [EFWS].

3.4.2.1.2. Assumptions for the analysis

The study is performed with conservative assumptions.

3.4.2.1.2.1. Initial conditions

The conservative initial conditions considered are presented in Sub-chapter 16.5 – Table 40.

The initial core power is 102% FP, namely 4590 MWth.

Thermal-hydraulic flow conditions are considered within the primary circuit (see Sub-chapter 16.5 - Table 40).

3.4.2.1.2.2. Decay Heat

A conservative decay heat curve is used {CCI Removed} [b].

3.4.2.1.2.3. Assumptions related to systems

The non-safety systems are not considered for the safety analysis unless their operation is conservative for the study.

The F1A systems taken into account are: PSV, VDA [MSRT], MSSV and ASG [EFWS] (see Sub-chapter 16.5 - Table 41).

3.4.2.1.2.4. Single failure and preventive maintenance

Neither single failure nor preventive maintenance are taken into account.

3.4.2.1.2.5. Specific assumptions related to this case

The RCP [RCS] pumps are stopped at the worst time; they are stopped when the reactor is tripped.

- If the RCP [RCS] pumps were stopped before RT, the RT would be initiated earlier due to a "Low RCP [RCS] pumps speed" signal and the RCP [RCS] would heat up for a shorter time. As a result, the energy balance would be more favourable to the mitigation of the transient.
- If the RCP [RCS] pumps were stopped after RT, this would support the performance of the steam generators and so efficiently remove the residual power for a longer time until the RCP [RCS] pumps stopped. As RCP [RCS] pumps contribute to the removal of the residual power, the longer they are running the more favourable the energy balance would be.

3.4.2.1.3. Method

Calculations for the overall plant, system and core behaviour are performed using the CATHARE 2 V2.5 computer code.

3.4.2.1.4. Acceptance criteria

The decoupled acceptance criterion for this study is that the core must remain covered.

The results presented in the ATWS cases for the loss of feedwater in sections 3.3.2.5 and 3.3.2.4 demonstrate that the number of fuel rods experiencing DNB is less than 10%. The additional loss of the reactor coolant pumps is beneficial for the DNBR calculation because of the moderator effect.

3.4.2.1.5. Results

Without the non-safety systems, the sequence of events is slightly different to the typical sequence of events. The variations of pressuriser pressure and level are not compensated for. Therefore, the pressuriser pressure increases gradually until the PSV opening setpoint is reached.

The sequence of events is provided in Sub-chapter 16.5 - Table 42.

The most representative parameters are presented in the following figures:

- Sub-chapter 16.5 – Figure 56
- Sub-chapter 16.5 – Figure 57
- Sub-chapter 16.5 – Figure 58

The loss of the normal feedwater leads to a decrease in heat removal and a primary temperature and pressure rise until the PSV opening setpoint is reached.

The first PSV opening lowers the pressure back to its normal value. The loss of cooling water through the PSV is sufficiently small for the core to remain covered until the reactor trip signal is generated on low SG level.

The signal causes the four ASG [EFWS] trains to be actuated and contribute to recovery of the SG level. The residual power is removed via the secondary side and the primary conditions are stabilised.

3.4.2.1.6. Conclusions

The controlled state is reached with the core remaining covered and without a significant peak cladding temperature. Therefore, the PCC-3/PCC-4 safety criteria are met.

Consequently, in such a case, the lower level safety function 'Core heat removal by RCP [RCS] natural circulation in shutdown mode' provides an efficient diverse mean to mitigate the event assuming the loss of the lower level safety function 'Core heat removal by RCP [RCS] forced flow in power mode'.

3.4.3. H3 – Transfer heat from the reactor coolant to the ultimate heat sink

The loss of main feedwater challenges the plant level safety function 'H3 – Transfer heat from the reactor coolant to the ultimate heat sink'. The lower level safety function 'heat removal via SG – emergency shutdown mode' is assumed to be unavailable, which leads to the total loss of feedwater event. The heat transfer from the RCP [RCS] to the steam generators is gradually reduced and the PLSF H3 is performed by the lower level safety function 'Heat removal by Low Head Emergency Core Cooling System (ECCS)' which mitigates the accident.

3.4.3.1. Loss of normal feedwater flow (loss of all ARE [MFWS] pumps and of the start-up and shutdown pump) without ASG [EFWS] (TLOFW)

3.4.3.1.1. Typical sequence of events

This transient is composed of two specific phases:

- Automatic phase: in the first phase of the event, operator actions are not considered. Only systems and I&C functions that are automatically actuated and safety classified are allowed.

- The manual phase: the second phase of the event starts 30 minutes after the reactor trip. After this grace period, operator actions are claimed.

With the total loss of SG feedwater, the decrease of the secondary side water inventory leads to a deterioration of the heat removal function. During this phase, the partial trip and the reactor trip are actuated to slow the decrease of the SG inventory. The RCV [CVCS] attempts to compensate for pressuriser level variations. Pressuriser heater/spray attempts to compensate for the RCP [RCS] pressure.

Feed and bleed is considered as an operation action to protect against total loss of feedwater due to common mode failure of the ASG [EFWS].

Note that diversity of electrical supply is provided between the ASG [EFWS] pumps in the EPR design since the pumps in safety divisions 1 and 4 operate at a different voltage to the pumps in safety divisions 2 and 3, which are also segregated within different parts of the safeguard building. In addition, the electrical supplies to the pumps in divisions 1 and 4 are backed-up by the SBO diesel generators.

The following automatic actions occur during the first phase:

- Partial reactor trip to 50% power (F2 classified), followed by reactor/turbine trip due to a mismatch between reactor power and ARE [MFWS] flow. This occurs about 15 seconds after the accident occurs.
- RCV [CVCS] charge/letdown compensates for the pressuriser level variations as long as the heat removal function is provided by the secondary side.
- Pressuriser heater/spray compensates for the RCP [RCS] pressure variations as long as the heat removal function is provided by the secondary side.
- MS bypass opening for heat removal after turbine trip: the heat removal function is maintained until the complete depletion of the secondary side water inventory.
- SG blowdown isolation when SG level is below MIN2.

During the second phase of the event, the pressuriser heaters are turned off, as soon as operator actions are possible, 30 minutes after RT. The steam generators continue to boil down, reducing their heat transfer capability. When the SG water level falls below the low level setpoint of 14% WR, the RCP [RCS] pumps are tripped to reduce the heat input to the RCP [RCS]. After about 1 hour, the SGs completely dry out and the heat up of the primary coolant begins. The increase in RCP [RCS] temperature causes the RCP [RCS] pressure to increase to the first PSV setpoint. The PSV opens/closes to maintain RCP [RCS] pressure while primary coolant inventory is continuously lost, and the RPV level decreases. When the core outlet temperature reaches 330°C coincident with a very low SG level, the operator initiates feed and bleed. The operator actuates the Primary Depressurisation System (PDS) and starts the RIS [SIS]. The fast depressurisation causes core voiding and clad heat-up. It also permits SI delivery, which stops the clad heat-up. Once the primary coolant inventory starts to increase and when the heat removal function is restored by the feed and bleed, the controlled state is reached.

In the second phase of the event, the assumed operator actions are:

- Pressuriser heaters turned off.
- RCP [RCS] pumps shutdown (on the criteria of no ASG [EFWS] injection and low level setpoint in SG (14% WR).
- Opening of the PDS when core outlet temperature reaches 330°C coincident with a very low SG level.
- Start-up of both RCV [CVCS] charging pumps immediately before opening the PDS.
- The manual start of the RIS [SIS].
- Secondary side cool down actuation.

It is noted that the actuation of feed and bleed by opening the PDS and start of the RIS [SIS] represents the RRC-A feature for this scenario.

The blowdown through the PDS increases the pressure and the temperature inside the containment and IRWST once the pressuriser relief tank bursting discs have failed.

This is mitigated by the EVU [CHRS] (if needed) and the IRWST is cooled via the LHSI/RHR.

3.4.3.1.2. Assumptions for the analysis

The study is performed with conservative assumptions.

3.4.3.1.2.1. Initial conditions

The conservative initial conditions considered are presented in Sub-chapter 16.5 - Table 43.

The initial core power is 102% FP, namely 4590 MWth.

Thermal-hydraulic flow conditions are considered within the primary circuit.

3.4.3.1.2.2. Decay heat

A conservative decay heat curve is used {CCI Removed} [b].

3.4.3.1.2.3. Assumptions related to systems

The non-safety systems are not considered for the safety analysis.

The F1A systems taken into account are: RIS [SIS], VDA [MSRT], and PSV (see Sub-chapter 16.5 - Table 44). Conservative I&C setpoints are used assuming uncertainties for degraded conditions.

The F2 systems taken into account are:

- Partial reactor trip,
- Primary Depressurisation System (PDS).

3.4.3.1.2.4. Single failure and preventive maintenance

Neither single failure nor preventive maintenance are taken into account.

3.4.3.1.2.5. Feed and bleed assumptions

The operator actuates feed and bleed heat removal when the high core outlet temperature (Tcot > 330°C) is reached.

The PDS is considered for the feed and bleed actuation.

3.4.3.1.2.6. Specific assumptions related to this case:

A partial reactor trip to 50% FP, followed by reactor/turbine trip, due to the mismatch between reactor power and ARE [MFWS] flow is considered.

The operator is assumed to start actions 30 minutes after the reactor trip.

The pressuriser heaters are shut off immediately at the end of the grace period.

The RCP [RCS] pumps are shut down by the operator on indication of a low low SG level (< 14% WR).

The analysis does not take into account the feedwater reserves in the feedwater system, including the feedwater tank, which would provide for an additional time delay before the opening of the pressuriser safety valves is required.

3.4.3.1.3. Method

Calculations for the overall plant, system and core behaviour are performed using the CATHARE 2 V2.5 computer code.

3.4.3.1.4. Results

Without the non-safety systems, the sequence of events is slightly different than the typical sequence of events. The variations of pressuriser pressure and level are not compensated for. Therefore, the pressuriser pressure increases gradually until the PSV opening setpoint is reached.

The most important events are listed in Sub-chapter 16.5 - Table 45.

The most representative parameters are presented in the following figures:

- Sub-chapter 16.5 – Figure 59
- Sub-chapter 16.5 – Figure 60
- Sub-chapter 16.5 – Figure 61
- Sub-chapter 16.5 – Figure 62

3.4.3.1.4.1. Sequence of events

The total loss of feedwater occurs at time 0 seconds. After 5 seconds, a partial trip to 50% FP is initiated due to the mismatch between the reactor power and the feedwater flow rate. A total reactor trip follows 10 seconds later. The turbine is tripped and the secondary pressure reaches the VDA [MSRT] setpoint (95.5 bar + 1.5 bar uncertainty). The secondary inventory starts to decrease.

At 1250 seconds, the SG level goes below the MIN2 level (40% WR to 35% WR with uncertainties) and the SG blowdown is isolated.

The heat removal safety function can be maintained for 50 minutes without SG feedwater supply before feed and bleed actions are required.

The first operator action occurs at 30 minutes. The pressuriser heaters are turned off.

At 2150 seconds, the SG level falls below 14% WR. The RCP [RCS] pumps are shut down by the operator. Consequently, the primary coolant starts to heat up. This leads to the first pressuriser safety valves opening (175 bar) at approximately 50 minutes.

The SGs are almost empty of water after approximately 1 hour. The heat removal capability of the SG is significantly reduced. Subsequently, decay heat is removed by the water or steam flow through the PSV. The primary coolant depletion and the core outlet temperature increase.

At 3350 seconds, the core outlet temperature reaches 330°C. The operator starts the feed and bleed safety feature by opening the PDS and actuating the RIS [SIS]. A partial cool down of the secondary side is also performed to reduce the SG stored energy.

The primary pressure rapidly decreases to 20 bar. This fast depressurisation allows the MHSI, and then the accumulators to inject coolant in the core to compensate for the water lost via the PDS.

As soon as the saturation temperature is reached within the core, vapour is generated and therefore, the fuel rods are no longer cooled. Consequently, the cladding temperature starts to increase. The RIS [SIS] injection at 3900 seconds restores the fuel rod cooling and stops the cladding heat up. The maximum cladding temperature reached is 379°C.

3.4.3.1.4.2. Final state

The “feed and bleed” allows the primary coolant temperature to decrease and restores the decay heat removal.

As the maximum temperature of the average fuel rod clad is only 379°C, the “Cladding temperature less than 1,200°C” criterion, for the hot rod is met with a significant margin.

After 6500 seconds, the average temperature remains below 100°C and the final pressure is around 5 bar. The primary coolant mass increases and reaches an equilibrium value of around 250 tons. Therefore, the RHR connecting conditions are met and the feed and bleed is no longer required to provide the decay heat removal function.

3.4.3.1.5. Conclusions

As the final state has been reached, the following conclusions can be drawn:

- Sub-criticality is initially reached shortly after reactor trip from "mismatch reactor power/ ARE [MFWS]-flow”. In the long term, sub-criticality is maintained by boration using the RBS [EBS] which can be started after 30 minutes. During the further 2 hours available, prior to feed and bleed initiation, this boration is sufficient to avoid a return to criticality. During this period, RBS [EBS] can inject approximately 35 tons of highly borated water before bleed is initiated. Following the actuation of feed and bleed the depressurisation, MHSI, accumulator injection, and LHSI provide significant additional boration to maintain sub-criticality.

- The RCP [RCS] pressure does not exceed the opening setpoint of the pressuriser safety valves and thus remains significantly below the acceptance limit of 130% of the design pressure.

- The core heat up maintains a significant margin to the acceptance criteria of a peak clad temperature (hot rod) of 1200°C as the average fuel rod clad is peak temperature is only 379°C. The feed and bleed action when core outlet temperature rises above 330°C guarantees sufficient heat removal, such that the final state with the balance between bleed (PDS) and feed (RIS [SIS]) and heat removal with LHSI/RRI [CCWS] via IRWST is reached.

- The mass and energy release into the containment from PSV and PDS operations is significantly lower than the design basis heat loads for the containment. Hence, any containment pressure build-up resulting from the PSV and PDS opening will be limited to values below the design pressure of 5.5 bar.

- The activity release during the accident is controlled because none of the barriers (fuel, RCP [RCS] and containment) is breached.

The calculation results show that, for the transient "Total Loss of Feedwater", the PCC-3/PCC-4 acceptance criteria are met and the required final state can be reached using the RRC-A feature “Pressuriser bleed”. This can be achieved assuming conservative initial conditions and only considering the operation of classified safety systems.

Consequently, in such a case, the lower level safety function ‘Heat removal by Low Head Emergency Core Cooling System (ECCS)’ provides an efficient diverse mean to mitigate the event assuming the loss of the lower level safety function ‘heat removal via SG – emergency shutdown mode’.

3.4.4. H4 – Maintain heat removal from fuel stored outside the reactor coolant system but within the site

3.4.4.1. Isolatable piping failure on a system connected to the spent fuel pool - Draining via the RCV [CVCS] letdown line, with failure of the manual isolation (state E)

In case of draining of the spent fuel pool via the RCV [CVCS] draining line in state E (incorrect alignment), the plant level safety function ‘H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site’ is challenged if the safety functional group ‘Manual isolation of the RCV [CVCS] unloading line’ cannot be achieved. The same lower level safety function is achieved by three diverse safety functional groups: ‘Automatic isolation of RIS/RRA [SIS/RHR] suction line’, ‘water make-up to the fuel pool by Classified Fire Fighting Water Supply System’ and ‘manual start-up of a PTR [FPCS] main train’.

3.4.4.1.1. Identification of causes

Following an incorrect alignment, the spent fuel pool is drained through the RCV [CVCS] discharge line. Failure of the manual isolation of the RCV [CVCS] drain line causes the draining to continue.

3.4.4.1.2. Typical sequence of events

If the RCV [CVCS] drain line is not isolated, the water level will continue to decrease.

When the water level in the spent fuel pool falls to {CCI} [a], the fuel pool cooling system operating pumps are automatically switched-off and the spent fuel is no longer cooled.

When the water level reaches {CCI} [a] in the reactor building transfer compartment, the RIS/RRA [SIS/RHR] suction line is automatically isolated by the closure of two redundant motorised valves. Only one of these valves isolates the RCV [CVCS] drain line but this is sufficient as the single failure criterion has already been applied.

Therefore, the draining is stopped with a water level in the spent fuel pool of {CCI} [a].

Water make-up is then performed using the Classified Fire Fighting Water Supply System (JAC/JPI [NIFPS]) to raise the fuel pool water level {CCI} [a], sufficient to start a PTR [FPCS] main train. The safe shutdown state is therefore reached.

3.4.4.1.3. Assumptions for the analysis

The study is performed with the assumptions used in PCC-4 study “Non-isolatable small break (< 50 mm) or isolatable break (< 250 mm) in RHR mode, spent fuel pool drainage aspects (state E)” presented in PCSR Sub-chapter 14.5, section 15.

3.4.4.1.4. Method

Decoupling criteria

For PCC events involving fuel pool draining, the decoupling criterion for the PTR [FPCS] design is to avoid pool boiling throughout the transient. Therefore, a maximum fuel pool water temperature of 97°C is assumed for this study. In the long term, following restoration of a PTR [FPCS] train, the water temperature in the fuel pool must not exceed 80°C.

Initial situation

Under normal operating conditions in this plant state, two PTR [FPCS] main trains, with one pump operating per train, are used to cool the fuel pool. The maximum heat load to be removed from the spent fuel pool occurs when the last fuel element has just been unloaded from the reactor vessel and placed inside the fuel pool. This conservatively calculated to be a decay heat of 20.23 MW and occurs at about 111 hours after shutdown.

The initial fuel pool water temperature is assumed to be 50°C. This value covers all potential operating states.

Grace period

For PCC events involving fuel pool draining, the grace period is calculated assuming a reduced fuel pool water volume of 1335 m^3. {CCI Removed} [a]

In this case, the fuel pool water temperature will reach 97°C in 3.4 hours after the loss of the cooling function assuming a decay heat of 20.23 MW.

It is conservatively assumed that only the water volume of the spent fuel pool is considered in the grace period calculations. In addition, the positive benefit from the make-up water temperature is not considered in the studies.

3.4.4.1.5. Results

The resulting fuel pool draining leads to an automatic shutdown of the PTR [FPCS] pumps {CCI Removed} [a] and therefore to a loss of the fuel pool cooling function.

The RIS/RRA [SIS/RHR] suction line is automatically isolated when the water level in the reactor building transfer compartment falls to {CCI} [a]. This stops the draining through the RCV [CVCS] drain line. With a RCV [CVCS] letdown line flow rate of 72 m^3/h, a drain time {CCI Removed} [a] of 0.4 hours is calculated, corresponding to a water loss of 27.

Therefore, the controlled state with the leakage stopped is reached automatically.

Water make-up is then undertaken using the Classified Fire Fighting Water Supply System (JAC/JPI [NIFPS]) at a minimum flow rate of approximately 150 m^3/h. This raises the water level {CCI} [a] in the fuel pool, sufficient to start a PTR [FPCS] main train.

The water make-up volume necessary to increase the level {CCI Removed} [a] is approximately 42 m^3 in state E.

If a delay to line up the JAC/JPI [NIFPS] system of 1 hour from the automatic shutdown of the pumps is assumed, the fuel pool cooling function is restored 1.7 hours after it was lost. The fuel pool water temperature does not exceed 74°C during this period. The calculation conservatively assumes a heat load of 20.23 MW. In addition, the calculation only considers the volume of the spent fuel pool at a water level of {CCI} [a]. The safe shutdown state is therefore reached and the fuel pool water temperature is stabilised at 64°C in the long term with one main PTR [FPCS] cooling train in operation. This value is calculated for a RRI [CCWS] design temperature of 38°C.

3.4.4.1.6. Conclusions

This study demonstrates that the diverse line is effective to fulfil the plant level safety function 'H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site' following draining of the spent fuel pool via the RCV [CVCS] letdown line in state E (incorrect alignment).

The safety criterion used for PCC events involving fuel pool drainage (no boiling) is therefore met.

3.5. CONTAINMENT SAFETY FUNCTION

3.5.1. C1 - Maintain integrity of the fuel cladding

The C1 plant level safety function is challenged by events that may lead to core boiling and damage of the fuel cladding. In particular, the following frequent PIEs are bounding for this PLSF:

- Spurious pressuriser spraying,
- Uncontrolled RCCA bank withdrawal at power,
- RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant.

If the reactor trip is not performed during these events, the fuel cladding integrity may be challenged. To clearly demonstrate that the C1 function does not result in violation of the safety criteria following failure of the reactor trip, the lower level safety function 'fast negative reactivity insertion' is assumed to be lost, either due to a mechanical blockage of the rods or due to a loss of the Protection System (RPR [PS]). Therefore, the events mentioned above are analysed as ATWS events.

3.5.1.1. ATWS by mechanical blockage of the rods – Spurious pressuriser spraying

3.5.1.1.1. Typical sequence of events

The sequence of events starts with the spurious actuation of the normal pressuriser spray.

The automatic pressure control system is not claimed during this transient, consequently the reactor coolant system pressure decreases due to the initiating event.

Reactor trip (RT) signal is generated from either a “Low pressuriser pressure (pressuriser pressure < MIN1)” signal, on low DNBR, or on a “low hot leg pressure” signal.

Due to a mechanical blockage of the RCCA, the reactor is not shut down, leading to an ATWS signal 20 seconds after the event.

Following the turbine trip on RT, the secondary side heat removal is maintained by the main steam relief trains, VDA [MSRT]) and by the MSSVs.

The Pressuriser Safety Valves (PSV) open to limit the primary side overpressure.

Without any additional actions, this state would be stable provided the steam generators have sufficient water inventory to remove the primary power (core power plus RCP [RCS] pump heat). If the inventory were significantly reduced the reduction in the heat transfer rate would result in a sharp increase in the primary temperature and pressure.

To avoid this potentially high pressure peak, a dedicated ATWS signal is implemented. The ATWS signal is triggered in the RPR [PS], from detection of "RT signal and rods out (or flux high) after a time delay". This ATWS signal, and the associated actions, is a RRC-A feature which is specifically implemented to protect against sequences with "ATWS by rods failure", and is F2 classified. It trips all the RCP [RCS] pumps following a “very low SG-water level” signal (SG level WR < MIN2) before the full SG depletion occurs. By this action, the reactor power is reduced by the moderator effect more smoothly with decreasing coolant flow rate, which leads to a reduced pressure increase on the primary side.

In response to the depletion of the SG mass, the ASG [EFWS] is actuated on a "SG level (WR) < MIN2" (F1A) signal.

The ATWS signal also automatically initiates RBS [EBS] injection of 7000 ppm enriched boron (corresponding to 11200 ppm natural boron), thus automatically maintaining core sub-criticality in the long term.

3.5.1.1.2. Acceptance criteria

For this demonstration, the following decoupled acceptance criteria are considered:

- the number of fuel rods experiencing DNB remains below 10%,
- the RCP [RCS] integrity is not impaired (as an acceptance criterion, the pressure at the worst point of the RCP [RCS] does not exceed 130% of the design pressure, i.e. 228.5 bar abs (Sub-chapter 3.4).

The analysis presented in this document demonstrates that the DNBR criterion is met. The maximum primary pressure reached will be given for information.

3.5.1.1.3. Assumptions for the analysis

The study is performed with the same assumptions used for the PCC-4 accident analysis.

3.5.1.1.3.1. Initial conditions

Conservative initial conditions are considered (see Sub-chapter 16.5 - Table 46).

The initial core power is 102% FP, i.e. 4590 MW.

Thermal-hydraulic flow conditions are considered within the primary circuit.

3.5.1.1.3.2. Neutronic data

The neutronic data are pessimised to minimise the moderator effect using BLX data.

3.5.1.1.3.3. Protection and mitigation actions

The following I&C functions provide protection and mitigation following pressuriser spurious spray followed by a reactor trip failure due to mechanical blockage of the rods:

- Reactor trip signal on "Pressuriser pressure < MIN1" (F1A),
- ATWS signal on reactor trip signal and high rods position (or high flux) after temporisation (F2),
- RBS [EBS] actuation on ATWS signal (F2),
- RCP [RCS] pumps trip on "SG level (wide range) < MIN2" if the ATWS signal has been generated (F2),
- VDA [MSRT] opening on "SG pressure > MAX1" (F1A),
- ASG [EFWS] actuation on "SG level (wide range) < MIN2" (F1A),
- SG blowdown isolation (APG [SGBS]).

In addition, three PSVs and two MSSVs per SG are available (F1A).

3.5.1.1.3.4. Assumptions related to systems

Therefore, the F1A systems assumed to operate are: ASG [EFWS], PSV, VDA [MSRT], MSSV. The F2 functions assumed to operate are: RBS [EBS] boration, RCP [RCS] pump trip.

Setpoints, delays and flow capacities are listed in Sub-chapter 16.5 – Tables 47 and 48.

3.5.1.1.3.5. Single failure and preventive maintenance

Neither single failure nor preventive maintenance are taken into account.

3.5.1.1.3.6. Specific assumptions related to this case

The ATWS signal is actuated 20 seconds after the reactor trip signal, starting the RBS [EBS] injecting 7000 ppm boric acid (corresponding to 11200 ppm natural boron). Once the SG level wide range falls below the MIN2 value, all the main coolant pumps are tripped.

Automatic boration via RCV [CVCS] pumps is not claimed in the analysis.

3.5.1.1.4. Method of analysis

The analysis is carried out using the coupled codes:

- MANTA V3.7 for overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), modelling F2/F1 systems operations,
- SMART V4.8 /FLICA-IIIF V3 for the neutronic and thermal-hydraulic behaviour of the core.

The DNBR calculation is performed with the FLICA III-F code.

3.5.1.1.5. Results

The most representative parameters are presented in the following figures:

- Sub-chapter 16.5 – Figure 63
- Sub-chapter 16.5 – Figure 64
- Sub-chapter 16.5 – Figure 65
- Sub-chapter 16.5 – Figure 66
- Sub-chapter 16.5 – Figure 67
- Sub-chapter 16.5 – Figure 68

The peak pressure downstream of the RCP [RCS] pumps is reached just after the initial opening of the first PSV.

The minimum DNBR occurs just before the turbine trip.

The VDA [MSRT] are actuated as a result of the turbine trip.

The MSSVs are briefly actuated during the transient.

The reactor trip signal is generated 111 seconds after the beginning of the transient.

The ATWS signal, generated 20 seconds after reactor trip, actuates the RBS [EBS] boration at 146.0 seconds.

The "SG level WR < MIN2" signal is generated at 225 seconds and all the RCP [RCS] pumps are tripped.

The ASG [EFWS] starts injecting at 240 seconds.

The pressuriser reaches 100% level at 338 seconds.

The decay heat is safely removed by ASG [EFWS] injection and VDA [MSRT]. The second and third PSV are not demanded during the transient.

The activity release during the accident is controlled as none of the barriers (fuel and RCP [RCS]) are breached.

The detailed sequence of events is given in Sub-chapter 16.5 - Table 49.

3.5.1.1.6. Conclusions

The peak pressure downstream of the RCP [RCS] pumps occurs for the 102% NP initial state. The value of the peak is significantly below the acceptance criterion of 130% design pressure.

The minimum DNBR during the transient is calculated to occur just before the turbine trip, with a value of 2.12 at 120 seconds. The initial DNBR is 2.15. As a result, no fuel rod enters DNB and the criterion is met.

The decay heat is safely removed via VDA [MSRT] and ASG [EFWS] in the SG.

The activity release during the accident is controlled as none of the barriers (fuel and RCP [RCS]) is breached.

The calculation results show that for the sequence "ATWS by rods failure – Spurious Spray", the acceptance criteria are met and the required final state is reached.

Consequently, in such a case, the lower level safety function ‘High concentrated and high boron injection’ provides an efficient diverse means to mitigate the event assuming the loss of the lower level safety function ‘fast negative reactivity insertion’.

3.5.1.2. ATWS by loss of RPR [PS] – Spurious pressuriser spraying

3.5.1.2.1. Typical sequence of events

The sequence of events starts with a spurious actuation of the pressuriser spray.

The automatic pressure control system is not claimed during this transient. Consequently the reactor coolant system pressure decreases due to the initiating event.

Prior to reaching the F1A RT signal setpoint, the transient is identical to the ATWS due to RCCA blockage scenario.

Following the generation of the F1A RT-signal, the transient differs from the ATWS due to RCCA mechanical blockage as follows:

- the absence of "ATWS signal" and associated actions (RBS [EBS] immediate actuation and all RCP [RCS] pumps trip on "SG-level < MIN2 Wide Range") as the F1A RT signal has not been triggered,

- the absence of actions associated with the failed F1A RT signal, i.e. turbine trip, ARE [MFW] high-load lines isolation, and VDA [MSRT] opening,

- and, if necessary to meet the safety criteria, the actuation of the dedicated RRC-A feature introduced to cope with the loss of F1A RT signal.

Once the transient reaches the “Hot leg Pressure (WR) < MIN2” diverse signal setpoint, the reactor trip is actuated and followed by the turbine trip and the isolation of the full load MFW line.

Without any additional actions, this state is stable whilst the steam generators have sufficient water inventory to remove the primary power (core power plus RCP [RCS] pump heat). This is performed by the low load MFW and the MSSV. If main feedwater is unavailable, the primary power can be removed by ASG [EFWS] (manual start up) and the MSSVs.

3.5.1.2.2. Acceptance criteria

For this demonstration, the following acceptance criteria are considered:

- the number of fuel rods experiencing DNB remains below 10%,

- the RCP [RCS] integrity is not impaired (as an acceptance criterion, the pressure at the worst point of the RCP [RCS] does not exceed 130% of the design pressure, i.e. 228.5 bar abs (PCSR Sub-chapter 3.4).

The analysis presented in this document provides the demonstration that DNBR criterion is met. The maximum primary pressure reached will be given for information.

3.5.1.2.3. Assumptions for the Analysis

3.5.1.2.3.1. Single failure and preventive maintenance

Single failure and preventive maintenance are not considered in the analysis. The event analyses covered herein assumes the loss of a safety function which is conservative.

3.5.1.2.3.2. Initial and boundary conditions

The initial operating conditions correspond to a thermal hydraulic flow rate at full power without plugging and fouling of the SG tubes, and with uncertainties included to pessimise RCP [RCS] pressure.

Initial values of the operating parameters are given in Sub-chapter 16.5 - Table 50.

3.5.1.2.3.3. Neutronic data

Conservative neutronic data assumed for the moderator effect at BLX.

3.5.1.2.3.4. Protection and mitigation actions

The following I&C functions provide protection and mitigation following spurious spray followed by the failure of the reactor trip signal from the reactor Protection System (RPR [PS]). The diverse reactor trip from the Protection System (RPR [PS]) considered for the transient is the following:

- Reactor trip on hot leg pressure (WR) < MIN2.

In addition, three PSVs and two MSSVs per SG are available (F1A).

3.5.1.2.3.5. Assumptions related to control channels

No control channels are considered.

3.5.1.2.3.6. Assumptions related to the systems

The F1A systems assumed to operate are: PSV, MSSV, and RCCA.

The F2 functions assumed to operate are: RIS [SIS] safety injection.

The setpoints, delays and flow capacities are listed in Sub-chapter 16.5 - Table 51 and Sub-chapter 16.5 - Table 52.

3.5.1.2.4. Method of analysis

The analysis is carried out using the internal coupling of:

- The MANTA V3.7 code for the overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), modelling F2/F1 systems operations,
- The SMART V4.8 /FLICA-IIIF V3 codes for neutronic and thermal-hydraulic behaviour of the core.

The DNBR calculation is performed with the FLICA III-F code.

3.5.1.2.5. Results

The most representative parameters are presented in the following figures:

- Sub-chapter 16.5 – Figure 69
- Sub-chapter 16.5 – Figure 70
- Sub-chapter 16.5 – Figure 71
- Sub-chapter 16.5 – Figure 72

The peak pressure downstream of the RCP [RCS] pumps is reached at the beginning of the transient.

The minimum DNBR is reached during the pressure drop caused by the spurious spray, in the period before the reactor trip.

The reactor trip signal is reached 188.1 seconds following a “Low Hot Leg Pressure” signal.

The decay heat is safely removed by the low load MFW and the MSSV.

There is no activity release during the accident as none of the barriers (fuel and RCP [RCS]) are breached.

The detailed sequence of events is given in Sub-chapter 16.5 - Table 53.

3.5.1.2.6. Conclusions

The lowest DNBR is obtained before the reactor trip and remains higher than 1.0 ($DNBR_{min}$ = 2.14).

The decay heat is safely removed via the SGs using the low ARE [MFWS] and MSSVs.

There is no activity release during the accident as none of the barriers (fuel and RCP [RCS]) are breached.

The calculation results show that for the sequence "ATWS by failure of the Protection System (RPR [PS]) – spurious spray", the acceptance criteria are met and the required final state is reached.

For information, the peak pressure downstream of the RCP [RCS] pumps is below the acceptance criterion of 130% design pressure, with a significant margin (157.98 bar abs is 89.8% of DP).

Consequently, in such a case, the lower level safety function ‘fast negative reactivity insertion’ is performed using efficient diverse means to mitigate the event.

3.5.1.3. ATWS by mechanical blockage of the rods – Forced decrease of reactor coolant flow (four pumps)

This section presents the forced decrease of reactor coolant flow (four pumps) event combined with a failure of the R2 Plant Level Safety Function (PLSF) leading to the ATWS.

For an ATWS, the PLSF C1 “Maintain integrity of the fuel cladding” is challenged. A complete failure of the reactor trip system on demand from the reactor Protection System (RPR [PS]) can result from:

- either a failure of the automatic reactor shutdown F1A signals (i.e. none of the signals sent by the RPR [PS] de-energises the rod drive coils),

- or failure of the control and shutdown rods to insert into the core following de-energising of their drive coils. In this case, actuation of the rods due to control or limitation signals also fails.

This section deals with the second case, due to rods failure.

3.5.1.3.1. Typical sequence of events

A complete loss of forced reactor coolant flow can be caused by a simultaneous fault in the power supplies to all the RCP [RCS] pumps caused by a drop in frequency on the external grid.

The bounding case studied corresponds to a supply frequency drop at 4 Hz per second to 0 Hz where it remains for an undetermined period.

A fast supply frequency drop leads to a reversal of the motor torque, which reduces the reactor coolant pumps speed and coolant flow more rapidly than a voltage drop transient, which is limited by the inertia of the flywheel.

If the reactor is at power at the time of the incident, the core power effectively remains constant. As the primary coolant flow decreases, the margin to nucleate boiling is reduced. This could result in DNB with subsequent fuel damage, if the reactor is not promptly tripped.

A Reactor Trip (RT) signal is generated following a “low RCP [RCS] pump speed” signal but, due to the mechanical blockage of the rods, no automatic shutdown occurs. A decrease in the DNBR occurs.

3.5.1.3.2. Safety criteria

For this demonstration, the following PCC-3/PCC-4 decoupling acceptance criteria are considered:

- Number of rods experiencing DNB not higher than 10% of the total core,
- Peak clad temperature must remain below 1482°C,
- Melted fuel at the hot spot must not exceed 10% by volume.

3.5.1.3.3. Results and conclusions

This transient is covered by the Loss of Off-site Power (LOOP) transient combined with the ATWS by mechanical blockage (see section 3.3.2.6).

In this case, once the RT signal is generated, the RCP [RCS] pumps are tripped. The flow decrease is then identical to the one during a LOOP transient.

The only difference is during the early phase of the transient, i.e. before the RT signal is generated. The reactor coolant flow decrease during a forced decrease of reactor coolant flow event is faster than that during a LOOP event but the RT occurs more rapidly.

The impact on the final results is therefore negligible.

3.5.1.4. ATWS by loss of TXS – Forced decrease of reactor coolant flow (four pumps)

This section presents the forced decrease of reactor coolant flow (four pumps) event combined with a failure in R2 Plan Level Safety Function (PLSF) leading to the ATWS.

For an ATWS, the PLSF C1 “Maintain integrity of the fuel cladding” is challenged. A complete failure of the reactor trip system on demand from the reactor Protection System (RPR [PS]) can result from:

- either a failure of the automatic reactor shutdown F1A signals (i.e. none of the signals sent by the RPR [PS] de-energises the rod drive coils),
- or failure of the control and shutdown rods to insert into the core following the de-energising of their drive coils. In this case, actuation of the rods due to control or limitation signals also fails.

This section deals with the first case, due to complete TXS failure.

3.5.1.4.1. Typical sequence of events

A complete loss of forced reactor coolant flow can be caused by a simultaneous fault in the power supplies to all the RCP [RCS] pumps caused by a drop in frequency on the external grid.

The bounding case studied corresponds to a supply frequency drop at 4 Hz per second to 0 Hz where it remains for an undetermined period.

A fast supply frequency drop leads to a reversal of the motor torque, which reduces the reactor coolant pumps speed and coolant flow more rapidly than a voltage drop transient which is limited by the inertia of the flywheel.

If the reactor is at power at the time of the incident, the core power effectively remains constant. As the primary coolant flow decreases, the margin to nucleate boiling is reduced. This could result in DNB with subsequent fuel damage, if the reactor is not promptly tripped.

As no automatic shutdown occurs due to the TXS platform failure, a decrease in the DNBR occurs.

3.5.1.4.2. Safety criteria

For this demonstration, the following PCC-3/PCC-4 decoupling acceptance criteria are considered:

- Number of rods experiencing DNB not higher than 10% of the total core,
- Peak clad temperature must remain below 1482°C,
- Melted fuel at the hot spot must not exceed 10% by volume.

3.5.1.4.3. Results and conclusions

Considering the significant margin available in this transient when assessed against the DNBR criteria when studied as a PCC-3 event (only 0.2% of rods experience DNB), a dedicated reactor trip will be implemented on low reactor coolant pump speed in a non TXS platform qualified at the appropriate standard to ensure that criterion of less than 10% of rods experiencing DNB is met.

3.5.1.5. ATWS by mechanical blockage of the rods – Uncontrolled RCCA bank withdrawal at power

This section presents the uncontrolled RCCA bank withdrawal combined with a failure of the R2 Plan Level Safety Function (PLSF) leading to the ATWS.

Following an ATWS, the PLSF C1 “Maintain integrity of the fuel cladding” is challenged. A complete failure of the reactor trip system on demand from the reactor Protection System (RPR [PS]) can result from:

- either a failure of the automatic reactor shutdown F1A signals (i.e. none of the signals sent by the RPR [PS] de-energises the rod drive coils),
- or failure of the control and shutdown rods to insert into the core after de-energising of their drive coils. In this case, actuation of the rods due to control or limitation signals also fails.

This section deals with the second case, due to rods failure.

3.5.1.5.1. Typical sequence of events

The uncontrolled RCCA bank withdrawal causes a rapid increase in core power, and a primary circuit temperature and pressure increase. On the secondary side, the secondary pressure increase is potentially limited by the GCT [MSB] opening.

During this phase, a Reactor Trip (RT) is actuated by either the low DNBR protection channel or the high neutron flux rate of change protection channel, which are both F1A classified. Reactor shutdown does not occur, nor does any other power reduction because of mechanical blockage of the rods, thus further RCCA withdrawal is stopped. Therefore power, temperature and pressure remain constant at a lower level than the one reached in ATWS by TXS platform failure transients.

An ATWS signal is generated on a RT signal combined with a “high rod position (or high flux) after an appropriate delay” signal. The ATWS signal automatically initiates RBS [EBS] injection with 7000 ppm enriched boron (corresponding to 11200 ppm natural boron), thus automatically providing core sub-criticality in the long term. The RCV [CVCS] is also available for this boration function but is not claimed in the study.

Following the reactor trip signal, the turbine trip is successfully achieved and the ARE [MFWS] is switched from high to low flow.

The ARE [MFWS] supply reduction leads to a decrease in secondary side heat removal, and further reduces the capacity of the primary coolant to remove heat from the core.

Consequently, primary and secondary pressures and temperatures increase more rapidly. The core power is reduced due to the moderator temperature feedback effect which decreases the overall reactivity.

The primary side pressure is controlled by:

- the pressuriser normal spray,
- three PSVs.

The secondary side pressure is controlled by:

- the GCT [MSB],
- the VDA [MSRT] and two MSSVs valves per SG.

When the wide range SG level MIN2 setpoint is reached, the ASG [EFWS] is started, the primary coolant pumps are tripped (by the ATWS signal) and the subsequent core heat-up causes a reactivity decrease from the moderator feedback effect.

The heat is removed through SGs using the ASG [EFWS] and the VDA [MSRT].

3.5.1.5.2. Safety criteria

For this demonstration, the following PCC-3/PCC-4 decoupling acceptance criteria are considered:

- Number of rods experiencing DNB not higher than 10% of the total core,
- Peak clad temperature must remain below 1482°C,
- Melted fuel at the hot spot must not exceed 10% by volume.

3.5.1.5.3. Results and conclusions

The RT signal stopped the RCCAs bank withdrawal. Thus the transients are bounded by the transients presented in section 3.5.1.6 below considering an ATWS by TXS platform failure.

3.5.1.6. ATWS by loss of TXS – Uncontrolled RCCA bank withdrawal at power

This section presents the Uncontrolled RCCA banks withdrawal combined with the complete failure of the TXS platform leading to the ATWS.

Following an ATWS, the PLSF C1 ‘Maintain integrity of the fuel cladding’ is challenged since the lower safety function ‘Negative reactivity fast insertion’ cannot be fulfilled by the TXS platform. Thus a diverse function is provided in a non TXS platform.

A complete failure of the reactor trip system on demand from the reactor Protection System (RPR [PS]) can result from:

- either a failure of the automatic reactor shutdown F1A signals (i.e. none of the signals sent by the RPR [PS] de-energises the rod drive coils),
- or failure of the control and shutdown rods to insert into the core after de-energising of their drive coils. In this case, actuation of the rods due to control or limitation signals also fails.

This section deals with the first case, due to a failure of the TXS platform.

3.5.1.6.1. Typical sequence of events

The uncontrolled RCCA bank withdrawal causes a rapid increase in core power and a primary circuit temperature and pressure increase. On the secondary side, the pressure increase is potentially limited by the opening of the GCT [MSB].

As no automatic shutdown signals are generated by the TXS platform, the withdrawal of the RCCA banks continues until a "diverse reactor trip" is triggered by a non TXS platform. The reactivity insertion and the resulting nuclear power, coolant temperature and pressure increase can be higher than the values obtained in the current PCSR with a reactor trip following a signal from the low DNBR channel.

The following non TXS diverse RT signals would occur:

- High ex-core neutron flux (MAX2),
- High Axial Offset (AO) (MAX1),
- High hot leg pressure (MAX1).

A diverse automatic shutdown signal is thus generated which ends the transient by the insertion of both the control and shutdown rods.

The shutdown margin guarantees core sub-criticality after reactor trip.

3.5.1.6.2. Safety criteria

For this demonstration, the following PCC-3/PCC-4 decoupling criteria are considered:

- Number of rods experiencing DNB not higher than 10% of the total core,
- Peak clad temperature must remain below 1482°C,
- Melted fuel at the hot spot must not exceed 10% by volume.

3.5.1.6.3. Assumptions for the analysis

The limiting conditions for the assessment of the challenge to the safety criteria depend on the following assumptions:

- The initial conditions
- Variation of the core neutronic parameters: FQ, FΔH and AO
- Variation of the RCP [RCS] parameters: core power, pressure, temperature

3.5.1.6.3.1. Initial conditions

The initial conditions are consistent with the initial conditions used for the PCC analysis. The initial conditions conservatively correspond to:

- Minimum initial DNBR,

- Maximum initial AO,
- Maximum initial RCCA insertion.

The initial conditions assumed are detailed in Sub-chapter 16.5 - Table 54.

3.5.1.6.3.2. Variation of the core neutronic parameters

The neutronic data corresponding to the RCCA withdrawal events are calculated with the SCIENCE nuclear code package at BLX and End Of Life (EOL) assuming the PCSR 18 month fuel management.

Sub-chapter 16.5 – Figure 73 presents the F_Q value as a function of core AO. As expected, a RCCA withdrawal from an AO of 12% at 100% NP leads to an increase of both F_Q and AO.

Bounding neutronic data are defined to be used in the assessment of the challenge to the safety criteria.

3.5.1.6.3.3. Variation of the reactor coolant pump parameters

A fast RCCA withdrawal speed of 75 steps/min is conservatively assumed. MANTA/SMART 3D coupled calculations are performed for this representative RCCA withdrawal transient.

The variation in core power is illustrated on Sub-chapter 16.5 – Figure 74.

The power level reaches 120% NP, the temperature has increased by 5°C and a reactor trip is actuated on either a “high hot leg pressure”, a “high ex-core neutron flux” or a “high AO” signal.

A conservative bounding transient has been defined on the basis of this representative transient. This bounding transient is used in the assessment of the challenge to the safety criteria.

3.5.1.6.3.4. Single failure and preventive maintenance

Single failure and preventive maintenance are not considered in the analysis.

3.5.1.6.4. Results and conclusions

3.5.1.6.4.1. Number of rods experiencing DNB calculation

The FLICA III-F code is used to perform the minimum DNBR calculations.

The following assumptions have been used for the FLICA III-F calculations:

- Top skewed axial power distribution with the AO at the reactor trip setpoint,
- Bounding FΔH. The initial FΔH is chosen such that the initial DNBR is equal to the DNBR limiting value. A bounding FΔH value is conservatively retained for the FLICA III-F calculations.
- Bounding thermal hydraulic conditions defined in section 3.5.1.6.3.3

Sub-chapter 16.5 - Table 55 summarises the FLICA III-F assumptions and Sub-chapter 16.5 – Table 56 summarises the results.

The number of rods experiencing DNB meets the acceptance criterion of 10%.

3.5.1.6.4.2. Clad temperature and fuel melted calculations

The COMBAT code is used to calculate the fraction of melted fuel at the hot spot and the maximum fuel clad temperature.

The following assumptions have been made for the COMBAT calculations:

- Bounding initial and final FQ defined in section 3.5.1.6.3.2.
- Bounding thermal hydraulic conditions defined in section 3.5.1.6.3.3.
- DNB assumed to occur at 110% NP.

Sub-chapter 16.5 - Table 55 summarises the COMBAT assumptions and Sub-chapter 16.5 - Table 56 summarises the results.

The maximum fuel clad temperature is lower than the acceptance criterion of 1482°C.

The maximum percentage of melted fuel is lower than the acceptance criterion of 10%.

3.5.1.7. ATWS by mechanical blockage of the rods – RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant

A complete failure of the reactor trip system on demand from the reactor Protection System (RPR [PS]) can result from:

- either a failure of the automatic reactor shutdown F1A signals, i.e. none of the signals sent by the RPR [PS] de-energises the rod drive coils,
- or failure of the control and shutdown rods to insert into the core after de-energising of their drive coils. In this case, actuation of the rods due to control or limitation signals also fails.

The current section deals with the second case, due to rod failure.

3.5.1.7.1. Typical sequence of events

The decrease of the boron concentration causes an increase of the core power, and consequently a primary circuit temperature and pressure increase. On the secondary side, the secondary pressure increase is potentially limited by the opening of the GCT [MSB].

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 conservatively performed without claiming these F2 classified channels.

A reactor trip signal occurs following a high pressuriser pressure signal. Reactor trip occurs, but control rod insertion does not occur, due to the assumed mechanical failure of the rods.

An ATWS signal is generated on a combination of a reactor trip signal and a “high rod position (or high flux) after an appropriate delay” signal. Following the ATWS signal, the RCV [CVCS] is isolated downstream of the RCV [CVCS] volume control tank. This stops the dilution, and the RBS [EBS] starts automatically to provide boron injection. In addition, the ATWS signal causes the primary coolant pumps to trip once the low SG level setpoint is reached.

Following the reactor trip signal, the turbine trips and the ARE [MFWS] full-load flow rate is switched from high to low flow.

The reduction in the ARE [MFWS] supply leads to a decrease in the secondary side heat removal rate. This, combined with the primary flow rate reduction due to the primary pump coast-down, cause an increase in the rate of increase in the primary and secondary pressure and temperature.

The pressure control is provided by the pressuriser safety valves on the primary side and by the GCT [MSB], the VDA [MSRT] and two MSSVs per SG on the secondary side.

The increase in temperature causes a reduction in reactivity and hence a power decrease via the moderator temperature feedback effect.

At the end of the transient, the dilution has been halted, reactivity is controlled by the RBS [EBS] and decay heat is removed via the steam generators using ASG [EFWS] injection.

3.5.1.7.2. Safety criteria

For this demonstration, the following decoupling criteria are considered:

- Number of rods experiencing DNB not higher than 10% of the total core,
- Peak clad temperature must remain below 1482°C,
- Melted fuel at the hot spot must not exceed 10% by volume.

3.5.1.7.3. Results and conclusion

The dilution is equivalent to an uncontrolled RCCA bank withdrawal at power, with a reactivity insertion rate of less than 2 pcm/second.

The consequences of the transient are therefore bounded by those calculated in the study of the uncontrolled RCCA bank withdrawal combined with ATWS by mechanical rod failure, section 3.5.1.5 above.

The moderator temperature feedback effect increases throughout the transient due to the decrease of the boron concentration. This does not occur for the uncontrolled RCCA bank withdrawal. When the turbine and the primary pumps are tripped, the higher moderator temperature coefficient causes a faster power decrease for the dilution, and hence a lower pressure peak and a lower temperature increase than in the case of an uncontrolled RCCA bank withdrawal.

3.5.1.8. ATWS by loss of TXS – RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant

A total failure of the automatic shutdown system of the reactor on demand from the reactor Protection System (RPR [PS]) can be caused by:

- either the failure of the automatic reactor shutdown F1A signals (i.e. none of the signals sent by the reactor Protection System (RPR [PS]) de-energise the rod drive coils),
- or failure of the control and shutdown rods after de-energising of the drive coils. In this case, the actuation of the rods due to control or limitation signals also fails.

This section deals with the first case due to a complete TXS platform failure.

3.5.1.8.1. Typical sequence of events

The decrease of the boron concentration causes an increase of core power, and consequently a primary circuit temperature and pressure increase. On the secondary side, the secondary pressure increase is potentially limited by the GCT [MSB] opening.

As neither alarm nor automatic shutdown signals are generated by the TXS platform, the dilution continues until a "backup reactor trip" is triggered by a non TXS platform. The reactivity insertion and the resulting nuclear power, coolant temperature and pressure increase can be higher than the values obtained in the current PCSR with a reactor trip on signals from the low DNBR channel or high pressuriser pressure.

The following non TXS back up RT could occur:

- low SG level wide range (MIN3)
- High ex-core neutron flux (MAX2)
- High hot leg pressure (MAX1)

A diverse automatic shutdown signal is thus generated which terminates the transient by the insertion of both the control and shutdown rods.

Following the RT signal, the RCV [CVCS] is isolated downstream of the RCV [CVCS] volume control tank and the RBS [EBS] is started to provide boron injection.

The guaranteed shutdown margin ensures core sub-criticality following the reactor trip.

3.5.1.8.2. Safety criteria

For this demonstration, the following decoupling criteria are considered:

- Number of rods experiencing DNB not higher than 10% of the total core
- Peak clad temperature must remain below 1482°C
- Melted fuel at the hot spot must not exceed 10% by volume

3.5.1.8.3. Results and conclusion

The dilution is equivalent to an uncontrolled slow withdrawal of RCCA banks at power, with a reactivity insertion rate of less than 2 pcm/second.

The consequences of the transient are therefore bounded by those calculated in the study of the uncontrolled RCCA bank withdrawal combined with ATWS by TXS failure.

The moderator temperature feedback effect increases throughout the transient due to the decrease of the boron concentration. This is not the case for the uncontrolled RCCA bank withdrawal. In particular when the turbine and the primary pumps are tripped, the higher moderator temperature coefficient causes a faster power decrease for the dilution, and hence a lower peak pressure and a lower temperature increase than in the case of an RCCA bank withdrawal.

3.5.2. C2 - Maintain integrity of the Reactor Coolant Pressure Boundary

The C2 plant level safety function is challenged in the case of overpressure event.

The lower level safety function ‘RCP [RCS] overpressure protection’ is provided by the pressuriser safety valves (PSVs). Relief diverse to the PSVs is not provided. However, other means of limiting the RCP [RCS] pressure can be claimed. These include the normal spray and, indirectly, the secondary side overpressure protection. The worst PIE for this PLSF is the inadvertent closure of four VIVs [MSIV]s. For this assessment, the failure of the three PSVs is assumed. This section demonstrates that there are no shortfalls regarding the diversity of the pressuriser safety valves as the diversity is provided by the secondary side.

A further ALARP demonstration for the design of the Pressuriser Safety Valves regarding the passive single failure has been performed [Ref-1]. The current design has diverse overpressure protection means and improves on previous plants, as presented below.

3.5.2.1. Inadvertent closure of four VIVs [MSIV]s without PSVs

3.5.2.1.1. Typical sequence of events

The sequence considered is initiated by the inadvertent closure of all the VIVs [MSIV]s at full power:

The closure of all VIVs [MSIV]s results in the termination of all main steam flows, which leads to an increase in the secondary pressure and temperature which in turn causes a primary pressure and temperature increase.

Reactor trip is initiated following a high pressuriser pressure signal in the PS [RPR].

The Main Feedwater Full Load line is automatically isolated following reactor trip. The pressuriser safety valves (PSV) fail to open to limit the primary side overpressure, this additional failure is considered for the functional diversity study. VDA [MSRT] actuation on the four main steam lines is initiated on high SG pressure. The peak primary pressure is reached.

Conservative assumptions are made for systems so that their efficiency regarding overpressure mitigation is reduced. The loss of main feedwater coincident with the initiating event is an additional decoupling criterion, which decreases the heat removal from the reactor coolant system via the secondary side, pessimising the reactor coolant system pressure transient.

Only Reactor Coolant System pressure control system via normal spray, when claimed and VDA [MSRT] actuation control the primary overpressure. Later on, the controlled state is reached, which is in this case the hot shutdown. The residual heat is removed via the main steam relief trains of all steam generators and the feedwater supply is provided by the emergency feedwater system.

3.5.2.1.2. Acceptance criteria

For this demonstration, the following acceptance criterion is considered: the Reactor Coolant System integrity is not challenged. As a decoupling criteria, the pressure at the worst point of the Reactor Coolant System should not exceed 130% of the Reactor Coolant System design pressure, i.e. 228.5 bar abs.

3.5.2.1.3. Assumptions for the analysis

3.5.2.1.3.1. Protection and mitigation actions

The following I&C functions provide protection and mitigation following the inadvertent closure of all VIVs [MSIV]s without PSVs:

- Reactor trip
- Pressuriser pressure control via normal spray if operational
- VDA [MSRT] actuation on "SG pressure > MAX1" (F1A)

In addition, two Main Steam Safety Valves per SG are available (F1A).

3.5.2.1.3.2. Methods of analysis

The analysis is carried out using the version V3.7 of MANTA code for the overall thermal-hydraulic behaviour of the main primary and secondary systems (Reactor Coolant System and SG).

The analysis is performed to show that the pressure in the reactor coolant system remains below 130% of the design pressure.

The analysis is carried out using a conservative approach. F1 classified systems operate with conservative characteristics. The neutronic data, the initial conditions and the control setpoints all assume conservative values. An additional failure of all PSVs is assumed. Therefore, the thermal hydraulic study of the transient includes the following steps:

- choice of the initial plant parameters,
- neutronic data consideration,
- transient computations,
- assessment of the challenge to the criterion.

3.5.2.1.3.3. Single failure and preventive maintenance

No preventive maintenance and single failure are considered for this analysis.

3.5.2.1.3.4. Initial and boundary conditions

The initial operating conditions correspond to a thermal hydraulic flow rate at full power and with uncertainties taken into account to maximise the RCP [RCS] pressure.

Initial values of the operating parameters are given in Sub-chapter 16.5 - Table 57.

Note that the impact of an increase in the initial pressuriser level is assessed in section 3.5.2.1.4.

3.5.2.1.3.5. Neutronic data

In order to maximise Reactor Coolant System pressure, the core power is assumed to be as high as possible prior to the reactor trip and assumptions on nuclear data are made to increase the neutron flux. In particular, minimum absolute values for the moderator density coefficient and the Doppler temperature coefficient are assumed as are maximum values for the Doppler power coefficients.

3.5.2.1.3.6. Assumptions related to control channels

SG level control is not relevant as the ARE [MFWS] is unavailable.

RCP [RCS] temperature control has no effect due to the speed of the primary pressure peak.

Pressuriser pressure control using the normal spray and heaters is claimed.

SG pressure control using the GCT [MSB] is not claimed.

Pressuriser level control is not modelled.

3.5.2.1.3.7. Assumptions related to F1 systems

The following F1 systems are available:

- Reactor trip on “high RCP [RCS] pressure” or “high SG pressure” signal. Reactor trip is triggered following a “high RCP [RCS] pressure (> MAX2)” or “high SG pressure (> MAX1)” signal. The related uncertainty included to delay the generation of the RT. The reactor trip characteristics are described in Sub-chapter 16.5 – Table 58.

- Main steam relief trains (VDA [MSRT]): The VDAs [MSRT]s automatically open once the secondary pressure reaches MAX1 threshold. Conservative values for their setpoint, delays and flow capacities are assumed as listed in Sub-chapter 16.5 - Table 59.

3.5.2.1.4. Results

The detailed sequence of events is given in Sub-chapter 16.5 - Table 60. The most representative parameters are presented in:

- Sub-chapter 16.5 – Figure 75
- Sub-chapter 16.5 – Figure 76

Following the closure of all the main steam isolation valves, the reactor coolant system pressure increases rapidly, due to the loss of heat removal via the secondary side, until actuation of the normal pressuriser spray, if claimed. The pressuriser pressure increase causes the actuation of the reactor trip following a high pressuriser pressure signal. The pressuriser safety valves are not claimed and are assumed to be failed closed. The main steam relief trains open when the SG pressure reaches the opening setpoint, hence controlling the pressure on both sides.

The results of the sensitivity without pressuriser normal spray are presented in:

- Sub-chapter 16.5 – Figure 77
- Sub-chapter 16.5 – Figure 78

The sequence of events is similar (see Sub-chapter 16.5 - Table 61), with the difference being a steeper increase of primary pressure due to the absence of the spray actuation.

Note that a further increase in the initial pressuriser level would have very little impact on the result. The margins to the acceptance criteria are sufficiently large to absorb the consequences of a potential pressure increase due to increased uncertainty in initial pressuriser level.

3.5.2.1.5. Conclusions

The maximum pressure at the worst point of the reactor coolant system, the reactor coolant pumps outlets, is equal to 203.3 bar abs for this transient of "Inadvertent closure of all main steam isolation valves at full power, without PSV available and with pressuriser spray". This value is lower than 130% design pressure (228.5 bar abs).

In the case of without normal spray, the pressure reaches a higher value, 209.3 bar abs. This value is also lower than 130% design pressure.

3.5.3. C3 - Limit the release of radioactive material from the reactor containment

The containment isolation performs the plant level safety function C3. An ALARP justification of the isolation is provided separately from this document.

3.5.4. C4 - Limit the release of radioactive waste and airborne radioactive material

The main steam isolation valve closure contributes to the plant level safety function C4 by isolating the steam generators. An ALARP justification of the design is provided separately from this document.

3.6. OTHER SAFETY FUNCTIONS

3.6.1. O1 – Prevent the failure or limit the consequences of failure of a structure, system or component whose failure could cause the impairment of a safety function

The plant level safety function O1 is applied to the main steam lines following an overpressure transient. The lower level safety function 'Essential component protection' is normally performed by the main steam relief valves. The main steam safety valves provide the diverse protection. The two MSSVs have a capacity equivalent to one VDA [MSRT]. The most onerous frequent initiating event for this plant level safety function is the closure of four VIVs [MSIV]s. The additional failure is the loss of the VDAs [MSRT]s.

3.6.1.1. Inadvertent closure of four VIVs [MSIV]s without VDAs [MSRT]s

3.6.1.1.1. Typical sequence of events

The sequence considered is initiated by the inadvertent closure of all VIV [MSIV]s at full power:

- The closure of all VIVs [MSIV]s results in the termination of all main steam flows, leading to an increase in the secondary pressure and temperature as well as an increase of the primary pressure and temperature.
- Reactor trip is initiated following a high SG pressure signal.
- The Main Steam Relief Trains (VDA [MSRT]) fail to open to limit the secondary side overpressure. This additional failure is considered for this functional diversity study only.
- PSV actuation
- MSSV actuation on the four main steam lines
- Secondary pressure peak is reached.
- The main feedwater full load line is isolated on reactor trip.

Only the Reactor Coolant System pressure control system via normal spray is considered, when claimed, PSV and MSSV actuation manage the secondary overpressure. Later on, the controlled state is reached, which for this case is hot shutdown.

3.6.1.1.2. Acceptance criteria

For this demonstration, the following acceptance criterion is considered: The secondary side integrity is not challenged if the maximum steam pressure in the SG does not exceed 130% of the design pressure, i.e. 129.7 bar abs.

3.6.1.1.3. Assumptions for the analysis

3.6.1.1.3.1. Protection and mitigation actions

The following I&C functions provide protection and mitigation following the inadvertent closure of all VIVs [MSIV]s without VDAs [MSRT]s:

- Reactor trip.
- Isolation of the Main Feedwater System Full Load line.
- Pressuriser pressure control via normal spray if operational.

In addition, three Pressuriser Safety Valves and two MSSVs per SG are available (F1A).

3.6.1.1.3.2. Methods of analysis

The analysis is carried out using the version V3.7 of the MANTA code for overall thermal-hydraulic behaviour of the main primary and secondary systems (Reactor Coolant System and SG).

The object of the analysis is to show that the pressure in the secondary system remains below 130% of the design pressure.

The analysis is carried out following a conservative approach. F1 classified systems operate with conservative characteristics. Conservative values for neutronic data, initial conditions and control setpoints are assumed. An additional failure of all the VDAs [MSRT]s is considered. Therefore, the thermal hydraulic study of the transient includes the following steps:

- choice of the initial plant parameters,
- neutronic data consideration,
- transient computations,
- assessment of the challenge to the criterion.

3.6.1.1.3.3. Single failure and preventive maintenance

No preventive maintenance and single failure are considered for this analysis.

3.6.1.1.3.4. Initial and boundary conditions

The initial operating conditions correspond to a thermal hydraulic flow rate at full power and with uncertainties taken into account to maximise the secondary side pressure.

Initial values of the operating parameters are given in Sub-chapter 16.5 - Table 62.

3.6.1.1.3.5. Neutronic data

In order to maximise the secondary side pressure, the core power is maximised until the reactor is tripped and assumptions on neutronic data are chosen to increase the neutron flux. Minimum values for the moderator density coefficient and the Doppler temperature coefficient are assumed, with maximum values for the Doppler power coefficients.

3.6.1.1.3.6. Assumptions related to control channels

SG level control is taken into account.

Reactor Coolant System temperature control is not considered due to the speed of the primary pressure increase.

Pressuriser pressure control via normal spray and heaters assumed.

SG pressure control via GCT [MSB] is not claimed.

Pressuriser level control is not taken into account.

3.6.1.1.3.7. Assumptions related to F1 systems

The following F1 systems are available:

- Reactor trip on high RCP [RCS] pressure or high SG pressure signal. Reactor trip is actuated following a “high RCP [RCS] pressure (> MAX2)” or “high SG pressure (> MAX1)” signal. The associated uncertainty is included to delay the RT. The reactor trip characteristics are described in Sub-chapter 16.5 - Table 63.

- PSVs. Their setpoint, delays and flow capacities are pessimised and listed in Sub-chapter 16.5 - Table 64.

- MSSVs. Their setpoint, delays and flow capacities are pessimised and listed in Sub-chapter 16.5 - Table 64.

3.6.1.1.4. Results

The detailed sequence of events is given in Sub-chapter 16.5 - Table 65. The most representative parameters are presented in:

- Sub-chapter 16.5 – Figure 79

- Sub-chapter 16.5 – Figure 80

Following the closure of all main steam isolation valves, the SG and Reactor Coolant System pressures increase rapidly due to the loss of heat removal via the secondary side. The, SG pressure reaches the reactor trip setpoint signal on high SG pressure and Reactor Coolant System pressure increase is stopped by the actuation of the normal pressuriser spray if claimed, and the PSVs. The VDA [MSRT] valves are assumed to be failed closed. The main steam safety valves open when the SG pressure reaches their actuation setpoint, limiting the pressure on the secondary side.

The results of a sensitivity without pressuriser normal spray are presented in:

- Sub-chapter 16.5 – Figure 81
- Sub-chapter 16.5 – Figure 82

The sequence of events is similar (see Sub-chapter 16.5 - Table 66), with the difference being a more rapid increase in primary pressure due to the absence of the spray actuation.

3.6.1.1.5. Conclusions

On the secondary side, the pressure remains lower than 130% design pressure. Even if the VDA [MSRT]s do not operate, the MSSVs can limit the SG pressure. The pressure in the secondary side, for both cases, shows a peak of 108.8 bar abs, which is below the maximum pressure allowed of 129.7 bar abs (130% design pressure).

3.6.2. Diversity to safe shutdown state

Diverse means exist to reach a non-hazardous stable state for all frequent PIEs in which the reactor is sub-critical, adequate heat removal is provided and radioactive releases are limited.

The reactor is in a non-hazardous stable state, in particular, in the case of:

- RHR connection, since, by design, one RRA [RHRS] train is sufficient to remove the residual heat at hot shutdown. This is demonstrated in PCSR Sub-chapter 6.3.
- Feed and bleed, as demonstrated in the cases of the SB LOCA with failure of the VDAs [MSRT]s (section 3.4.1.3) and Total Loss of Feedwater (section, 3.4.3)
- Heat removal via the secondary side: two ASG [EFWS] trains are sufficient to provide heat removal in cold shutdown for 24 hours before emptying the emergency feedwater tanks. Regarding heat removal, this case is bounded by the feedwater line break case (see PCSR Sub-chapter 14.5).

The first two final states ensure adequate heat removal, adequate boration (via RIS [SIS]/RHR boron injection) and limited radioactive releases. The third state presented must be reached in conjunction with adequate boration.

The three non-hazardous stable states listed above are the most probable and cover the following frequent PIEs in conjunction with a common cause failure of a lower level safety function:

- RCV [CVCS] malfunction causing increase in reactor coolant inventory
- RCV [CVCS] malfunction causing decrease in reactor coolant inventory (state A)
- Inadvertent opening of a pressuriser safety valve
- Feedwater malfunction – causing a reduction in feedwater temperature
- Feedwater malfunction - causing an increase in feedwater flow rate
- Excessive increase in steam flow

- Inadvertent opening of a SG relief train (state A)
- Turbine trip
- Loss of condenser vacuum
- Loss of normal feedwater flow
- Small feedwater system piping failure
- Inadvertent closure of one or all main steam isolation valves
- Short-term loss of off-site power
- Uncontrolled RCCA bank withdrawal at power
- Uncontrolled RCCA bank withdrawal from HZP
- RCCA misalignment up to rod drop
- Start-up of an inactive reactor coolant loop at an incorrect temperature
- RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant
- Uncontrolled single RCCA withdrawal

However, several specific conditions have not been detailed:

- Small break (not greater than DN 50) including a break occurring on the Extra Boration System injection line (State A)

  In the event of failure of the MHSI to stop (manual), the MHSI and EVU [CHRS] can be used to ensure that the core remains covered and to provide heat removal from the IRWST. Moreover, the secondary side is also available to remove RCP [RCS] heat. This case is similar to the case of the SB LOCA without LHSI. The case of the SB LOCA without LHSI is presented in PCSR Sub-chapter 16.1, section 3.8, and is discussed below.

- Steam generator tube rupture

  As the VDAs [MSRT]s are not available, the SG pressure will stabilise at the MSSV pressure setpoint. The pressure equilibrium phase will result in the primary pressure reaching the MSSV pressure setpoint in order to establish a pressure balance between the RCP [RCS] and the affected SG. The RCV [CVCS] charging flow is isolated following a partial cooldown complete signal and SGa level > MAX2. Although, the VDAs [MSRT]s are stuck closed, the partial cooldown complete signal is actuated as soon as the setpoint is below 60 bar (final PCD setpoint). This results in the isolation of the RCV [CVCS] charging flow and leak termination.

All the non-hazardous stable states presented here are bounded by RRC-A sequences, presented in PCSR Sub-chapter 16.1. These sequences are considered in the load combinations to ensure mechanical integrity of the components (Sub-chapter 6.2).

3.6.2.1. Cooldown of the RCP [RCS] with an isolated Steam Generator

Some fault conditions require the isolation of an affected SG. In such circumstances, if the reactor coolant pumps have been tripped and cooling is provided by natural circulation then there is scope for the flow in the affected loop to stagnate and for temperatures in the loop to remain relatively hot. Rapid depressurisation of the RCP [RCS] by the operator in such conditions could result in the formation of steam in the “inactive” loop.

A modification has been introduced to prevent this phenomenon. The isolation of an affected SG could be required due to a small feedwater line break or a small steam line break coincident with the loss of off-site power, leading to an empty depressurised SG. Under such circumstances, considering single failure and preventive maintenance, up to two RCP [RCS] loops could be “inactive”. A modification has been included in the UK EPR design to implement motorised valves on the ASG [EFWS] pump header. Opening of the ASG [EFWS] pump discharge line header enables the SGs to be supplied by any of the ASG [EFWS] pumps, and prevents the formation of isolated loops. The motorised valves can be actuated from the main control room 30 minutes after reactor trip. This modification ensures that at least two SGs are available and ensure circulation of fluid in the RCP [RCS]. Two SGs are sufficient to ensure that safety of the plant is not impaired in the long term.

3.6.2.2. Decrease in RCP [RCS] inventory faults with failure of the LHSI

The diversity case for reaching a safe shutdown state following a SB LOCA fault is considered. This claims that, in the event of failure of the LHSI, the MHSI and EVU [CHRS] together with the VDA [MSRT] and the ASG [EFWS] on the secondary side are capable of providing adequate long-term cooling.

The results presented in PCSR Sub-chapter 16.1 are summarised here, in Sub-chapter 16.5 – Table 75, and show that the SB LOCA with loss of LHSI has fewer consequences than the SB LOCA without MHSI, despite the differences in the assumed initial reactor power. Confidence in these results stems from additional results based on FA3 studies performed at 4300 MWth. In these analyses, conservative assumptions are considered for the dominant parameters. Similar results are exhibited. In the case of failure of the LHSI, the core remains covered and the heat is removed in the long term via the EVU [CHRS], which ensures adequate IRWST temperature, and the ASG [EFWS]. In contrast, in the case of failure of MHSI, the transient leads to some cladding temperature rise due to core uncovery. Thus, the analysis for this sequence is bounding for the UK EPR.

A long-term study has been performed for the Flamanville 3 detailed design for the containment pressure and temperature transients in the case of SB LOCA without LHSI [Ref-1]. This study is referenced for information only, to provide an example of a study carried out for the Flamanville detailed design. This study deals with the qualification of in-containment material, whereas PCSR Sub-chapter 6.2 deals with the containment qualification.

The analysis is carried out with an initial core power of 4500 MW but the initial conditions in the containment, the residual heat and the characteristics of the RRI/SEC [CCWS/ESWS] trains are conservative.

The pressure and temperature (Tsat(Pvap)) transients calculated in the containment are reproduced below in Sub-chapter 16.5 – Figure 100 [Ref-1]. Because of the unavailability of the LHSI, no cooled water injection is considered. The manual actuation of the two available EVU [CHRS] trains at 12 hours makes the pressure and temperature conditions at 24 hours comparable to those in the case of the same accident but studied under PCC conditions (Ptot = 1.9 bar abs, Tsat(Pvap) = 87°C). However, the pressure and temperature conditions in the containment remain constant until the recovery of the LHSI trains, which makes this transient more onerous during the period 24 hours to 100 hours. This result demonstrates the capacity to extract heat using the EVU [CHRS]. Additional margins could be obtained regarding the pressure and temperature inside the containment by relaxing the time at which the EVU [CHRS] trains are assumed to be actuated.

3.7. CONCLUSIONS

Section 3 analyses the response of the plant to a failure in a PLSF in detailed transient studies.

The diverse protection line shows that the required acceptance criteria are met for all frequent faults. The following reactor trip signals must be provided on a non TXS platform to mitigate the events analysed here:

- Low SG level,
- Low hot leg pressure,
- Low cold leg temperature,
- High hot leg pressure,
- High axial offset,
- Low reactor coolant pump speed,
- High neutron flux.

The last four signals will be allocated to a sufficiently classified non TXS platform.

4. CONCLUSIONS

Following this analysis, the adequacy of the functional diversity in the EPR design is demonstrated for all frequent events.

This analysis is based on the selection of the bounding transients for all the plant level safety functions after a comprehensive review of the postulated initiating events.

Transient analyses, following the first step, show that the safety criteria are met for all the cases analysed using conservative assumptions.

SUB-CHAPTER 16.5 - TABLE 1

Safety Functions Used in Fault Schedule

Main Safety Functions	Plant Level Safety Functions	Lower Level Safety Functions
Reactivity Control	R1 - Maintain Core reactivity control	Control of boron concentration - slow variation
	R2 - Shutdown and maintain core sub-criticality	Negative reactivity fast insertion
		Highly concentrated boron injection
	R3 - Prevention of uncontrolled positive reactivity insertion into the core	Anti-dilution protection
		Ensure minimum boron concentration of water injected into RCP [RCS]
		RCP [RCS] overcooling protection
	R4 - Maintain sufficient sub-criticality of fuel stored outside the reactor coolant system but within the site	Dry fuel racks sub-criticality control
		Underwater fuel racks sub-criticality control
Heat removal	H1 - Maintain sufficient Reactor Coolant System water inventory for core cooling	Fast water injection into the RCP [RCS]
		Low head injection into the RCP [RCS]
		Medium head injection into the RCP [RCS]
		Prevention of RCP [RCS] drainage through auxiliary lines
		Prevention of RCP [RCS] leakage through MCP seals
		RCP [RCS] pressuriser level control
		Water storage for residual heat removal

SUB-CHAPTER 16.5 - TABLE 1 (CONT'D)

Safety Functions Used in Fault Schedule

Main Safety Functions	Plant Level Safety Functions	Lower Level Safety Functions
Heat removal	H2 - Remove heat from the core to the reactor coolant	Core heat removal by RCP [RCS] forced flow in power mode
		Core heat removal by RCP [RCS] forced flow in shutdown mode
		Core heat removal by RCP [RCS] natural circulation in shutdown mode
		RCP [RCS] pressure decrease by energy discharge from pressuriser
		RCP [RCS] pressure decrease by steam cooling in saturated pressuriser
		RCP [RCS] pressure stabilisation - pressuriser two-phase
		RCP [RCS] pressure stabilisation - pressuriser single-phase
	H3 - Transfer heat from the reactor coolant to the ultimate heat sink	Heat removal by steam generators - Emergency shutdown mode
		Heat removal by steam generators - Normal shutdown mode
		Heat removal by steam generators - Power mode
		Heat removal from containment by Containment Heat Removal system (CHRS)
		Heat removal from containment by Low Head Emergency Core Cooling System (ECCS)
		Heat removal in shutdown mode by Residual Heat Removal system (RHRS)
		Reduction of heat generation in RCP [RCS] by limiting auxiliary thermal power source
	H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site	Fuel pool heat removal

SUB-CHAPTER 16.5 - TABLE 1 (CONT'D)

Safety Functions Used in Fault Schedule

Main Safety Functions	Plant Level Safety Functions	Lower Level Safety Functions
Containment	C1 - Maintain integrity of the fuel cladding	Control of the core power distributions
		Prevention of unacceptable core power distributions
	C2 - Maintain integrity of the Reactor Coolant Pressure Boundary	Prevention of RCP [RCS] pressurised thermal shock (PTS)
		RCP [RCS] overpressure protection
	C3 - Limit the release of radioactive material from the reactor containment	Containment building isolation
		Heat removal from containment by Containment Heat Removal system (CHRS)
		Limitation of mass/energy release inside containment
		Participation in containment of secondary system inside RB
		Water storage for residual heat removal
	C4 - Limit the release of radioactive waste and airborne radioactive material	Prevention of radioactive release outside containment from radioactive auxiliary systems
		Prevention of radioactive release outside containment from radioactive steam generator
Other	O1 - Prevent the failure or limit the consequences of failure of a structure, system or component whose failure would cause the impairment of a safety function	Essential component protection

SUB-CHAPTER 16.5 - TABLE 2

List of PCC-2 Events

PCSR Sub-chapter 14.3 section	Event
1	ARE [MFWS] malfunction causing a reduction in feedwater temperature
2	ARE [MFWS] malfunction causing an increase in feedwater flow
3	Excessive increase in secondary steam flow
4	Turbine trip
5	Loss of condenser vacuum
6	Short-term loss of off-site power (≤ 2 hours)
7	Loss of normal feedwater flow (loss of all ARE [MFWS] pumps and of the start-up and shutdown pump)
8	Partial loss of core coolant flow (Loss of one reactor coolant pump)
9	Uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power
10	Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from hot zero power conditions
11	RCCA misalignment up to rod drop, without limitation
12	Start-up of an inactive reactor coolant loop at an incorrect temperature
13	RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant
14	RCV [CVCS] malfunction causing increase or decrease in reactor coolant inventory
15	Primary side pressure transients (spurious pressuriser spraying, spurious pressuriser heating)
16	Uncontrolled RCP [RCS] level drop (states C, D)
17	Loss of one cooling train of the RIS/RRA [SIS/RHRS] in RHR mode (states C, D)
18	Loss of one train of the fuel pool cooling system (PTR [FCPS]) or of a supporting system (state A)
19	Spurious reactor trip (state A)

SUB-CHAPTER 16.5 - TABLE 3

List of PCC-3 Events

PCSR Sub-chapter 14.4 section	Event	Frequency
1	Small steam or feedwater system piping failure (≤ DN 50) including break of connecting lines (≤ DN 50) to SG	2 $x10^{-3}$ per reactor per year
2	Long-term loss of off-site power (> 2 hours)	1 $x10^{-3}$ per reactor per year
3	Inadvertent opening of a pressuriser safety valve	1.6 $x10^{-3}$ per reactor per year
4	Inadvertent opening of a SG relief train or of a safety valve (state A)	> 1 $x10^{-2}$ per reactor per year
5	Small break LOCA (< DN 50) including a break occurring on the extra boration system injection line (states A and B)	6 $x10^{-4}$ per reactor per year
6	Steam generator tube rupture (1 tube)	1 $x10^{-3}$ per reactor per year
7	Inadvertent closure of one/all main steam isolation valves	1.8 $x10^{-2}$ per reactor per year (for 1 VIV [MSIV])
8	Inadvertent loading and operation of a fuel assembly in an improper position	2 x 10^{-3} per reactor per year
9	Forced decrease of reactor coolant flow (four pumps)	Assumed to be frequent despite the lack of data
11	Loss of primary coolant outside the containment	Some V-LOCA precursors are frequent faults
13	Uncontrolled single control rod withdrawal	1.35 $x10^{-3}$ per reactor per year
14	Long-term loss of off-site power (> 2 hours), fuel pool cooling aspect (state A)	1 $x10^{-3}$ per reactor per year
15	Loss of one train of the fuel pool cooling system (PTR [FCPS]) or of a supporting system (State F)	> 1 $x10^{-3}$ per reactor per year
16	Isolable piping failure on a system connected to the fuel pool (states A to F)*	1 $x10^{-3}$ per reactor per year

* Among the events identified in Sub-chapter 14.4, section 16, only the following transients are considered frequent faults due to their frequency:

- Draining via the RCV [CVCS] draining line (state E)
- Voluntary draining of the reactor building pool, spent fuel pool not isolated (state D or state F)
- Inadequately prepared transfer between the loading pit and the fuel building transfer compartment (state A to D)

SUB-CHAPTER 16.5 - TABLE 4

Summary for Decrease in RCP [RCS] Inventory Events

Events	Reactivity Control				Heat removal				Containment			
Decrease in RCP [RCS] inventory	R1	R2	R3	R4	H1	H2	H3	H4	C1	C2	C3	C4
RCV [CVCS] malfunction causing decrease in reactor coolant inventory	N/A	Covered by ATWS SB LOCA	Not bounding for event	N/A	Covered by SB LOCA without MHSI or PCD	N/A	Covered by SB LOCA without MHSI	N/A	N/A	N/A	Covered by SB LOCA	N/A
Small break (< DN 50) including a break occurring on the Extra Boration System injection line (State A)	N/A	**ATWS SB LOCA**	Not bounding for event	N/A	**SB LOCA without MHSI SB LOCA without partial cooldown**	N/A	SB LOCA without MHSI - covered by TLOFW (see §2.5.3.14**)**	N/A	N/A	N/A	**Containment isolation demonstration**	N/A
Steam Generator Tube Rupture (1 tube)	N/A	Covered by ATWS SB LOCA	SGTR without MFW FL isolation	N/A	Covered by SB LOCA without MHSI or PCD	N/A	C Covered by SB LOCA without MHSI	N/A	N/A	N/A	**SGTR without MFW FL isolation**	**ALARP justification of VIV [MSIV] failure to close (SGTR without VIV [MSIV])**

SUB-CHAPTER 16.5 - TABLE 5

Summary for Increase in RCP [RCS] Inventory Events

Events	**Reactivity Control**				**Heat removal**				**Containment**			
Increase in RCP [RCS] inventory	R1	R2	R3	R4	H1	H2	H3	H4	C1	C2	C3	C4
RCV [CVCS] malfunction causing an increase in reactor coolant inventory	N/A	Function not bounding	Function not bounding	N/A	N/A	N/A	Function not bounding	N/A	N/A	Function not bounding	N/A	N/A

SUB-CHAPTER 16.5 - TABLE 6

Summary of Decrease in Heat Removal Events

Events	**Reactivity Control**				**Heat removal**				**Containment**			
Decrease in heat removal	R1	R2	R3	R4	H1	H2	H3	H4	C1	C2	C3	C4
Loss of condenser vacuum	N/A	Covered by Loss of normal feedwater	Function not bounding	N/A	N/A	N/A	Covered by Loss of normal feedwater	N/A	N/A	Covered by closure of all VIVs [MSIV]s	N/A	N/A
Loss of normal feedwater	N/A	**ATWS Loss of normal feedwater**	Function not bounding	N/A	N/A	N/A	**Total loss of feedwater**	N/A	N/A	Covered by closure of all VIVs [MSIV]s	N/A	N/A
Loss of off-site power	N/A	**ATWS LOOP**	Function not bounding	N/A	N/A	**Loss of feedwater+ loss of reactor coolant pumps**	Covered by Total loss of feedwater	N/A	N/A	Covered by closure of all VIVs [MSIV]s	N/A	N/A
Small feedwater system piping failure	N/A	Covered by Loss of normal feedwater	Function not bounding	N/A	N/A	N/A	Covered by Loss of normal feedwater	N/A	N/A	Covered by closure of all VIVs [MSIV]s	N/A	N/A
Inadvertent closure of 1/all VIV [MSIV]s	N/A	Covered by Loss of normal feedwater	Function not bounding	N/A	N/A	N/A	Covered by Loss of normal feedwater	N/A	N/A	**Justification for PSVs (Closure of 4 VIVs [MSIVs] without PSVs)**	N/A	N/A

SUB-CHAPTER 16.5 - TABLE 7

Summary of Increase in Heat Removal Event

Events	Reactivity Control				Heat removal				Containment			
Decrease in heat removal	R1	R2	R3	R4	H1	H2	H3	H4	C1	C2	C3	C4
Feedwater malfunction causing a reduction in feedwater temperature	N/A	Covered by excessive increase in steam flow	Covered by excessive increase in steam flow	N/A	N/A	N/A	Function not bounding	N/A	N/A	N/A	N/A	N/A
Feedwater malfunction causing an increase in feedwater flow	N/A	Covered by excessive increase in steam flow	Covered by excessive increase in steam flow	N/A	N/A	N/A	Function not bounding	N/A	N/A	N/A	N/A	N/A
Excessive increase in steam flow	N/A	**ATWS excessive increase in steam flow**	**Excessive increase in steam flow without VIV [MSIV]**	N/A	N/A	N/A	Function not bounding	N/A	Excessive increase in steam flow without VIV [MSIV]	N/A	N/A	N/A
Small steam system piping failure	N/A	Covered by excessive increase in steam flow	Covered by excessive increase in steam flow	N/A	N/A	N/A	Function not bounding	N/A	N/A	N/A	N/A	N/A

SUB-CHAPTER 16.5 - TABLE 8

Summary for Reactivity Events

Events	**Reactivity Control**				**Heat removal**				**Containment**			
Reactivity events	R1	R2	R3	R4	H1	H2	H3	H4	C1	C2	C3	C4
Uncontrolled RCCA bank withdrawal at power	N/A	**ATWS URBWP**	Covered by ATWS dilution	N/A	N/A	N/A	N/A	N/A	**ATWS URBWP**	N/A	N/A	N/A
Uncontrolled RCCA bank withdrawal from hot zero power conditions	N/A	Covered by ATWS URBWP and ATWS dilution	Covered by ATWS dilution	N/A	N/A	N/A	N/A	N/A	Covered by ATWS URBWP and ATWS dilution	N/A	N/A	N/A
Start-up of an inactive reactor coolant pump at an incorrect temperature,	N/A	Covered by ATWS URBWP and ATWS dilution	Covered by ATWS dilution	N/A	N/A	N/A	N/A	N/A	Covered by ATWS URBWP and ATWS dilution	N/A	N/A	N/A

Events	**Reactivity Control**				**Heat removal**				**Containment**			
RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant,	N/A	**ATWS Dilution**	**ATWS dilution**	N/A	N/A	N/A	N/A	N/A	**ATWS Dilution**	N/A	N/A	N/A
Uncontrolled single control rod withdrawal	N/A	Covered by ATWS URBWP and ATWS dilution	Covered by ATWS dilution	N/A	N/A	N/A	N/A	N/A	Covered by ATWS URBWP and ATWS dilution	N/A	N/A	N/A
Partial loss of core coolant flow (Loss of one reactor coolant pump)	N/A	Covered by ATWS URBWP and ATWS dilution	Covered by ATWS dilution	N/A	N/A	N/A	N/A	N/A	Covered by ATWS URBWP and ATWS dilution	N/A	N/A	N/A
RCCA misalignment up to rod drop	N/A	Covered by ATWS URBWP and ATWS dilution	Covered by ATWS dilution	N/A	N/A	N/A	N/A	N/A	Covered by ATWS URBWP and ATWS dilution	N/A	N/A	N/A

SUB-CHAPTER 16.5 - TABLE 9

Summary for Fuel Pool Events

Events	**Reactivity Control**				**Heat removal**				**Containment**			
	R1	R2	R3	R4	H1	H2	H3	H4	C1	C2	C3	C4
Loss of one train of the fuel pool cooling system (PTR [FPCS]) or of a supporting system (state A)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	Covered by Draining via the RCV [CVCS] unloading line (state E)	N/A	N/A	N/A	N/A
Long-term LOOP, fuel pool cooling aspects (state A)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	Covered by Draining via the RCV [CVCS] unloading line (state E)	N/A	N/A	N/A	N/A
Draining via the RCV [CVCS] unloading line (state E)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	**Draining via the RCV [CVCS] unloading line (state E)**	N/A	N/A	N/A	N/A

Events	Reactivity Control				Heat removal				Containment			
	R1	R2	R3	R4	H1	H2	H3	H4	C1	C2	C3	C4
Voluntary draining of the reactor building pool, Spent Fuel Pool not isolated (state D)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	Covered by Draining via the RCV [CVCS] unloading line (state E)	N/A	N/A	N/A	N/A
Inadequately prepared transfer between the loading pit and the fuel building transfer compartment (state A to D)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	Covered by Draining via the RCV [CVCS] unloading line (state E)	N/A	N/A	N/A	N/A

SUB-CHAPTER 16.5 - TABLE 10

Summary for Miscellaneous Events

Events	Reactivity Control				Heat removal				Containment			
Miscellaneous	R1	R2	R3	R4	H1	H2	H3	H4	C1	C2	C3	C4
Spurious pressuriser spray	N/A	**ATWS spurious spray**	Function not bounding	N/A	N/A	N/A	Function not bounding	N/A	**ATWS spurious spray**	N/A	N/A	N/A
Spurious pressuriser heaters	N/A	Covered by spurious pressuriser spray	Function not bounding	N/A	N/A	N/A	Function not bounding	N/A	Function not bounding	**Spurious Pressuriser heaters**	N/A	N/A

SUB-CHAPTER 16.5 - TABLE 11

Summary of Cases to Be Analysed

Family of Events	Reactivity Control				Heat removal				Containment			
	R1	R2	R3	R4	H1	H2	H3	H4	C1	C2	C3	C4
Decrease in RCP [RCS] inventory	N/A	ATWS SB LOCA	SGTR without MFW FL isolation	N/A	SB LOCA without MHSI SB LOCA without PCD SB LOCA without VDAs [MSRT]s		Already in H1 (Feed and Bleed)	N/A	N/A	N/A	ALARP justification (containment isolation)	ALARP justification for VIV [MSIV] failure to close (SGTR with failure of VIV [MSIV]a)
Increase in RCP [RCS] inventory	N/A	Function not bounding	Function not bounding	N/A	N/A	N/A	Function not bounding	N/A	N/A	N/A	Function not bounding	N/A
Decrease in heat removal	N/A	ATWS Loss of feedwater ATWS LOOP	Function not bounding	N/A	N/A	Loss of feedwater + loss of reactor coolant pumps	Total loss of feedwater	N/A	N/A	Justification for PSVs (Closure of 4 VIVs [MSIV]s without PSVs)	N/A	N/A

Family of Events	**Reactivity Control**				**Heat removal**				**Containment**			
	R1	R2	R3	R4	H1	H2	H3	H4	C1	C2	C3	C4
Increase in heat removal	N/A	ATWS excessive increase in steam flow	Excessive increase in steam flow without VIV [MSIV]	N/A	N/A	N/A	Function not bounding	N/A	Excessive increase in steam flow without VIV [MSIV]	N/A	N/A	N/A
Reactivity events	N/A	ATWS URBWP ATWS dilution	ATWS dilution (RPR [PS])	N/A	N/A	N/A	N/A	N/A	ATWS URBWP ATWS dilution	N/A	N/A	N/A
Fuel pool transients	N/A	N/A	N/A	N/A	N/A	N/A	N/A	Draining via the RCV [CVCS] unloading line (state E)	N/A	N/A	N/A	N/A
Miscellaneous	N/A	ATWS Spurious pressuriser spray	N/A	N/A	N/A	N/A	N/A	N/A	ATWS Spurious pressuriser spray	Spurious pressuriser heaters	N/A	N/A

SUB-CHAPTER 16.5 - TABLE 12

Limiting Transients per PLSF

Main **Safety Function**	**Plant Level Safety Functions**	**Limiting transients**
Reactivity Control	R1 - Maintain core reactivity control	Loss of RCV [CVCS] event after reactor trip
	R2 - Shutdown and maintain core sub-criticality	ATWS by mechanical blockage of the rods and by loss of Protection System: – SB Loss of coolant accident (< DN 50) – Loss of main feedwater – Excessive increase in steam flow – Loss of off-site power – Rod drop faults
	R3 - Prevention of uncontrolled positive reactivity insertion into the core	– Excessive increase in steam flow without Main Steam Isolation Valve closure – ATWS by failure of Protection System: dilution
	R4 - Maintain sufficient sub-criticality of fuel stored outside the reactor coolant system but within the site	Not challenged by fuel pool events
Heat removal	H1 - Maintain sufficient Reactor Coolant System water inventory for core cooling	– SB Loss of coolant accident (< DN 50) without MHSI – SB Loss of coolant accident (< DN 50) without partial cooldown signal – SB Loss of coolant accident with failure of the VDAs [MSRT]s

Main Safety Function	**Plant Level Safety Functions**	**Limiting transients**
	H2 - Remove heat from the core to the reactor coolant	Loss of feedwater with loss of reactor coolant pumps
	H3 - Transfer heat from the reactor coolant to the ultimate heat sink	Total loss of feedwater
	H4 - Maintain heat removal from fuel stored outside the reactor coolant system but within the site	Draining via the RCV [CVCS] unloading line with failure of the manual isolation (State E)
Containment	C1 - Maintain integrity of the fuel cladding	ATWS by mechanical blockage of the rods and by loss of Protection System: – RCCA bank withdrawal – Dilution – Spurious pressuriser spray – Forced decrease in reactor coolant flow
	C2 - Maintain integrity of the Reactor Coolant Pressure Boundary	Justification of Pressuriser Safety Valves design
	C3 - Limit the release of radioactive material from the reactor containment	ALARP justification of containment isolation
	C4 - Limit the release of radioactive waste and airborne radioactive material	ALARP justification concerning VIV [MSIV] closure
Other	O1 - Prevent the failure or limit the consequences of failure of a structure, system or component whose failure would cause the impairment of a safety function	Closure of 4 VIVs [MSIV]s without VDAs [MSRT]s

SUB-CHAPTER 16.5 - TABLE 13

R2 – ATWS SB LOCA – 1: Initial State SB LOCA

Parameters	Units	Nominal values	Uncertainties	Used values
Primary side				
Core power	%FP	100	2	102
Core power	MW	4500	2%	4590
Primary flow rate	m^3/h/loop	TH: 27185	-	27185
Average temperature	°C	312,8	2,5	315,3
Primary pressure	bar abs	155	2,5	157,5
PZR level	%MR	56	+5	61
Secondary side				
SG level	%NR	49	-	49
MFWS Temperature	°C	230	-	230
ASG Temperature	°C	10/50	-	50

SUB-CHAPTER 16.5 - TABLE 14

R2 - ATWS SB LOCA - 1: Systems Characteristics

Parameter	Value
Reactor trip / turbine trip signals on pressuriser pressure < MIN2	
Setpoint Delay	135 bar -3 bar uncertainty 1.2 s
ATWS signal (reactor trip signal and high rod position (or high flux) after delay	
Delay (after RT signal)	20 s
RBS [EBS] boration:	
Delay (after ATWS signal)	15 s
Capacity per RBS [EBS] train	2.8 kg/s
RBS [EBS] boron concentration (natural boron)	11200 ppm
All reactor coolant pumps trip on signal ATWS and SG level (wide range) < MIN2	
Setpoint Delay (after RT signal)	40% - 5% uncertainty 1.5 s + 0.15 s
MS relief train	
Setpoint (with uncertainty) Delay Capacity	95.5 bar + 1.5 bar 0.5 + 1.5 = 2 s 1150 te/h under 100 bar (50% of full load flow rate/SG)
MS safety valves	
Setpoint Accumulation Capacity	105 bar 3% 575 te/h under 100 bar, 2 per SG

Parameter	Value
Pressuriser safety valves	
Setpoints Closing pressure dead time opening time Capacity	175 / 178 / 181 bar 147/149.5/152 bar 0.5 s 0.1 s steam 290 te/h, liquid 623 te/h under 176 bar
ASG [EFWS] actuation on SG level (wide range cold side) < MIN2	
Setpoint Delay Capacity 4 ASG [EFWS] tanks water content	40% - 5% uncertainty 1.5 + 15 = 16.5 s 90 te/h per SG at 97 bar 1680 te
RIS [SIS] and partial cooldown actuation	
Setpoint Delay	115 bar -3 bar uncertainty 0.9 s
Automatic partial cooldown	
Setpoint cooling rate	60 bar + 1.5 bar uncertainty -250°C/h
MHSI	
Pump actuation Injection pressure Injection temperature natural boron concentration	15 s 85 bar 70°C 2450 ppm

Parameter	Value
Accumulators	
Initial injection pressure	45 bar
Volume	35 m^3
temperature	50°C
natural boron concentration	2450 ppm

SUB-CHAPTER 16.5 - TABLE 15

R2 - ATWS SB LOCA - 2: Sequence of events

Time (s)	Events
0	Beginning of calculation (break initiation)
79	Reactor trip with no RCCA dropping
79	Turbine trip
87	High SG pressure, VDA [MSRT] and MSSV opening
99	ATWS signal
114	RBS [EBS] injection
177	RCP [RCS] pumps trip (SG level < 35% WR + ATWS signal)
194	ASG [EFWS] actuation (SG level < 35% WR)
211	First PSV opening
825	Pressuriser heaters switch off (pressuriser level < 12% MR)
1055	RIS [SIS] signal and beginning of partial cooldown
1316	MHSI injection
1508	End of partial cooldown
2588	Accumulators injection
5000	End of transient

SUB-CHAPTER 16.5 - TABLE 16

R2 - ATWS LOFW - 1: Initial Conditions

Parameter	Units	Nominal Values	Uncertainties	Used values
Primary Side				
Power	% FP	100	2	102
Power	MW	4500	2%	4590
RCP [RCS] flow rate	m^3/h/loop	T/H 27,185		27,185
Average temperature	°C	312.7	2.5	315.2
Pressure	Bar abs	155	(-) 2.5	152.2
Pressuriser level	% MR	51% (nom)		51%
Secondary side				
ARE [MFWS] inlet temperature	°C	230		230
SG level	% NR	49		49

SUB-CHAPTER 16.5 - TABLE 17

R2 - ATWS LOFW - 2: Systems Characteristics

PARAMETER	VALUE
Reactor trip / turbine trip signals on pressuriser pressure > MAX1 (NR)	
Setpoint	165 bar (166.5 - 1.5) pessimised to advance turbine trip
ATWS signal (reactor trip signal and high rod position (or high flux) after delay, actuates RBS [EBS] boration:	
Capacity (per RBS [EBS] train)) Actuation delay RBS [EBS] boron concentration (natural boron)	2.8 kg/s 20 s 11200 ppm
ASG [EFWS] actuation on SG level (wide range cold side) < MIN2	
Setpoint Actuation delay Capacity	38% (40%-2%) pessimised to delay ASG [EFWS] actuation 15 s 90 te/h per SG at 97 bar
VDA [MSRT] opening on SG pressure > MAX1	
Setpoint Actuation delay Capacity	95.5 + 1.5 bar (pessimised to delay VDA [MSRT] opening) 2 s 1150 te/h under 100 bar (50% of full load flow rate/SG)
MSSVs	
Setpoint Capacity	105 +1.5 bar (pessimised to delay MSSV opening) 575 te/h under 100 bar, 2 per SG (2 x 25% of full load flow rate/SG)
Pressuriser safety valves	
Setpoints Capacity Hysteresis	175 - 1.5 / 178 -1.5 / 181 -1.5 bar, pessimised to advance PSV opening steam 290 te/h, under 176 28 / 28.5 / 29 bar

SUB-CHAPTER 16.5 - TABLE 18

R2 - ATWS LOFW - 3: Sequence of events

Time (s)	Event
3	DNB minimum for the transient
5	ARE [MFWS] cut off
37	Reactor trip signal on pressuriser pressure > MAX1 (rods drop not effective)
37.3	Turbine trip (valves closed)
42/67.5	PSV1 opening/closing
57	ATWS signal / RBS [EBS] actuation
43.5	Primary pressure peak (177 bar at pressuriser)
44/65.3	PSV2 opening/closing
45.2	VDA [MSRT] opening
63.5/100.8	MSSV
91.5	Reactor coolant pumps trip
115	ASG [EFWS] injection on SG level < MIN2
247.5	Primary pressure peak (173.8 bar)
248/301	PSV1 opening / closing
259/285.5	Pressuriser filled
330/951.5	Pressuriser filled
~600	RBS [EBS] Boron arrival in the core

SUB-CHAPTER 16.5 - TABLE 19

R2 - ATWS LOFW w/o RPR [PS] - 1: Initial Conditions

Parameter	Units	Nominal Values	Uncertainties	Used values
Primary Side				
Power	% FP	100	2	102
Power	MW	4500	2%	4590
RCP [RCS] flow rate	m^3/h/loop	T/H 27,185		27,185
Average temperature	°C	312.7	2.5	315.2
Pressure	Bar abs	155	- 2.5	152.5
Pressuriser level	% MR	51% (nom)		51%
Secondary side				
ARE [MFWS] inlet temperature	°C	230		230
SG level	% NR	49		49

SUB-CHAPTER 16.5 - TABLE 20

R2 - ATWS LOFW w/o RPR [PS] - 2: Systems Characteristics

PARAMETER	VALUE
Reactor trip / turbine trip signals on high HL pressure	
Setpoint	173 bar (preliminary value)
Reactor coolant pumps trip on low low SG levels	
Setpoint	14% WR
MSSVs	
Setpoint	105 +1.5 bar (pessimised to delay MSSV opening)
Accumulation	3%
Capacity	575 te/h under 100 bar, 2 per SG (2 x 25% of full load flow rate/SG)
Pressuriser safety valves	
Setpoints	175 - 1.5 / 178 -1.5 / 181 -1.5 bar abs., pessimised to advance PSV opening
Capacity	steam 290 te/h, under 176
Hysteresis	28 / 28.5 / 29 bar
Actuation Delay	0.5 s dead time on opening 0.1 s opening time 5 s dead time on closing 1 s closing time

SUB-CHAPTER 16.5 - TABLE 21

R2 - ATWS LOFW w/o RPR [PS] - 3: Sequence of Events

Time (s)	Event
3	DNB minimum for the transient
5	ARE [MFWS] cut off
60	RT on high hot leg pressure
61.5	Primary pressure peak (173.4 bar)
62/84	PSV 1 opening/closing
84	MSSV opening
720	Reactor coolant pumps trip
957/974	PSV1 opening/closing
1164/1180	PSV1 opening/closing
1313/1327	PSV1 opening/closing
~1400	SGs are empty
1430	PSV1 opening/closing
1860	The operator starts emergency operating procedures

SUB-CHAPTER 16.5 - TABLE 22

R2 - ATWS LOOP - 1: Initial Conditions

Parameter	Initial value
RCP [RCS] flow rate	T/H design flow rate
Power	102% (100+2%)NP (pessimised)
Pressure	152.5 (155-2.5) bar (pessimised)
Average temperature	315.2 (312.7+2.5) °C (pessimised)
Pressuriser level	51% (pessimised)
SG pressure	81.1 bar (depending on primary temperature)
ARE [MFWS] inlet temperature	230°C (nom)
SG level	49% NR (nom)

SUB-CHAPTER 16.5 - TABLE 23

R2 - ATWS LOOP - 2: Systems Characteristics

PARAMETER	VALUE
Reactor trip / turbine trip signals on reactor coolant pumps speed < MIN	
Setpoint (% of nominal speed)	91
RBS [EBS] actuation on ATWS signal (reactor trip signal and high rod position(or high flux) after delay	
Capacity (per RCP [RCS] loop) Actuation delay RBS [EBS] boron concentration (natural boron)	2.8 kg/s 46 s 11200 ppm
ASG [EFWS] actuation on SG level (wide range cold side) < MIN2	
Setpoint Actuation delay Capacity	38% (40% - 2%) pessimised to delay ASG [EFWS] actuation 50 s 90 te/h per SG at 97 bar
VDA [MSRT] opening on SG pressure > MAX1	
Setpoint Capacity	95.5 + 1.5 bar (pessimised to delay VDA [MSRT] opening) 1150 te/h under 100 bar (50% of full load flow rate/SG)
MSSVs	
Setpoint Capacity	105 +1.5 bar 575 te/h under 100 bar, 2 per SG (2 x 25% of full load flow rate/SG)
Pressuriser safety valves	
Setpoints Capacity Hysteresis	175 – 1.5 / 178 -1.5 / 181 -1.5 bar, pessimised to advance PSV opening steam 290 te/h, under 176 28 / 28.5 / 29 bar

SUB-CHAPTER 16.5 - TABLE 24

R2 - ATWS LOOP - 3: Sequence of events

Time (s)	Event
5	LOOP:
5	Turbine trip ARE [MFWS] cut off RCP [RCS] pumps trip
7	Reactor trip signal (RCP [RCS] pump speed < MIN1)
14/36.5	1st PSV opening/closing
14.3	VDA [MSRT] opening
15.75	Primary pressure peak (at pressuriser) (176.2 bar)
16/34	2nd PSV opening
27.5	RBS [EBS] actuation
75	RBS [EBS] injection in RCP [RCS] loops
398/449	1st PSV opening / closing
470	ASG [EFWS] injection
482.5	Pressuriser filled
~500	RBS [EBS] boron arrival in the core

SUB-CHAPTER 16.5 - TABLE 25

R2 - ATWS LOOP w/o RPR [PS] - 1: Initial conditions

Parameter	Units	Nominal Values	Uncertainties	Used values
Primary Side				
Power	% FP	100	2	102
Power	MW	4500	2%	4590
RCP [RCS] flow rate	m^3/h/loop	T/H 27,185		27,185
Average temperature	°C	312.7	2.5	315.2
Pressure	Bar abs	155	- 2.5	152.5
Pressuriser level	% MR	51% (nom)		51%
Secondary side				
ARE [MFWS] inlet temperature	°C	230		230
SG level	% NR	49		49

SUB-CHAPTER 16.5 - TABLE 26

R2 - ATWS LOOP w/o RPR [PS] - 2: Systems Characteristics

PARAMETER	VALUE
Reactor trip / turbine trip signals on high HL pressure	
Setpoint	173 bar abs. (preliminary value)
MSSVs	
Setpoint Accumulation	105 +1.5 bar (pessimised to delay MSSV opening) 3%
Capacity	575 te/h under 100 bar, 2 per SG (2 x 25% of full load flow rate/SG)
Pressuriser safety valves	
Setpoints Capacity Hysteresis Actuation delay	175 - 1.5 / 178 -1.5 / 181 -1.5 bar, pessimised to advance PSV opening steam 290 te/h, under 176 28 / 28.5 / 29 bar 0.5 s dead time on opening 0.1 s opening time 5 s dead time on closing 1 s closing time

SUB-CHAPTER 16.5 - TABLE 27

R2 - ATWS LOOP w/o RPR [PS] - 3: Sequence of events

Time (s)	Event
5	Loss of off-site power occurrence ARE [MFWS] cut off
	Reactor coolant pumps trip
	Turbine trip
7	DNB minimum for the transient
13.7	RT on high HL pressure
14/31.5	PSV1 opening/closing
15.5	Primary pressure peak (175.8 bar)
19.5	MSSV opening
1814	The operator starts emergency operating procedures

SUB-CHAPTER 16.5 - TABLE 28

R2 - ATWS EISF - 1: Initial Conditions

Parameter	Initial value
RCP [RCS] flow rate	T/H design flow rate
Power	102% NP (pessimised)
Pressure	152.5 abs bar (pessimised)
Average temperature	315.2°C (pessimised)
Pressuriser level	51% (pessimised)
SG pressure	result of the code
ARE [MFWS] inlet temperature	230°C (nom)
SG level	49% NR (nom)

SUB-CHAPTER 16.5 - TABLE 29

R2 - ATWS EISF - 2: Systems Characteristics

PARAMETER	VALUE
Reactor trip / turbine trip signals on SG level < MIN1 (NR)	
Setpoint	20% (13.8 m) – 2% pessimism
Delay	1.5 s
ATWS signal (reactor trip signal and high rod position (or high flux) after delay, actuates RBS [EBS] boration:	
Delay (after RT signal) for ATWS signal	20 s
Delay (after ATWS signal) for RBS [EBS] boration	15 s
Capacity (per RCP [RCS] loop)	2.8 kg/s
RBS [EBS] boron concentration (natural boron)	11200 ppm
ASG [EFWS] actuation on SG level (wide range cold side) < MIN2	
Setpoint	7.85 m (40%) – 2% pessimism
Delay	1.5 + 15 = 16.5 s
Capacity	90 te/h per SG at 97 bar
4 ASG [EFWS] tanks water content	1680 te
MS relief train actuation on SG pressure > MAX1	
Setpoint	95.5 abs bar – 1.5 bar pessimism
Delay	0.9 + 1.5 + 0.3 = 2.7 s
VDA [MSRT] closing delay	40 s
Capacity	1270 te/h under 100 bar
MS safety valves	
Setpoint	105 bar – 1.5 bar pessimism
Accumulation	3%
Capacity	865 te/h under 100 bar, 2 per SG (2 x 27.5% of full load flow rate/SG)
Pressuriser safety valves	
Setpoints	175 – 1.5 / 178 – 1.5 / 181 – 1.5 bar (pessimised)
dead time	0.5 s
opening time	0.1 s
Capacity	steam 360 te/h, under 176 bar
	liquid 450 te/h under 176 bar
Hysteresis (preliminary value)	28 – 28.5 – 29 bar

SUB-CHAPTER 16.5 - TABLE 30

R2 - ATWS EISF - 3: Sequence of Events

Time (s)	Event
5.0	Increase of secondary steam flow
313.3	Reactor trip signal on SG level < MIN1
315.5	Turbine trip
327.6	VDA [MSRT] opening
328.3 / 357.3	PSV1 opening/closing
328.3	Primary pressure peak (175.03 bar)
333.3	ATWS signal / RBS [EBS] actuation
348.9	RBS [EBS] Boron arrival in the core
397.1	Reactor coolant pumps trip
397.5	Secondary pressure peak (104.08 bar)
412.0	ASG [EFWS] activation

SUB-CHAPTER 16.5 - TABLE 31

R2 - ATWS EISF w/o RPR [PS] - 1: Initial Conditions

Parameter	Initial value
Initial state = 100%	
RCP [RCS] flow rate	T/H design flow rate
Power	102% NP (100% + 2%)
Pressure	152.5 abs bar (155 bar abs– 2.5 bar abs)
Average temperature	315.2°C (312.7°C + 2.5°C)
Pressuriser level	51% (56% - 5%)
SG pressure	result of the code
ARE [MFWS] inlet temperature	230°C (nominal)
SG level	49% NR (nominal)

SUB-CHAPTER 16.5 - TABLE 32

R2 - ATWS EISF w/o RPR [PS] - 2: Systems Characteristics

PARAMETER	VALUE
Diversify reactor trip + turbine trip signals on SG level	
Setpoint	40% GL – 5% pessimism
RT Delay	1.5 s
Delay between RT and turbine trip	2.2 s
ARE [MFWS] parameters	
ARE [MFWS] high load isolation	15 s
ARE [MFWS] low load isolation	15 s
MS safety valves	
Setpoint	105 bar – 1.5 bar pessimism
Accumulation	3%
Capacity	635 te/h under 100 bar, 2 per SG (2 x 25% of full load flow rate/SG)
Pressuriser safety valves	
Setpoints	175 – 1.5 / 178 – 1.5 / 181 – 1.5 bar (pessimised)
Dead time	0.5 s
Opening time	0.1 s
Capacity	steam 290 te/h, under 176 bar
	liquid 450 te/h under 176 bar
Hysteresis (preliminary value)	28 – 28.5 – 29 bar

SUB-CHAPTER 16.5 - TABLE 33

R2 - ATWS EISF w/o RPR [PS] - 3: Sequence of Events

Time (s)	Event
5.0	Increase of secondary steam flow
928.5	Diversify reactor trip on low SG level
930.7	Turbine trip
1230	MSSV opening and secondary pressure stabilisation
1274.0	Secondary pressure peak (103.7 bar)
1471.5	Pressuriser pressure peak (171.8 bar)
2728.5	Beginning of operator actions (30 min after RT)

SUB-CHAPTER 16.5 - TABLE 34

H1 - SB LOCA w/o MHSI - 1: Initial Conditions

Parameters	Units	Nominal values	Uncertainties	Values used
Primary side				
Core power	%FP	100	2	102
Core power	MW	4500	2%	4590
Primary flow rate	m^3/h/loop	TH: 27185	-	27185
Average temperature	°C	312.8	2.5	315.3
Primary pressure	bar abs	155	2.5	157.5
PZR level	%MR	56	+5	61
Secondary side				
SG level	%NR	49	-	49
MFWS Temperature	°C	230	-	230
ASG Temperature	°C	10/50	-	50

SUB-CHAPTER 16.5 - TABLE 35

H1 - SB LOCA w/o MHSI - 2: Characteristics of systems

Parameter	Value
Reactor trip / turbine trip signals on pressuriser pressure < MIN2	
Setpoint	135 bar -3 bar uncertainty
Delay	1.2 s
MS relief train	
Setpoint (with uncertainty)	95.5 bar + 1.5 bar
Delay	0.5 + 1.5 = 2 s
Capacity	1150 te/h under 100 bar (50% of full load flow rate/SG)
ASG [EFWS] actuation on SG level (wide range cold side) < MIN2	
Setpoint	40% - 5% uncertainty
Delay	1.5 + 15 = 16.5 s
Capacity	90 te/h per SG at 97 bar
4 ASG [EFWS] tanks water content	1680 te
LHSI	
Injection pressure	20 bar
Injection temperature	70°C
Accumulators	
Initial injection pressure	45 bar
Volume	35 m^3
Temperature	50°C

SUB-CHAPTER 16.5 - TABLE 36

H1 - SB LOCA w/o MHSI - 3: Typical Sequence of Events

Time (s)	**Event**
0	Beginning of calculation (break initiation)
79	Reactor trip
79	Turbine trip
89	High SG pressure, VDA [MSRT] opening
110	Pressuriser heaters switch off (pressuriser level < 12% MR)
172	RIS [SIS] signal and beginning of partial cooldown
534	RCP [RCS] pumps trip (ΔP over three pumps < 75%)
625	End of partial cooldown
1013	ASG [EFWS] actuation (SG level < 35% WR)
1972	Actuation of “Fast Cooldown” by the operator
2056	Accumulators injection
2239	LHSI injection
4000	End of transient

SUB-CHAPTER 16.5 - TABLE 37

Table - H1 - SB LOCA w/o PCD - 1: Initial conditions

Parameters	Units	Nominal values	Uncertainties	Used values
Primary side				
Core power	%FP	100	2	102
Core power	MW	4500	2%	4590
Primary flow rate	m^3/h/loop	TH: 27185	-	27185
Average temperature	°C	312.8	2.5	315.3
Primary pressure	bar abs	155	2.5	157.5
PZR level	%MR	56	+5	61
Secondary side				
SG level	%NR	49	-	49
MFWS Temperature	°C	230	-	230
ASG Temperature	°C	10/50	-	50

SUB-CHAPTER 16.5 - TABLE 38

Table - H1 - SB LOCA w/o PCD - 2: Characteristics of systems

Parameter	Value
Reactor trip / turbine trip signals on pressuriser pressure < MIN2	
Setpoint	135 bar – 3 bar uncertainty
Delay	1.2 s
MS relief train	
Setpoint (with uncertainty) Delay Capacity	95.5 bar + 1.5 bar 0.5 + 1.5 = 2 s 1150 te/h under 100 bar (50% of full load flow rate/SG)
ASG [EFWS] actuation on SG level (wide range cold side) < MIN2	
Setpoint Delay Capacity 4 ASG [EFWS] tanks water content	40% - 5% uncertainty 1.5 + 15 = 16.5 s 90 te/h per SG at 97 bar 1680 te
MHSI	
Injection pressure Injection temperature	85 bar 70°C
Accumulators	
Initial injection pressure Volume Temperature	45 bar 35 m^3 50°C

SUB-CHAPTER 16.5 - TABLE 39

H1 - SB LOCA w/o PCD - 3: Sequence of Events

Time (s)	Event
0	Beginning of calculation (break initiation)
79	Reactor trip
79	Turbine trip
89	High SG pressure, VDA [MSRT] opening
110	Pressuriser heaters switch off (Pressuriser level < 12% MR)
172	RIS [SIS] signal, failure of the partial cooldown signal
684	RCP [RCS] pumps trip (ΔP over three pumps < 75%)
1972	Actuation of “ Manual partial cooldown” by the operator
2058	ASG [EFWS] actuation (SG level < 35% WR)
2145	MHSI injection
2424	End of “Manual partial cooldown”
3510	Accumulators injection
4000	End of transient

SUB-CHAPTER 16.5 - TABLE 40

H2 - LOFW w/o Reactor Coolant Pumps - 1: Initial Conditions

Primary side	
Core power	102% (= 100% + 2%) / 4590 MW
Power through the SG	1154 MW (depending on primary power ; including reactor coolant pumps)
Mean RCP [RCS] temperature	315.2°C (= 312.7°C + 2.5°C)
Pressuriser pressure	157.5 bar (= 155 bar + 2.5 bar)
Pressuriser level	61% (= 56% + 5%)
Loop flow rate	TH flow rate (27,185 m^3/h)
Core bypass flow rate	5.5%
Decay heat	{CCI Removed} [b]
Secondary side	
Secondary mass	Depending on operating point
SG level	54% (= 49% + 5%)
SG pressure	Depending on operating point
SG temperature	Depending on operating point

SUB-CHAPTER 16.5 - TABLE 41

H2 - LOFW w/o Reactor Coolant Pumps - 2: Characteristics of Systems

Parameter	Value
Reactor trip / turbine trip signals on pressuriser pressure < MIN2	
Setpoint	135 bar - 3 bar uncertainty
Delay	1.2 s
MS relief train	
Setpoint (with uncertainty)	95.5 bar + 1.5 bar
Delay	0.5 + 1.5 = 2 s
Capacity	1150 te/h under 100 bar (50% of full load flow rate/SG)
ASG [EFWS] actuation on SG level (wide range cold side) < MIN2	
Setpoint	40% - 2% uncertainty
Delay	1.5 + 15 = 16.5 s
Capacity	90 te/h per SG at 97 bar
MSSV	
Setpoint	105 + 1.5 = 106.5 bar abs

SUB-CHAPTER 16.5 - TABLE 42

H2 - LOFW w/o Reactor Coolant Pumps - 3: Sequence of events

Time (s)	Event
0	Loss of main feedwater
36	Opening 1st PSV
46	SG level < MIN1 (RT threshold)
46	RCP [RCS] pumps stop
48	RT
48	Turbine trip
49	Automatic blowdown isolation
55	VDA [MSRT] opening
56	Closing 1st PSV
302	SG2 level < MIN2
307	SG3 level < MIN2
318	Start-up ASG [EFWS]2
323	Start-up ASG [EFWS]3
323	SG4 level < MIN2
324	SG1 level < MIN2
339	Start-up ASG [EFWS]4
339	Start-up ASG [EFWS]1
~4000	All parameters are stabilised The controlled state is reached
10000	End of short-term calculation

SUB-CHAPTER 16.5 - TABLE 43

H3 - TLOFW - 1: Initial conditions

Parameters	Units	Nominal values	Uncertainties	Values used
Primary side				
Core power	%FP	100	2	102
Core power	MW	4500	2%	4590
Primary flow rate	m^3/h/loop	TH: 27185	-	27185
Average temperature	°C	312.8	2.5	315.3
Primary pressure	bar abs	155	2.5	157.5
PZR level	%MR	56	-5	51
Secondary side				
SG level	%NR	49	-5	44
MFW Temperature	°C	230	-	230

SUB-CHAPTER 16.5 - TABLE 44

H3 - TLOFW - 2: Characteristics of systems

Parameter	Value
Primary depressurisation system (PDS)	
Capacity	900 t/h of saturated steam under 176 bar
MS relief train	
Setpoint (with uncertainty) Delay Capacity	95.5 bar + 1.5 bar 0.5 + 1.5 = 2 s 1150 te/h under 100 bar (50% of full load flow rate/SG)
Pressuriser safety valves	
Setpoints dead time opening time Capacity Hysteresis (preliminary value)	175 / 178 / 181 bar 0.5 s 1.5 s steam 290 te/h, liquid 450 te/h under 176 bar 28 bar
MHSI	
Injection pressure Injection temperature	85 bar 50°C
Accumulators	
Initial injection pressure Volume Temperature	45 bar 35 m^3 50°C

SUB-CHAPTER 16.5 - TABLE 45

H3 - TLOFW - 3: Sequence of events

Time (s)	Event
0	ARE [MFWS] cut off
5	Partial trip to 50% FP
5	Turbine trip
15	Total reactor trip
978	SG level < 40% WR: Blowdown isolation
1815	Beginning of operators' actions
1815	Pressuriser heaters cut off
1852	SG level < 14% WR: Manual RCP [RCS] pumps trip
2557	Primary pressure > 175 bar: PSV opening
2935	Core outlet temperature > 330°C: Feed and Bleed actuation
2935	PDS opening
3558	MHSI injection
3905	Accumulators injection
5726	LHSI injection
8000	End of transient

SUB-CHAPTER 16.5 - TABLE 46

C1 - ATWS spray - 1: Initial conditions

Parameters	Unit	Nominal value	Value used for study
Core power	MWth	4500	4590
Core power level	% NP	100	102
Primary average temperature	°C	312.7	315.2
Pressuriser pressure	Bar abs	155.0	152.5
Pressuriser level	% NR	56	51
Primary flow rate	m^3/h	27,185	27,185
Steam generator outlet pressure	Bar abs	78.0	Calculated by code
Steam generator level	% NR	49	49
ARE temperature	°C	230	230

SUB-CHAPTER 16.5 - TABLE 47

C1 - ATWS spray - 2: Protection System

Signal	**Setpoint**	**Time delays**
Low pressuriser pressure	133.5 bar abs (including -1.5 bar uncertainties)	Delay of 0.9 s between pressure threshold and reactor trip on Low pressuriser pressure signal.
High SG pressure	97 bar abs (including +1.5 bar uncertainties)	Delay of 0.9 s between pressure threshold and high SG pressure signal.

SUB-CHAPTER 16.5 - TABLE 48

C1 - ATWS spray - 3: Systems characteristics

PSVs actuation		
PARAMETER		**VALUE**
Opening pressure thresholds	Min	173.5 abs, 176.5 bar abs and 179.5 bar abs (including -1.5 bar uncertainty
Valve opening dead time	operational	0.5 s
Valve opening time	operational	0.1 s
VDA [MSRT]s actuation		
Opening pressure threshold	Maximum	97.0 bar abs (including + 1.5 bar uncertainty)
Flow rate	Minimum	1150 t/h of saturated steam at 100 bar abs
Valve opening dead time	Maximum	1.5 s
Valve opening time	Maximum	0.5 s
MSSVs actuation		
Opening pressure threshold	Maximum	106.5 bar abs (including + 1.5 bar uncertainty)
Accumulation	Maximum	3%
Flow rate	Minimum	575 t/h of saturated steam at 100 bar abs
ASG [EFWS] actuation		
Opening level threshold	Minimum	0.38 (including -2% bar uncertainty)
Temperature	Nom	50°C
Flow rate	Minimum	25 kg/s
Normal Spray actuation		
Opening pressure threshold		Stuck open at the beginning of the transient
Flow rate	Maximum	35 kg/s per line

SUB-CHAPTER 16.5 - TABLE 49

C1 - ATWS spray - 4: Sequence of events

Time (s)	Event
0.0	Steady State-Initial conditions
5.0	Pressuriser spurious spray- Beginning of the transient
111.0	Reactor trip signal
113.4	Turbine trip
125.0	VDA [MSRT] opening
131.0	ATWS signal
138.5	1^{st} PSV threshold reached
146.0	RBS [EBS] injection
146.0	MSSV threshold reached
225.0	Reactor coolant pump trip signal
240.5	ASG [EFWS] actuation
338.0	100% pressuriser level reached

SUB-CHAPTER 16.5 - TABLE 50

C1 - ATWS spray w/o RPR [PS] - 1: Initial conditions

Parameters	Unit	Nominal value	Uncertainty	Used value
Primary side				
Core power	MWth	4500	-	4590
Core power level	% NP	100%	+2%	102%
Boron concentration	ppm	-	-	1610
Primary average temperature	°C	312.7	+2.5	315.2
Pressuriser pressure	bar abs	155.0	-2.5	152.5
Pressuriser level	% NR	56%		51%
Primary flow rate	m^3/h	27,185	-	27,185
Secondary side				
Steam generator outlet pressure	Bar abs	78.0	-	80.37
Steam generator level	% NR	49%	-5%	44%
ARE temperature	°C	230	-	230

SUB-CHAPTER 16.5 - TABLE 51

C1 - ATWS spray w/o RPR [PS] - 2: Protection systems and boundary conditions

Signal	Setpoint	Time delays
Low Hot Leg pressure	117.5 bar abs (including -2.5 bar uncertainties)	Delay of 0.9 s between pressure threshold crossing and reactor trip on low hot leg pressure signal. Then, delay of 0.4 s between reactor trip on low hot leg pressure signal and rods drop.
Turbine isolation	2.4 s	
ARE isolation	/	Conservative approach, ARE isolation right after the signal of RT

SUB-CHAPTER 16.5 - TABLE 52

C1 - ATWS spray w/o RPR [PS] - 3: Systems actuation

PSVs actuation		
PARAMETER		**VALUE**
Opening pressure thresholds	Minimum	173.5 abs, 176.5 bar abs and 179.5 bar abs (including -1.5 bar uncertainty)
Valve opening dead time	Maximum	0.5 s
Valve opening time	Maximum	0.1 s
Flow rate	Minimum	300 t/h of saturated steam at 176 bar abs
MSSVs actuation		
Opening pressure threshold	Maximum	106.5 bar abs (including + 1.5 bar uncertainty)
Accumulation	Maximum	3%
Flow rate	Minimum	575 t/h of saturated steam at 100 bar abs
Normal Spray actuation		
Opening pressure threshold		Stuck open at the beginning of the transient
Flow rate	Maximum	35 kg/s

SUB-CHAPTER 16.5 - TABLE 53

C1 - ATWS spray w/o RPR [PS] - 4: Sequence of events

Time (s)	Event
0.0	Steady State-Initial conditions
5.0	Pressuriser spurious spray- Beginning of the transient
53.0	Minimum DNBR
188.1	Reactor trip signal
188.2	MFW isolation
190.5	Turbine trip
560.0	MSSV threshold reached

SUB-CHAPTER 16.5 - TABLE 54

C1 - ATWS URBWP w/o TXS - 1: Initial conditions

Parameter	Fast withdrawal
RCP [RCS] flow rate	Thermal hydraulic (108,720 m^3/h)
Bypass	5.5%
Thermal Power	100% NP
Core Pressure	155 bar
Core Inlet temperature	295.6°C
Axial Offset	12%
RCCA positions	Limit of insertion
F_Q	2.88

SUB-CHAPTER 16.5 - TABLE 55

C1 - ATWS URBWP w/o TXS - 2: Bounding conditions for safety criteria verification

Parameter	Fast withdrawal
RCP [RCS] flow rate	Thermal hydraulic ($108{,}720\ m^3/h$)
Bypass	5.5%
Thermal Power	120% NP
Transient duration[3]	40 s
Start time of DNB	20 s
Core Pressure	165 bar
Core Inlet temperature	305°C
Axial Offset	30%
RCCA positions	ARO
F_Q	3.0
$F_{\Delta H}$	1.91

[3] As COMBAT calculation depends on the integral below the ramp of power and heat flux hot channel factor, a bounding transient has to take longer time to reach the final state than real transients.

SUB-CHAPTER 16.5 - TABLE 56

C1 - ATWS URBWP w/o TXS - 3: Results and safety criteria verification

	Fast withdrawal	Acceptance criteria verification
% of rods experiencing DNB	9.2%	< 10%
% of melted fuel in the hot spot	5.6%	< 10%
Maximum Clad Temperature reached	1206°C	< 1482°C

SUB-CHAPTER 16.5 - TABLE 57

C2 - 4 VIV [MSIV]s w/o PSVs - 1: Initial conditions

Parameters	Unit	Nominal value	Uncertainty	Used value
Core power	MWth	4500		4590
Core power level	% NP	100%	+2%	102%
Primary average temperature	°C	312.7	-2.5	310.2
Pressuriser pressure	Bar abs	155	-2.5	152.5
Pressuriser level	% NR	56%	+5%	61%
Primary flow rate	m^3/h	27,185		27,185
Steam generator outlet pressure	Bar abs	78.0		Code computation
Steam generator level	% NR	49%	-5%	44%
ARE [MFWS] temperature	°C	230	-	230

SUB-CHAPTER 16.5 - TABLE 58

C2 - 4 VIV [MSIV]s w/o PSVs - 2: Reactor trips

Signal	Setpoint	Time delays
High Pressuriser pressure	168 bar abs (including +1.5 bar uncertainties)	Delay of 0.9 s between pressure threshold crossing and reactor trip on high pressuriser pressure signal. Then, delay of 0.4 s between reactor trip on high pressuriser pressure signal and rods drop.
High SG pressure	97 bar abs (including +1.5 bar uncertainties)	Delay of 0.9 s between pressure threshold crossing and reactor trip on high SG pressure signal. Then, delay of 0.4 s between reactor trip on high SG pressure signal and rods drop.
ARE [MFWS] isolation	/	Conservative approach, ARE [MFWS] isolation on the initiating event.

SUB-CHAPTER 16.5 - TABLE 59

C2 - 4 VIV [MSIV]s w/o PSVs - 3: F1 Systems actuation

PARAMETER		VALUE
PSVs actuation		
Opening pressure thresholds	Non operational	176.5 bar abs, 179.5 bar abs and 182.5 bar abs (including + 1.5 bar uncertainty)not operational
Valve opening dead time	Non operational	0.5 s
Valve opening time	Non operational	0.1 s
Flow rate	Non operational	300 t/h of saturated steam at 176 bar abs
VDA [MSRT]s actuation		
Opening pressure threshold	Maximum	97.0 bar abs (including + 1.5 bar uncertainty)
Flow rate	Minimum	1150 t/h of saturated steam at 100 bar abs
Valve opening dead time	Maximum	1.5 s
Valve opening time	Maximum	0.5 s
MSSVs actuation		
Opening pressure threshold	Maximum	106.5 bar abs (including + 1.5 bar uncertainty)
Accumulation	Maximum	3%
Flow rate	Minimum	575 t/h of saturated steam at 100 bar abs
Normal Spray actuation (when operating)		
Opening pressure threshold	Maximum	160 bar abs
Opening time	Maximum	2 s
Flow rate	Minimum	23 kg/s

SUB-CHAPTER 16.5 - TABLE 60

C2 - 4 VIV [MSIV]s w/o PSVs - 4: Sequence of events with pressuriser normal spray

Time (s)	Event
0	Steady State-Initial conditions
5	Inadvertent closure of all VIV [MSIV]s - Beginning of the transient
5.1	Isolation of the ARE
12.0	‘’High Pressuriser Pressure’’ threshold reached
13.3	Beginning of the rods drop
13.8	‘’High SG Pressure ’’ threshold reached
13.8	VDA [MSRT]s actuation threshold reached
16.3	Opening of the VDA [MSRT]s valves
18.6	Primary pressure peak
18.4	Pressuriser pressure peak

SUB-CHAPTER 16.5 - TABLE 61

C2 - 4 VIV [MSIV]s w/o PSVs - 5: Sequence of events without pressuriser normal spray

Time (s)	Event
0	Steady State-Initial conditions
5	Inadvertent closure of all VIVs [MSIV]s - Beginning of the transient
5.1	Isolation of the ARE [MFWS]
11.4	‘’High Pressuriser Pressure ’’ threshold reached
12.7	Beginning of the rods drop
13.8	‘’ High SG Pressure’’ threshold reached
13.8	VDAs [MSRT]s actuation threshold reached
16.3	Opening of the VDA [MSRT] valves
18.5	Primary Pressure peak
18.0	Pressuriser Pressure peak

SUB-CHAPTER 16.5 - TABLE 62

O1 - 4 VIVs [MSIV]s w/o VDAs [MSRT]s - 1: Initial conditions

Parameters	Unit	Nominal value	Uncertainty	Used value
Core power	MWth	4500		4590
Core power level	% NP	100%	+2%	102%
Boron concentration	ppm	0		0
Primary average temperature	°C	312.7	+2.5	315.2
Pressuriser pressure	Bar abs	155	-2.5	152.5
Pressuriser level	% NR	56%	-5%	51%
Primary flow rate	m^3/h	27,185		27,185
Steam generator outlet pressure	Bar abs	78.0	Code result	80.35
Steam generator level	% NR	49%	+5%	54%
ARE [MFWS] temperature	°C	230	-	230

SUB-CHAPTER 16.5 - TABLE 63

O1 - 4 VIVs [MSIV]s w/o VDAs [MSRT]s - 2: Reactor trips

Signal	Setpoint	Time delays
High pressuriser pressure	168 bar abs (including +1.5 bar uncertainties)	Delay of 0.9 s between pressure threshold crossing and reactor trip on high pressuriser pressure signal. Then, delay of 0.4 s between reactor trip on high pressuriser pressure signal and rods drop.
High SG pressure	97 bar abs (including +1.5 bar uncertainties)	Delay of 0.9 s between pressure threshold crossing and reactor trip on high SG pressure signal. Then, delay of 0.4 s between reactor trip on high SG pressure signal and rods drop.

SUB-CHAPTER 16.5 - TABLE 64

O1 - 4 VIVs [MSIV]s w/o VDAs [MSRT]s - 3: Systems actuation

PARAMETER		VALUE
PSVs actuation		
Opening pressure thresholds	Maximum	176.5 bar abs, 179.5 bar abs and 182.5 bar abs (including + 1.5 bar uncertainty)not operational
Valve opening dead time	Maximum	0.5 s
Valve opening time	Maximum	0.1 s
Flow rate		300 t/h of saturated steam at 176 bar abs
VDAs [MSRT]s actuation		
Opening pressure threshold	Non operational	97.0 bar abs (including + 1.5 bar uncertainty)
Flow rate	Non operational	1150 t/h of saturated steam at 100 bar abs
Valve opening dead time	Non operational	1.5
Valve opening time	Non operational	0.5
MSSVs actuation		
Opening pressure threshold	Maximum	106.5 bar abs (including + 1.5 bar uncertainty)
Accumulation	Maximum	3%
Flow rate	Minimum	575 t/h of saturated steam at 100 bar abs
Normal Spray actuation (when operating)		
Opening pressure threshold	Maximum	160 bar
Opening time	Maximum	2 s
Flow rate	Minimum	23 kg/s
Main Feedwater System		
Isolation time	Maximum	15 s (Valves closing delay)
Low load Flow rate	Maximum	30% of FL ARE flow rate
ARE [MFWS] FL isolation on “rt_checkback” signal		I&C delay: 0.9 s FL Valves closing delay: 15 s, on a step basis.

SUB-CHAPTER 16.5 - TABLE 65

O1 - 4 VIVs [MSIV]s w/o VDAs [MSRT]s - 4: Sequence of events with operational spray

Time (s)	Event
0	Steady State - Initial conditions
5	Inadvertent closure of all VIVs [MSIV]s - Beginning of the transient
9.4	‘’High SG Pressure’’ threshold reached
10.9	Beginning of the rods drop
13.9	MSSVs ’actuation threshold reached
14.9	1st PSV open
Not reached	2nd PSV open
Not reached	3rd PSV open
15.4	Pressuriser Pressure peak
18.3	SG Pressure peak

SUB-CHAPTER 16.5 - TABLE 66

O1 - 4 VIVs [MSIV]s w/o VDAs [MSRT]s - 5: Sequence of events with operational spray

Time (s)	Event
0	Steady State-Initial conditions
5	Inadvertent closure of all VIVs [MSIV]s - Beginning of the transient
9.4	''High SG Pressure'' threshold reached
10.8	Beginning of the rods drop
13.9	MSSVs actuation threshold reached
13.9	1st PSV open
Not reached	2nd PSV open
Not reached	3rd PSV open
15.3	Pressuriser Pressure peak
18.2	SG Pressure peak

SUB-CHAPTER 16.5 - TABLE 67

R2 – ATWS EISF w/o TXS - 1: Initial conditions

Parameter	Initial value
Initial state= 100%	
RCP [RCS] flow rate	T/H design flow rate
Power	102% NP (100% + 2%)
Pressure	152.5 abs bar (155 bar abs – 2.5 bar abs)
Average temperature	315.2°C (312.7°C + 2.5°C)
Pressuriser level	51% (56% - 5%)
SG pressure	result of the code calculation
ARE [MFWS] inlet temperature	230°C (nominal)
SG level	49% NR (nominal)

SUB-CHAPTER 16.5 - TABLE 68

R2 – ATWS EISF w/o TXS - 2: System characteristics

PARAMETER	VALUE
Diverse reactor trip / turbine trip signals on SG level	
Setpoint	40% GL – 5% conservatism
RT delay	1.5 s
Delay between RT and TT	2.2 s
ARE [MFWS] parameters	
ARE [MFWS] high load isolation	15 s
ARE [MFWS] low load isolation	15 s
MSSVs	
Setpoint	105 bar – 1.5 bar pessimised
Accumulation	3%
Capacity	635 te/h under 100 bar, 2 per SG (2 x 25% of full load flow rate/SG)
Pressuriser safety valves	
Setpoints	175 – 1.5 / 178 – 1.5 / 181 – 1.5 bar (conservative values)
Dead time	0.5 s
Opening time	0.1 s
Capacity	steam 290 te/h, under 176 bar
	liquid 450 te/h under 176 bar
Hysteresis (preliminary values)	28 – 28.5 – 29 bar

SUB-CHAPTER 16.5 - TABLE 69

R2 – ATWS EISF w/o TXS - 3: Sequence of events

Time (s)	Event
5	Increase of secondary steam flow
860	Diverse reactor trip on low SG level
862.2	Turbine trip

SUB-CHAPTER 16.5 - TABLE 70

H1 – SB LOCA w/o VDAs [MSRT]s at 102% FP - 1: Initial Conditions

Parameters	Units	Nominal values	Uncertainties	Values used
Primary side				
Core power	%FP	100	2	102
Core power	MW	4500	2%	4590
Primary flow rate	m^3/h/loop	TH: 27185	-	27185
Average temperature	°C	312.8	2.5	315.3
Primary pressure	bar abs	155	2.5	157.5
PZR level	%MR	56	+5	61
Secondary side				
SG level	%NR	49	-	49
MFWS Temperature	°C	230	-	230
ASG Temperature	°C	10/50	-	50

SUB-CHAPTER 16.5 - TABLE 71

H1 – SB LOCA w/o VDAs [MSRT]s at 102% FP- 2: System Characteristics

Parameter	Value
Reactor trip / turbine trip signals on PZR pressure < MIN2	
Setpoint	135 bar - 3 bar uncertainty
Delay (I&C delay + RT breakers opening)	1.2 s (0.9 + 0.3)
Primary depressurization system (PDS)	
Capacity	900 t/h of saturated steam under 176 bar
ASG [EFWS] actuation on SG level (wide range cold side) < MIN2	
Setpoint Delay (I&C delay + full flowrate) Capacity 4 ASG [EFWS] tanks water content	40% - 5% uncertainty 16.5 s (1.5 + 15) 90 te/h per SG at 97 bar 1680 te
MHSI	
Injection pressure Injection temperature	85 bar 70°C
Accumulators	
Initial injection pressure Volume temperature	45 bar 35m3 50°C

SUB-CHAPTER 16.5 - TABLE 72

H1 –SB LOCA w/o VDAs [MSRT]s at 102%FP – 3: Sequence of events

Time (s)	Event
0	Beginning of calculation (break initiation)
79	Reactor trip
79	Turbine trip
125	Pressuriser heaters switch off (pressuriser level < 12% MR)
216	RIS [SIS] signal, failure of the partial cooldown signal
682	RCP [RCS] pumps trip (ΔP over three pumps < 75%)
2508	Low loop level reached : Feed and Bleed actuation
2644	MHSI
2924	Accumulators injection
3230	LHSI injection
5000	End of transient

SUB-CHAPTER 16.5 - TABLE 73

R2 – ATWS ROD DROP w/o TXS –1: Initial conditions

Parameter	**Rod drop**	
Burn up	BLX	EOL
RCP flow rate Thermal hydraulic	108 720 m^3/h	108 720 m^3/h
By pass	5.5%	5.5%
Thermal Power	100% NP	100% NP
Core Pressure	155 bar	155 bar
Core Inlet temperature	295.0°C	295.0°C
Axial Offset	9.5%	7.5%
RCCA positions	Insertion limits	Insertion limits
DNBR	1.6	1.6

SUB-CHAPTER 16.5 - TABLE 74

R2 – ATWS ROD DROP w/o TXS - 2: Selection of Dropped Rods

Drop of 3 rods					
Conservative case	**Core burn up**	**rods**	**Δrho (pcm)**	**ΔFΔH (%)**	**T2[4]**
ΔFΔH	**BLX**	N15+C13+R05	287	19	0.78
ΔFΔH	**EOL**	D14+P14+D04	432	12	0.75

[4] T2 represents the second maximum of the flux seen by the Power Range Detector (PRD). It is minimised to increase the response of the average temperature control.

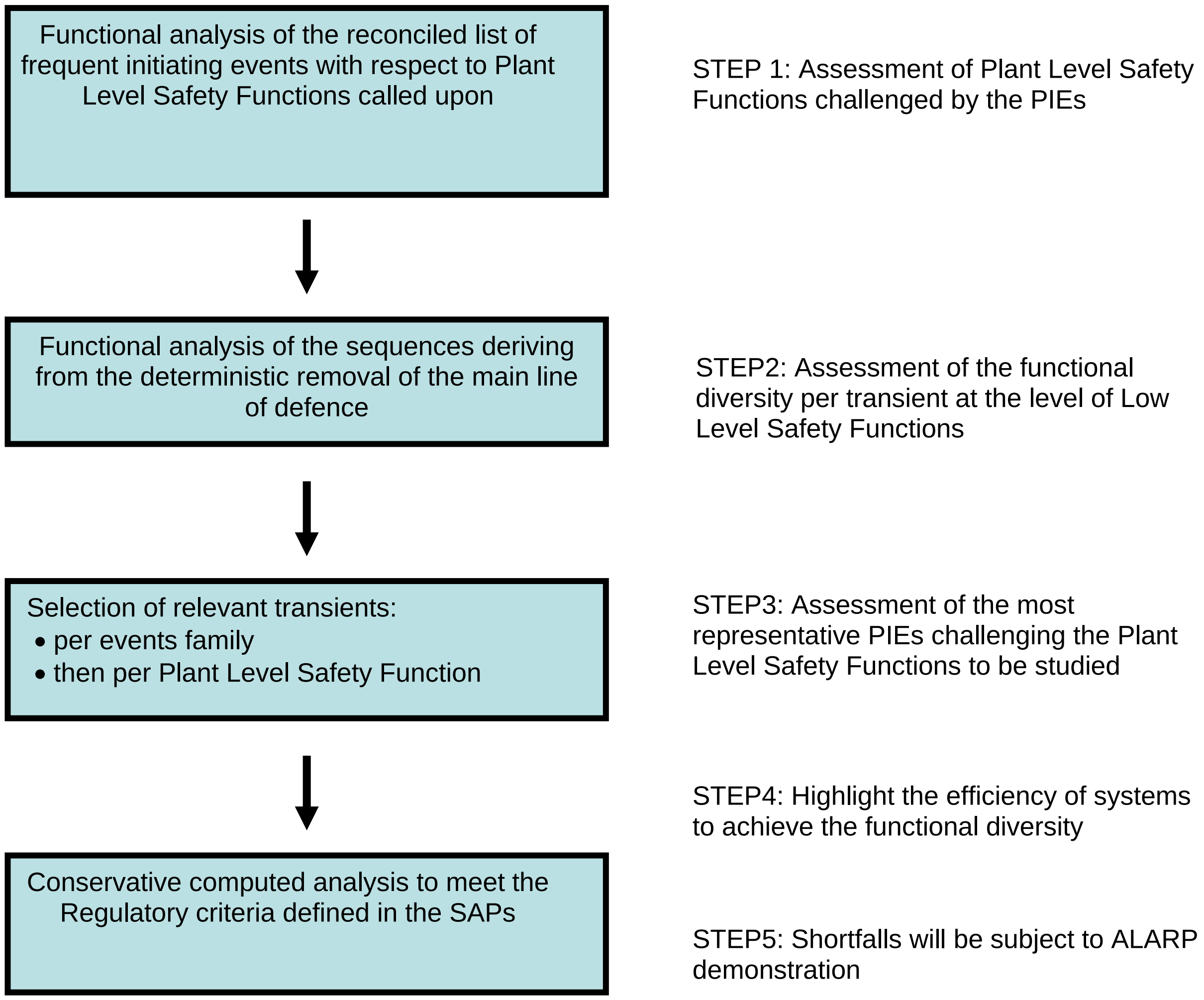
SUB-CHAPTER 16.5 - TABLE 75

Other safety functions - Diversity to safe shutdown state
1: Comparison of results for LOCA cases without MHSI or LHSI

LOCA (BREAK SIZE UP TO 20 CM^2) WITHOUT MHSI (STATE A)	LOCA (BREAK SIZE UP TO 20 CM^2) WITHOUT LHSI (STATE A)
PCSR Sub-chapter 16.1 – section 3.7	PCSR Sub-chapter 16.1 – section 3.8
Calculation at 4250 MW with BE parameters Argumentation at 4500 MW	Calculation at 4900 MW (in Appendix 16B) with BE parameters Argumentation at 4500 MW
Fuel cladding temperature 420°C (4250 MW) Estimated to be 435°C at 4500 MW	Fuel cladding temperatures remain at saturated conditions with no cladding rupture
Core uncovery (plenum level = 0) at 4250 MW	No core uncovery at 4900 MW

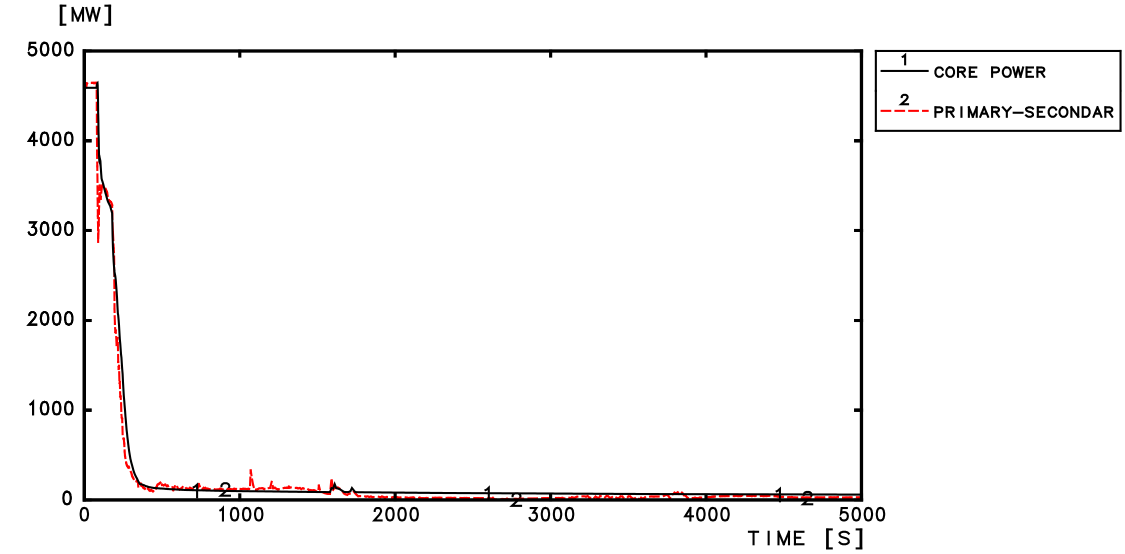
SUB-CHAPTER 16.5 – FIGURE 1

Functional Analysis Diagram

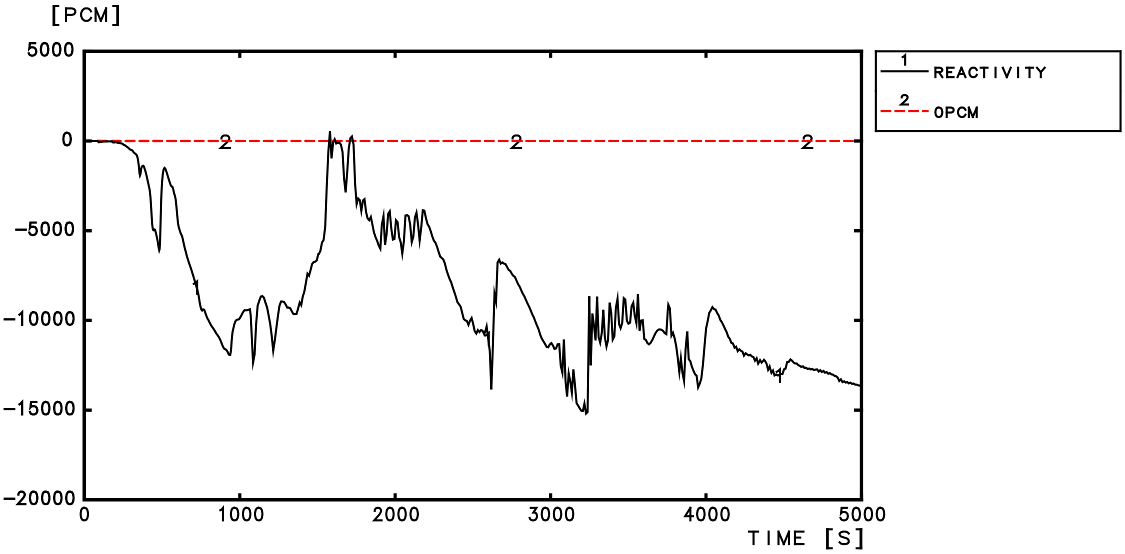


SUB-CHAPTER 16.5 – FIGURE 2

R2 - ATWS SB LOCA - 1: Core reactivity and power



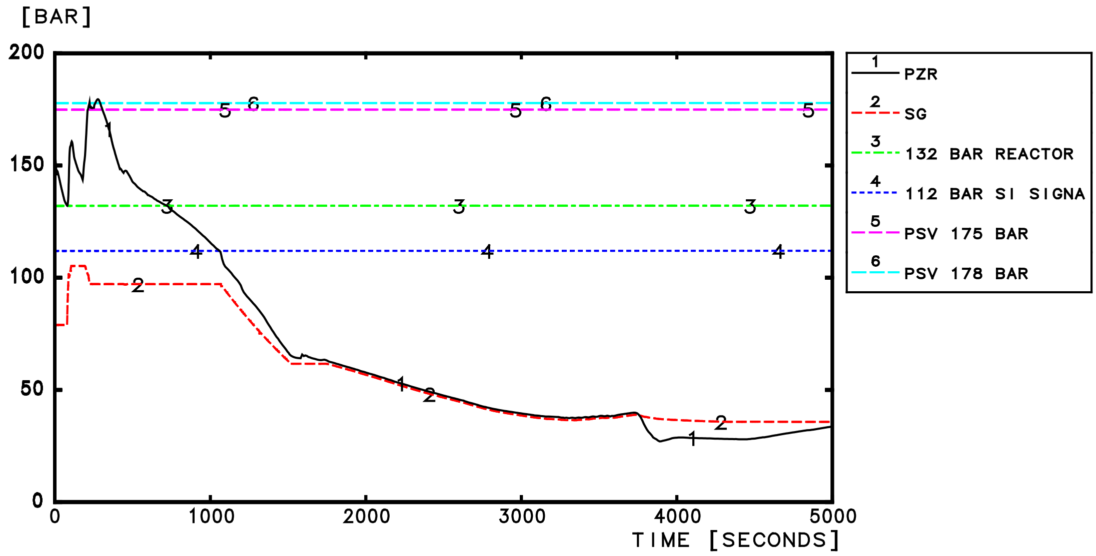
POWER



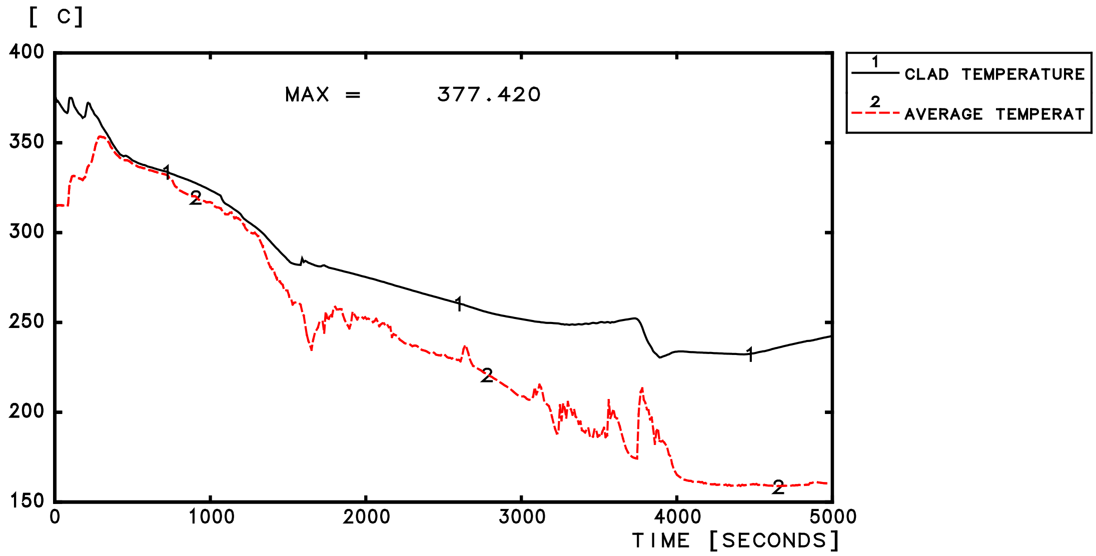
CORE REACTIVITY

SUB-CHAPTER 16.5 – FIGURE 3

R2 - ATWS SB LOCA - 2: Primary/secondary pressure and core temperatures



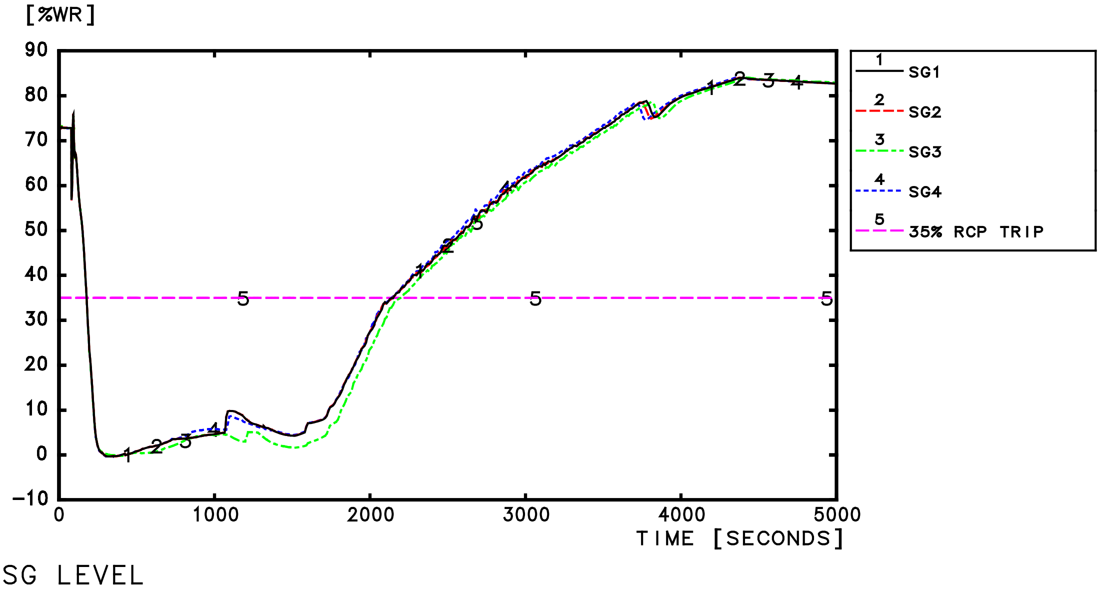
PRIMARY PRESSURE

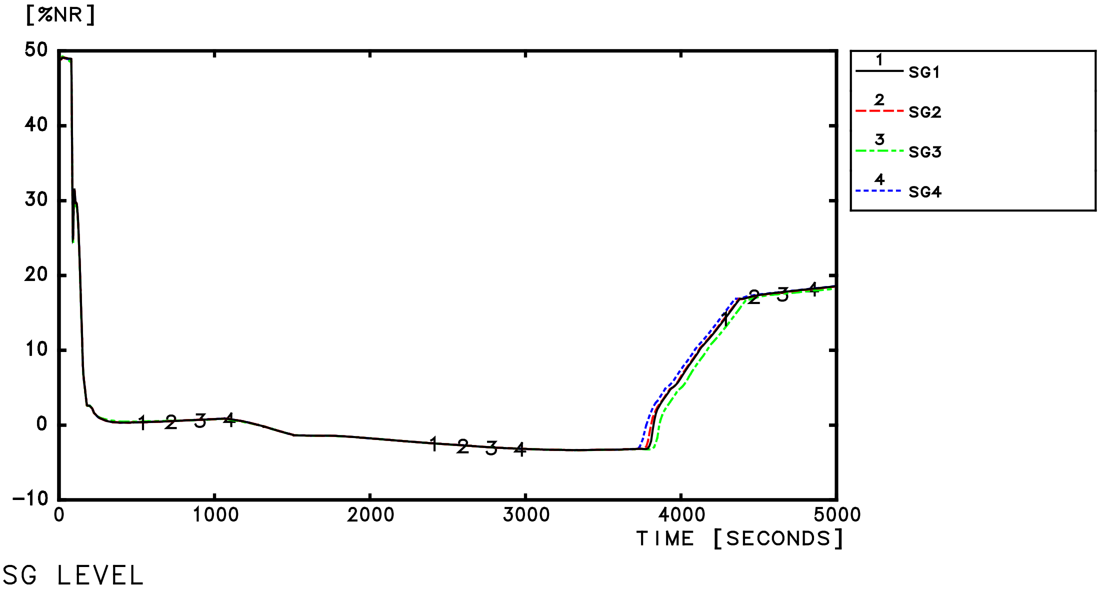


CORE TEMPERATURES

SUB-CHAPTER 16.5 – FIGURE 4

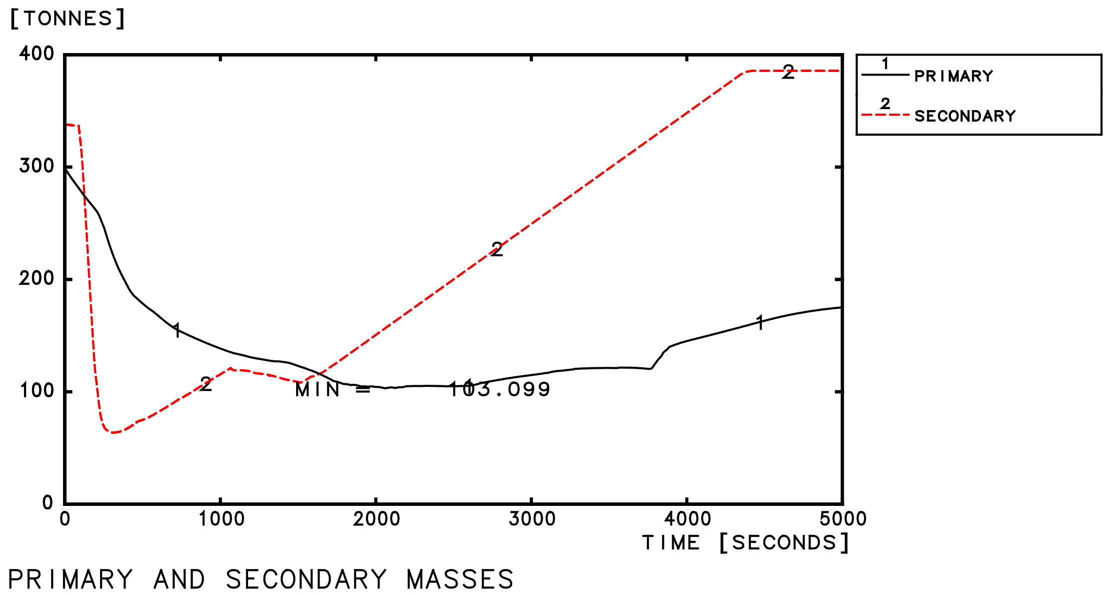
R2 - ATWS SB LOCA - 3: SG wide and narrow range levels

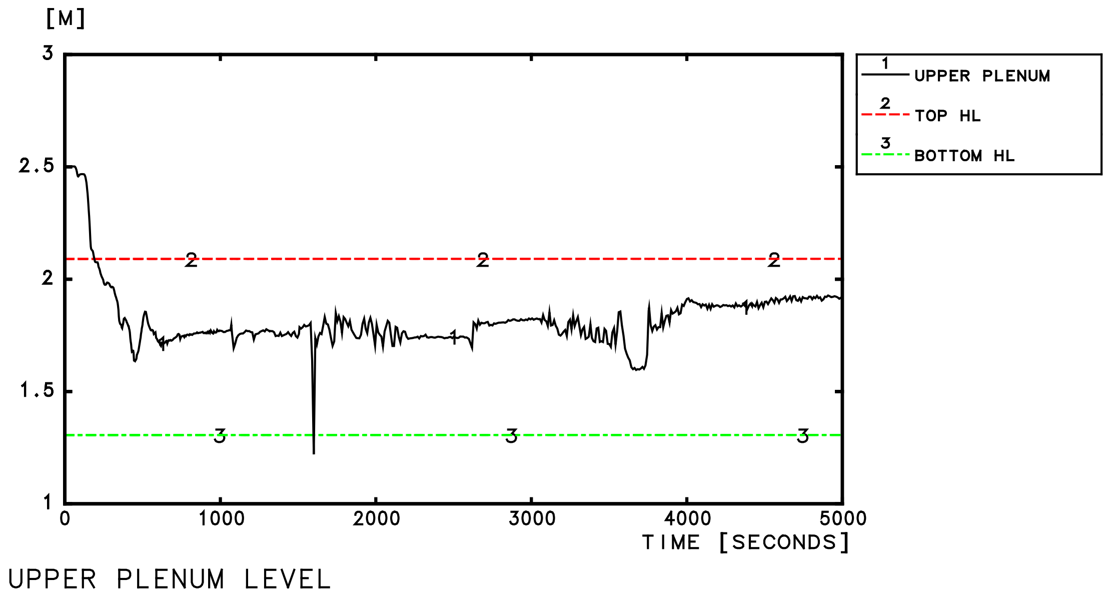




SUB-CHAPTER 16.5 – FIGURE 5

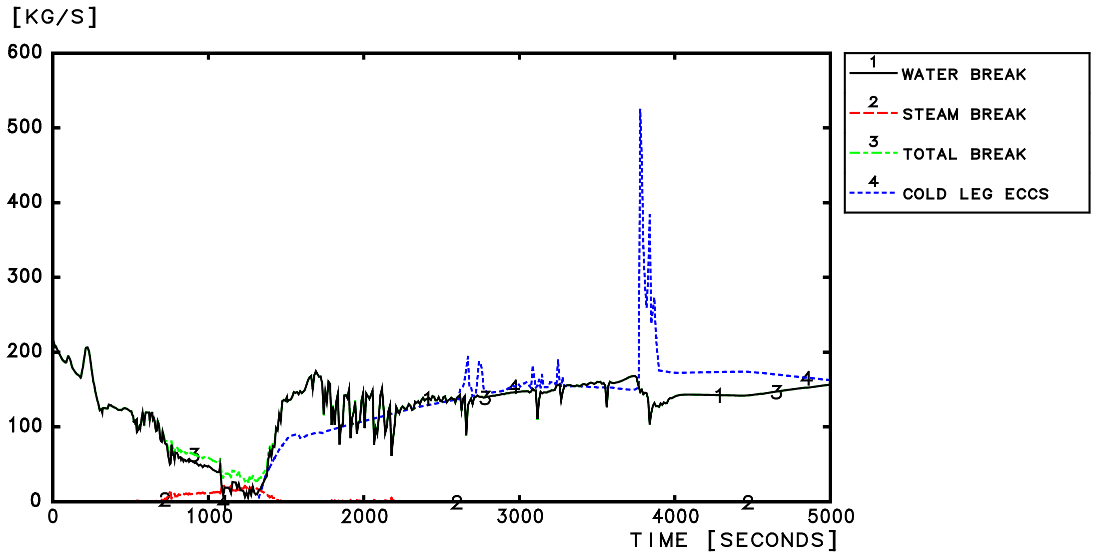
R2 - ATWS SB LOCA - 4: Primary and secondary masses and upper plenum liquid level



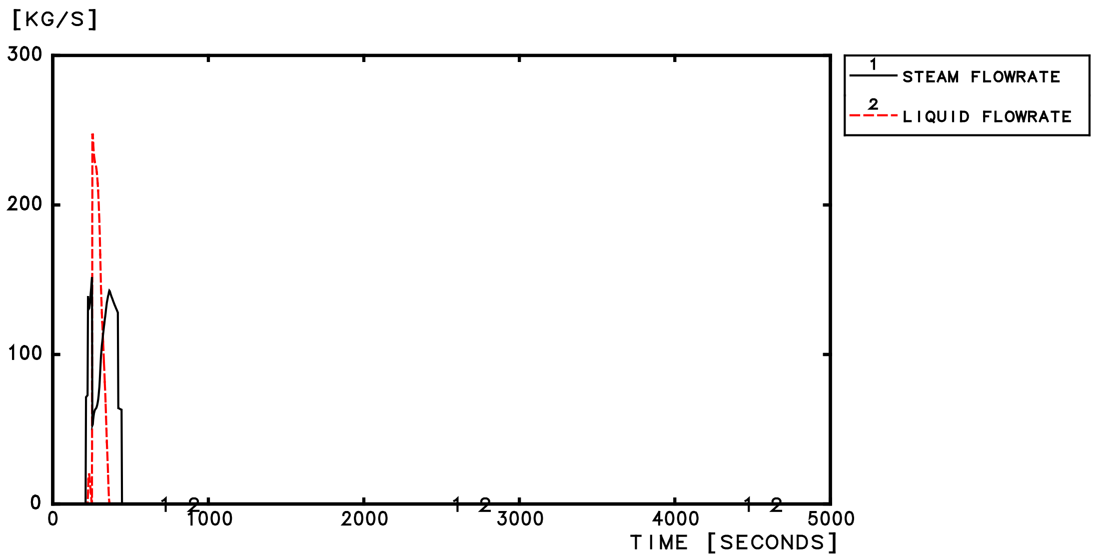


SUB-CHAPTER 16.5 – FIGURE 6

R2 - ATWS SB LOCA - 5: RIS [SIS] and break flow rates and PSV flow rates



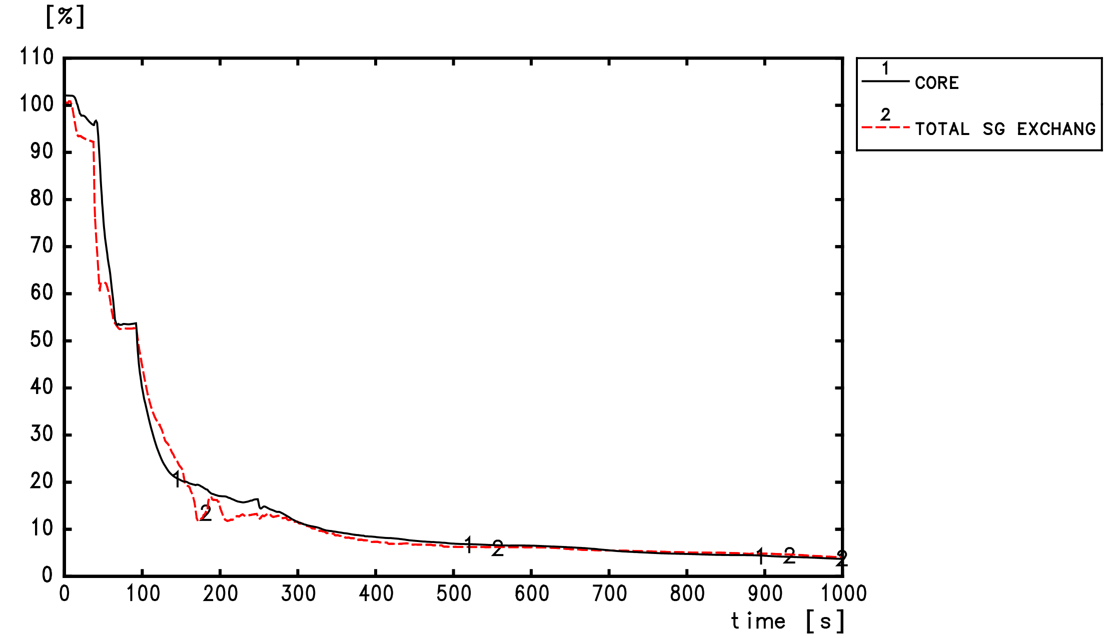
BREAK FLOWRATE AND TOTAL ECCS INJECTION RATE



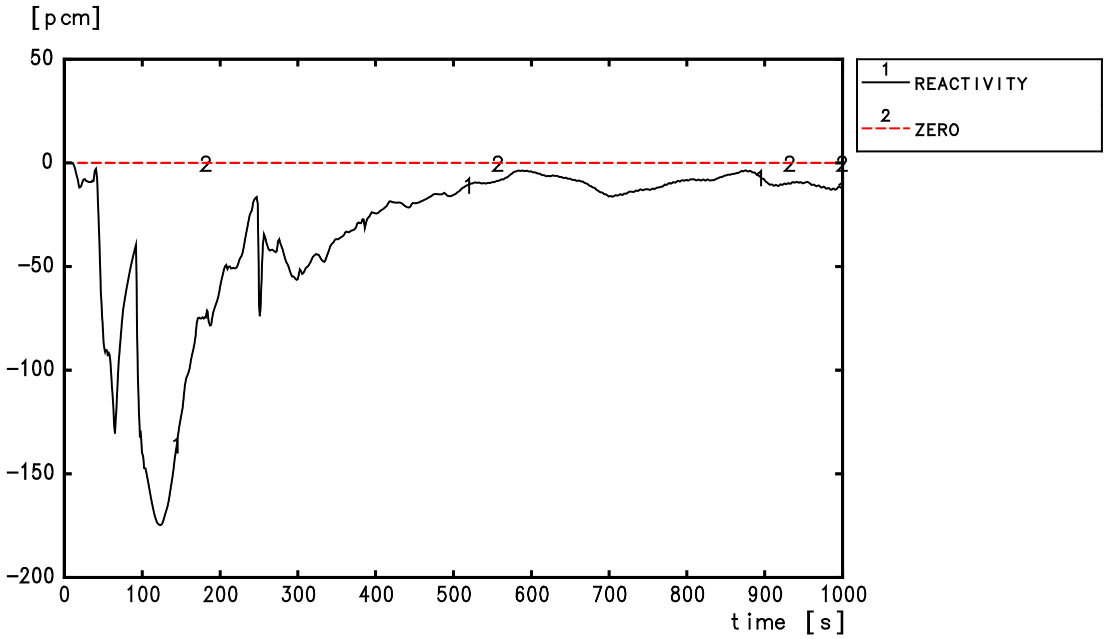
PSV FLOWRATE

SUB-CHAPTER 16.5 – FIGURE 7

R2 - ATWS LOFW - 1: Core power and reactivity



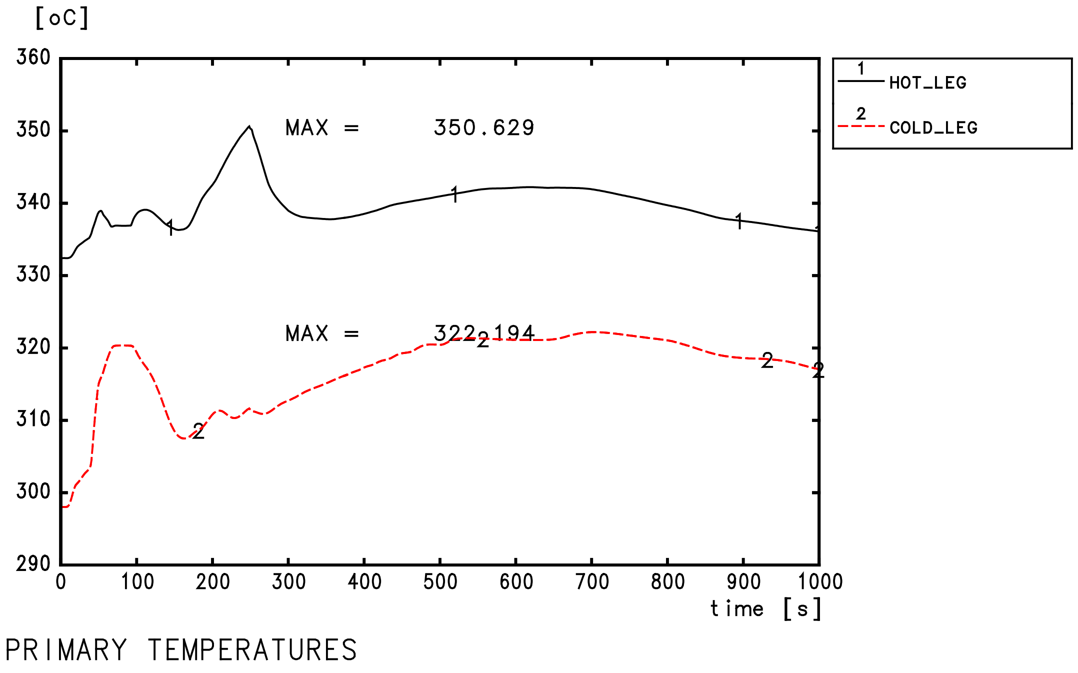
CORE POWER

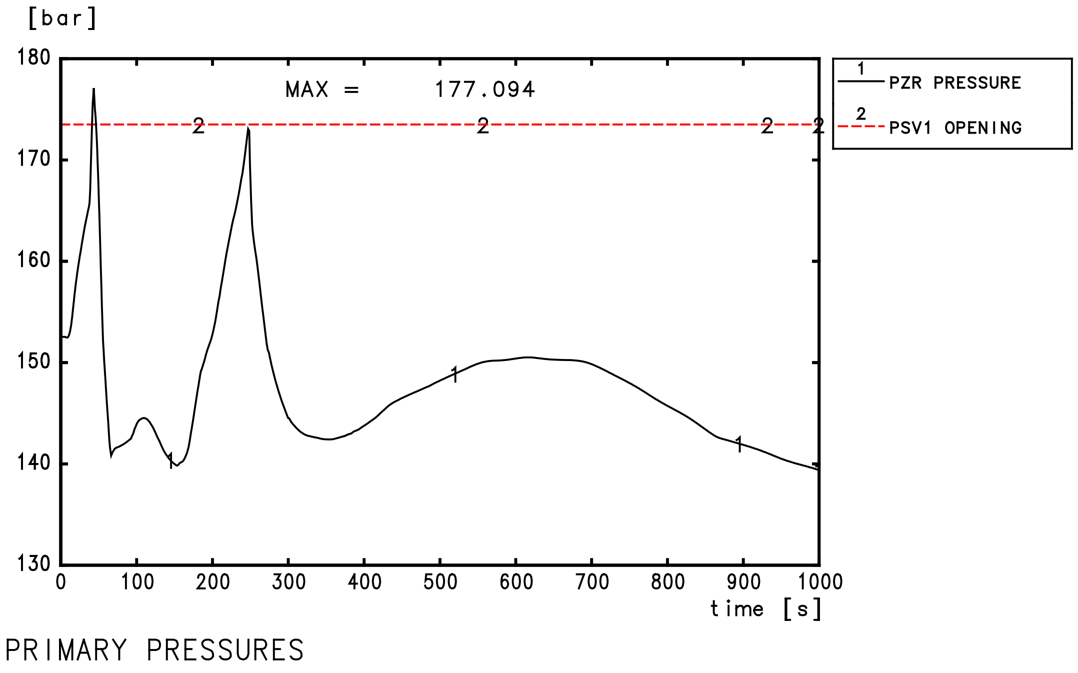


REACTIVITY

SUB-CHAPTER 16.5 – FIGURE 8

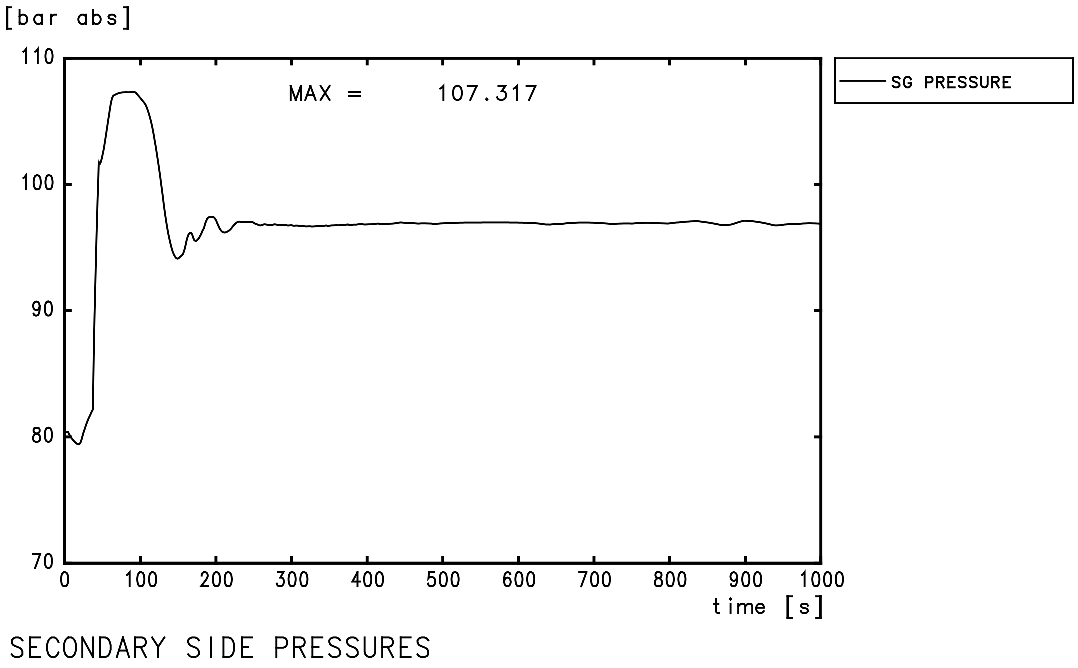
R2 - ATWS LOFW - 2: RCP [RCS] temperatures and pressuriser pressure

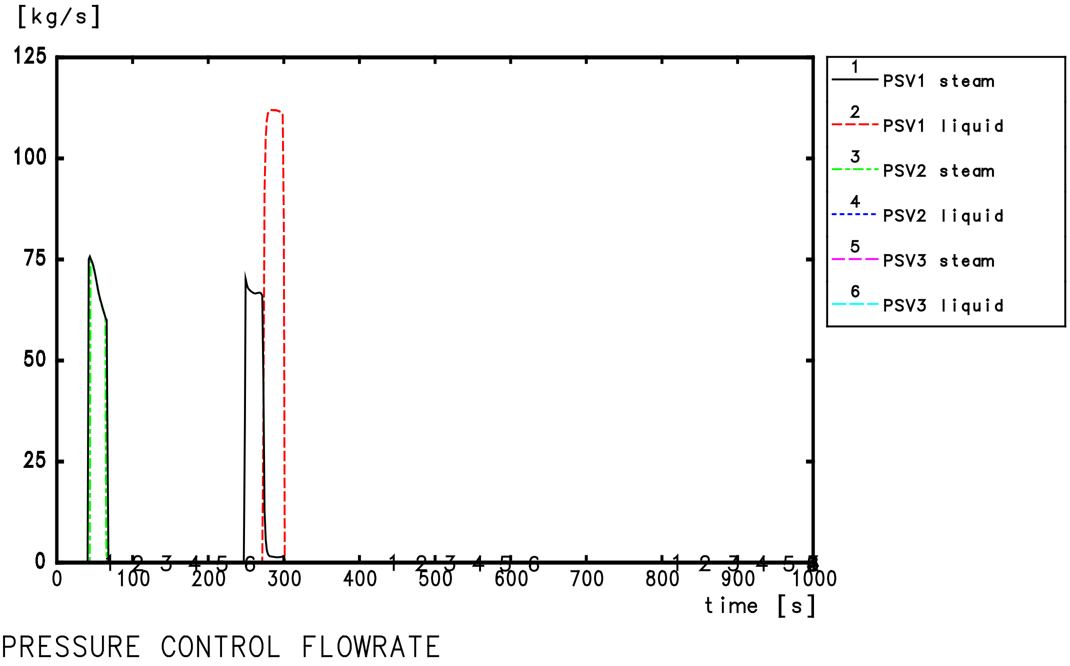




SUB-CHAPTER 16.5 – FIGURE 9

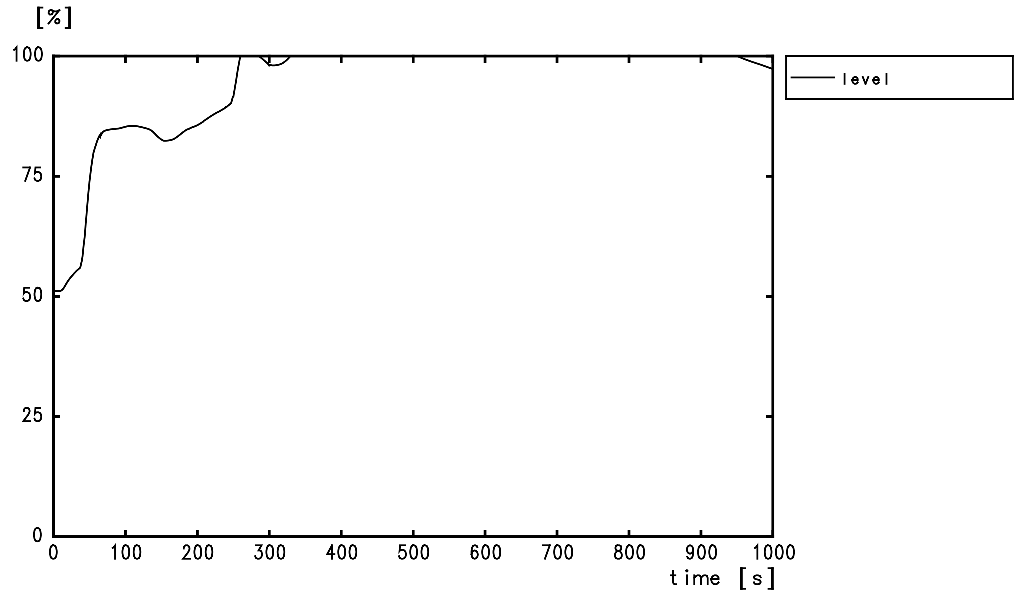
R2 - ATWS LOFW - 3: Secondary side pressure and PSV flow rates



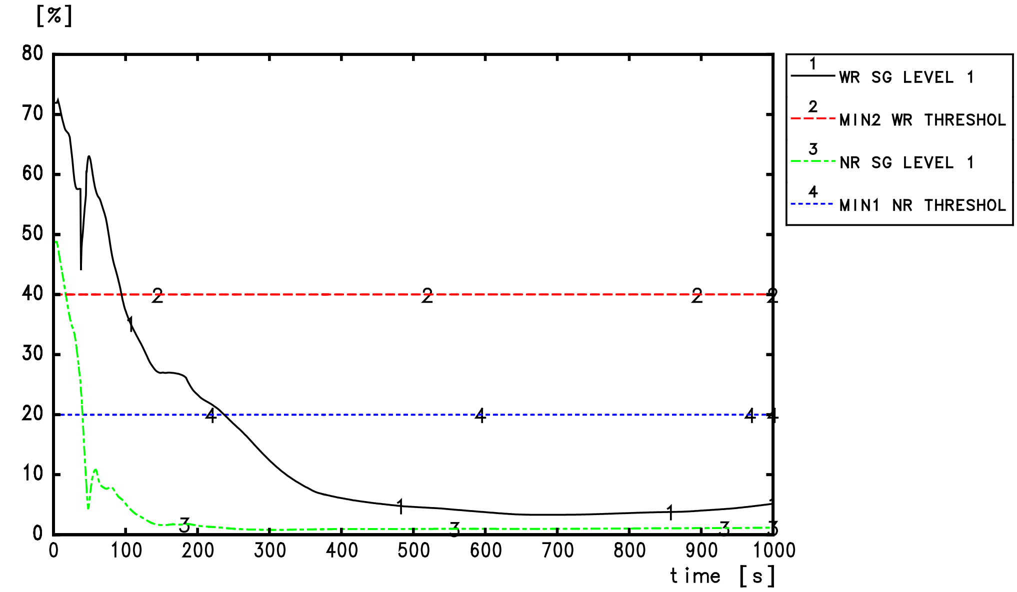


SUB-CHAPTER 16.5 – FIGURE 10

R2 - ATWS LOFW - 4: Pressuriser and SG levels



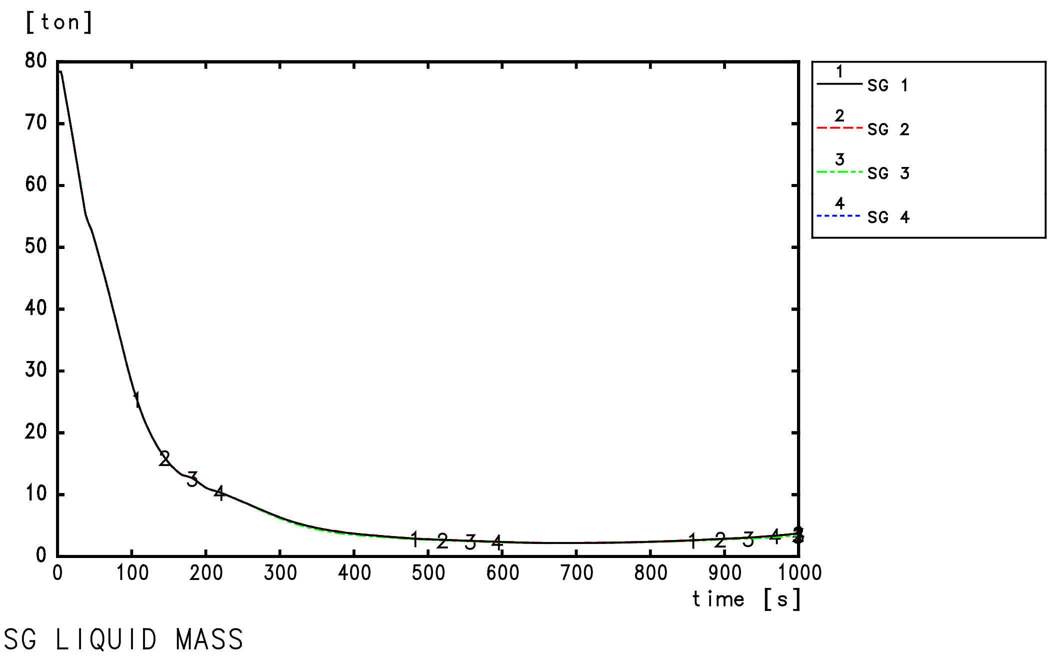
PZR LEVEL

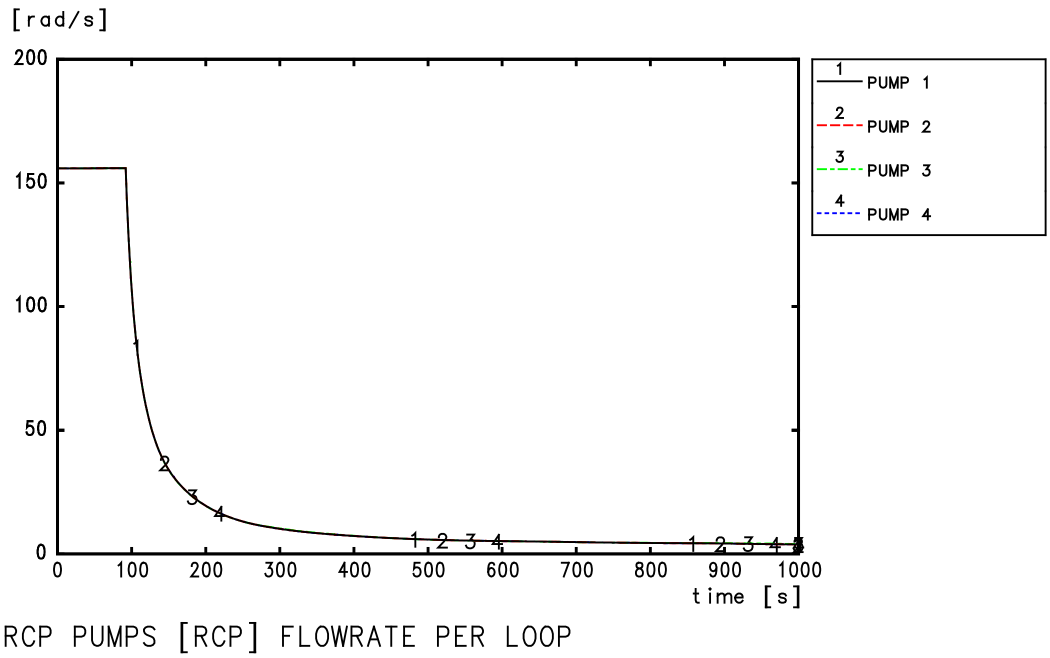


WIDE RANGE AND NARROW RANGE SG LEVEL

SUB-CHAPTER 16.5 – FIGURE 11

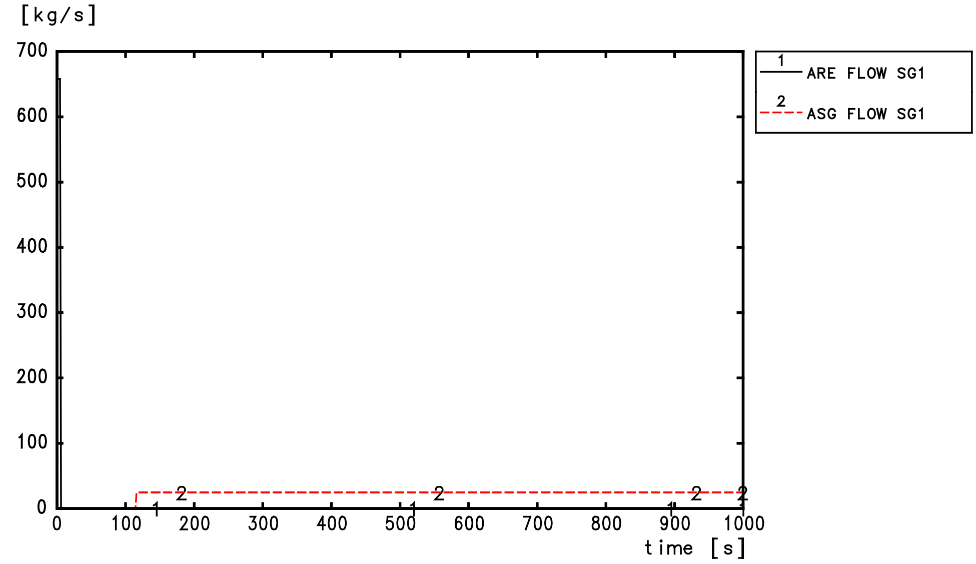
R2 - ATWS LOFW - 5: SG liquid mass and reactor coolant pump speed



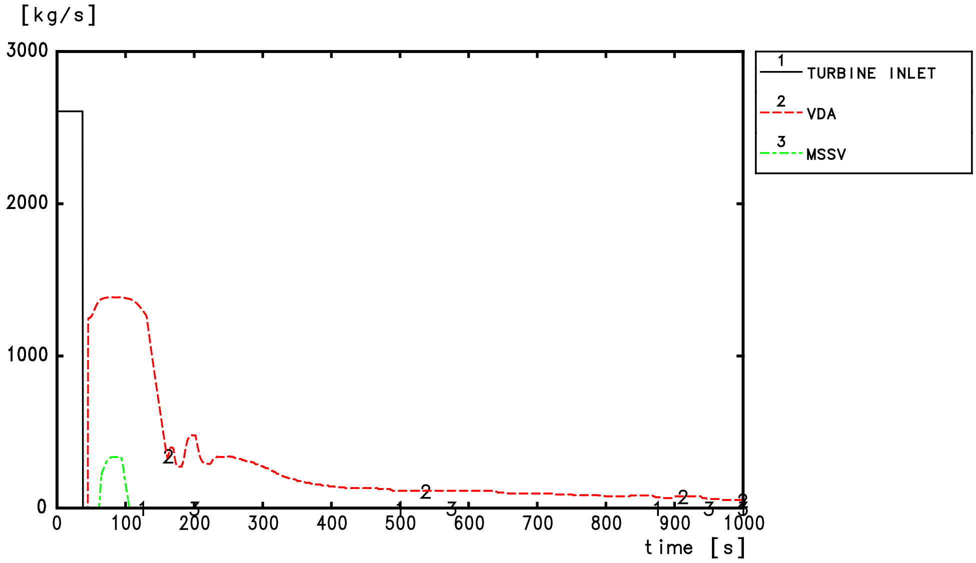


SUB-CHAPTER 16.5 – FIGURE 12

R2 - ATWS LOFW - 6: SG Feedwater and secondary side steam flow rate



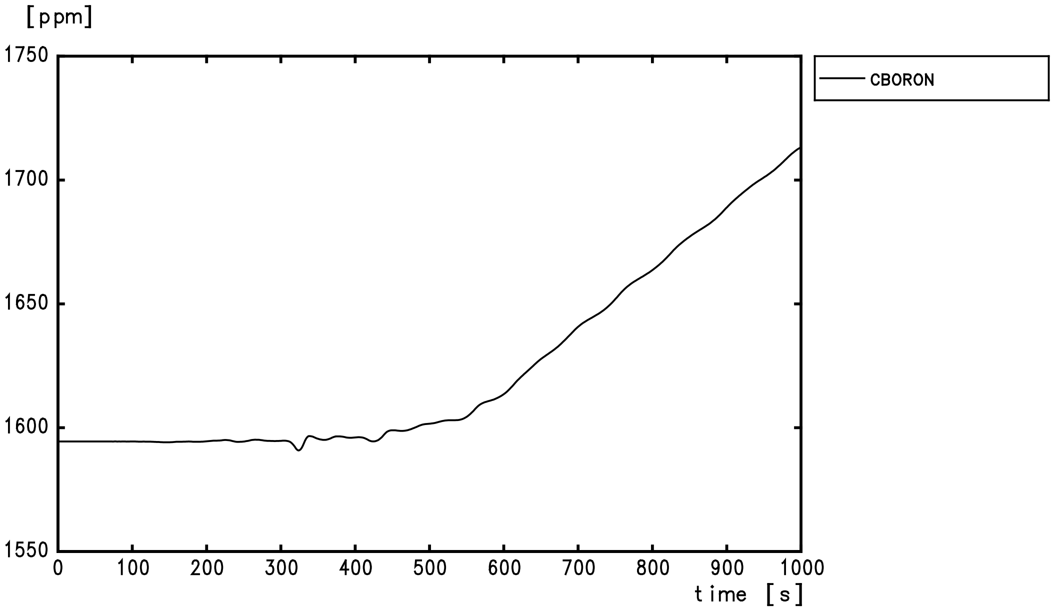
MAIN FEEDWATER FLOWRATE PER LOOP



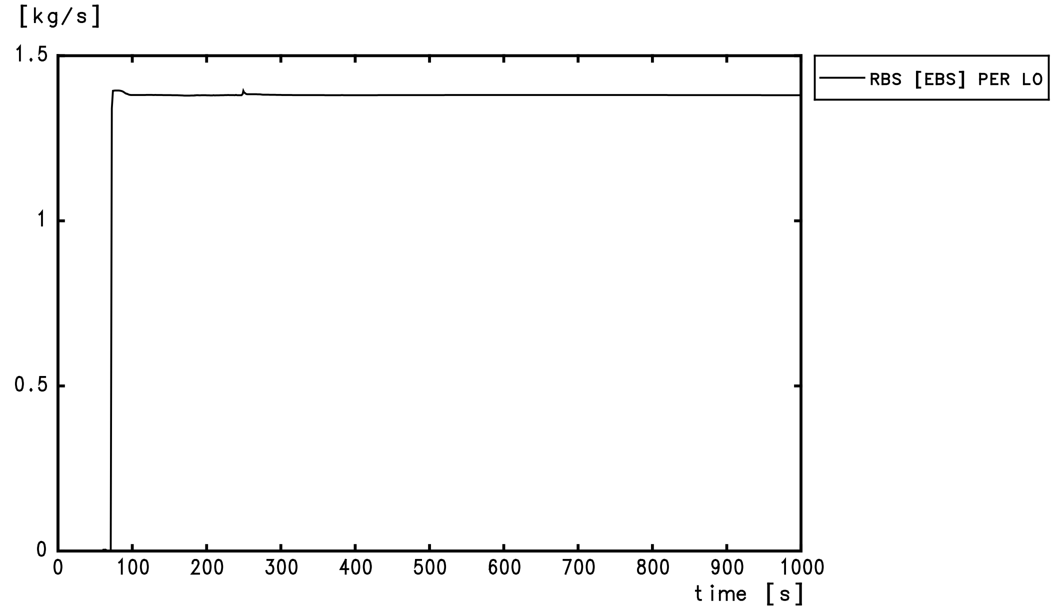
STEAM FLOWRATE

SUB-CHAPTER 16.5 – FIGURE 13

R2 - ATWS LOFW - 7: Core boron concentration and RBS [EBS] flow rate



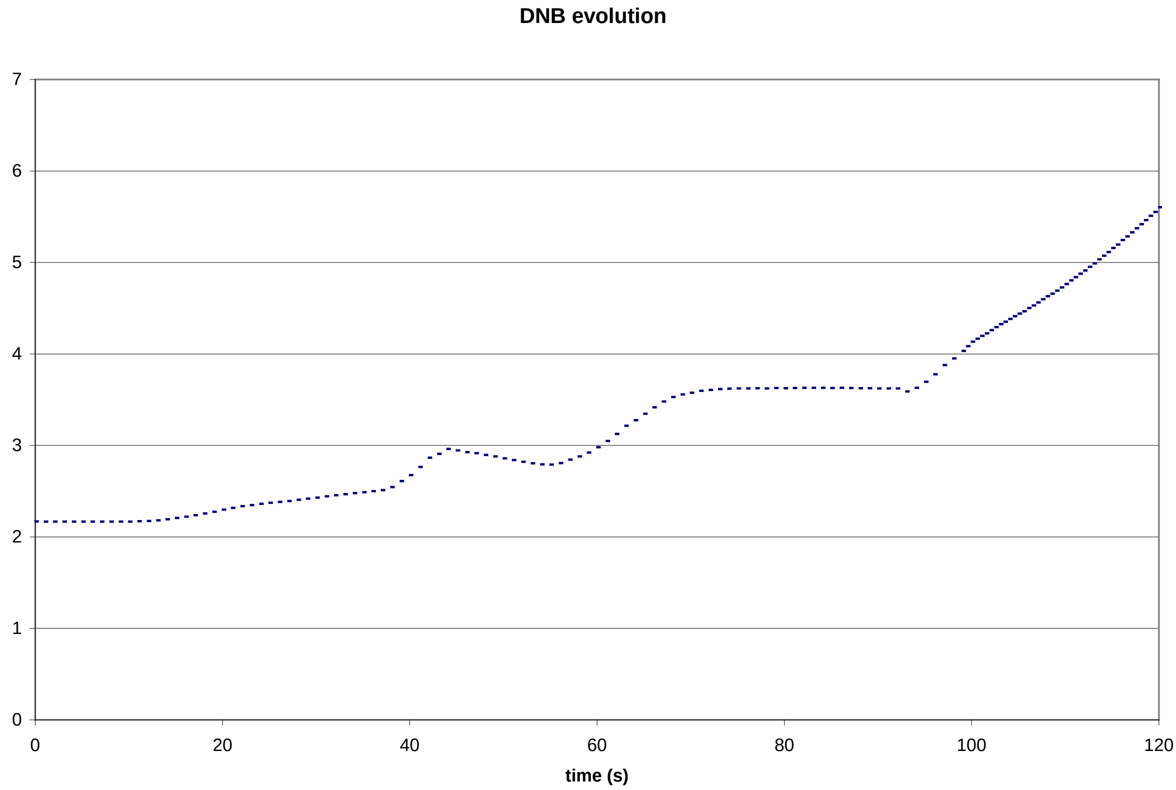
BORON CONCENTRATION



RBS [EBS] FLOWRATE

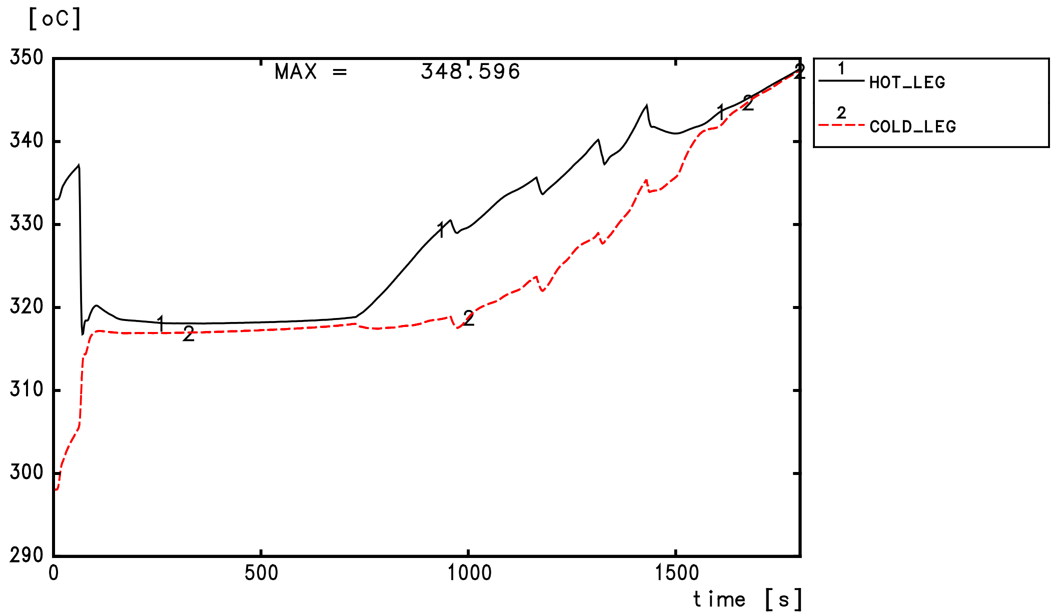
SUB-CHAPTER 16.5 – FIGURE 14

R2 - ATWS LOFW - 8: DNBR transient

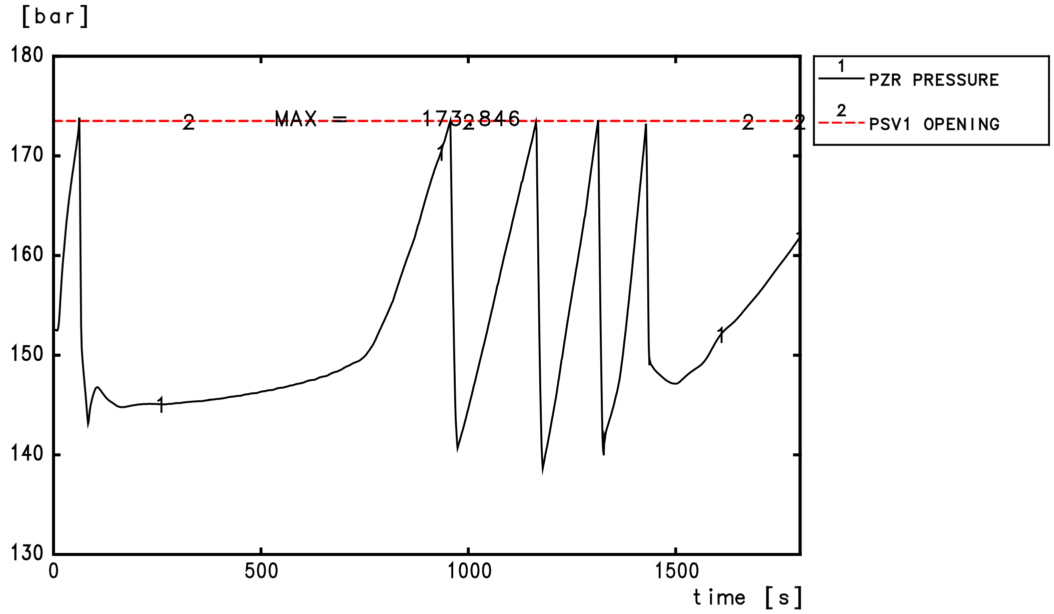


SUB-CHAPTER 16.5 – FIGURE 15

R2 - ATWS LOFW w/o RPR [PS] - 1: RCP [RCS] temperature and pressuriser pressure



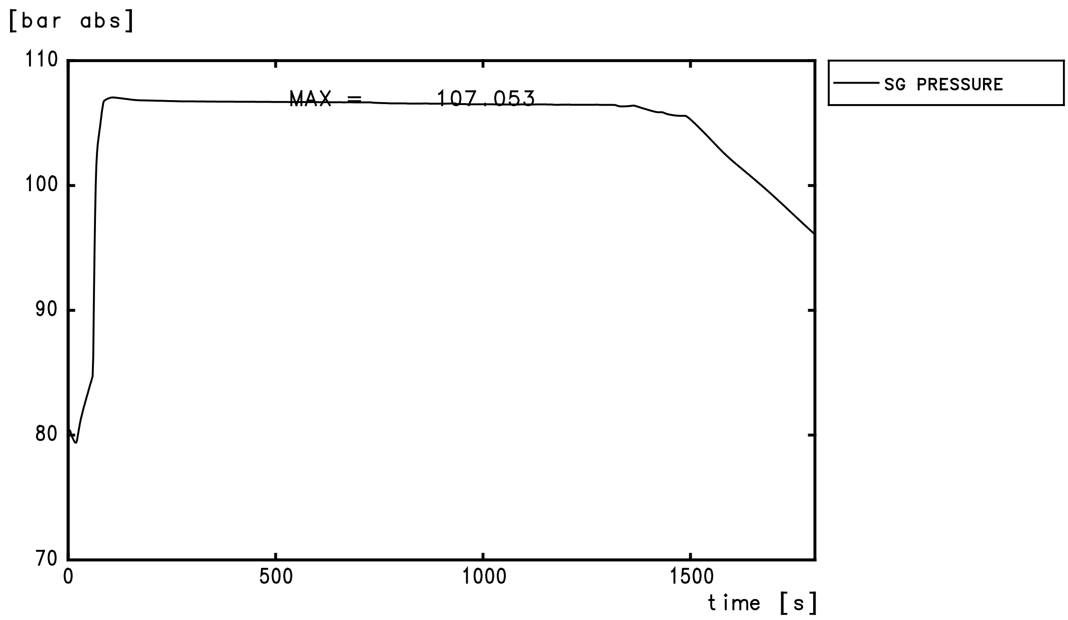
PRIMARY TEMPERATURES



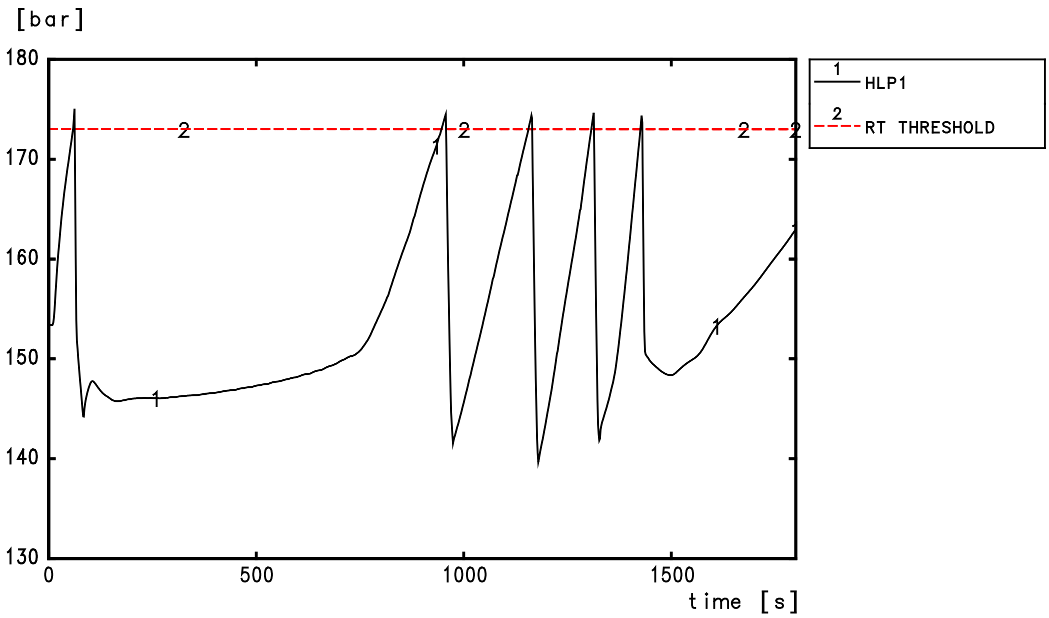
PRIMARY PRESSURE

SUB-CHAPTER 16.5 – FIGURE 16

R2 - ATWS LOFW w/o RPR [PS] - 2: Hot leg pressure and secondary side pressure



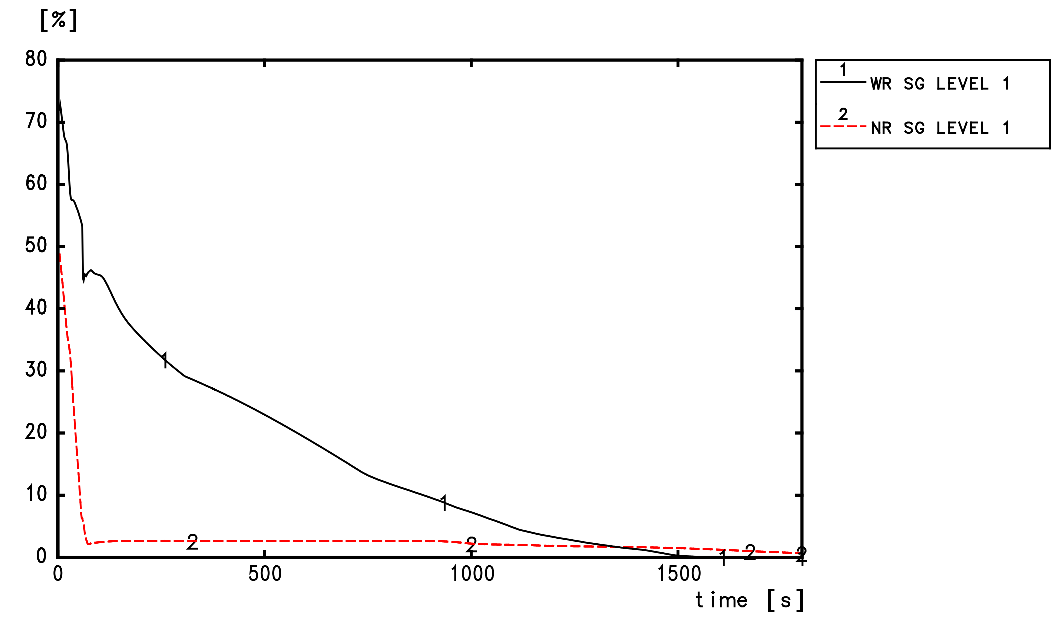
SECONDARY SIDE PRESSURES



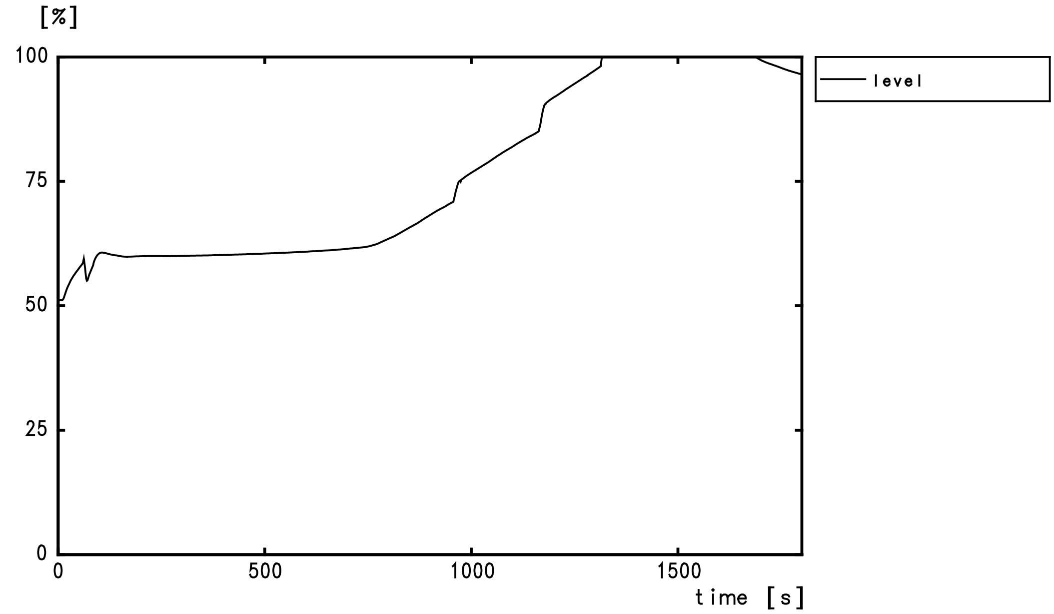
HL PRESSURE EVOLUTION

SUB-CHAPTER 16.5 – FIGURE 17

R2 - ATWS LOFW w/o RPR [PS] - 3: SG and pressuriser levels



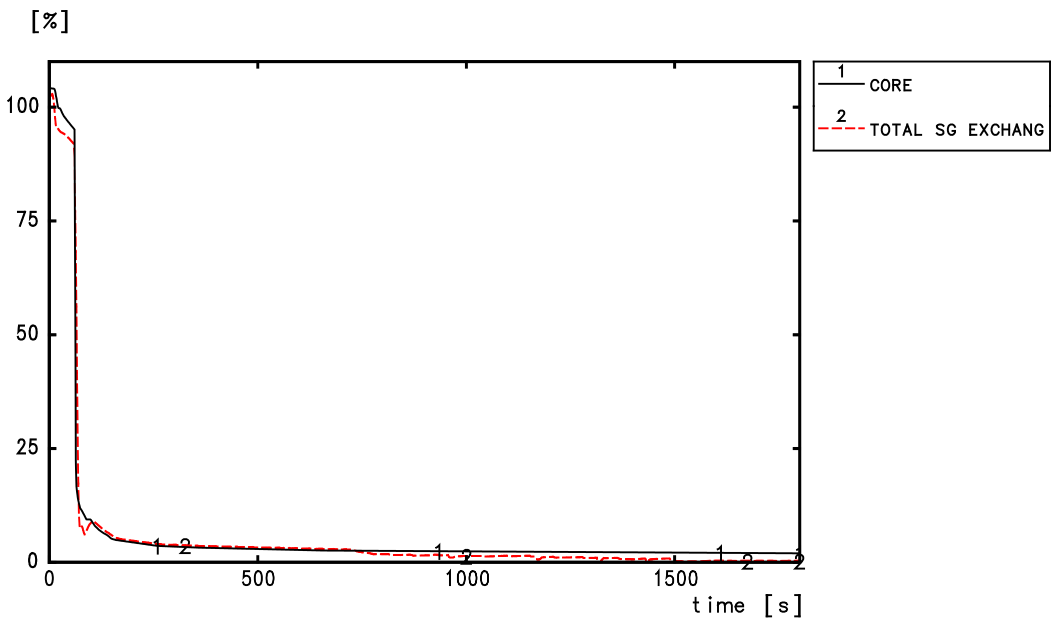
WIDE RANGE AND NARROW RANGE SG LEVEL



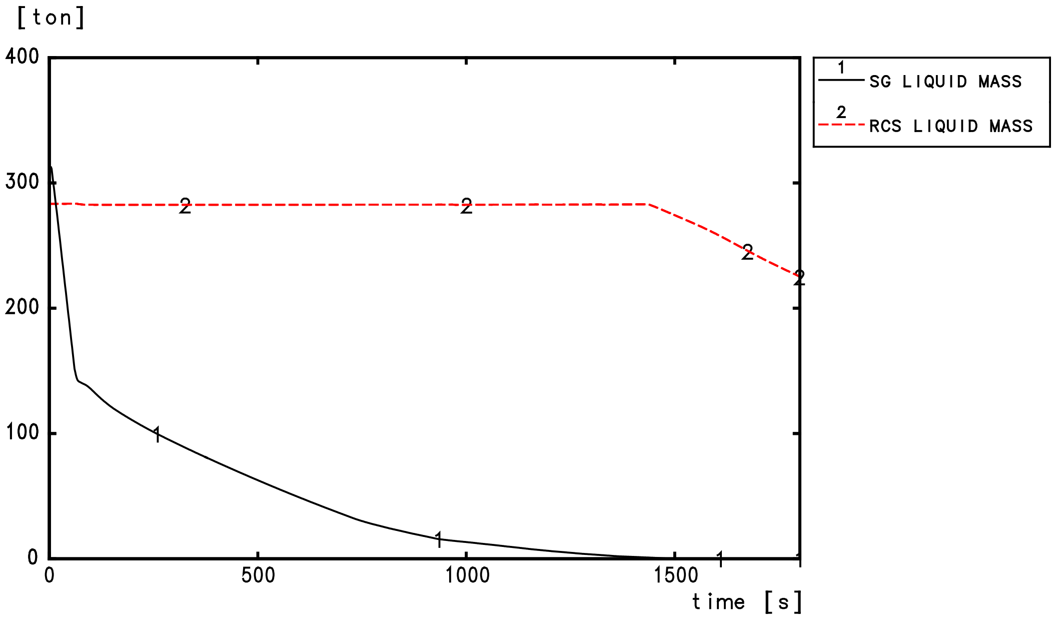
PZR LEVEL

SUB-CHAPTER 16.5 – FIGURE 18

R2 - ATWS LOFW w/o RPR [PS] - 4:
Core and SG exchanged powers and RCP [RCS] and SG liquid masses



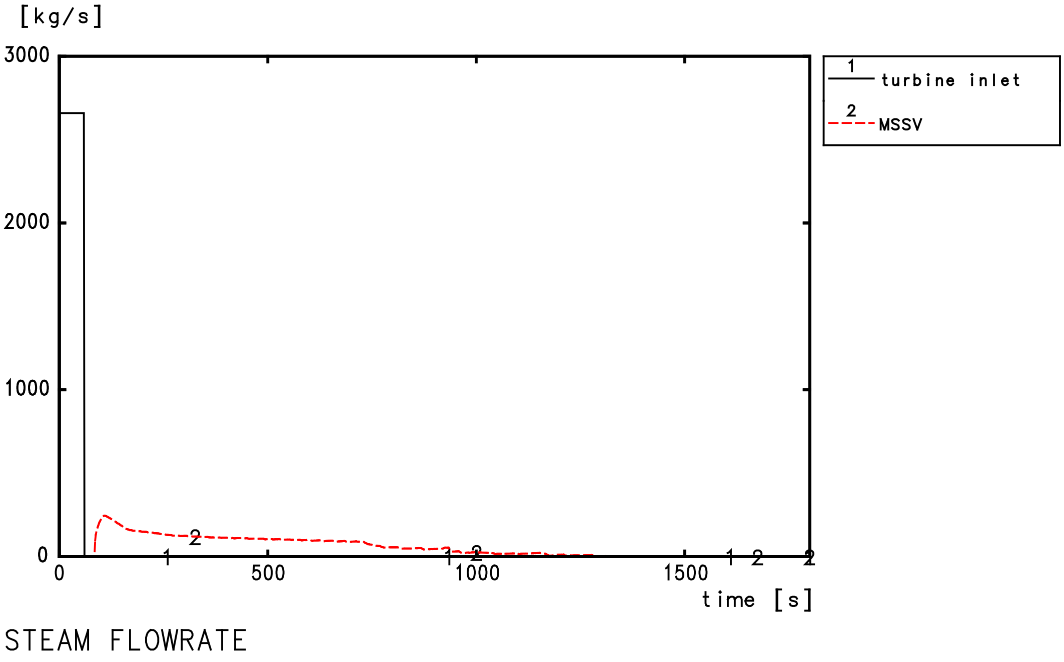
CORE POWER

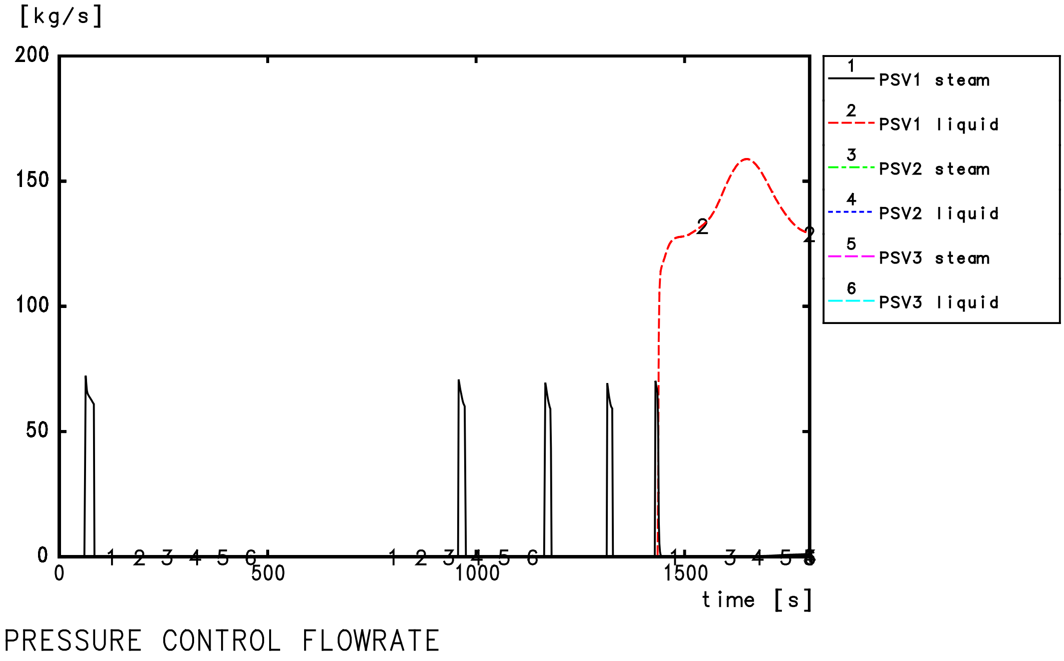


PRIMARY AND SECONDARY LIQUID MASS

SUB-CHAPTER 16.5 – FIGURE 19

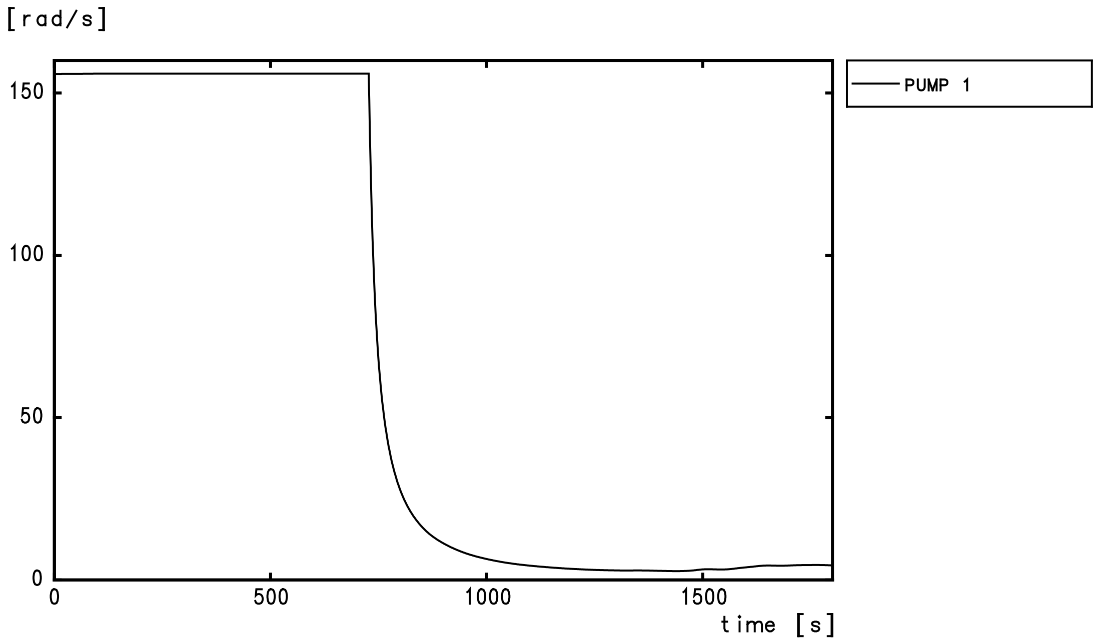
R2 - ATWS LOFW w/o RPR [PS] - 5: Secondary steam and PSV flow rates





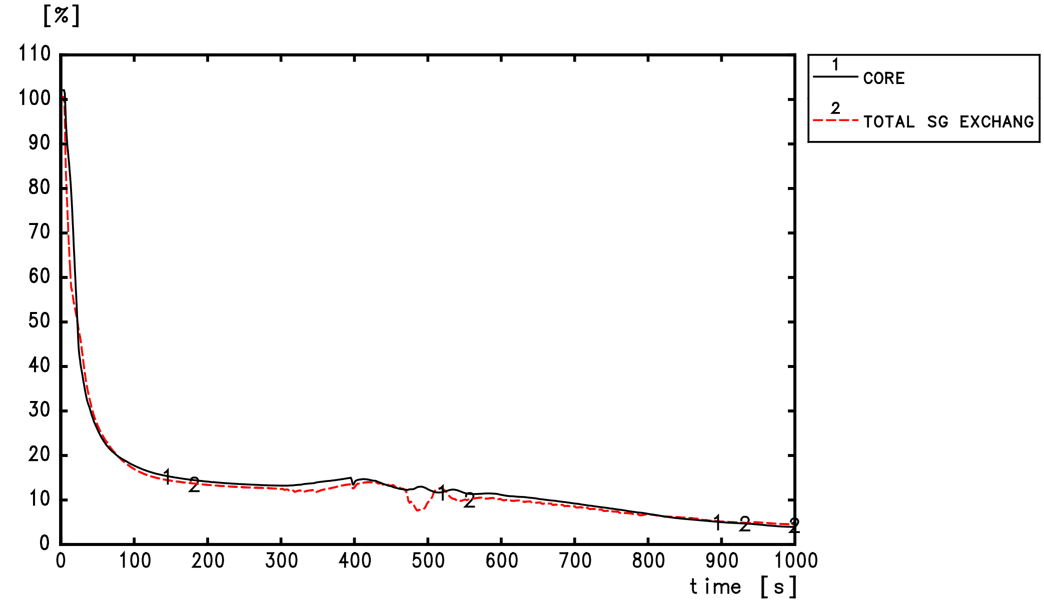
SUB-CHAPTER 16.5 – FIGURE 20

R2 - ATWS LOFW w/o RPR [PS] - 6: Reactor coolant pump speed

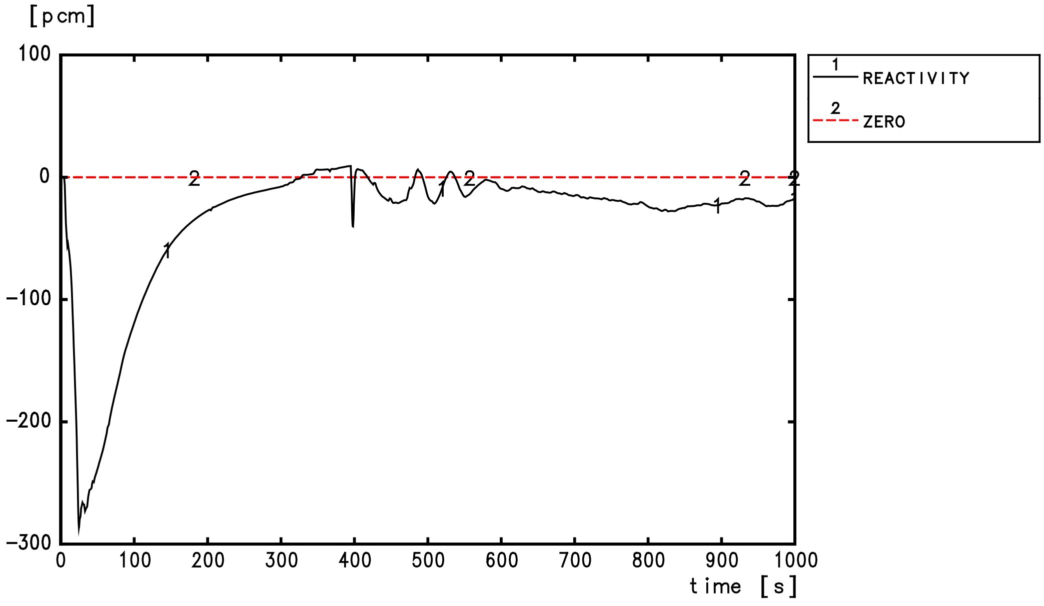


SUB-CHAPTER 16.5 – FIGURE 21

R2 - ATWS LOOP - 1: Core power and reactivity



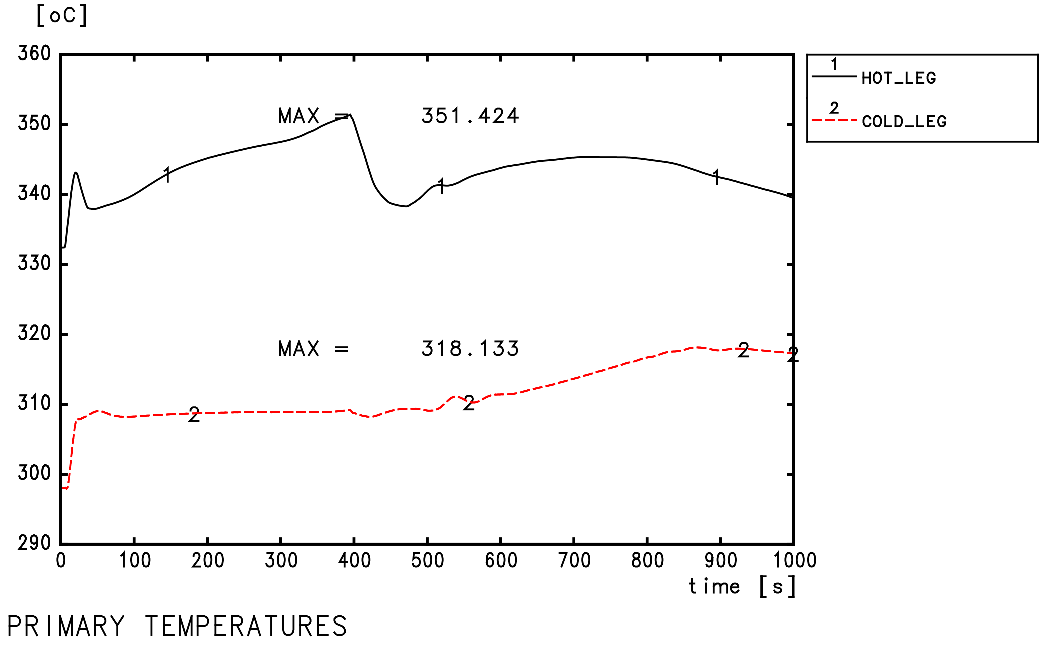
CORE POWER

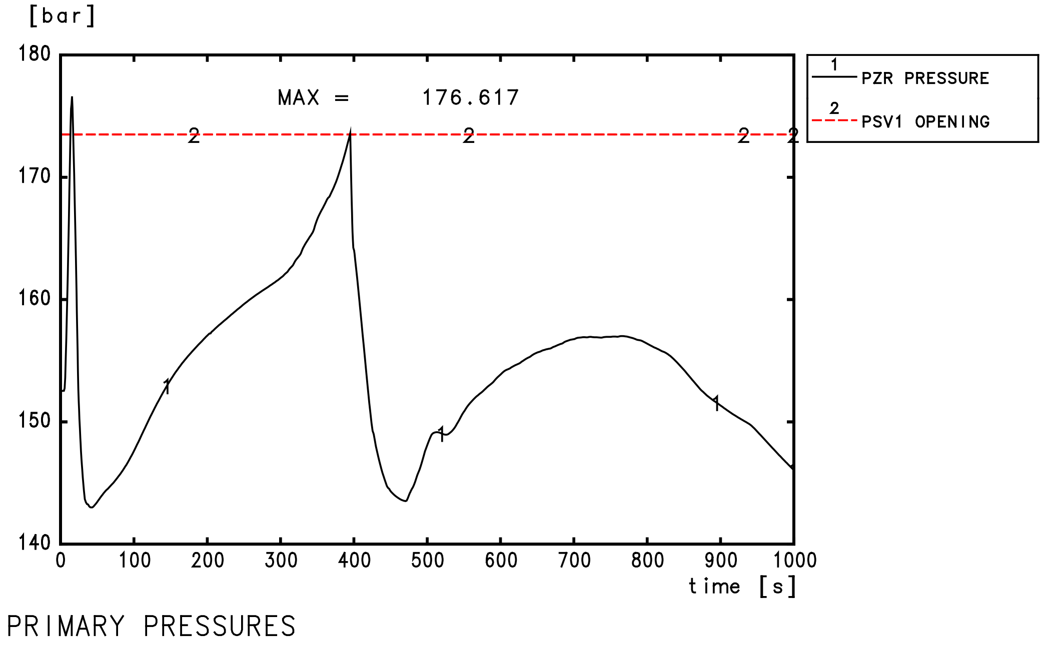


REACTIVITY

SUB-CHAPTER 16.5 – FIGURE 22

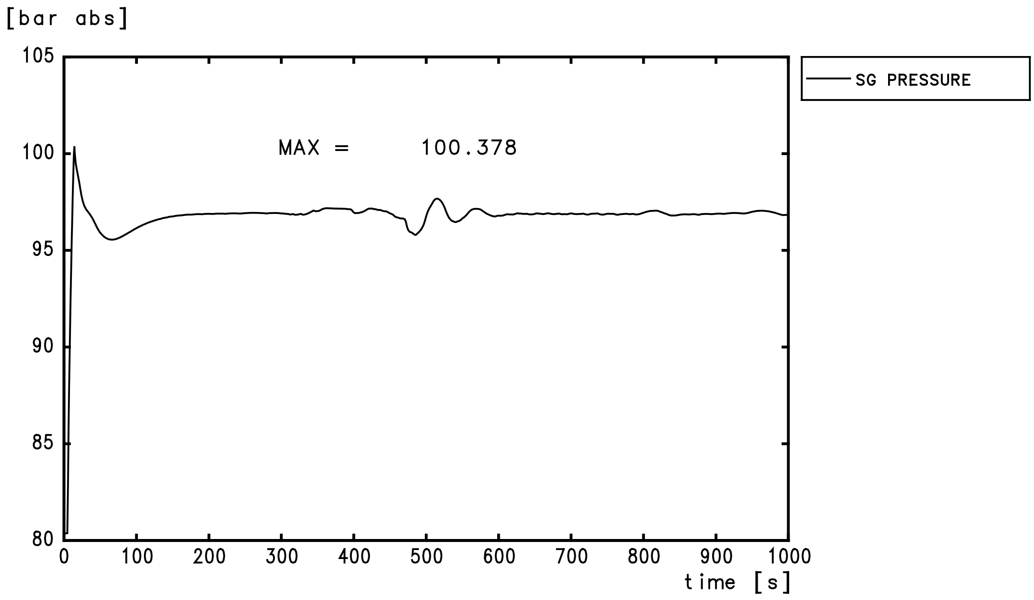
R2 - ATWS LOOP - 2: RCP [RCS] temperatures and pressuriser pressure



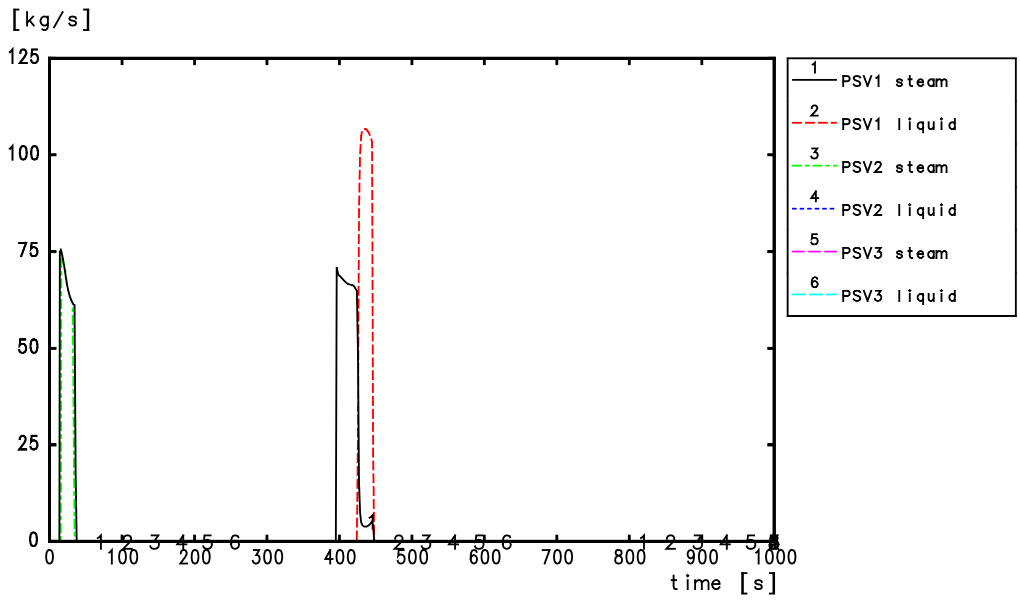


SUB-CHAPTER 16.5 – FIGURE 23

R2 - ATWS LOOP - 3: Secondary side pressure and PSV flow rates



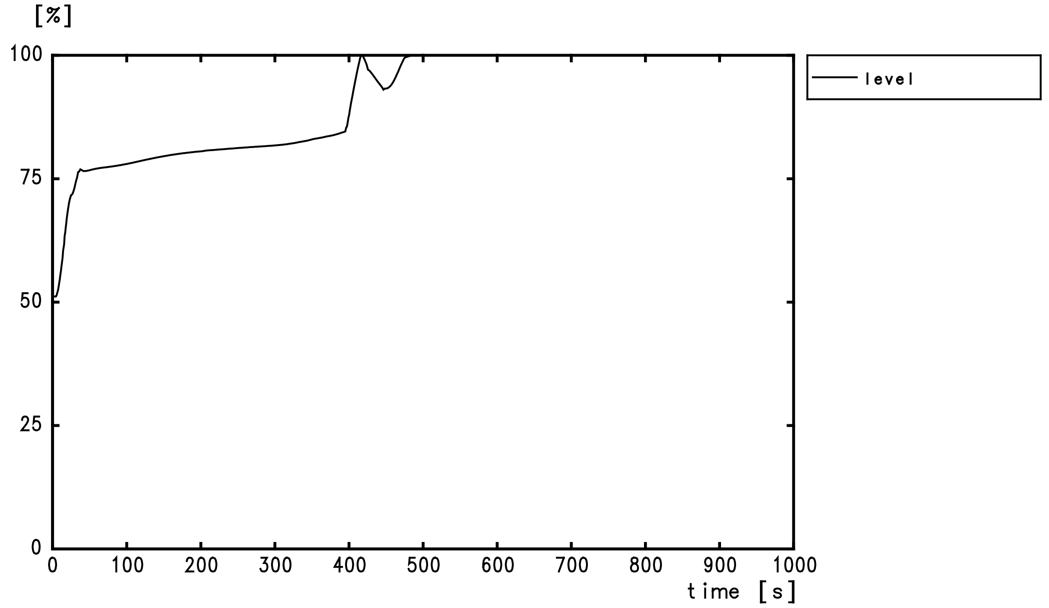
SECONDARY SIDE PRESSURES



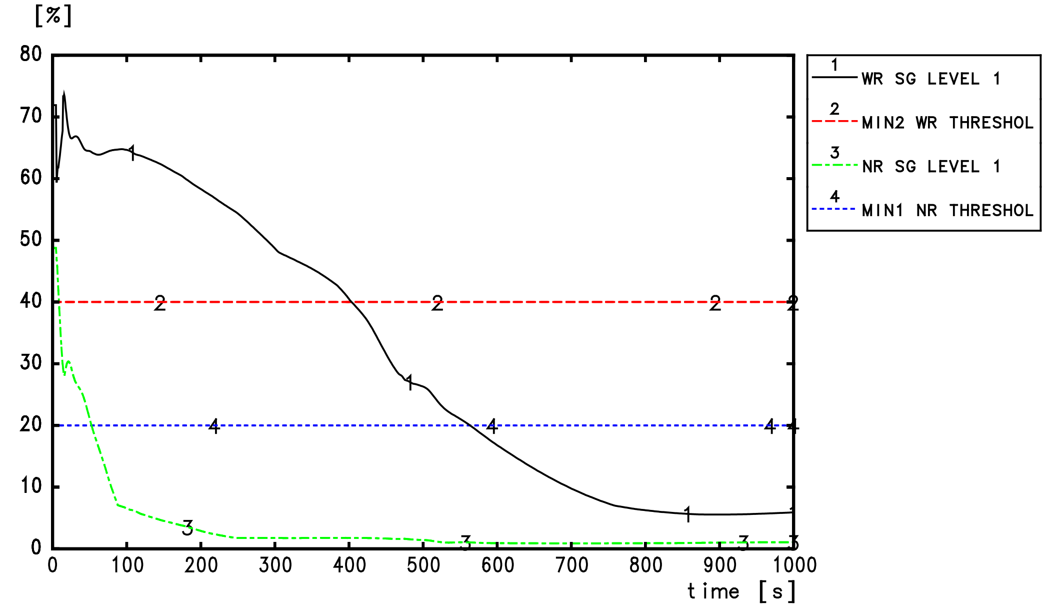
PRESSURE CONTROL FLOWRATE

SUB-CHAPTER 16.5 – FIGURE 24

Figure - R2 - ATWS LOOP - 4: SG and pressuriser levels



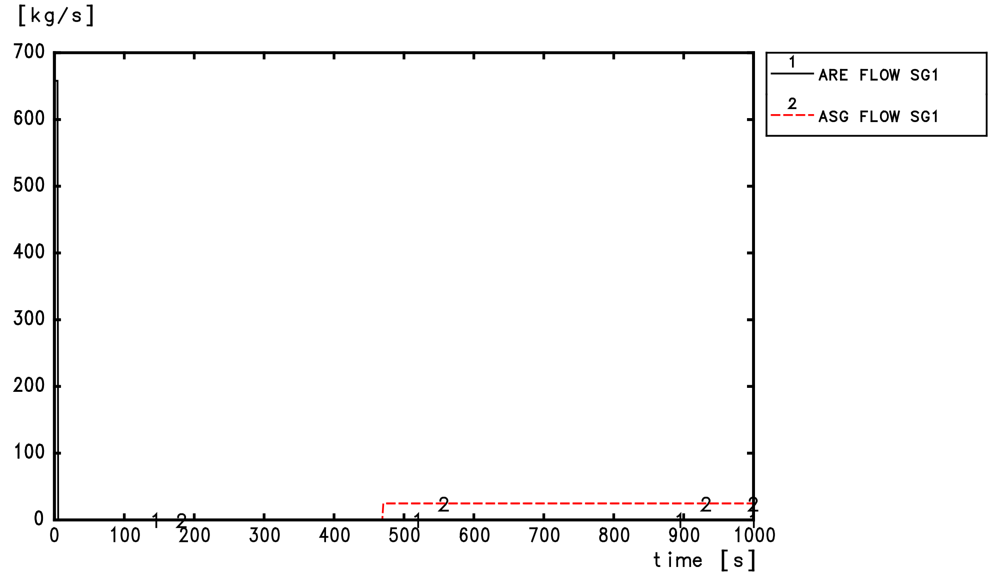
PZR LEVEL



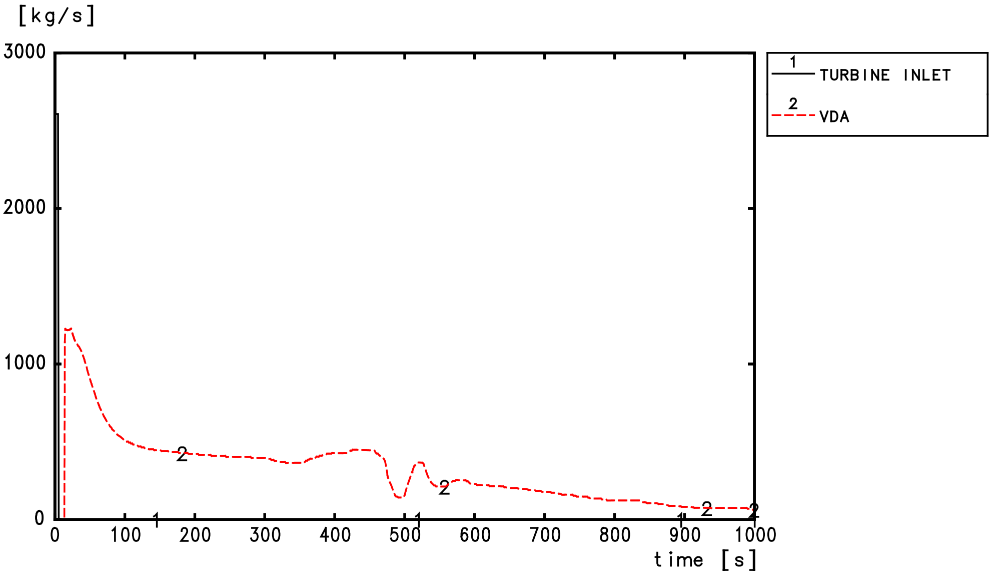
WIDE RANGE AND NARROW RANGE SG LEVEL

SUB-CHAPTER 16.5 – FIGURE 25

R2 - ATWS LOOP - 5: Feedwater flow rate and secondary side steam flow rate



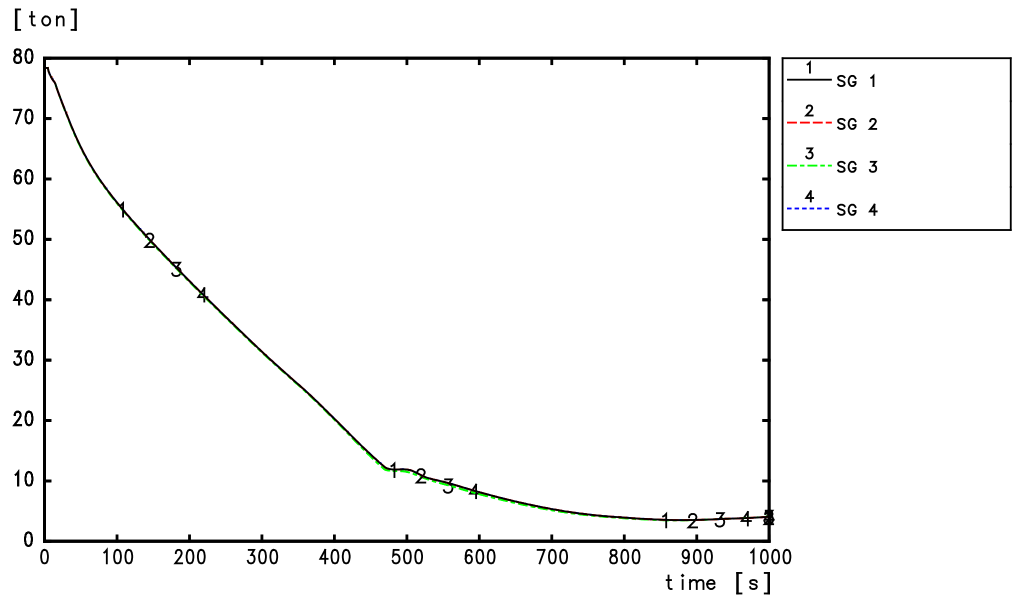
MAIN FEEDWATER FLOWRATE PER LOOP



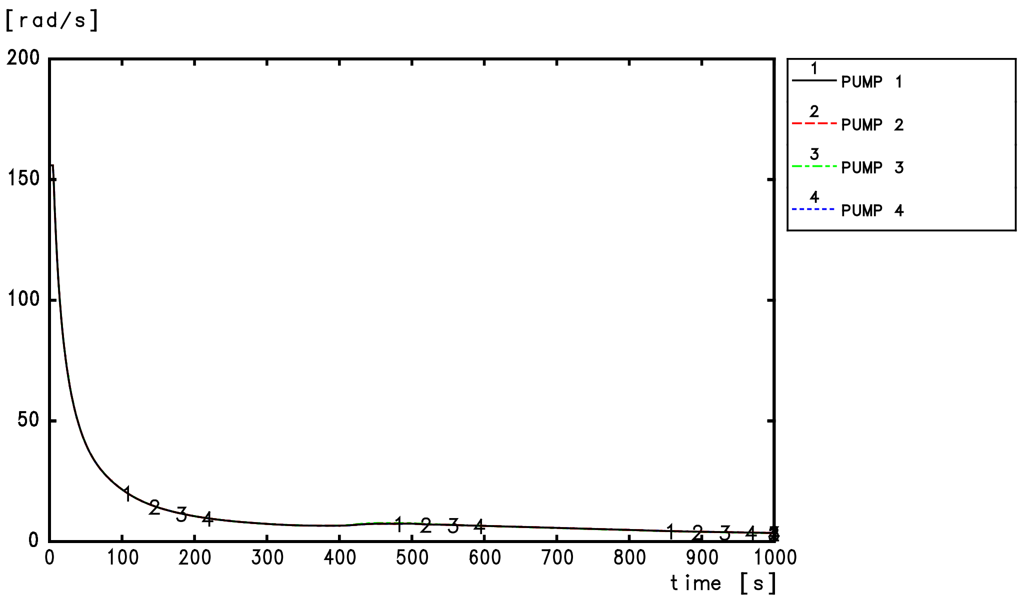
STEAM FLOWRATE

SUB-CHAPTER 16.5 – FIGURE 26

R2 - ATWS LOOP - 6: Reactor coolant pump speed and SG liquid mass



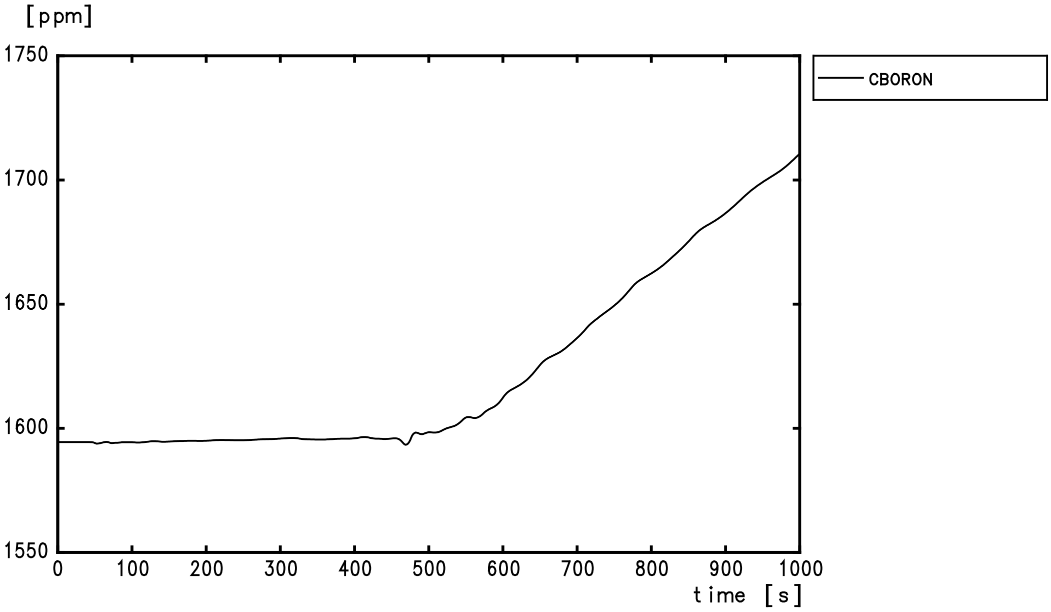
SG LIQUID MASS



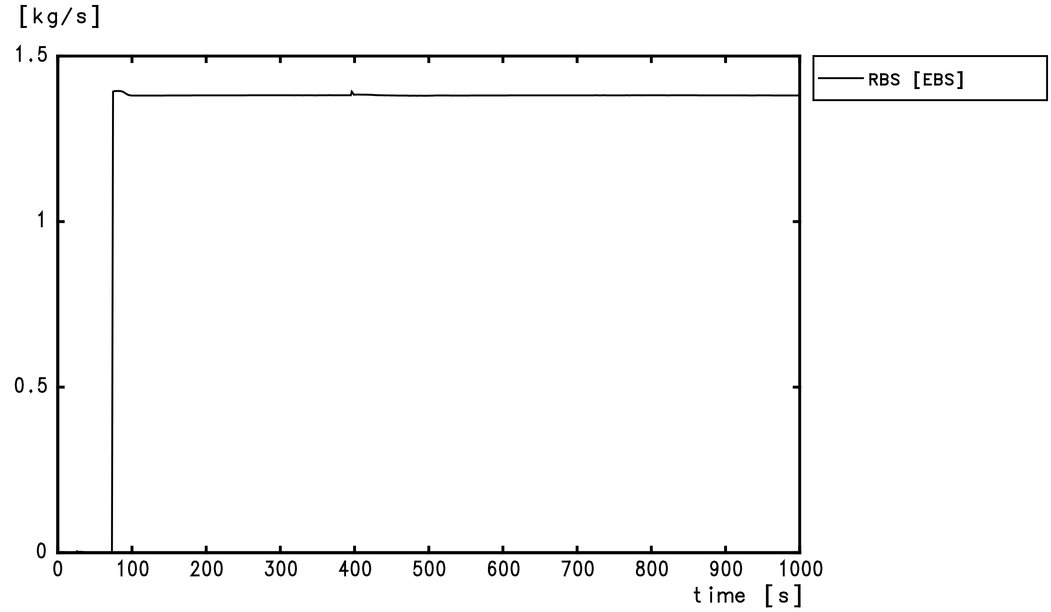
RCP PUMPS [RCP] FLOWRATE PER LOOP

SUB-CHAPTER 16.5 – FIGURE 27

R2 - ATWS LOOP - 7: Core boron concentration and RBS [EBS] flow rate



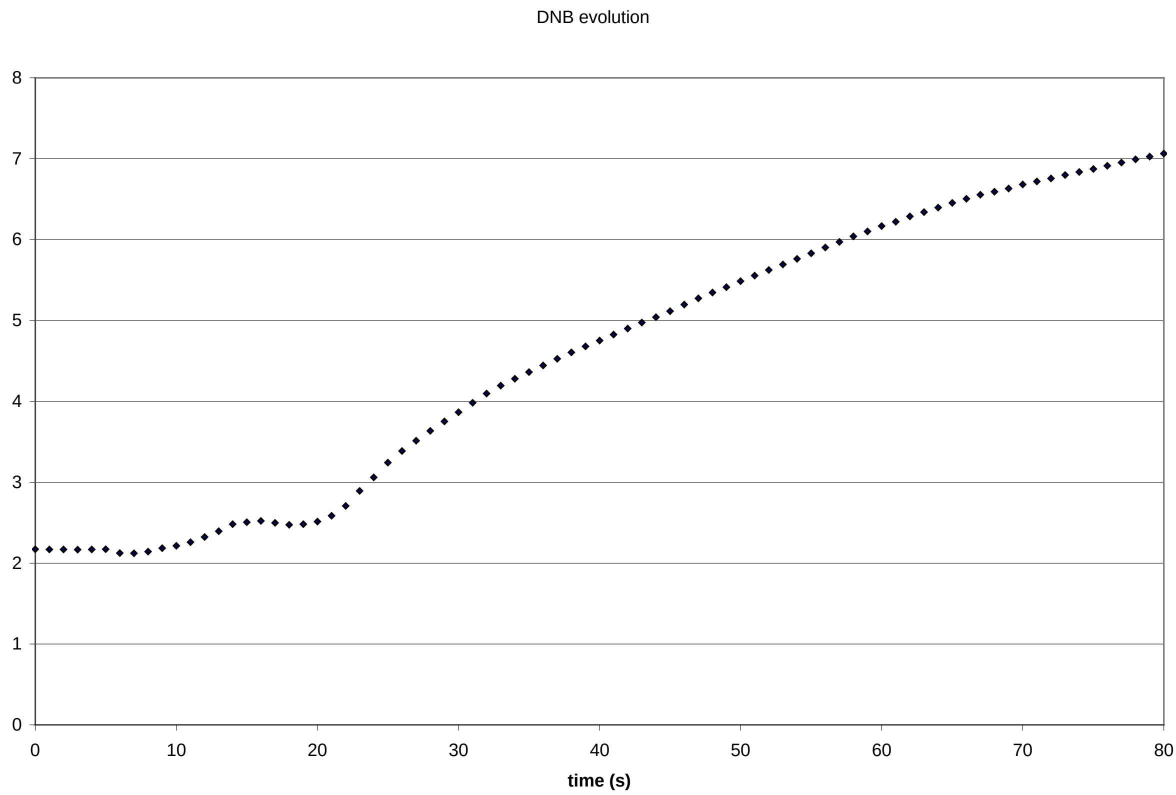
BORON CONCENTRATION



RBS [EBS] FLOWRATE PER LOOP

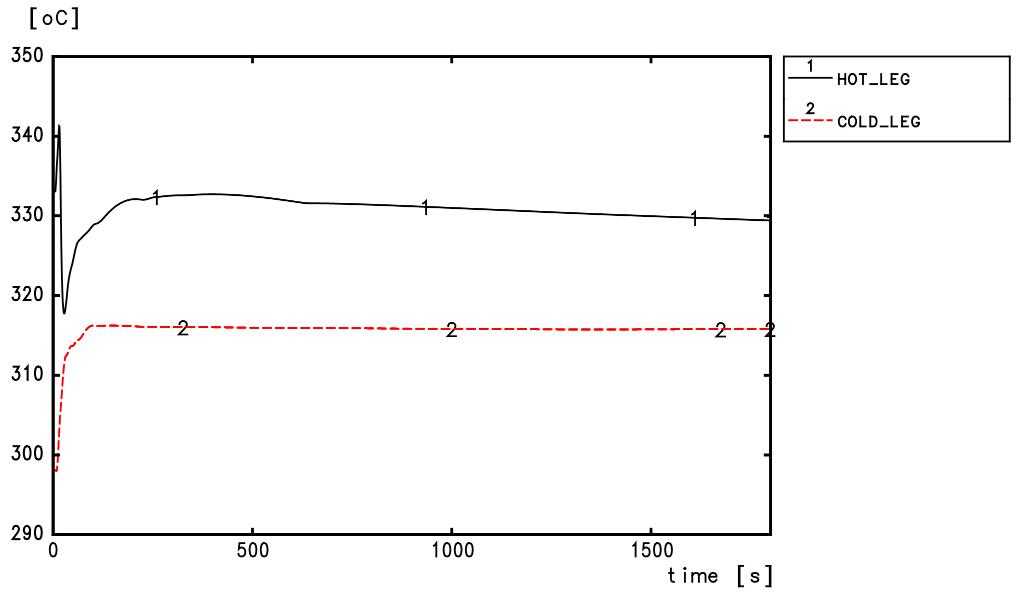
SUB-CHAPTER 16.5 – FIGURE 28

R2 - ATWS LOOP - 8: DNBR transient

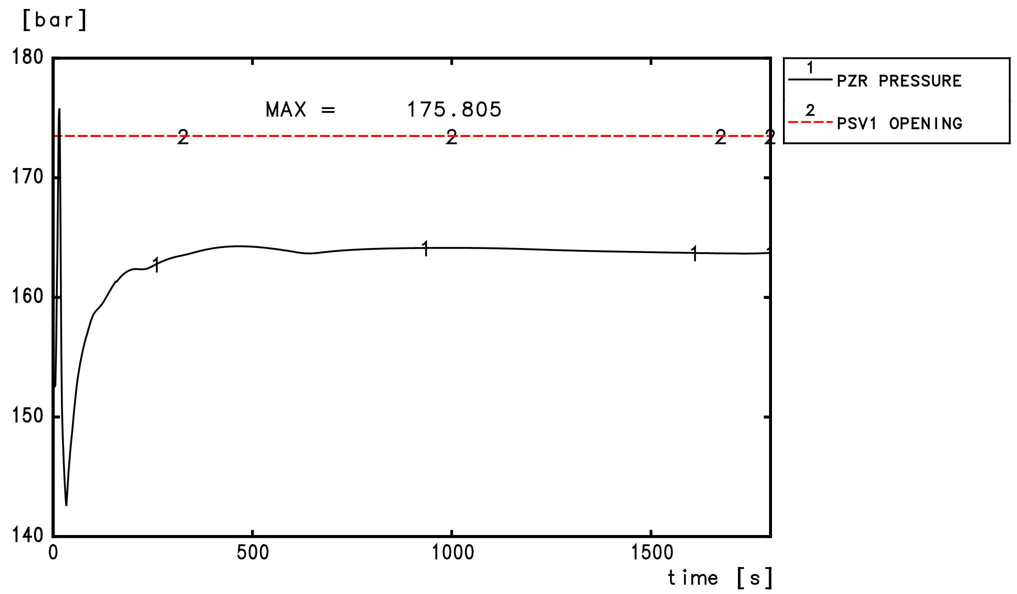


SUB-CHAPTER 16.5 – FIGURE 29

R2 - ATWS LOOP w/o RPR [PS] - 1: RCP [RCS] temperatures and pressuriser pressure



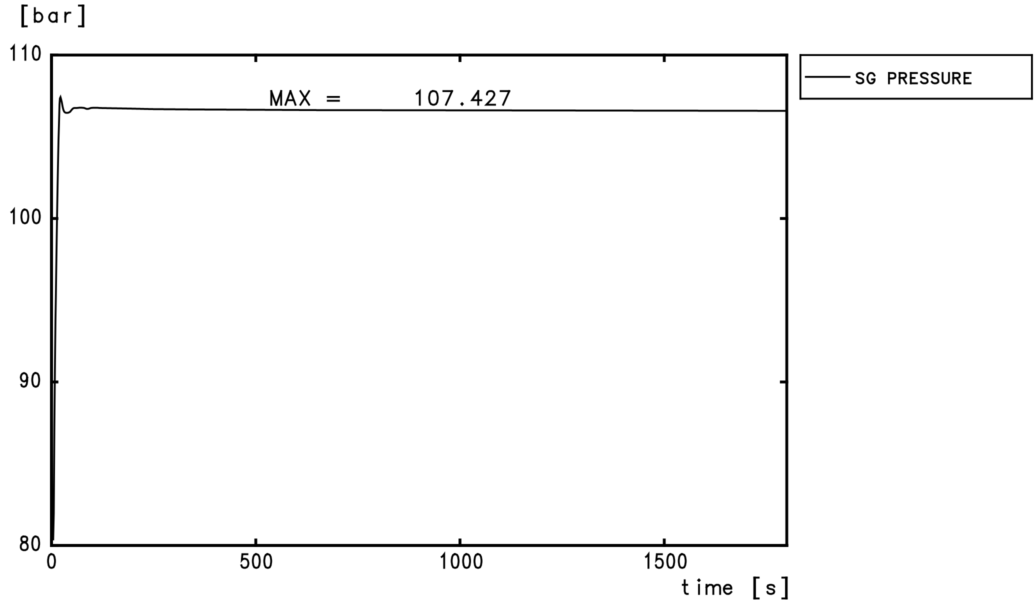
PRIMARY TEMPERATURES



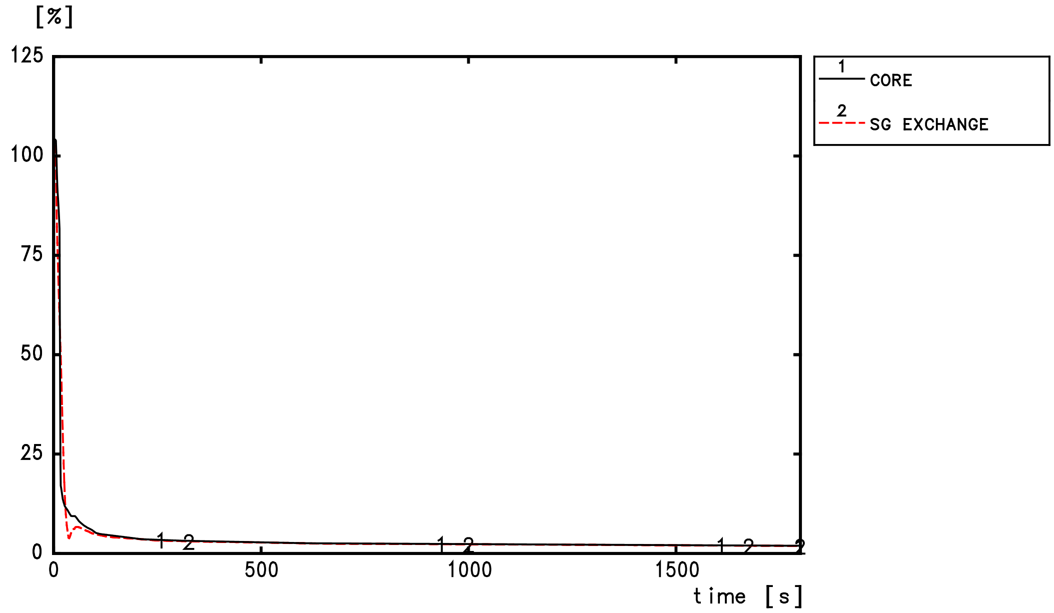
PRIMARY PRESSURE

SUB-CHAPTER 16.5 – FIGURE 30

R2 - ATWS LOOP w/o RPR [PS] - 2: Secondary side pressure and core power



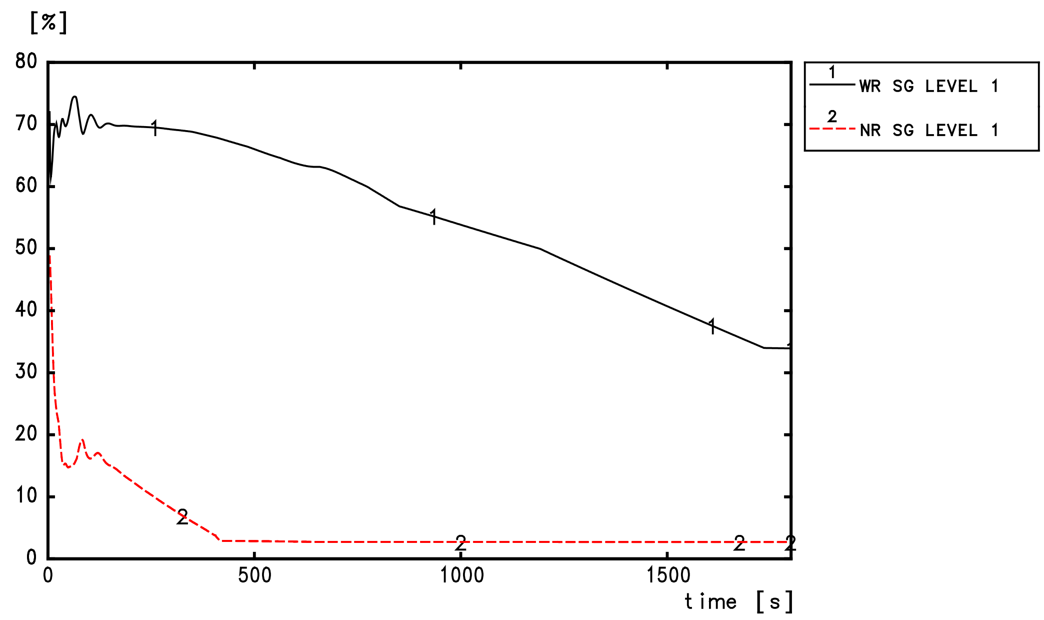
SECONDARY SIDE PRESSURES



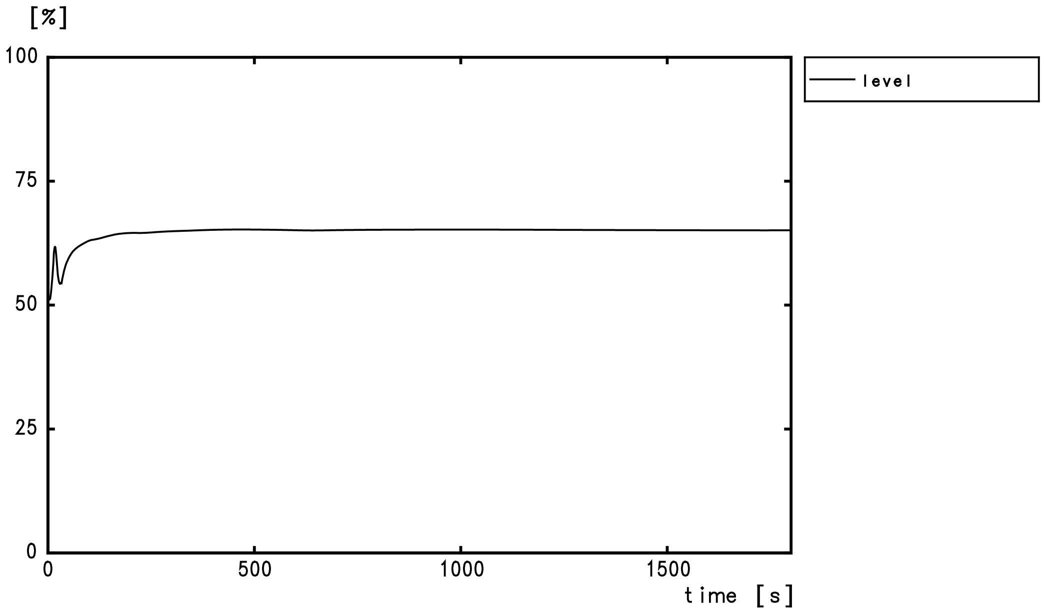
CORE POWER

SUB-CHAPTER 16.5 – FIGURE 31

R2 - ATWS LOOP w/o RPR [PS] - 3: Pressuriser and SG levels



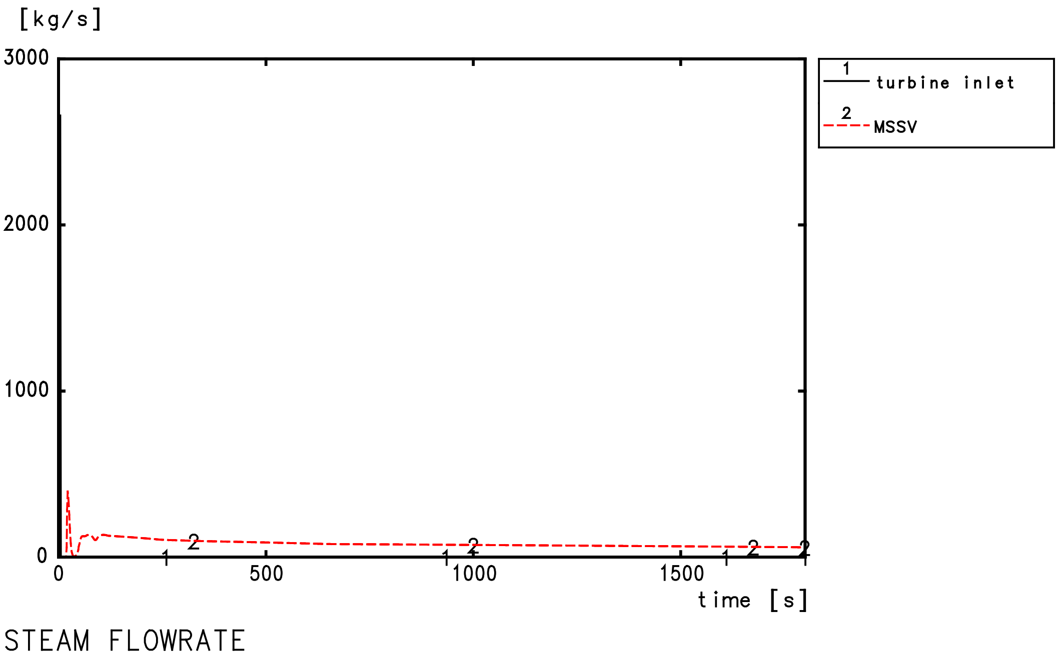
WIDE RANGE AND NARROW RANGE SG LEVEL

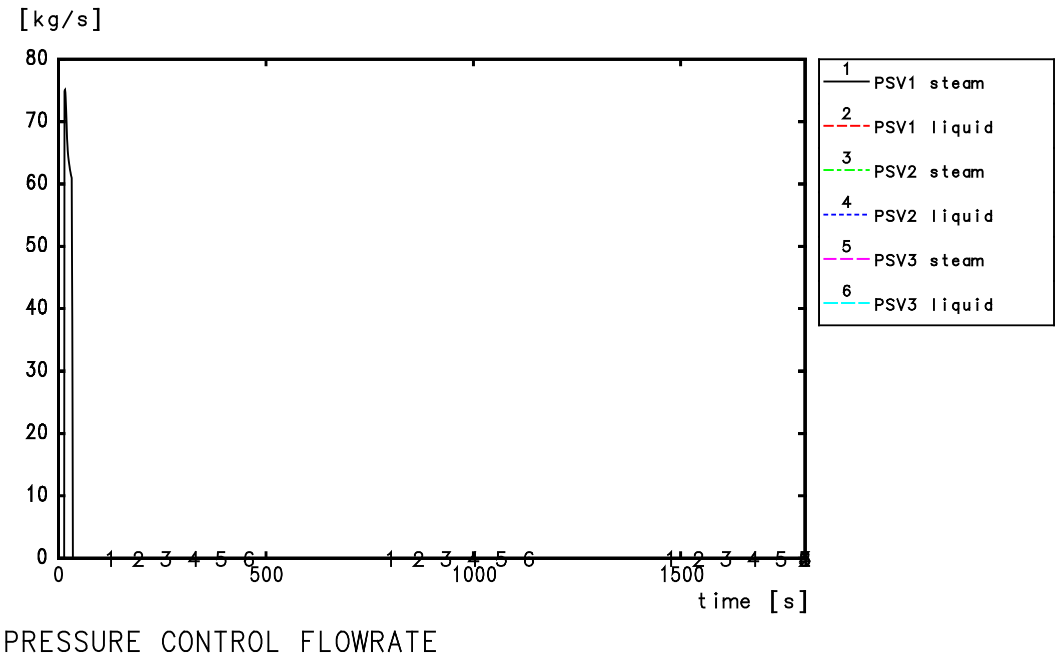


PZR LEVEL

SUB-CHAPTER 16.5 – FIGURE 32

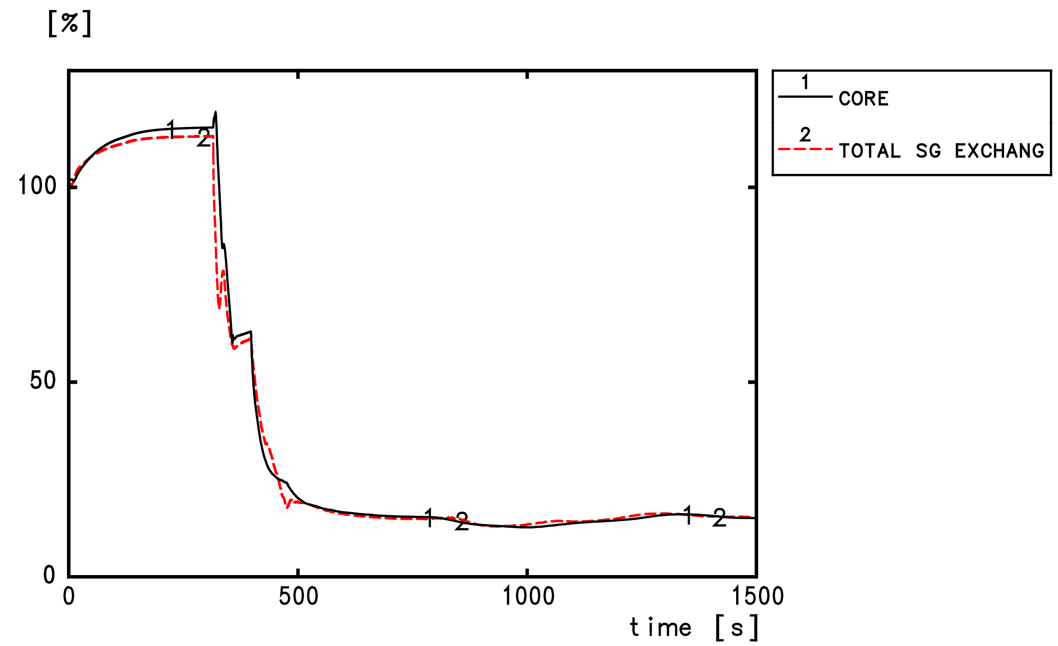
R2 - ATWS LOOP w/o RPR [PS] - 4: Secondary steam flow rate and PSV flow rate



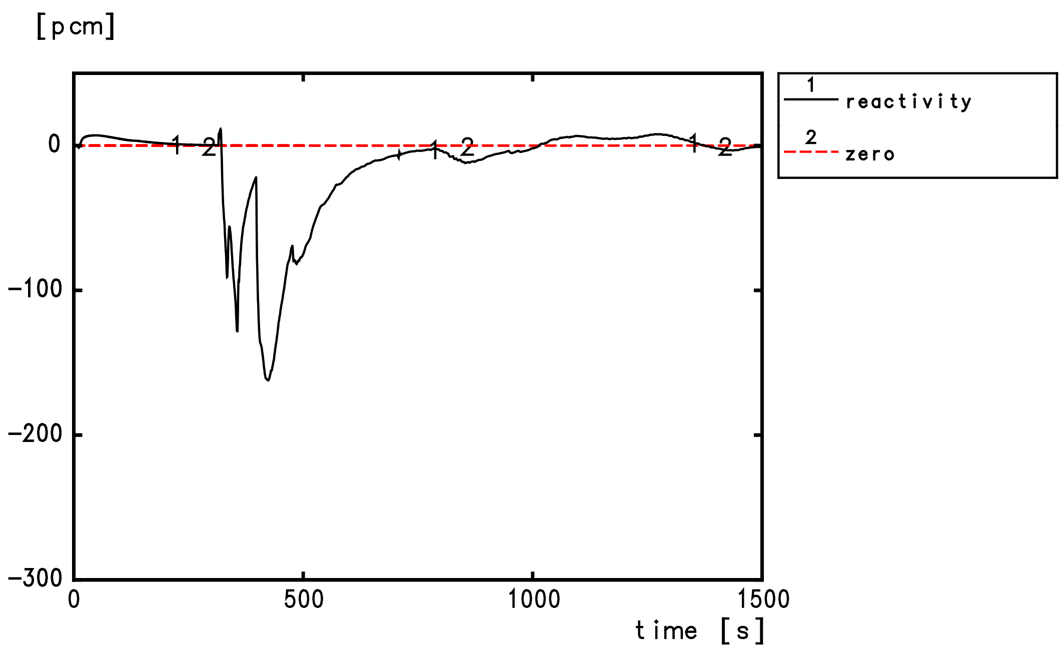


SUB-CHAPTER 16.5 – FIGURE 33

R2 - ATWS EISF - 1: Core power and SG exchanged power and reactivity



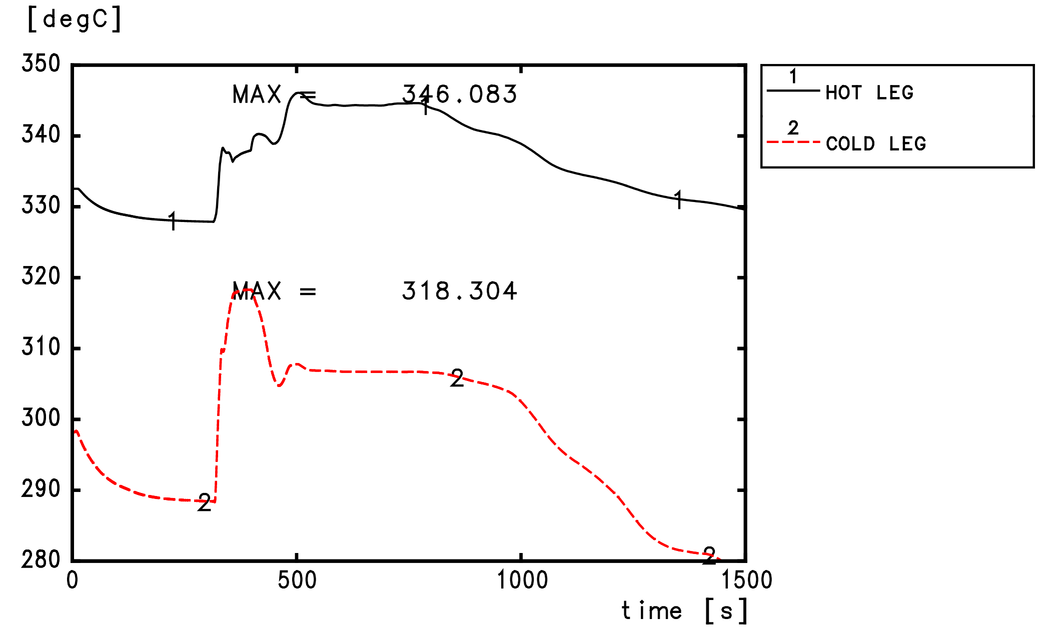
CORE POWER AND SG EXCHANGE



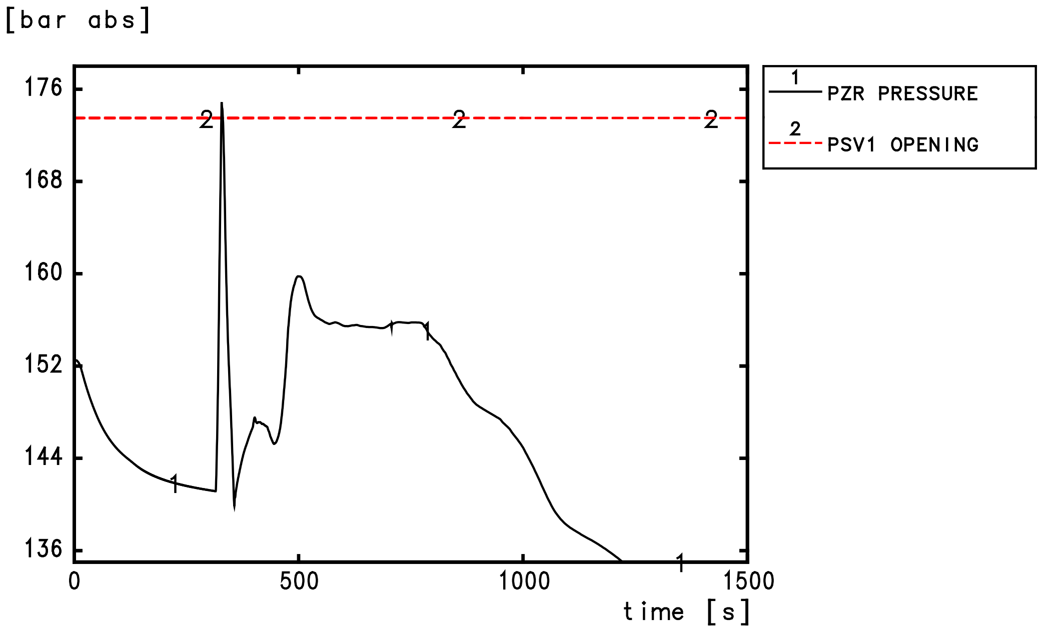
REACTIVITY

SUB-CHAPTER 16.5 – FIGURE 34

R2 - ATWS EISF - 2: RCP [RCS] temperatures and RCP [RCS] pressure



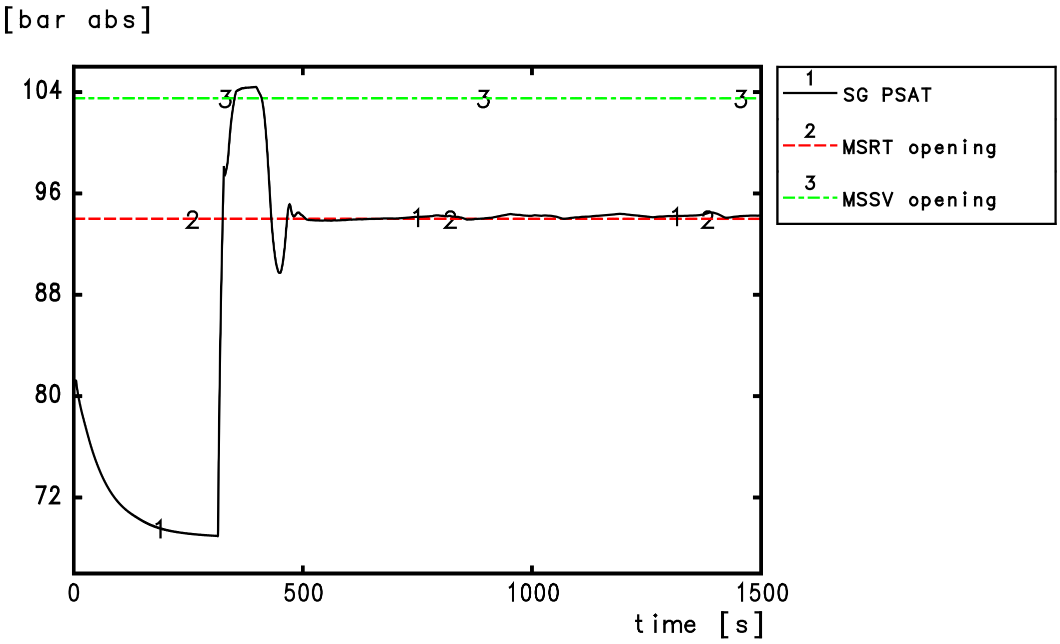
PRIMARY TEMPERATURES



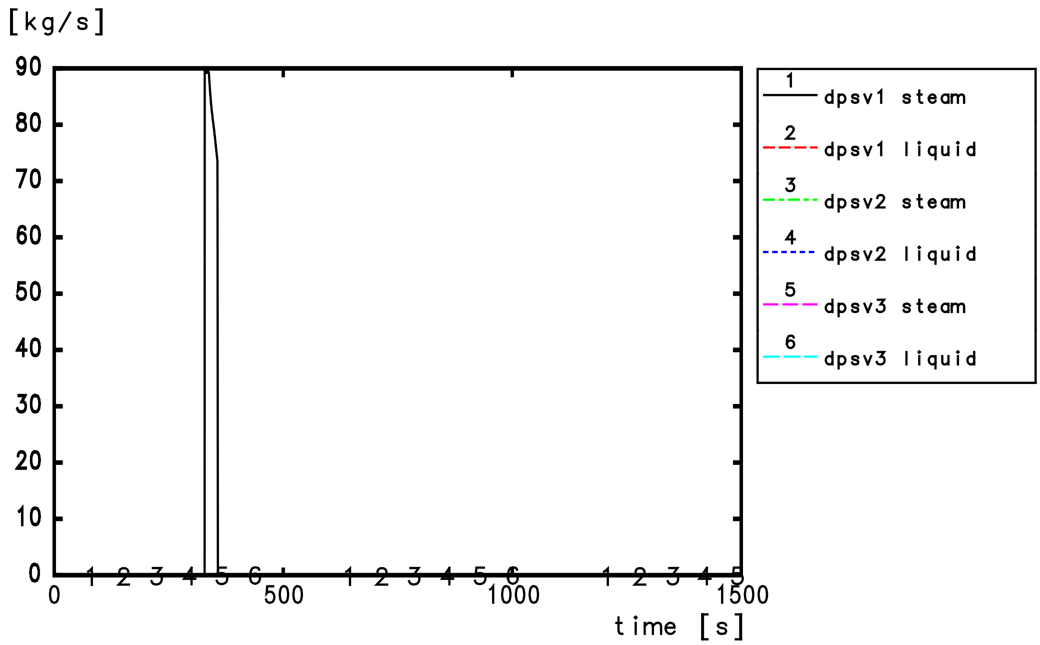
PRIMARY PRESSURE

SUB-CHAPTER 16.5 – FIGURE 35

R2 - ATWS EISF - 3: Secondary side pressure and pressuriser safety valves flow rate



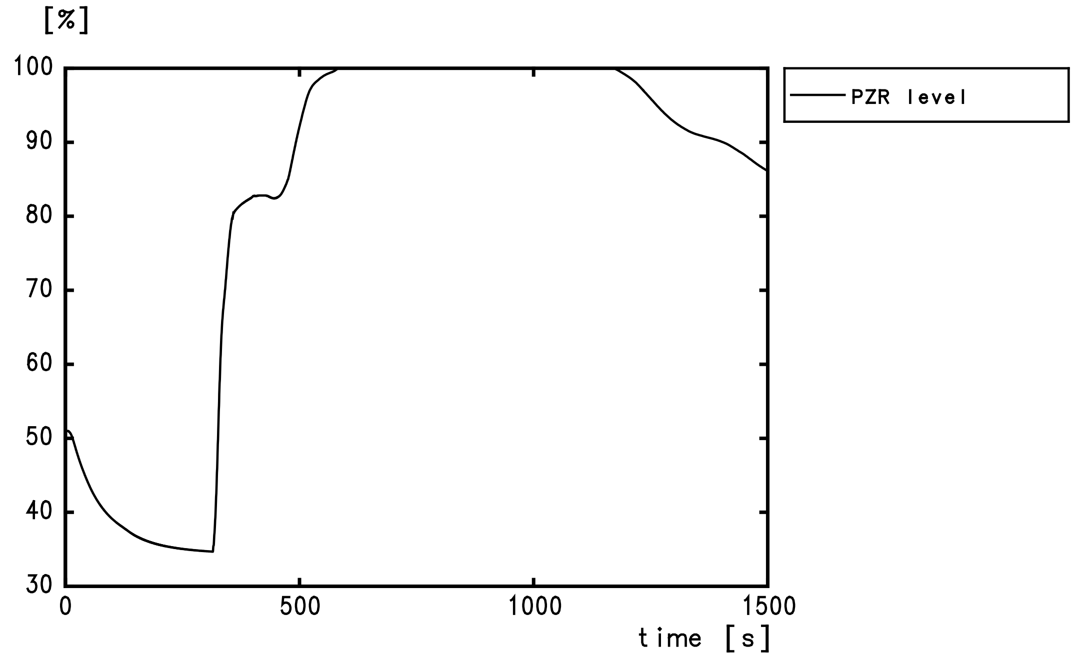
SG pressure



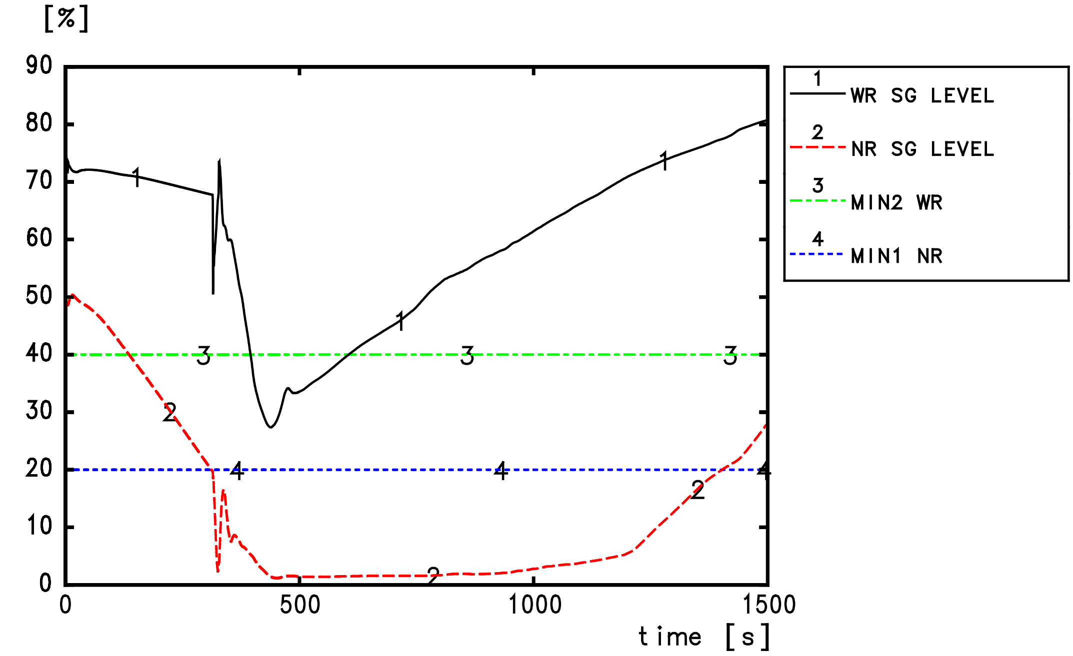
PRESSURE CONTROL FLOWRATES

SUB-CHAPTER 16.5 – FIGURE 36

R2 - ATWS EISF - 4: Pressuriser and SG levels



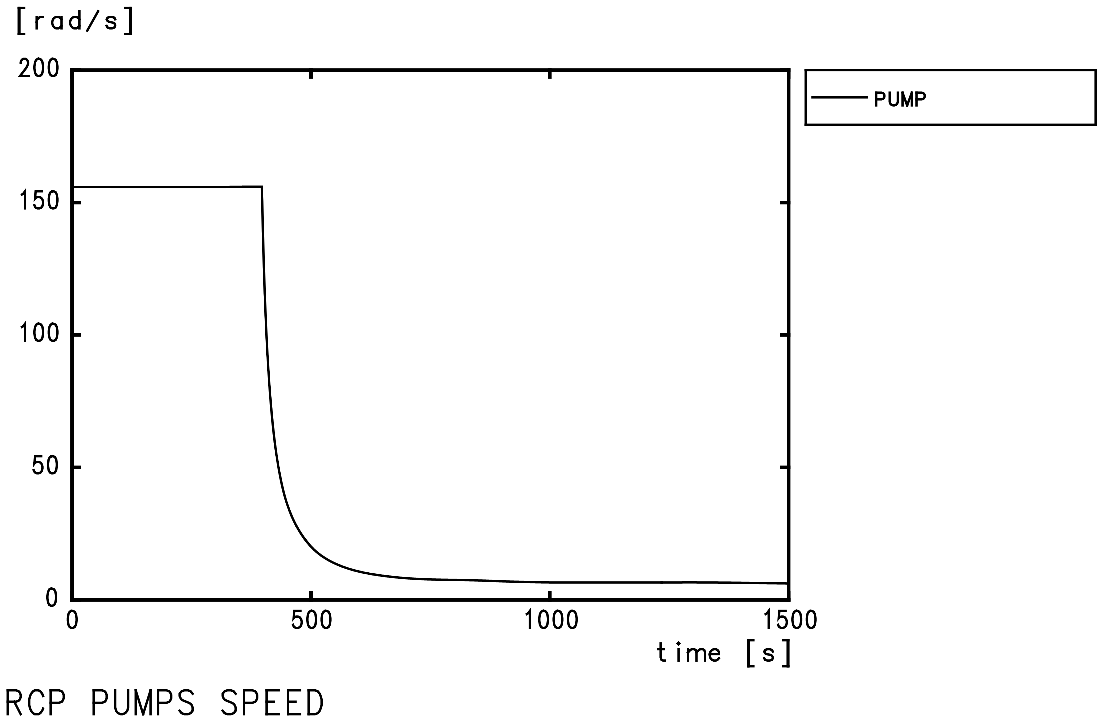
PZR LEVEL



WIDE RANGE AND NARROW RANGE SG LEVELS

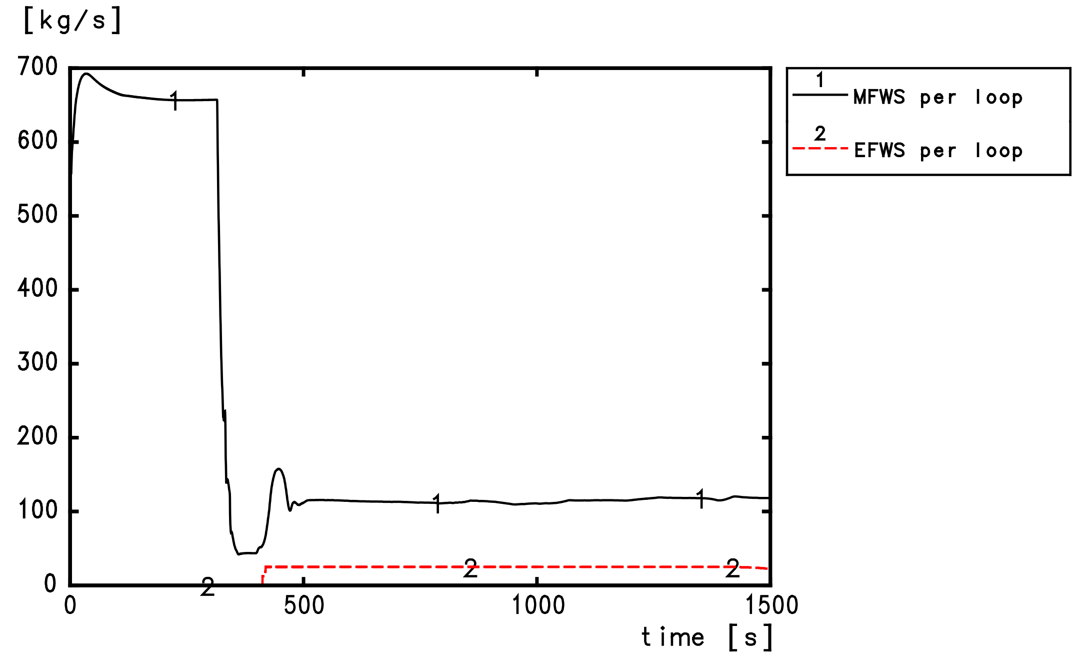
SUB-CHAPTER 16.5 – FIGURE 37

R2 - ATWS EISF - 5: Reactor coolant pump speed

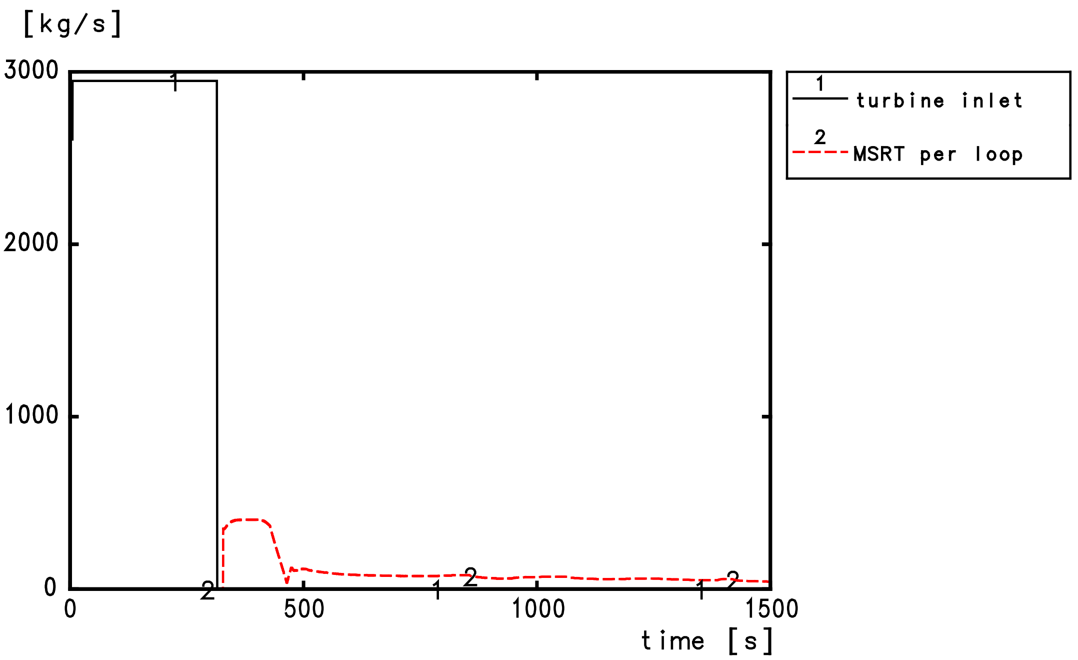


SUB-CHAPTER 16.5 – FIGURE 38

R2 - ATWS EISF - 6: Main and emergency feedwater flow rates and steam flow rates



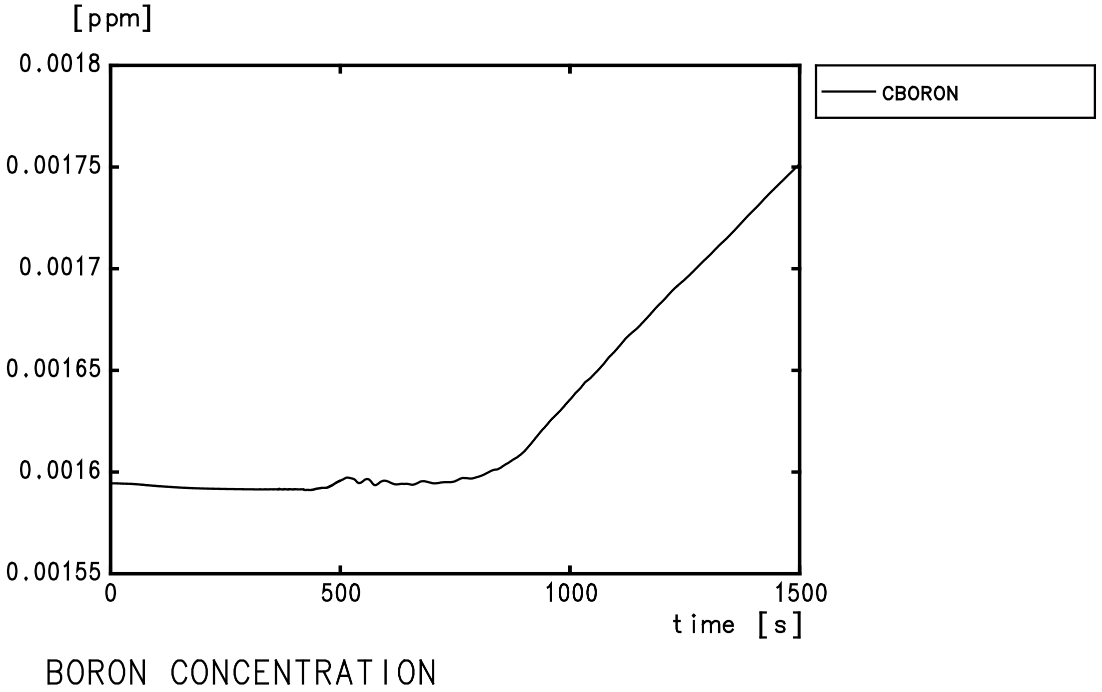
MAIN AND EMERGENGY FEEDWATER FLOWRATES

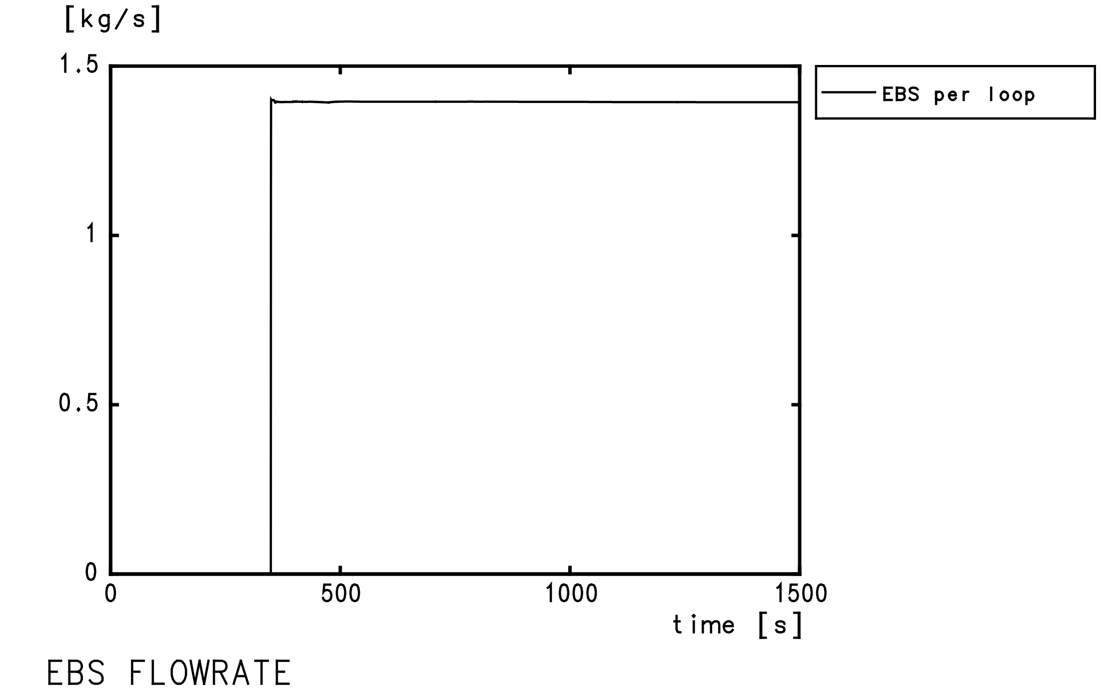


STEAM FLOWRATE

SUB-CHAPTER 16.5 – FIGURE 39

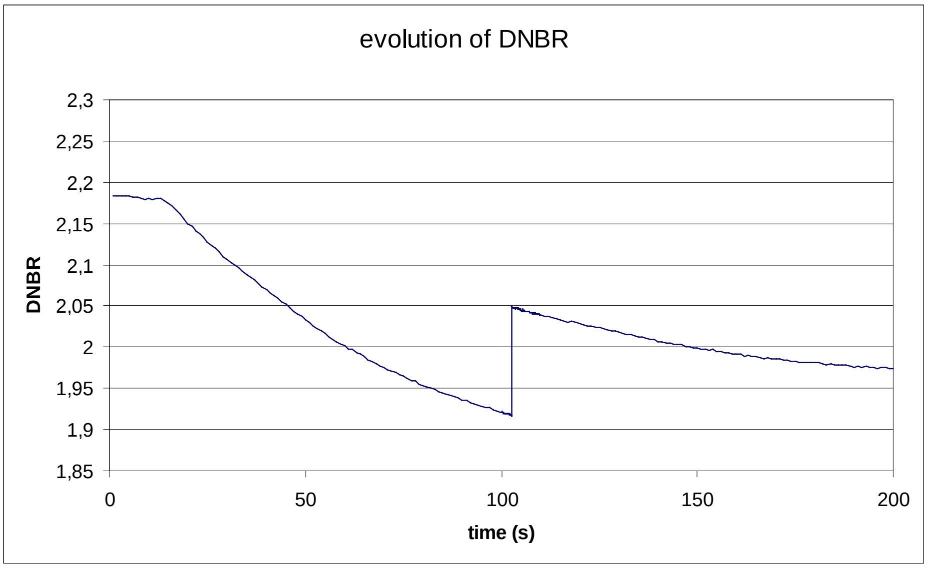
R2 - ATWS EISF - 7: Core boron concentration and RBS [EBS] flow rate





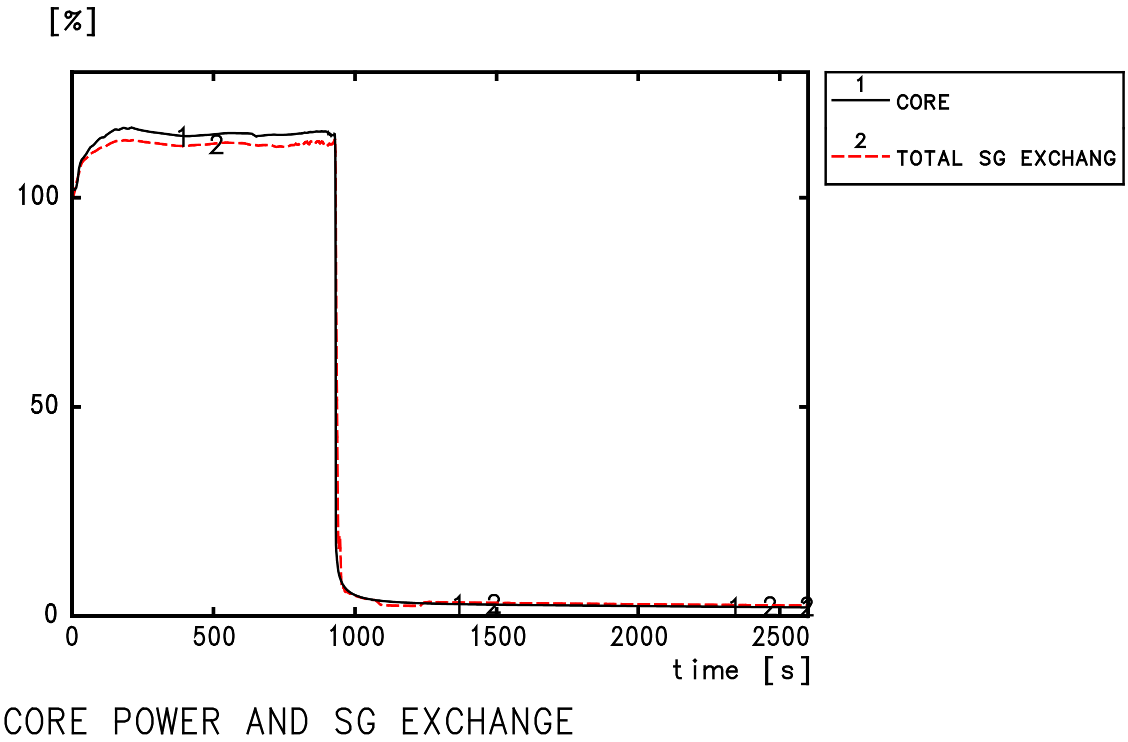
SUB-CHAPTER 16.5 – FIGURE 40

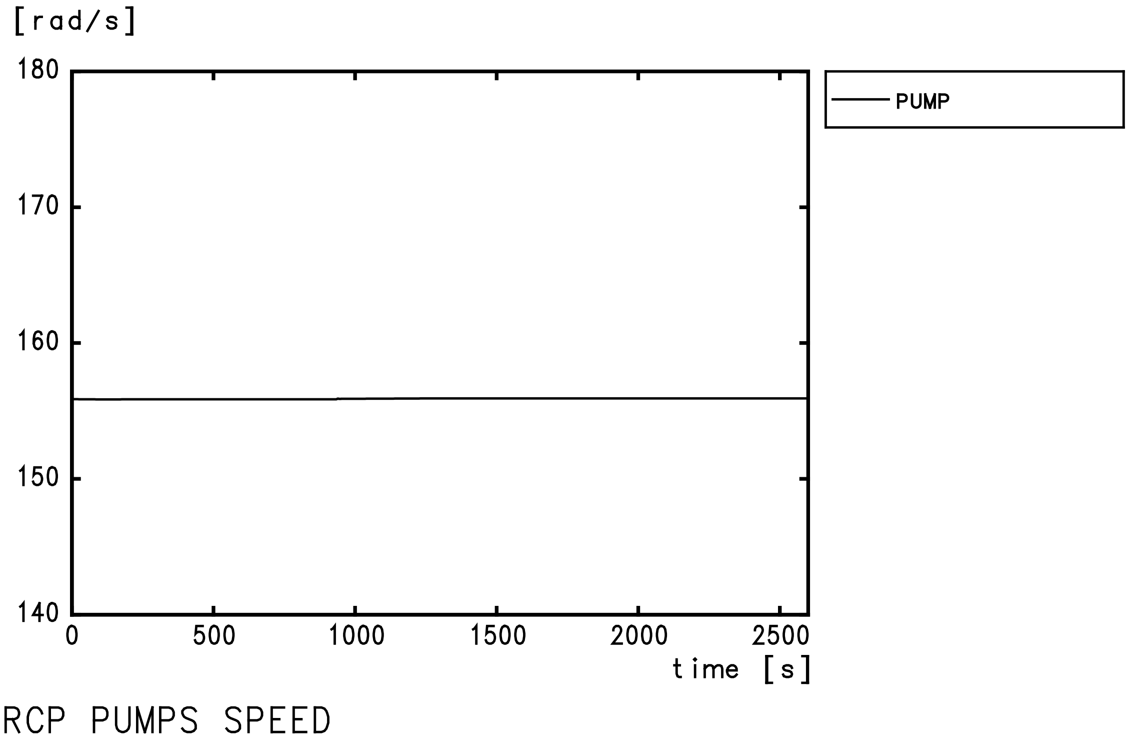
R2 - ATWS EISF - 8: DNBR Transient



SUB-CHAPTER 16.5 – FIGURE 41

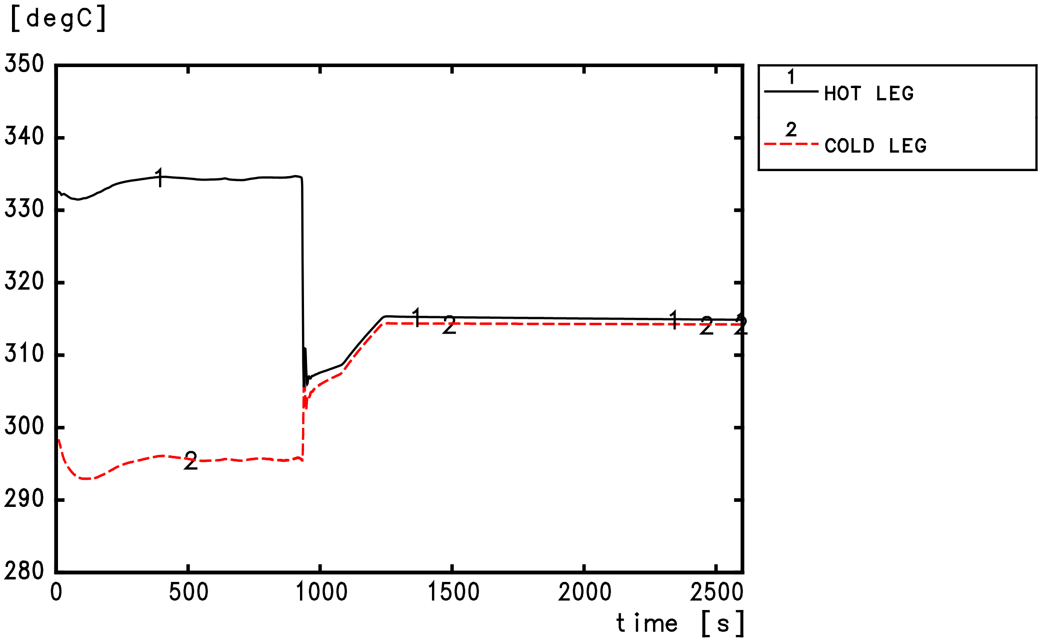
R2 - ATWS EISF w/o RPR [PS] - 1: Core power and reactor coolant pump speed



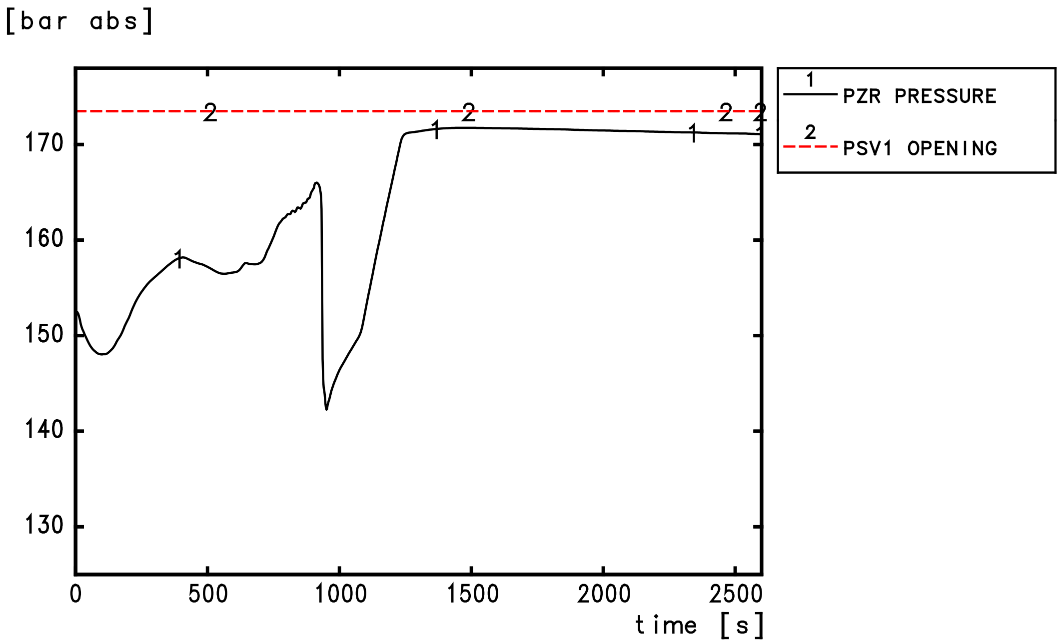


SUB-CHAPTER 16.5 – FIGURE 42

R2 - ATWS EISF w/o RPR [PS] - 2: RCP [RCS] temperatures and pressuriser pressure



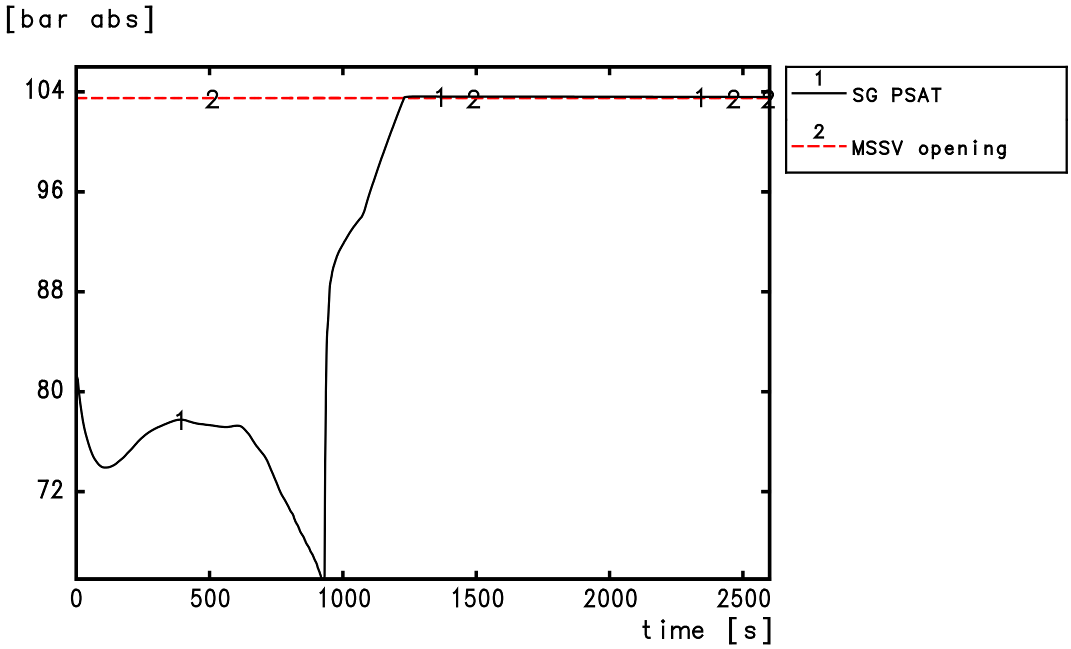
PRIMARY TEMPERATURES



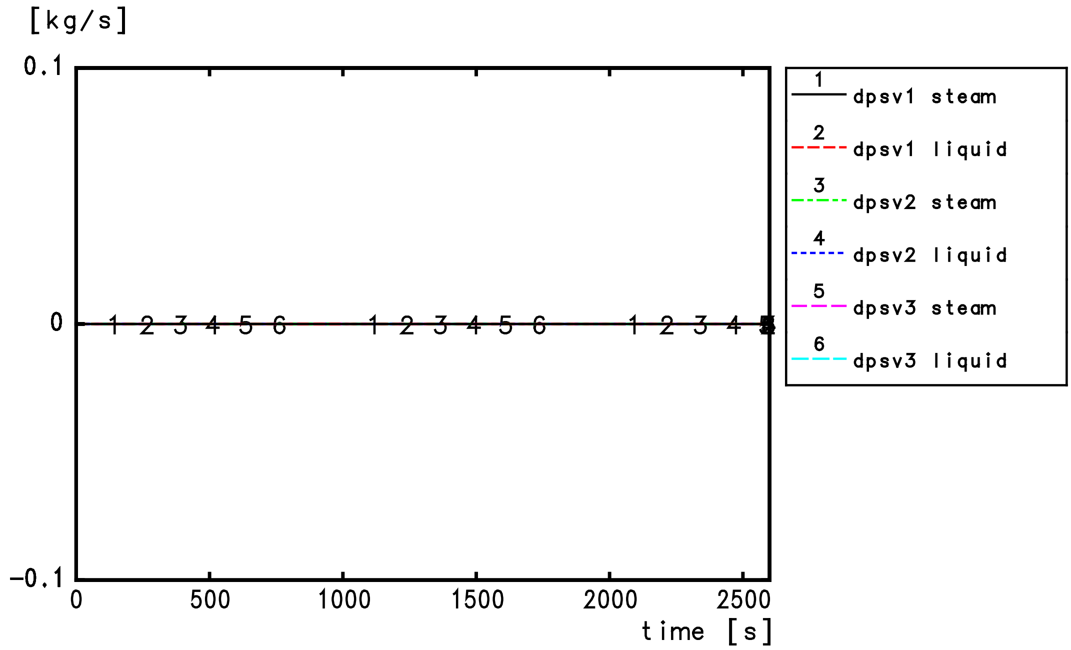
PRIMARY PRESSURE

SUB-CHAPTER 16.5 – FIGURE 43

R2 - ATWS EISF w/o RPR [PS] - 3: SG pressure and PSV flow rate



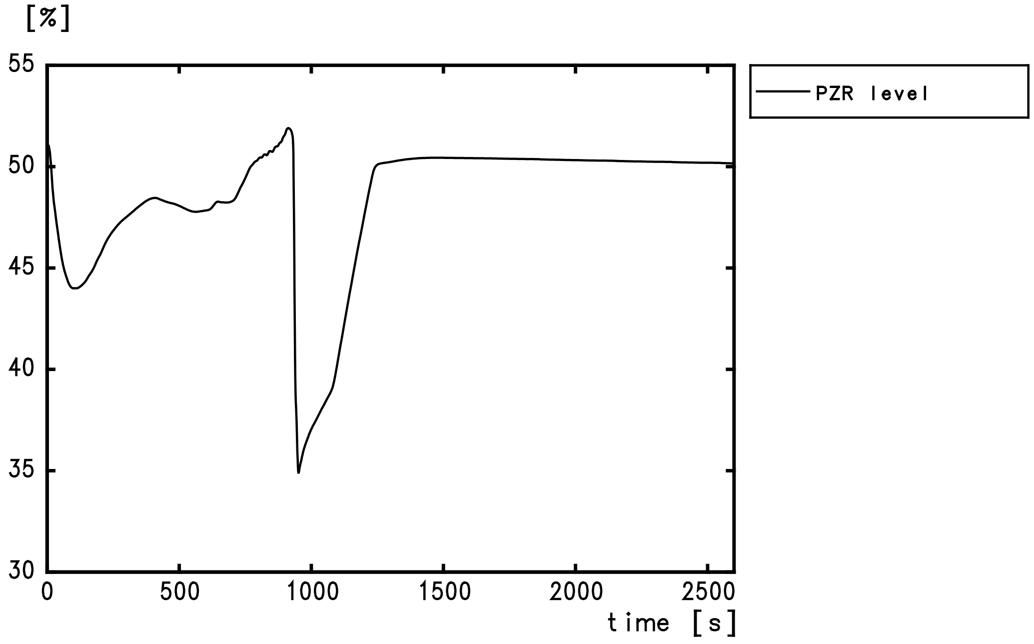
SG pressure



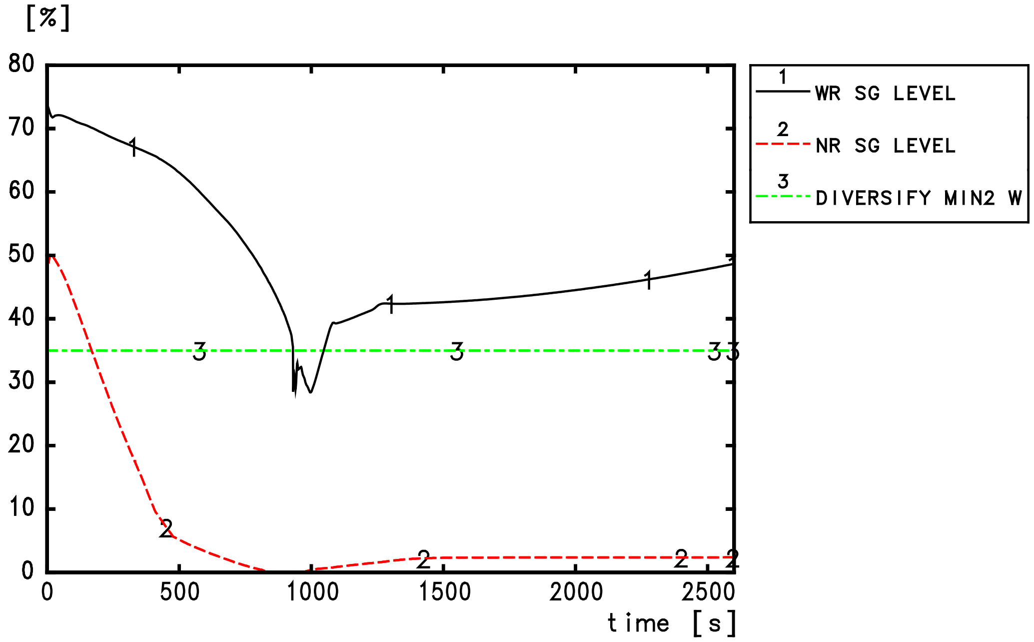
PRESSURE CONTROL FLOWRATES

SUB-CHAPTER 16.5 – FIGURE 44

R2 - ATWS EISF w/o RPR [PS] - 4: Pressuriser and SG levels



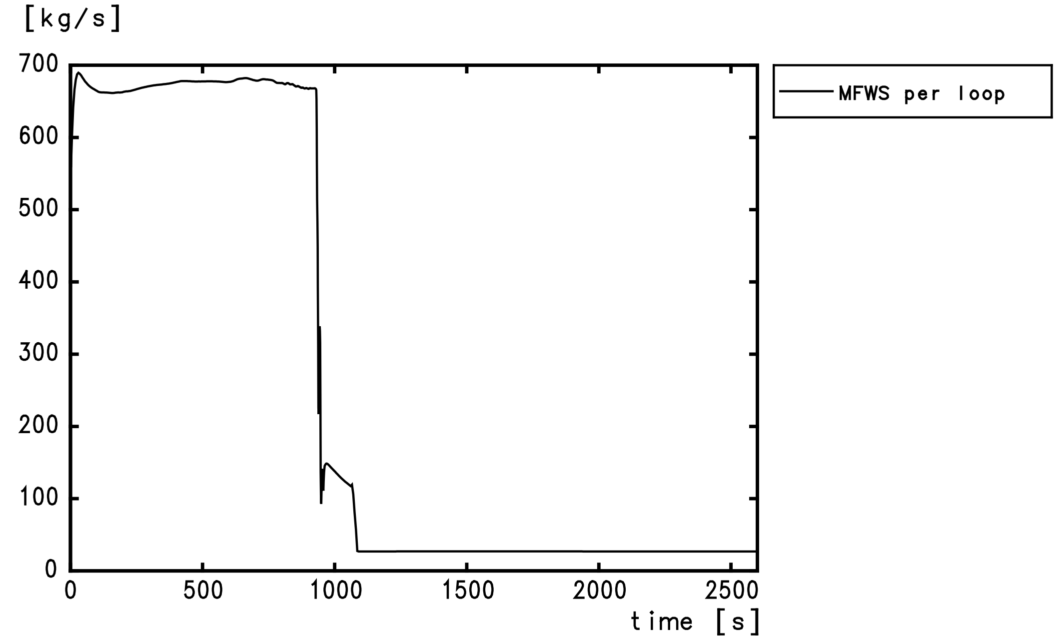
PZR LEVEL



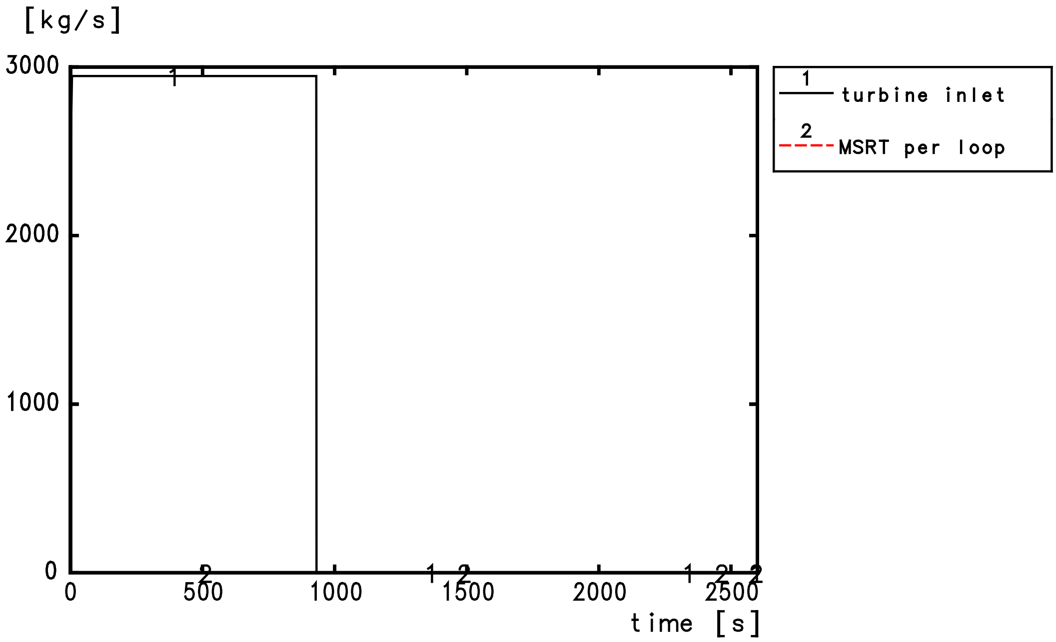
WIDE RANGE AND NARROW RANGE SG LEVELS

SUB-CHAPTER 16.5 – FIGURE 45

R2 - ATWS EISF w/o RPR [PS] - 5: Main feedwater flow rate and steam flow rate



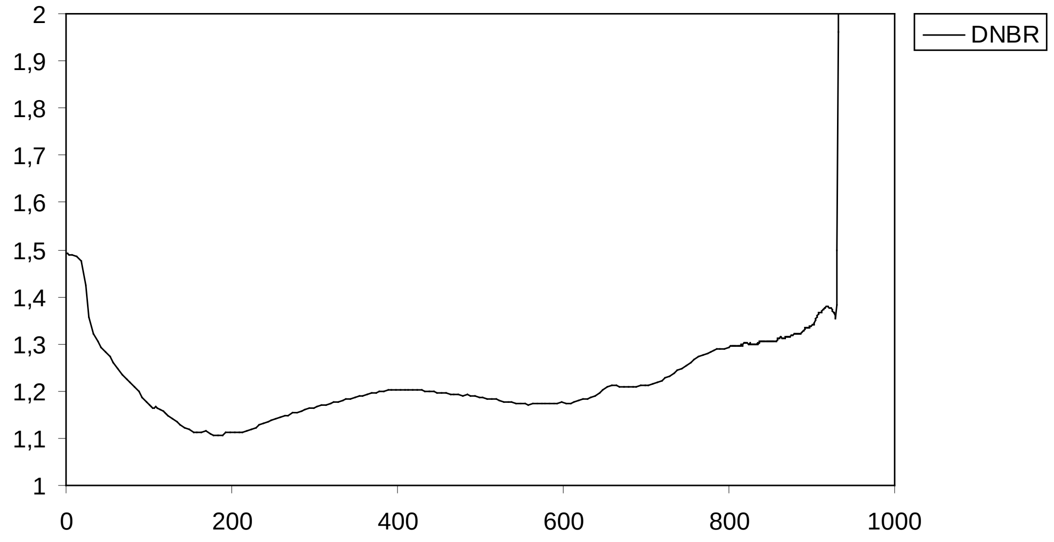
MAIN FEEDWATER FLOWRATES



STEAM FLOWRATE

SUB-CHAPTER 16.5 – FIGURE 46

R2 - ATWS EISF w/o RPR [PS] - 6: DNBR transient



DNBR EVOLUTION

SUB-CHAPTER 16.5 – FIGURE 47

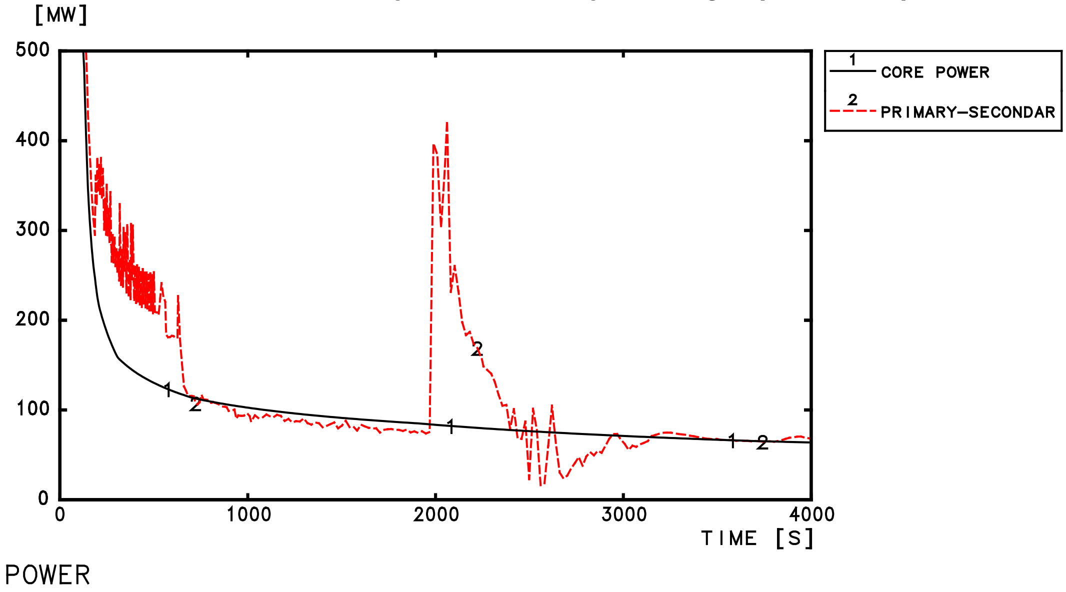
H1 - SB LOCA w/o MHSI - 1: Decay heat curve, A+B+C terms

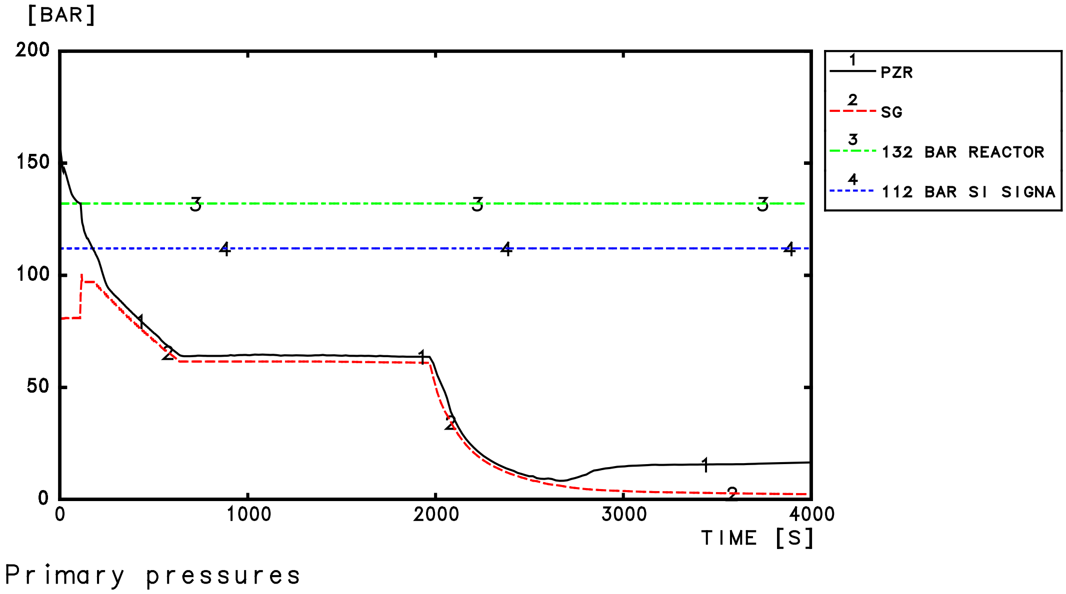
{CCI Removed}

b

SUB-CHAPTER 16.5 – FIGURE 48

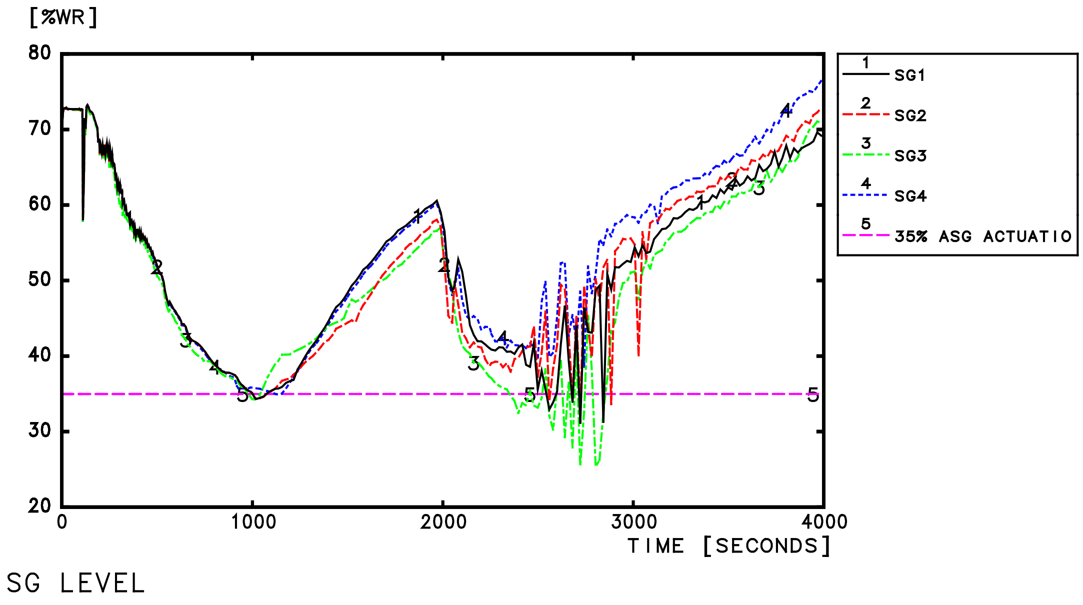
H1 - SB LOCA w/o MHSI - 2: Primary and secondary exchanged power and pressures

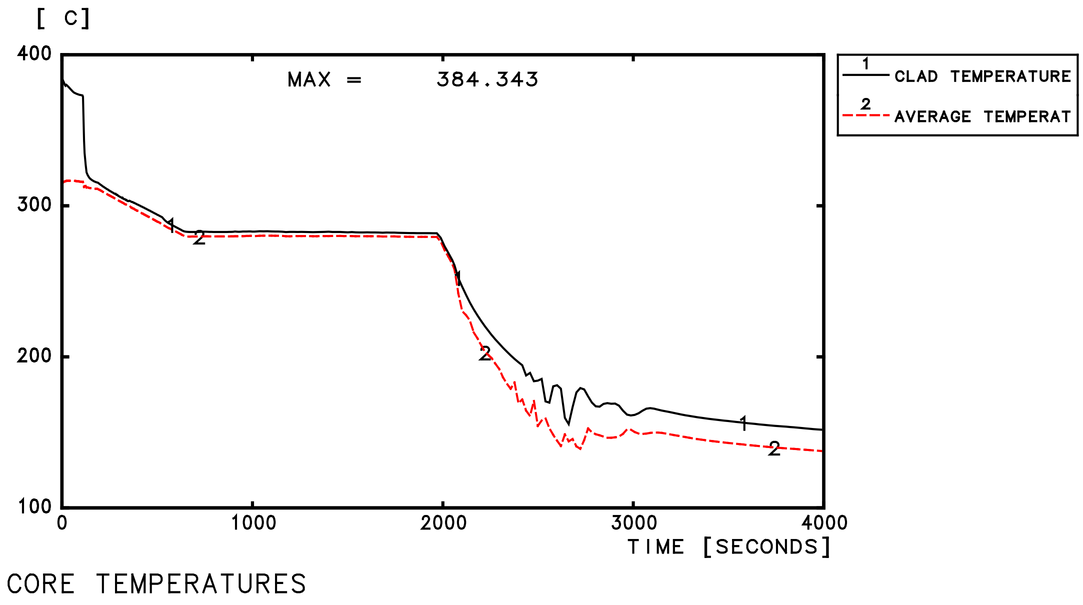




SUB-CHAPTER 16.5 – FIGURE 49

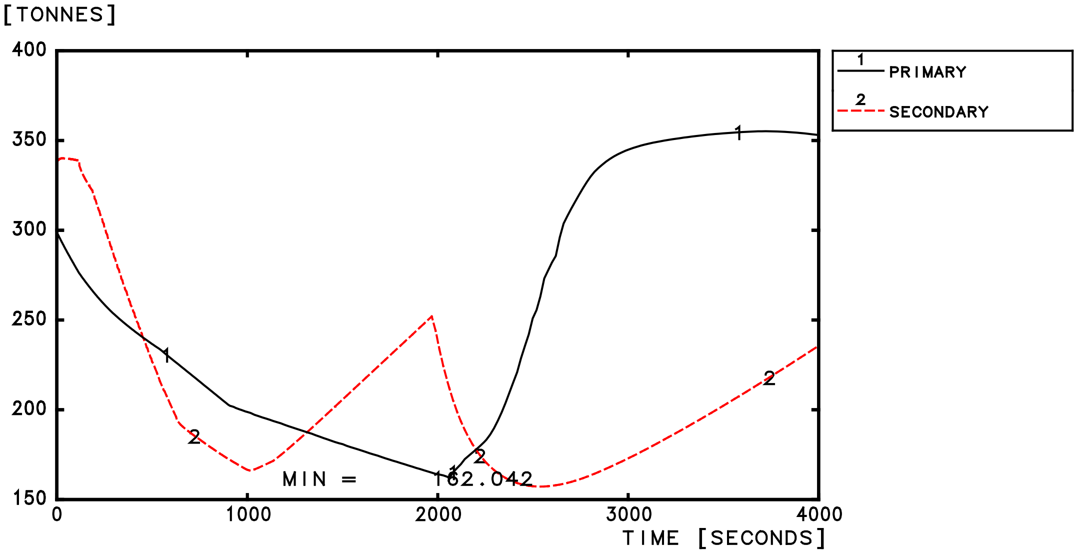
Figure - H1 - SB LOCA w/o MHSI - 3: Wide range SG levels and core temperatures



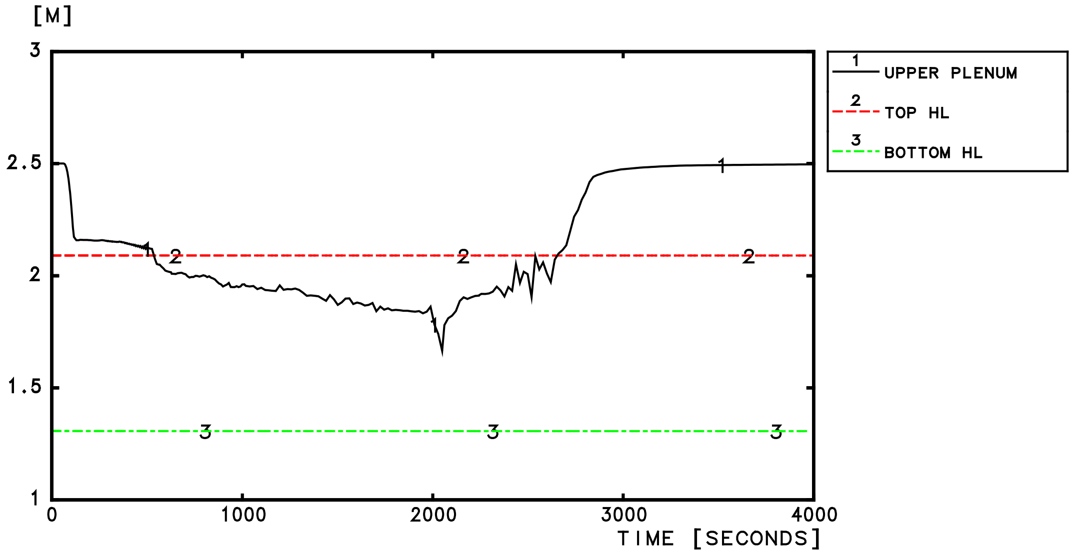


SUB-CHAPTER 16.5 – FIGURE 50

H1 - SB LOCA w/o MHSI - 4:
Primary and secondary masses and upper plenum liquid level



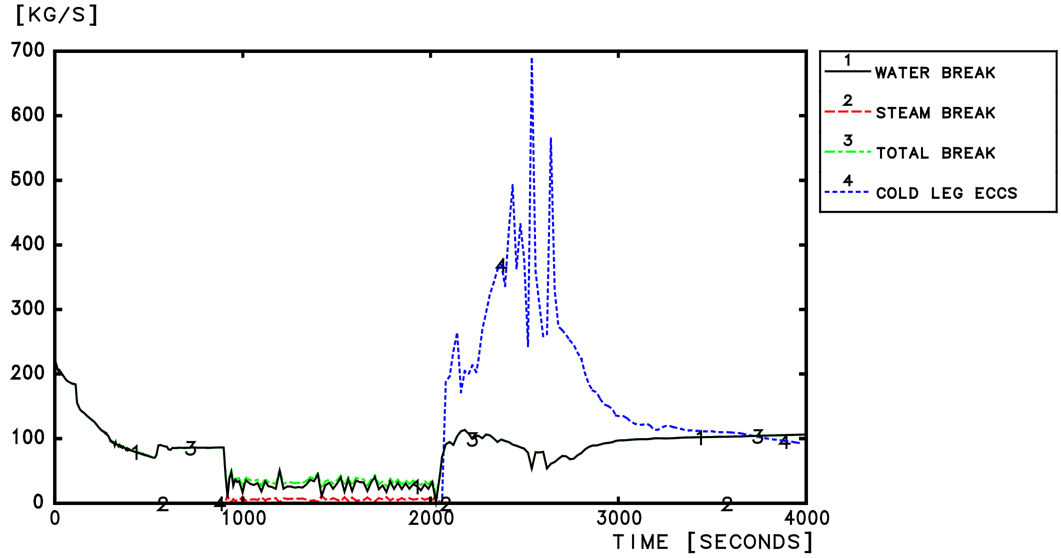
Primary and secondary Masses



Upper plenum level

SUB-CHAPTER 16.5 – FIGURE 51

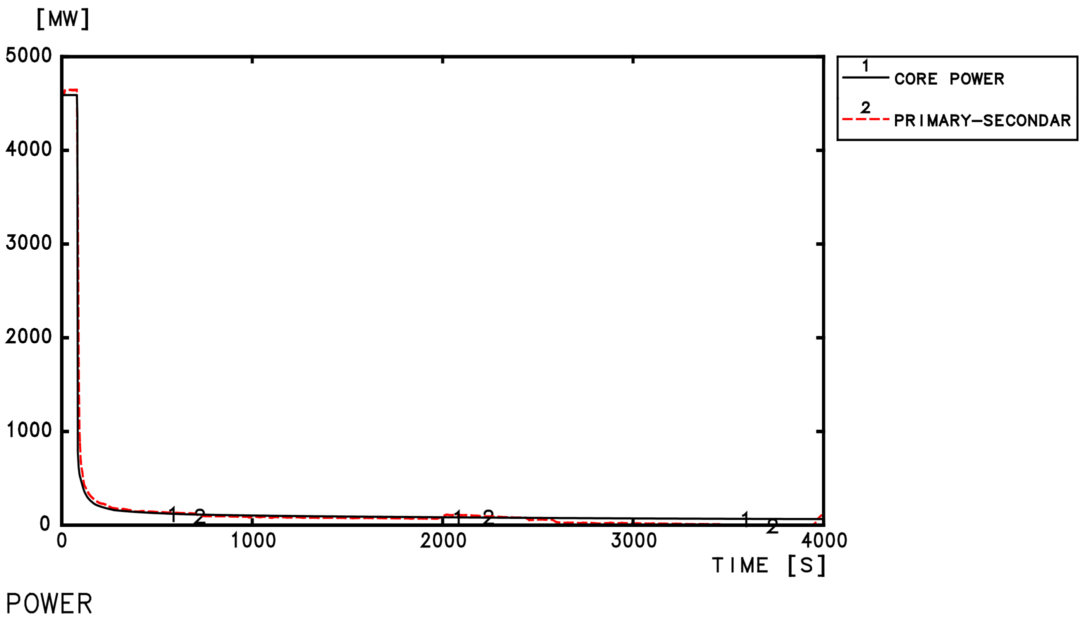
H1 - SB LOCA w/o MHSI - 5: RIS [SIS] and break flow rates

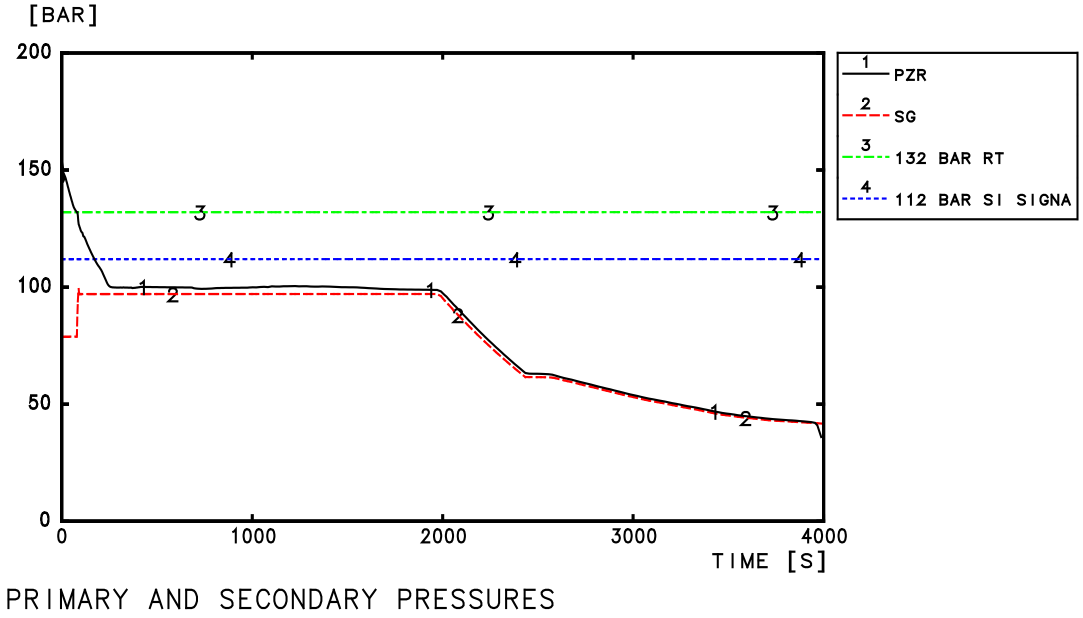


Break flowrate and total ECCS injection rate

SUB-CHAPTER 16.5 – FIGURE 52

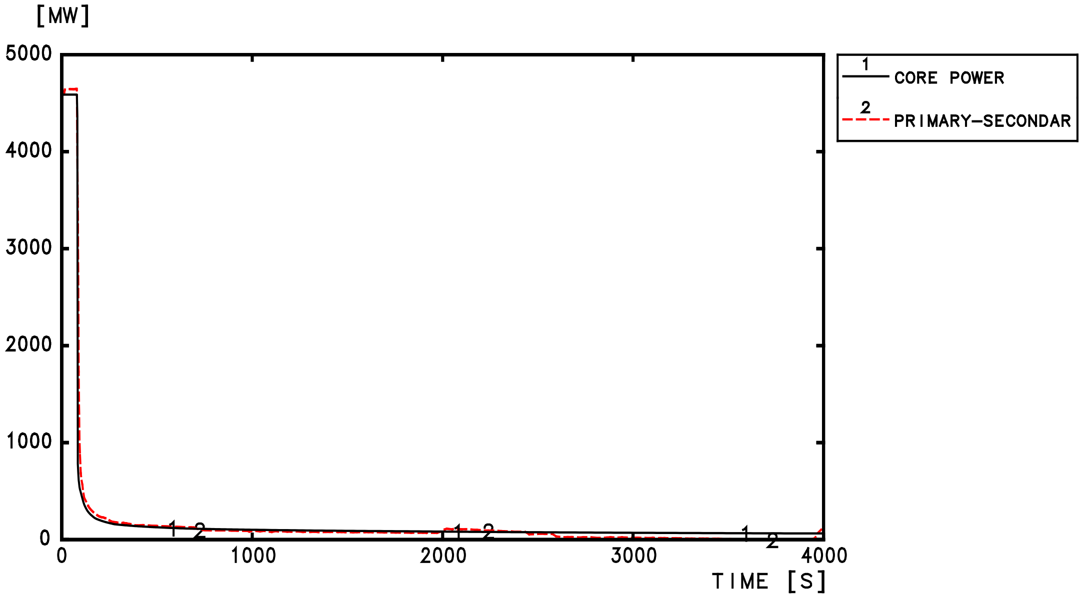
H1 - SB LOCA w/o PCD - 1: Primary and secondary exchanged power and pressures



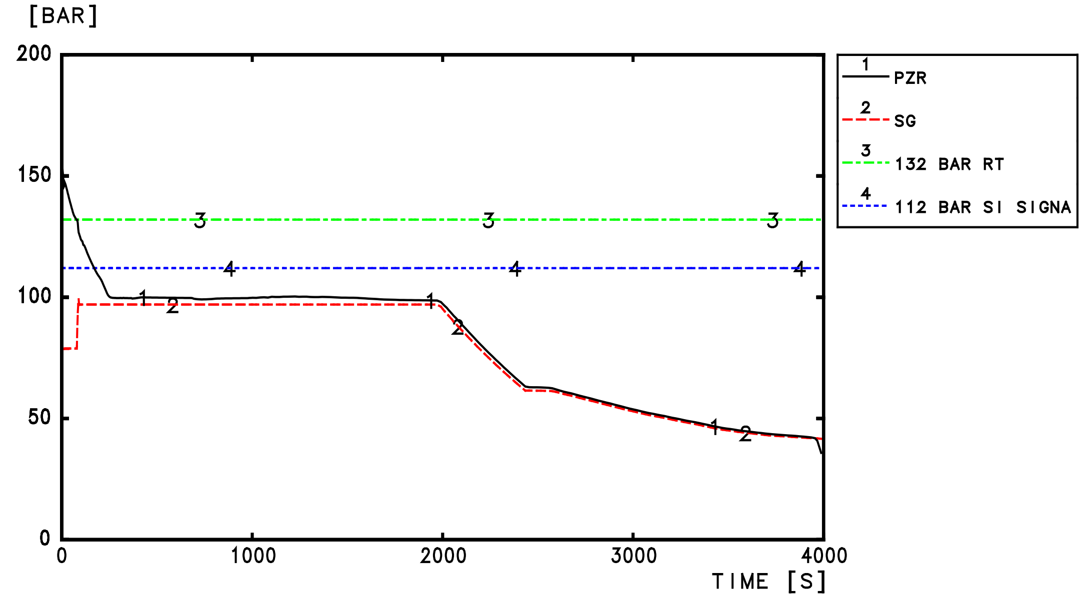


SUB-CHAPTER 16.5 – FIGURE 53

H1 - SB LOCA w/o PCD - 2: SG wide range levels and core temperatures



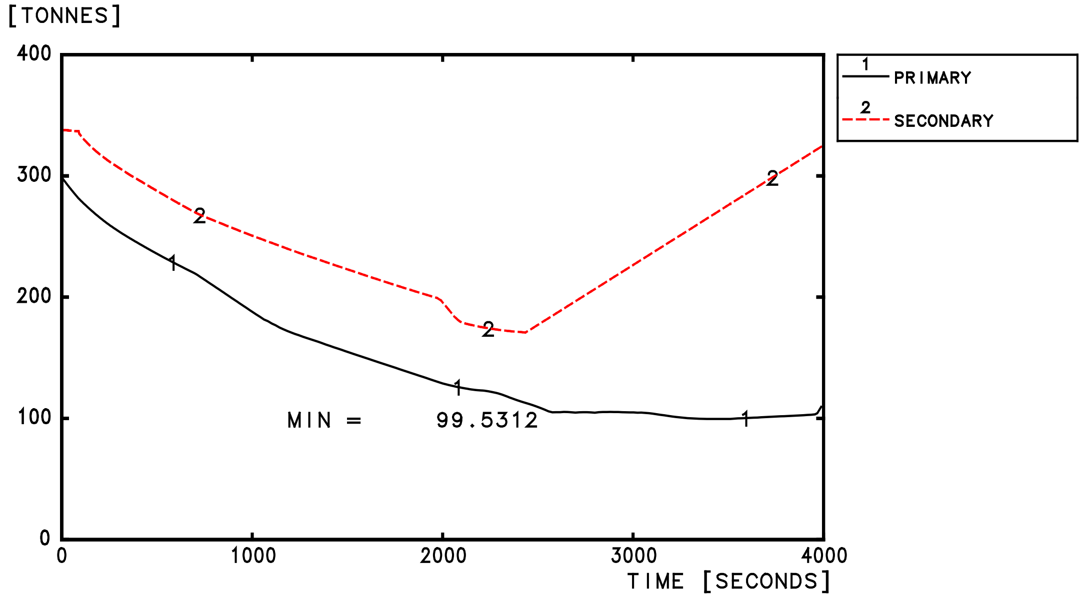
POWER



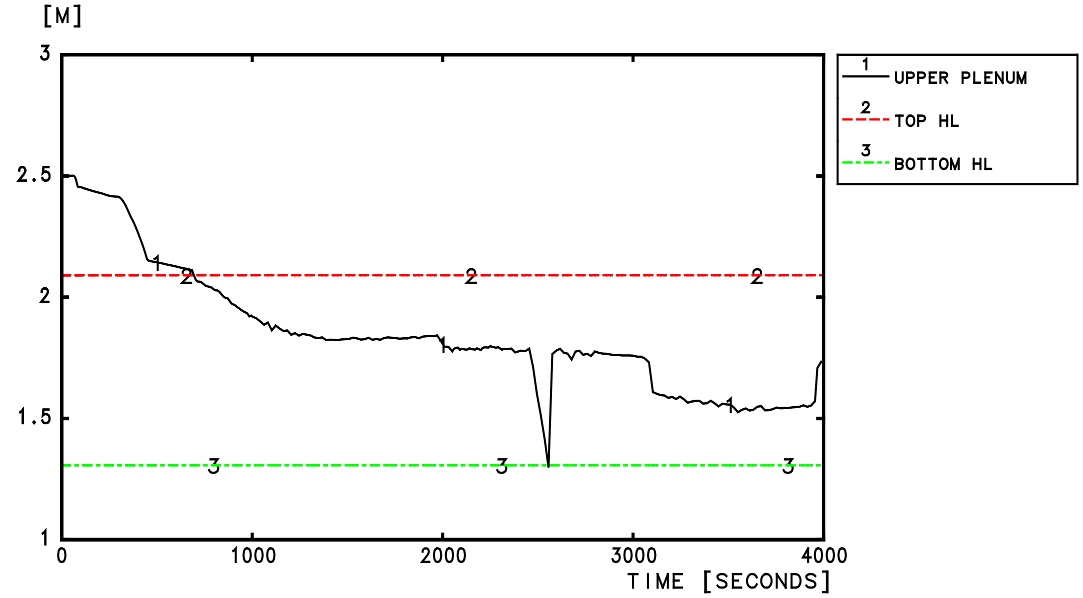
PRIMARY AND SECONDARY PRESSURES

SUB-CHAPTER 16.5 – FIGURE 54

H1 - SB LOCA w/o PCD - 3: Primary and secondary masses and upper plenum liquid level



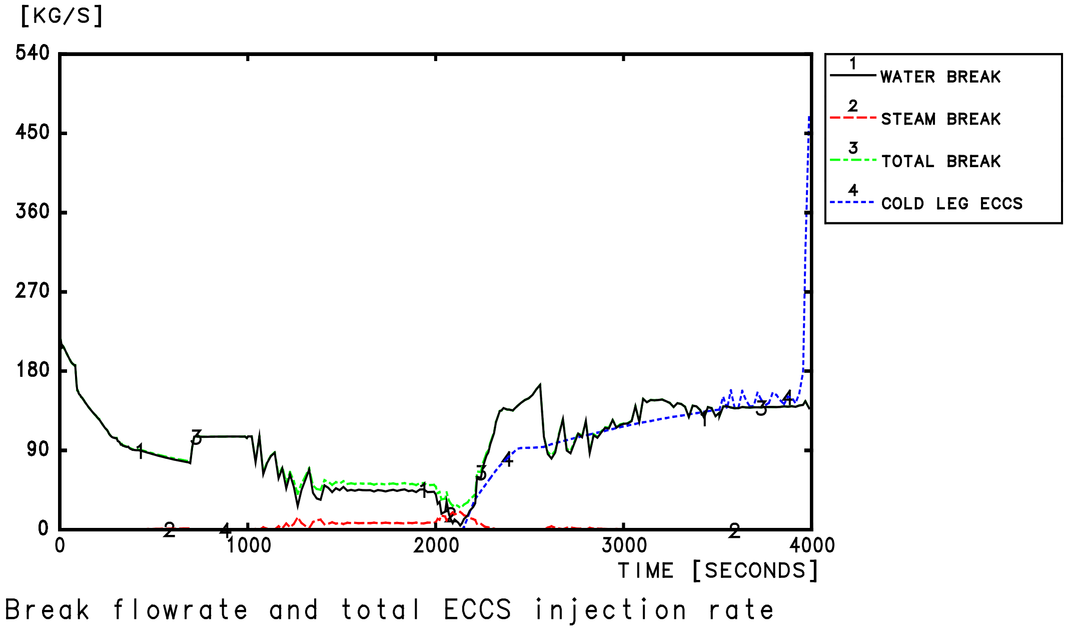
Primary and secondary Masses



Upper plenum level

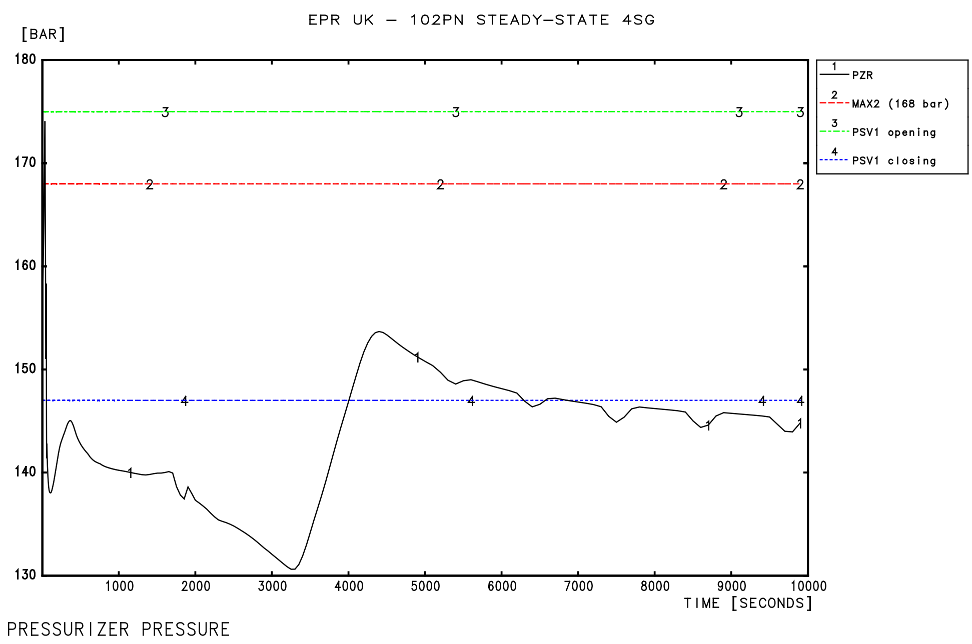
SUB-CHAPTER 16.5 – FIGURE 55

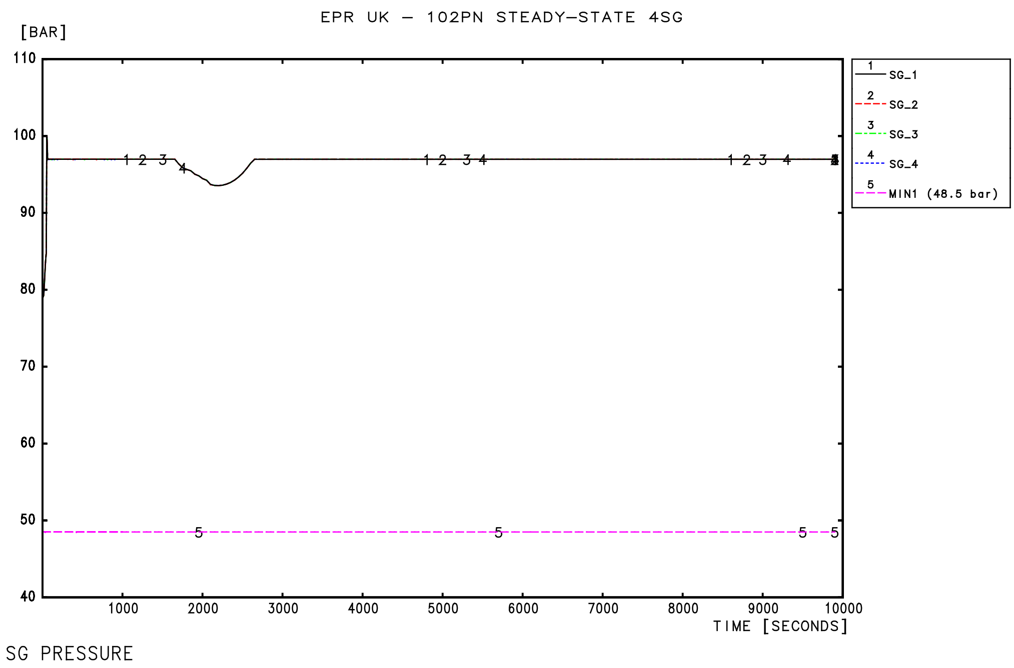
H1 - SB LOCA w/o PCD - 4: RIS [SIS] and break flow rates



SUB-CHAPTER 16.5 – FIGURE 56

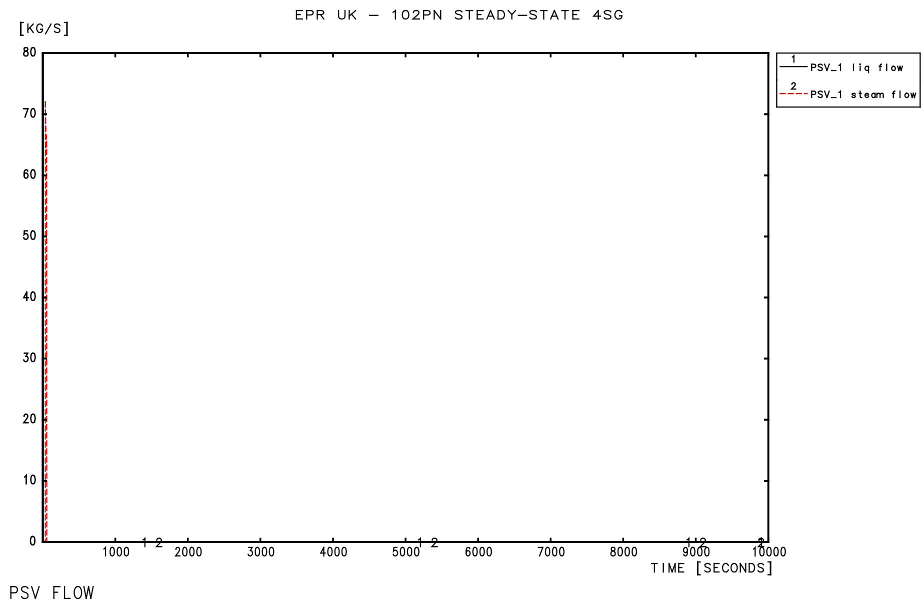
H2 - LOFW w/o reactor coolant pumps - 1: Primary and secondary pressures

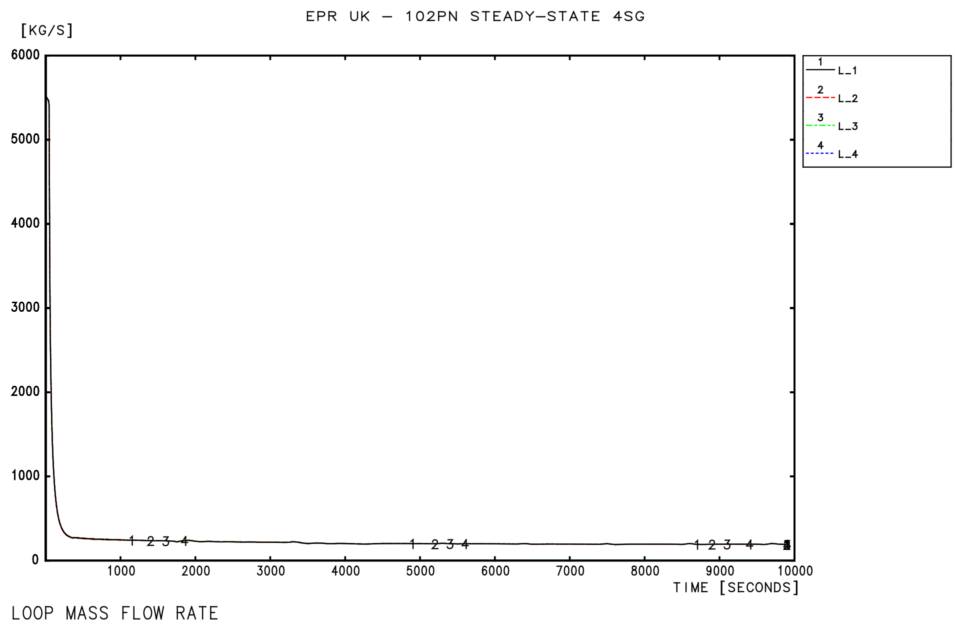




SUB-CHAPTER 16.5 – FIGURE 57

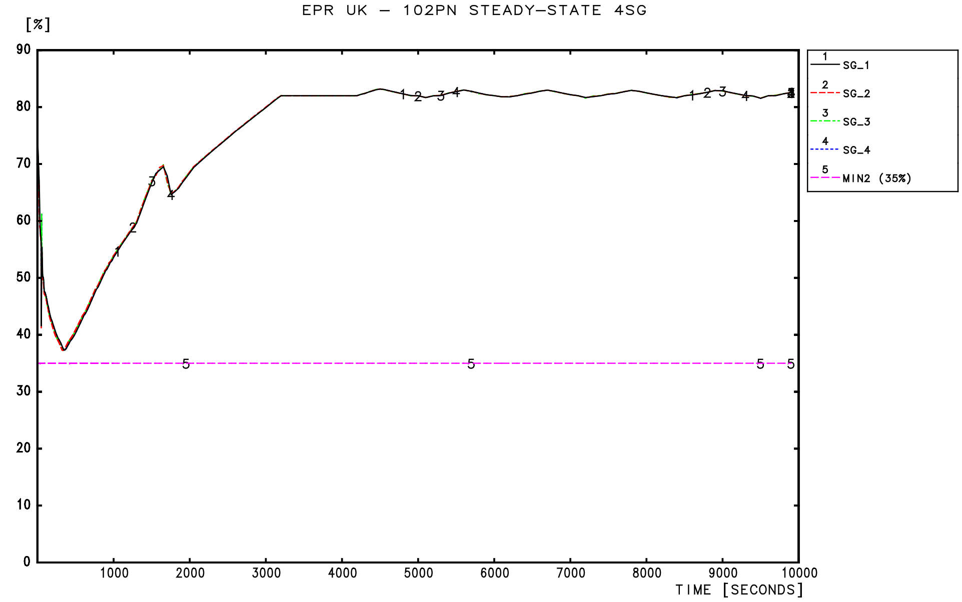
H2 - LOFW w/o reactor coolant pumps - 2: PSV flow rates and loop mass flow rate



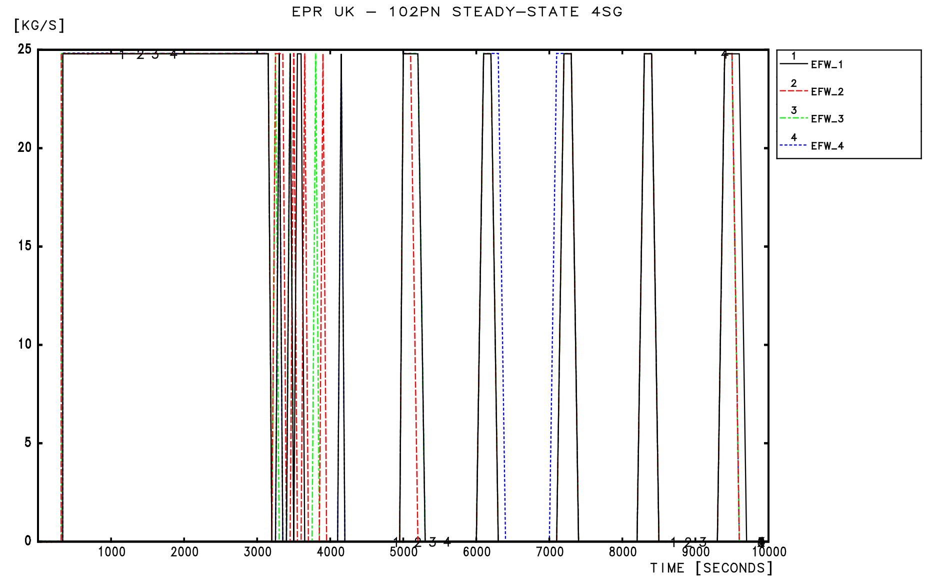


SUB-CHAPTER 16.5 – FIGURE 58

H2 - LOFW w/o reactor coolant pumps - 3: Wide range SG levels and ASG [EFWS] flow rate



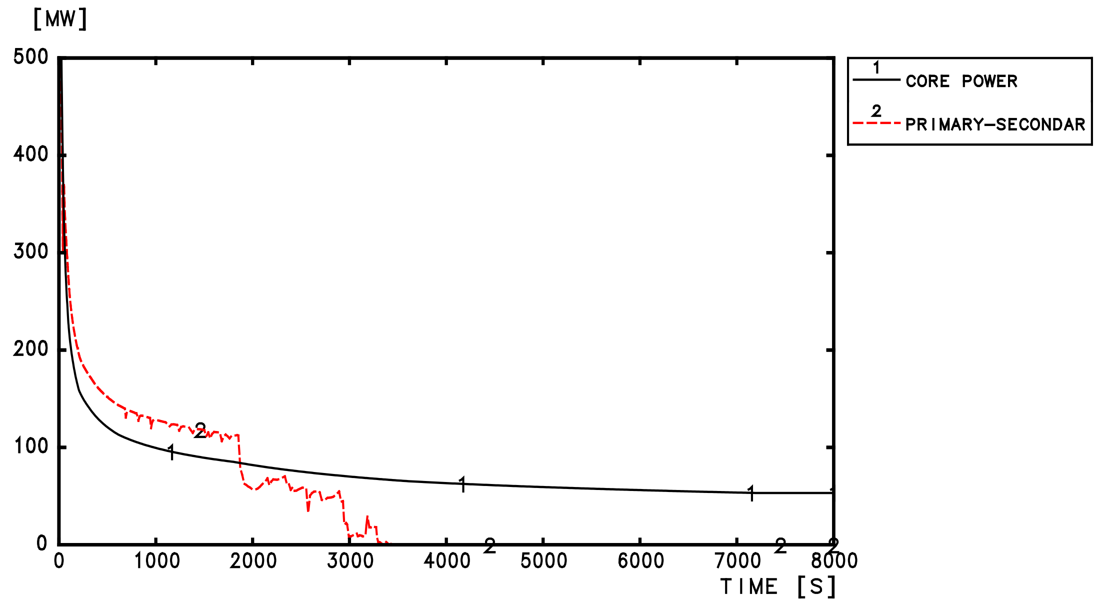
WIDE RANGE SG MEASURED LEVEL



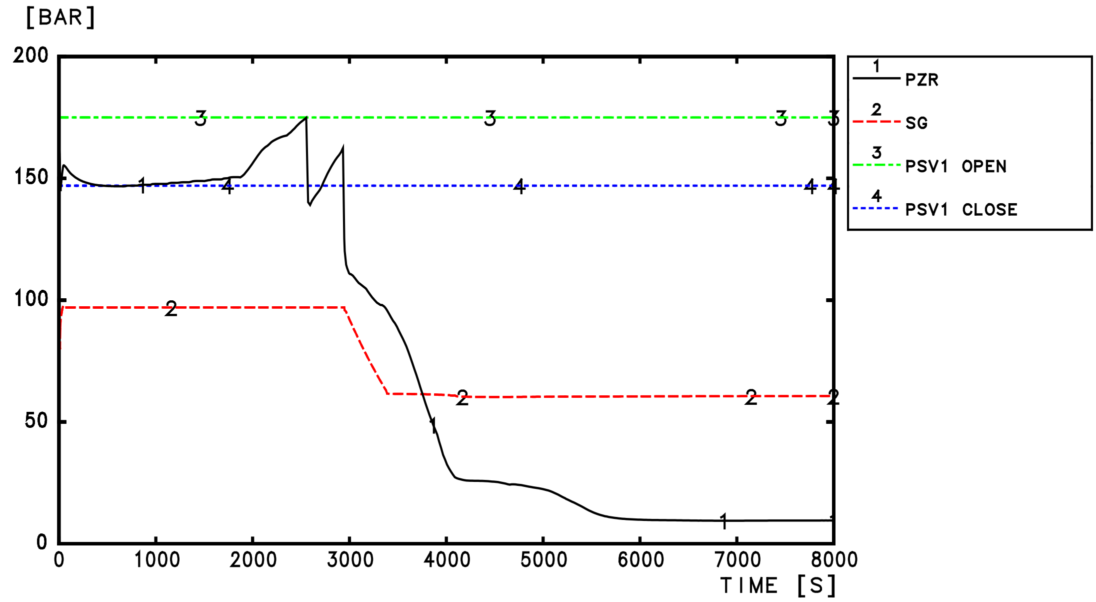
EFWS FLOW

SUB-CHAPTER 16.5 – FIGURE 59

H3 - TLOFW - 1: Primary and secondary exchanged power and pressures



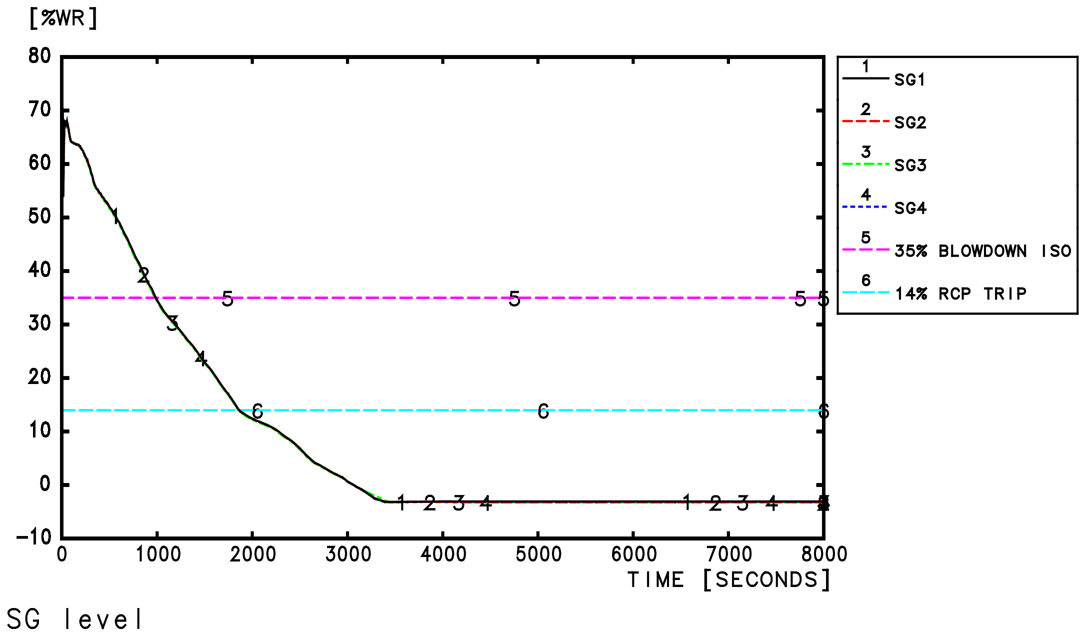
POWER

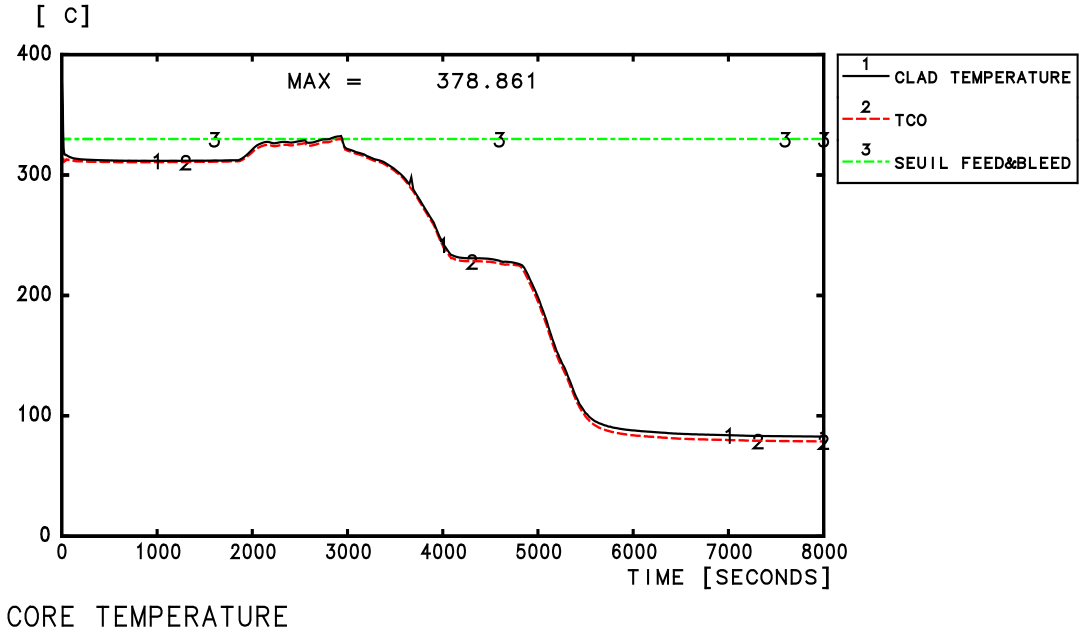


Primary pressures

SUB-CHAPTER 16.5 – FIGURE 60

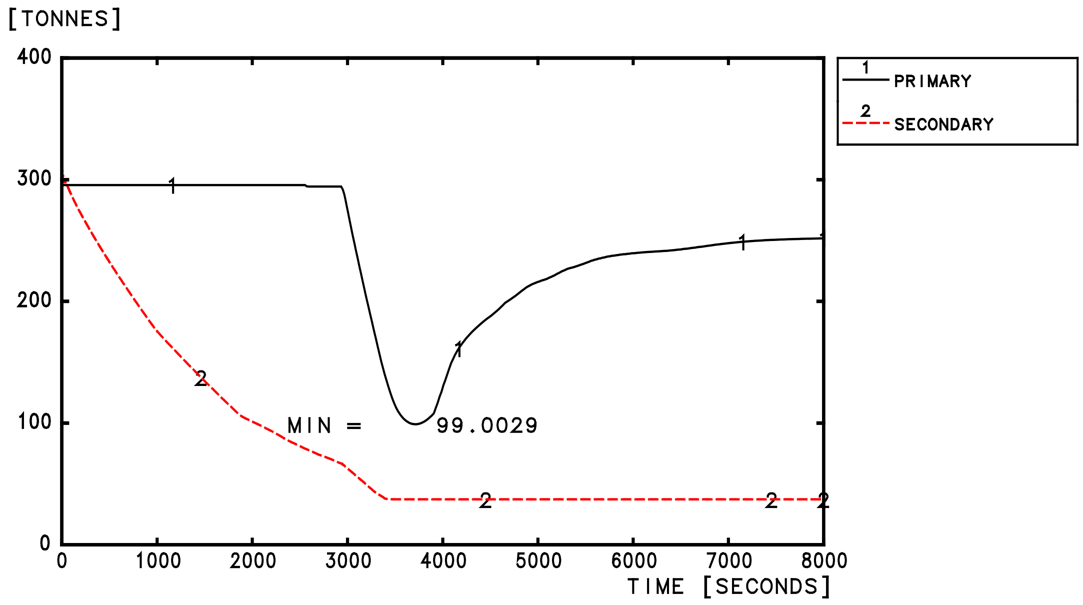
H3 - TLOFW - 2: SG wide range levels and core temperatures



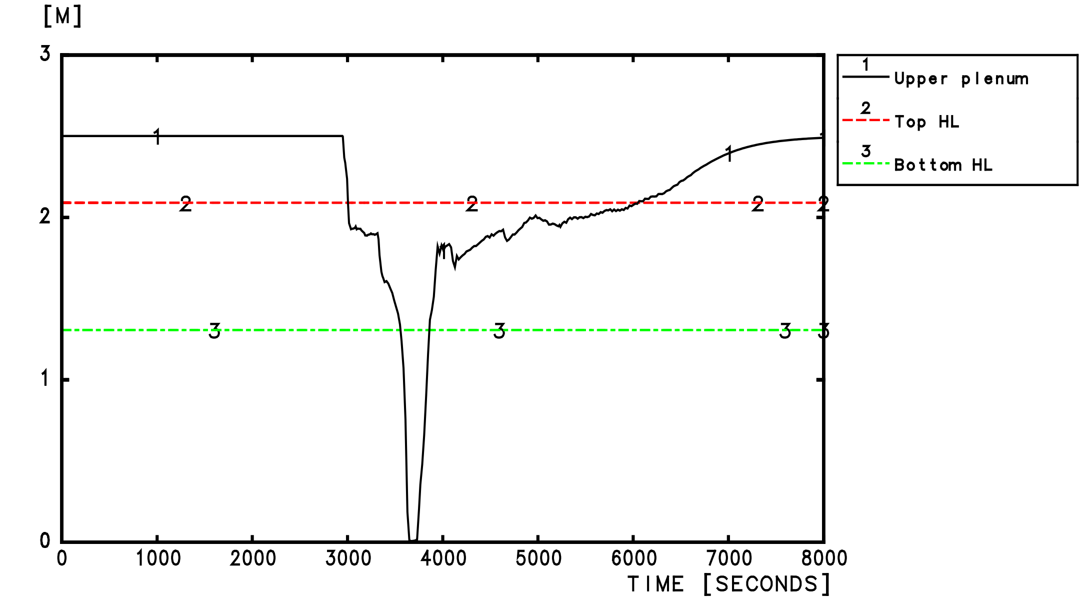


SUB-CHAPTER 16.5 – FIGURE 61

H3 - TLOFW - 3: Primary and secondary masses and upper plenum liquid level



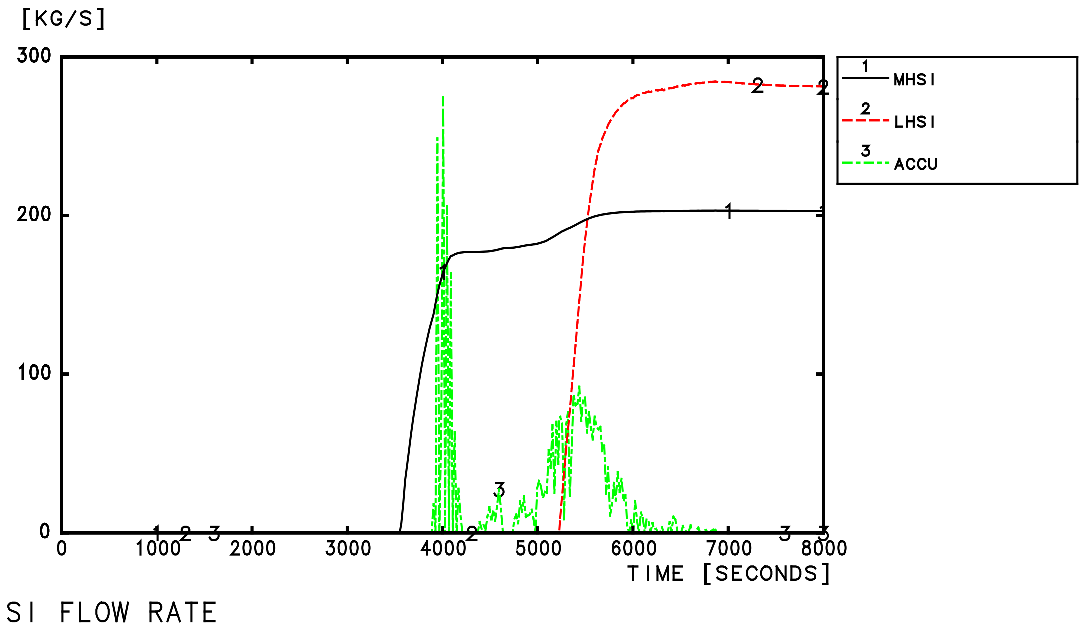
Primary and secondary Masses

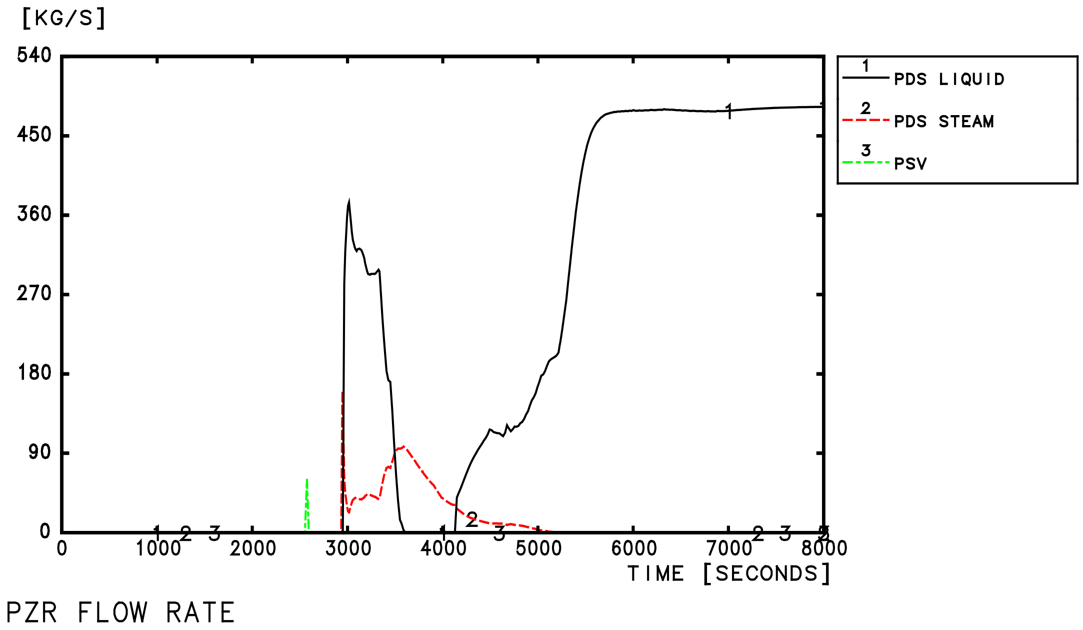


Upper plenum level

SUB-CHAPTER 16.5 – FIGURE 62

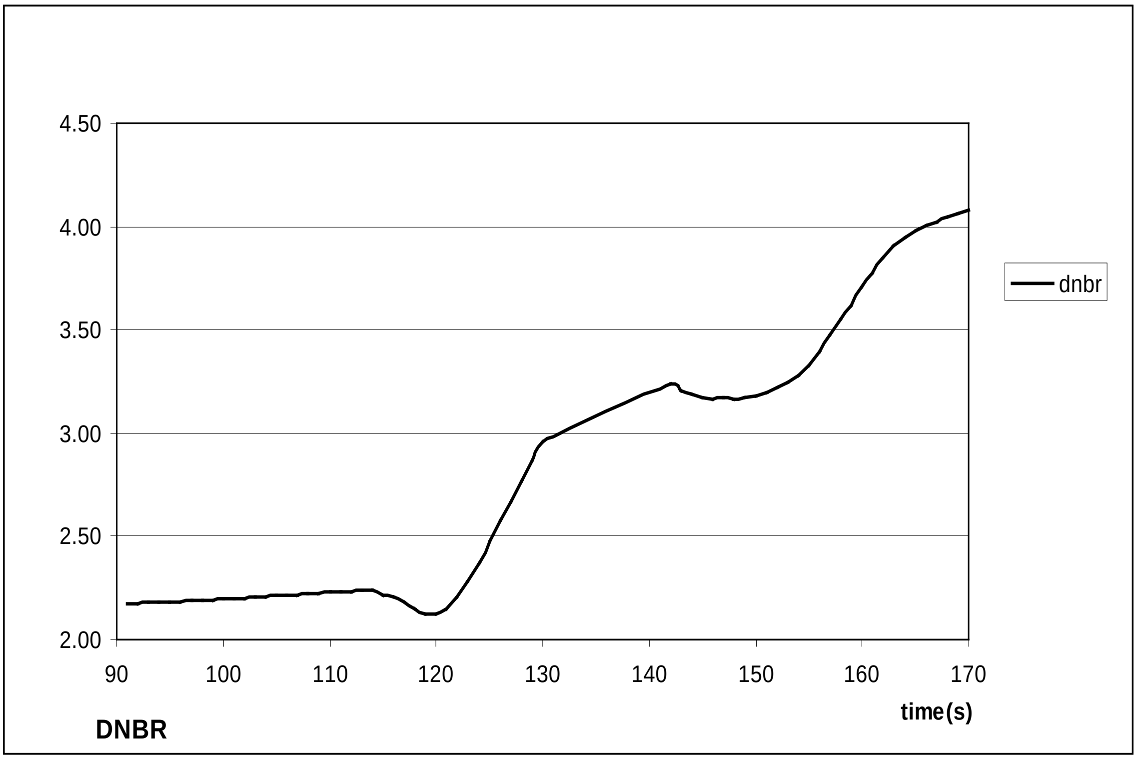
H3 - TLOFW - 4: RIS [SIS], PSV and PDS flow rate





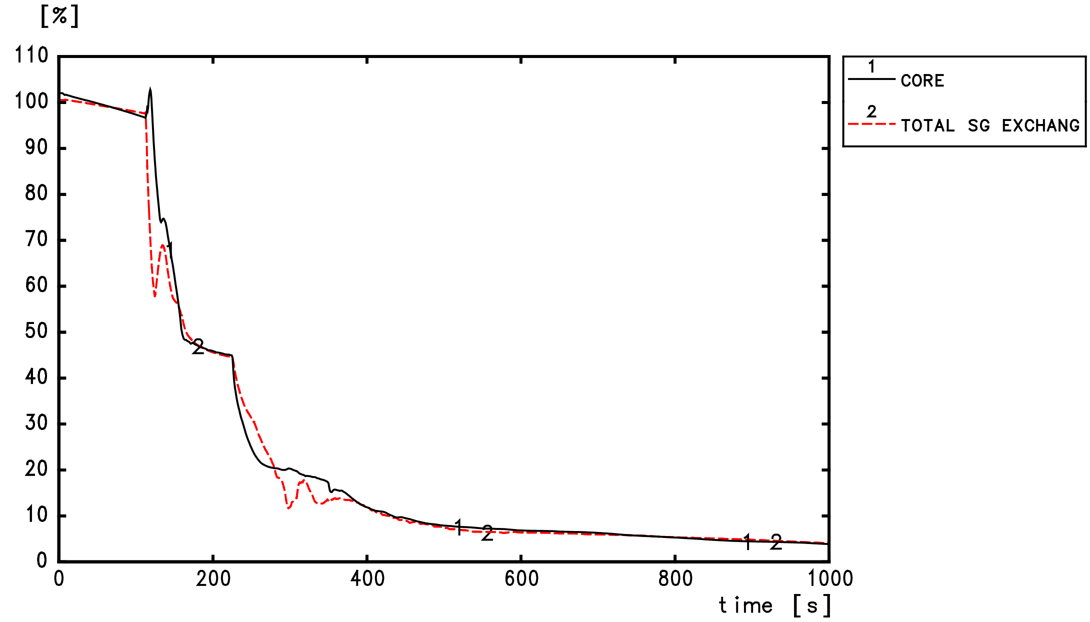
SUB-CHAPTER 16.5 – FIGURE 63

C1 - ATWS spray - 1: DNBR transient

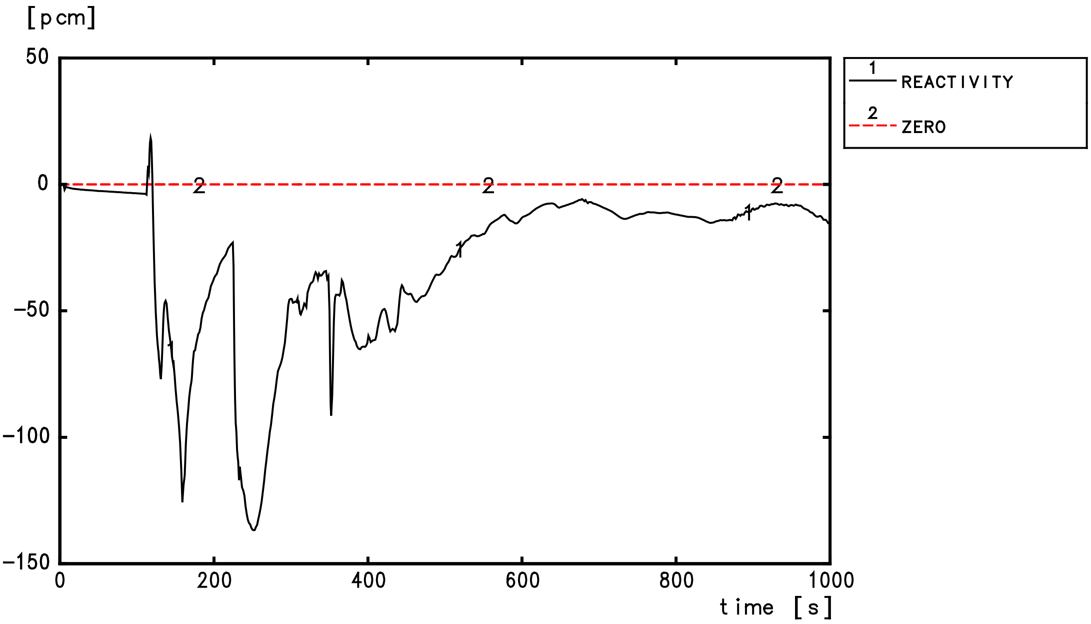


SUB-CHAPTER 16.5 – FIGURE 64

C1 - ATWS spray - 2: Core power and reactivity



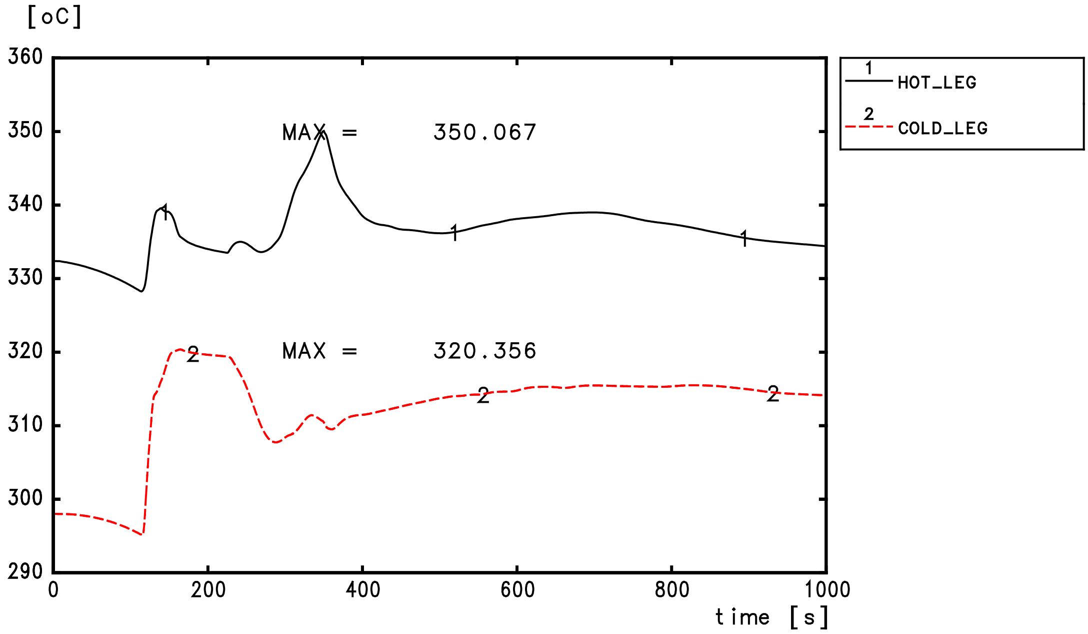
CORE POWER



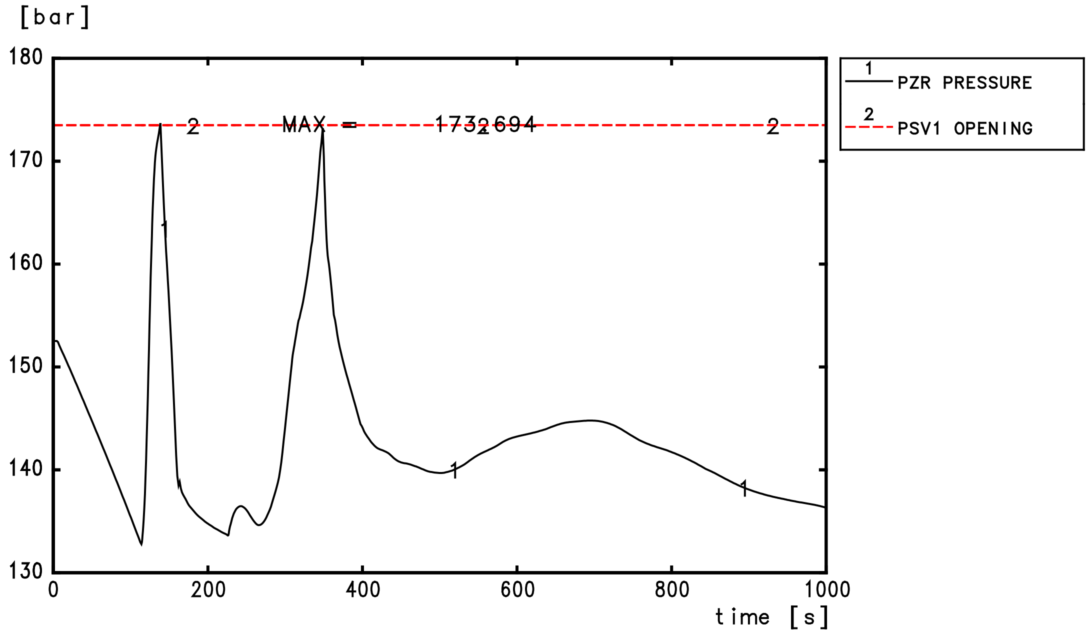
REACTIVITY

SUB-CHAPTER 16.5 – FIGURE 65

C1 - ATWS spray - 3: Primary temperatures and pressuriser pressure



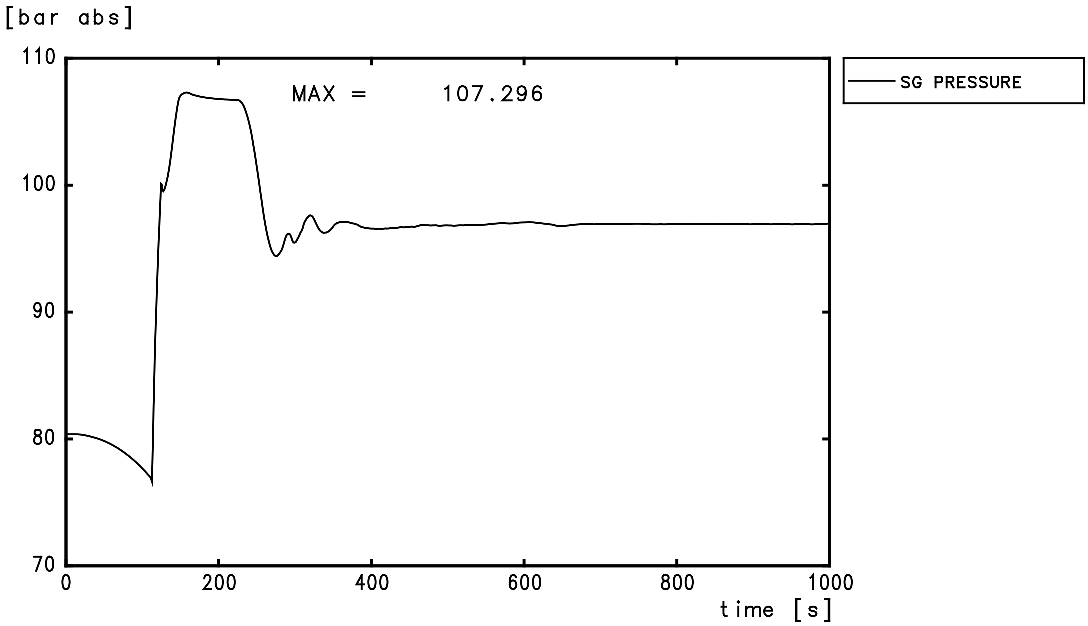
PRIMARY TEMPERATURES



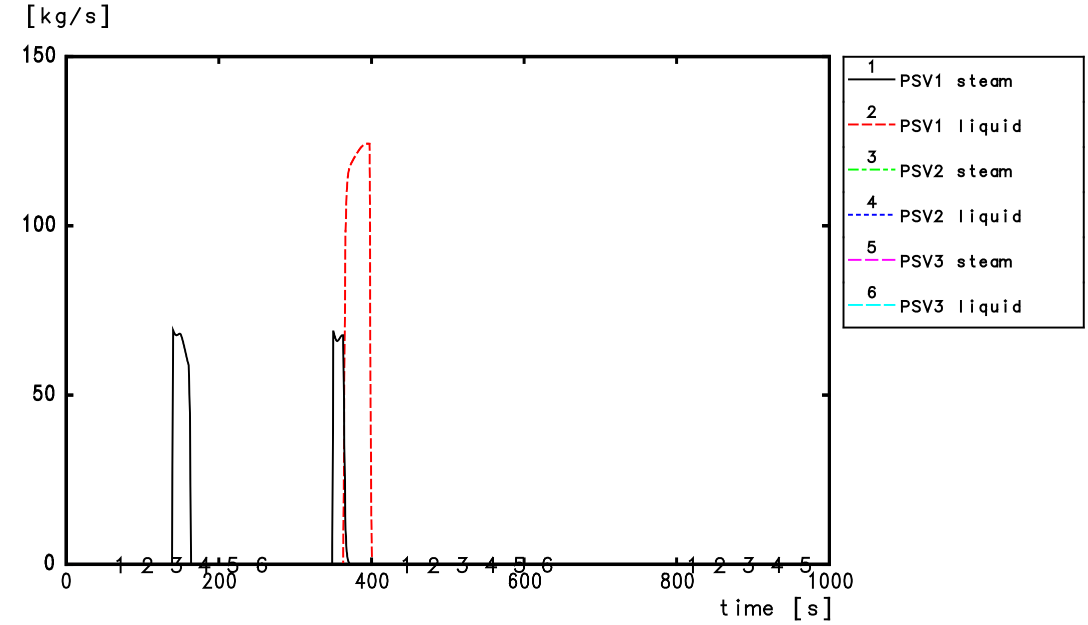
PZR PRESSURE

SUB-CHAPTER 16.5 – FIGURE 66

C1 - ATWS spray - 4: Secondary side pressure and PSV flow rate



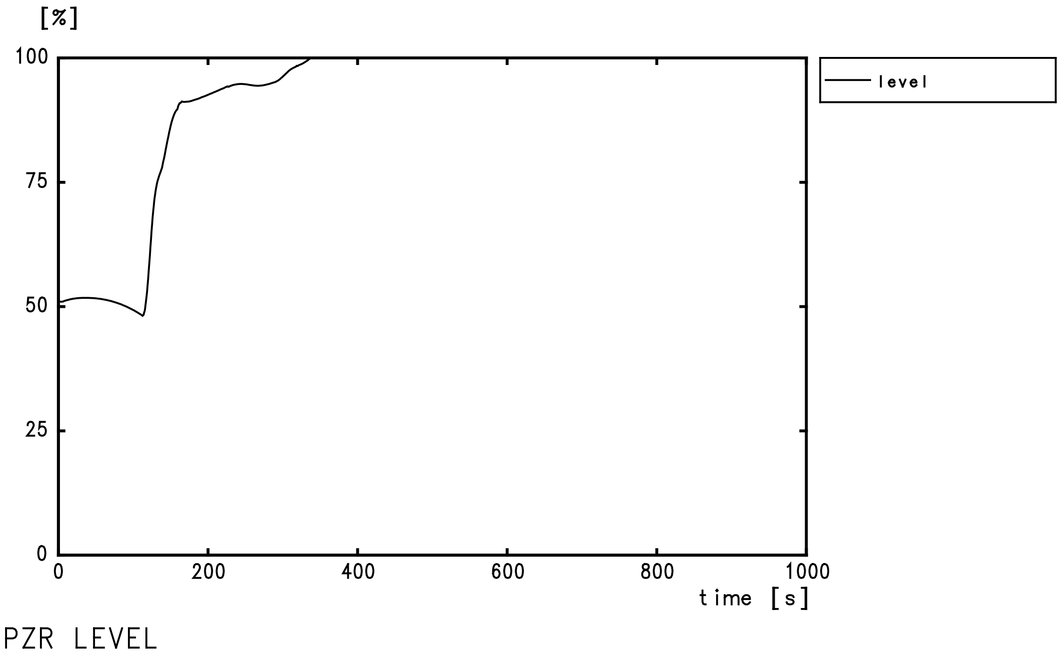
SECONDARY SIDE PRESSURES

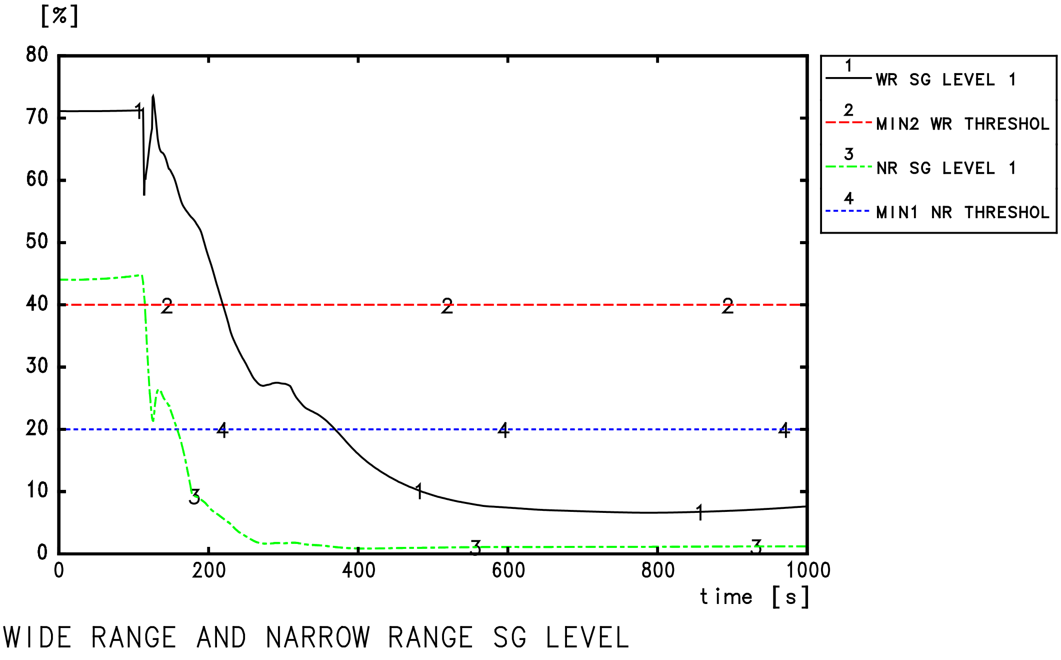


PSV FLOWRATES

SUB-CHAPTER 16.5 – FIGURE 67

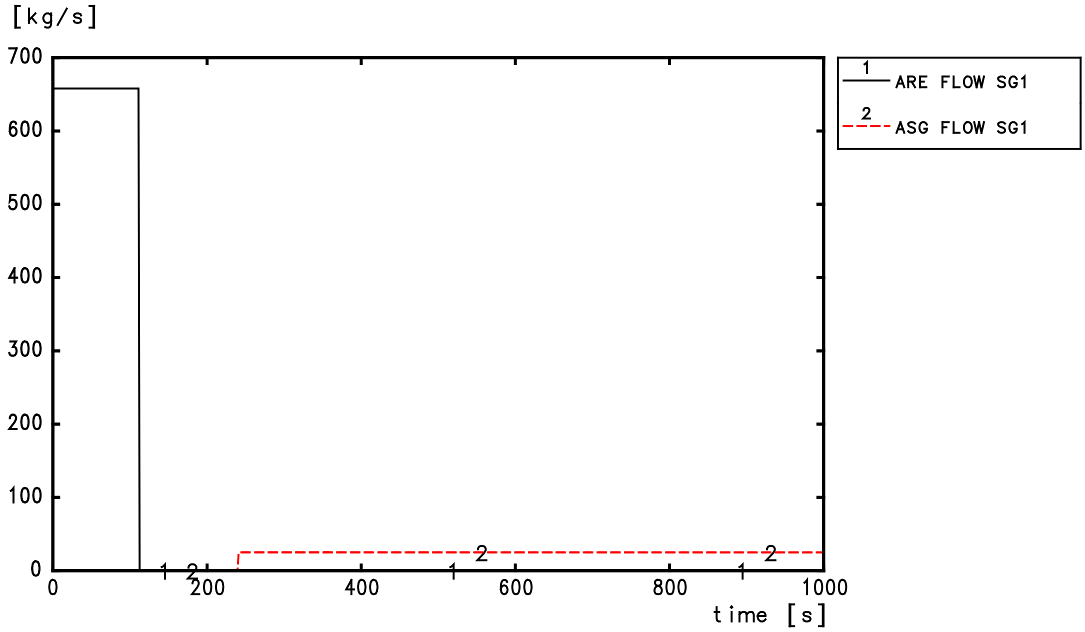
C1 - ATWS spray - 5: Pressuriser and SG levels



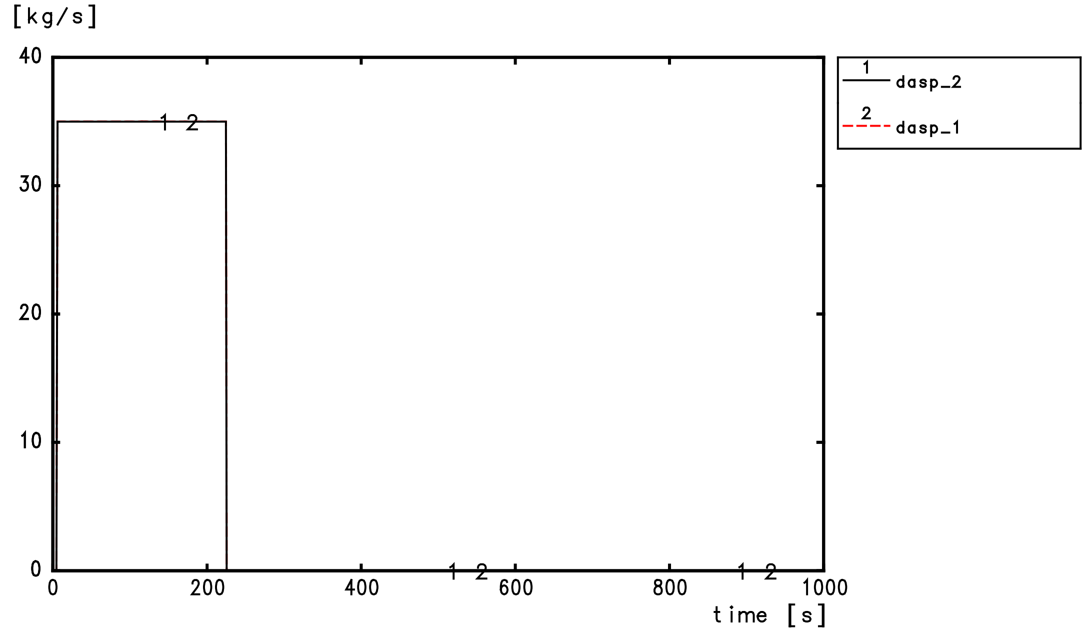


SUB-CHAPTER 16.5 – FIGURE 68

C1 - ATWS spray - 6: Feedwater flow rate and spray flow rate



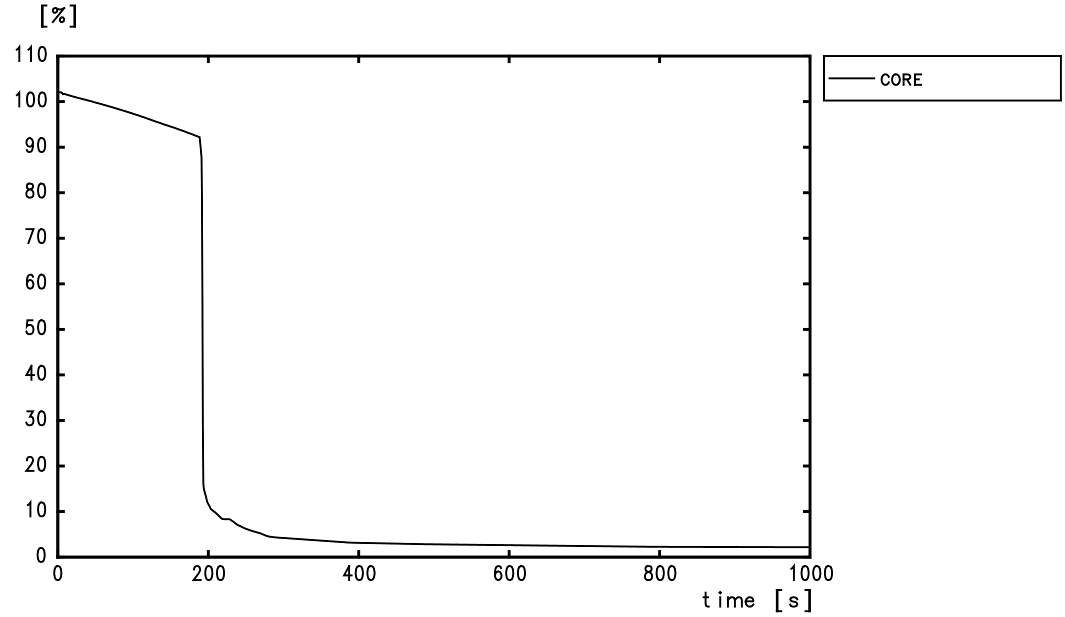
FEEDWATER FLOWRATE



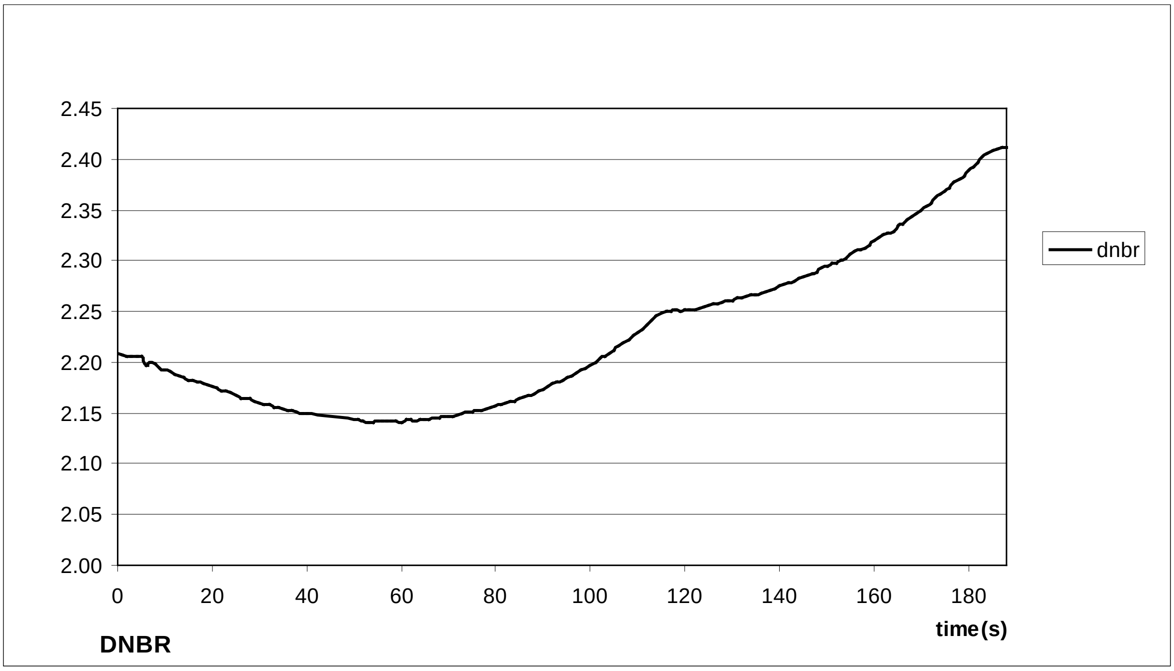
SPRAY FLOWRATE

SUB-CHAPTER 16.5 – FIGURE 69

Figure - C1 - ATWS spray w/o RPR [PS] - 1: Core power and DNBR

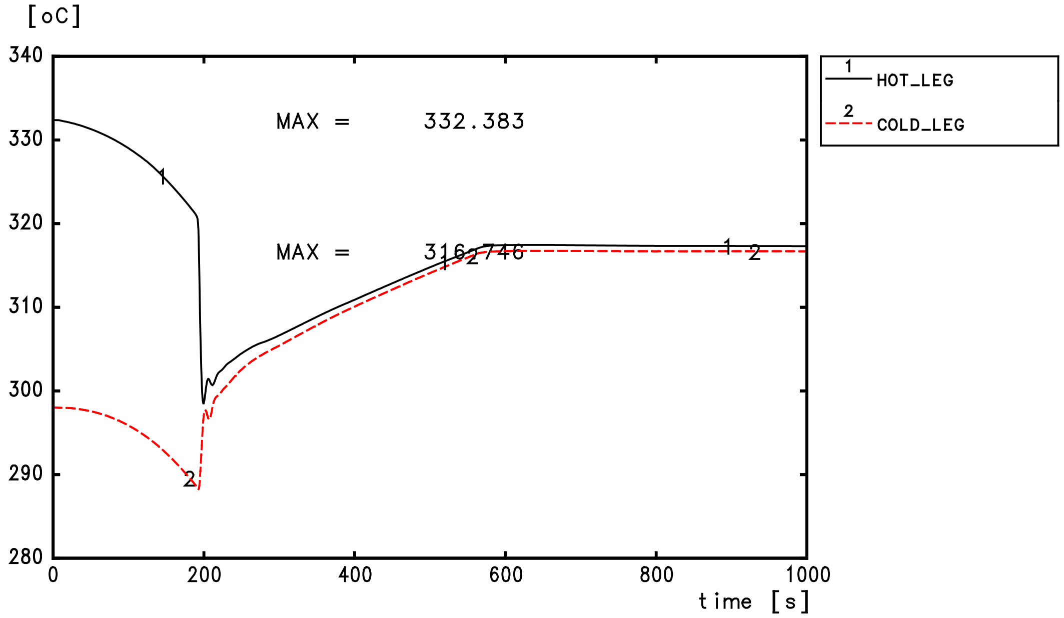


CORE POWER

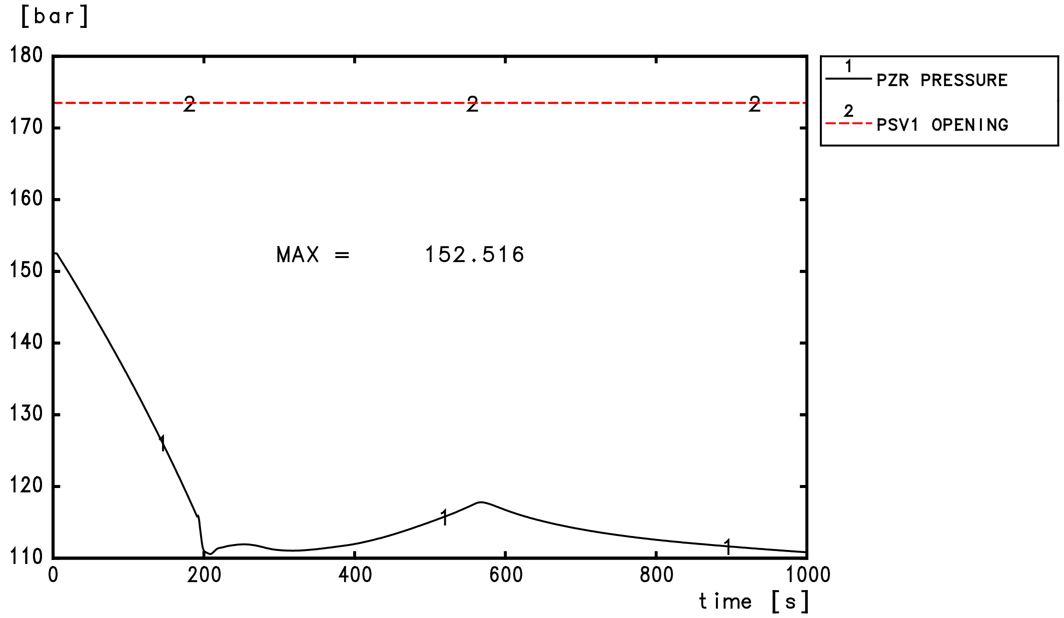


SUB-CHAPTER 16.5 – FIGURE 70

C1 - ATWS spray w/o RPR [PS] - 2: RCP [RCS] temperatures and pressuriser pressure



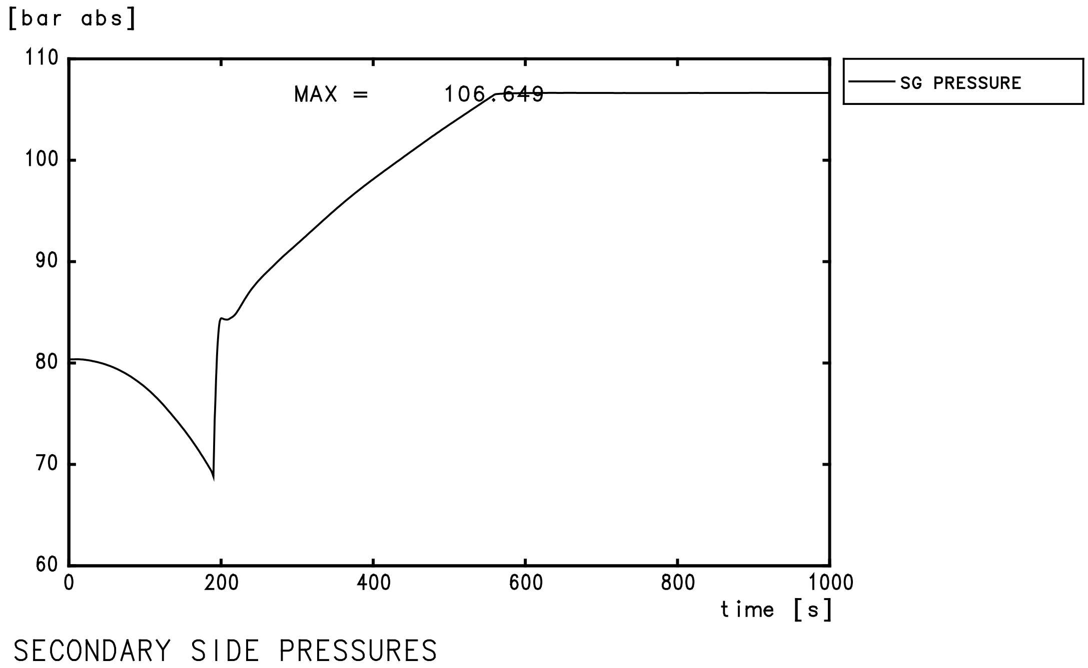
PRIMARY TEMPERATURES

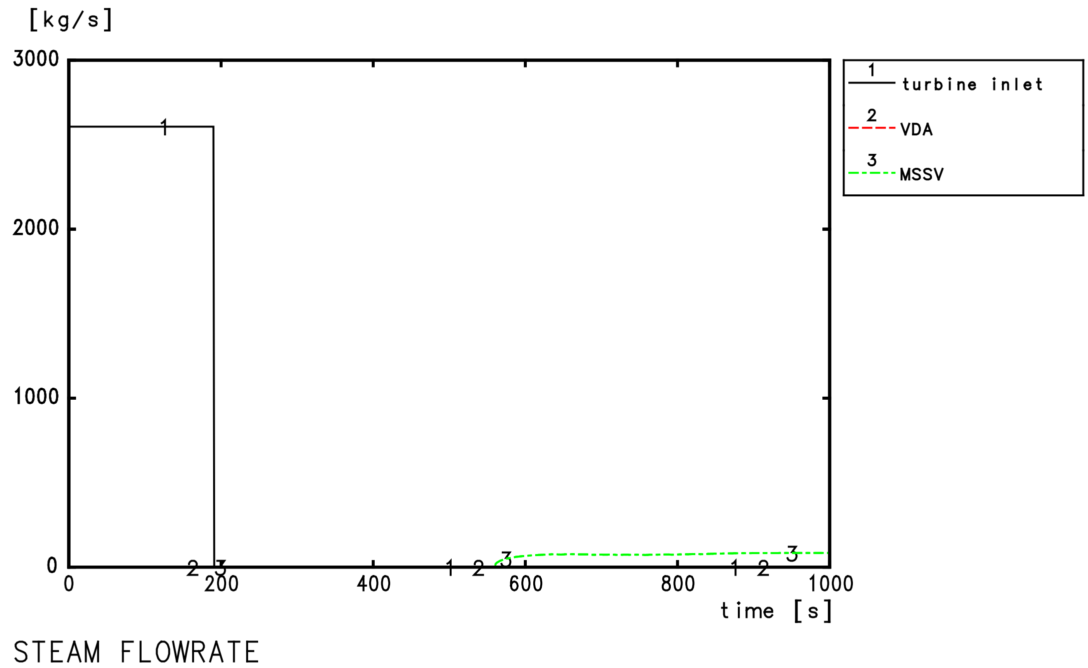


PZR PRESSURE

SUB-CHAPTER 16.5 – FIGURE 71

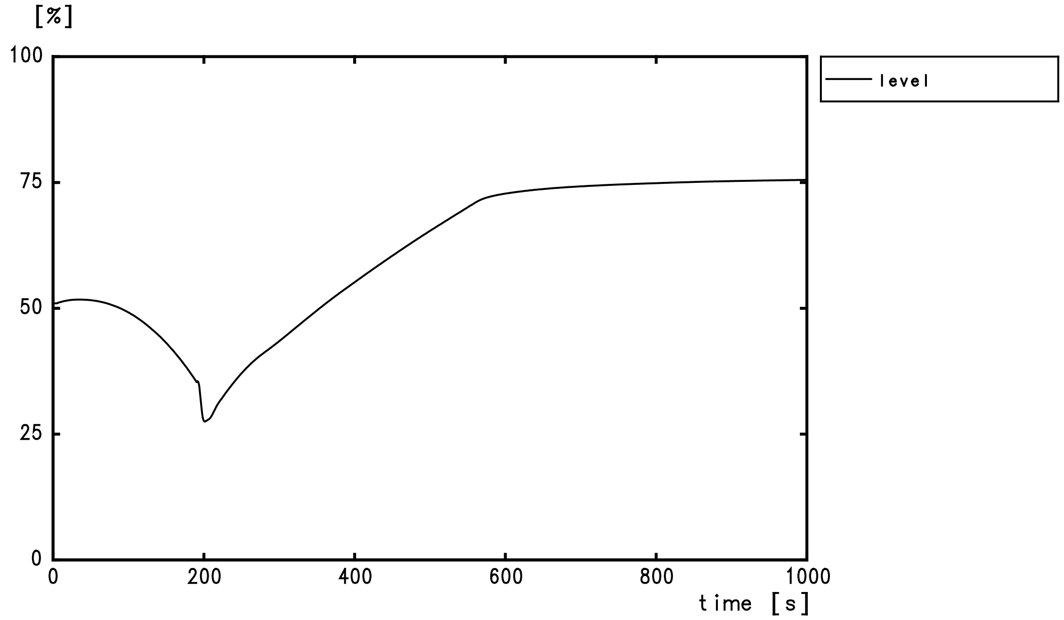
C1 - ATWS spray w/o RPR [PS] - 3: secondary side pressure and steam flow rate



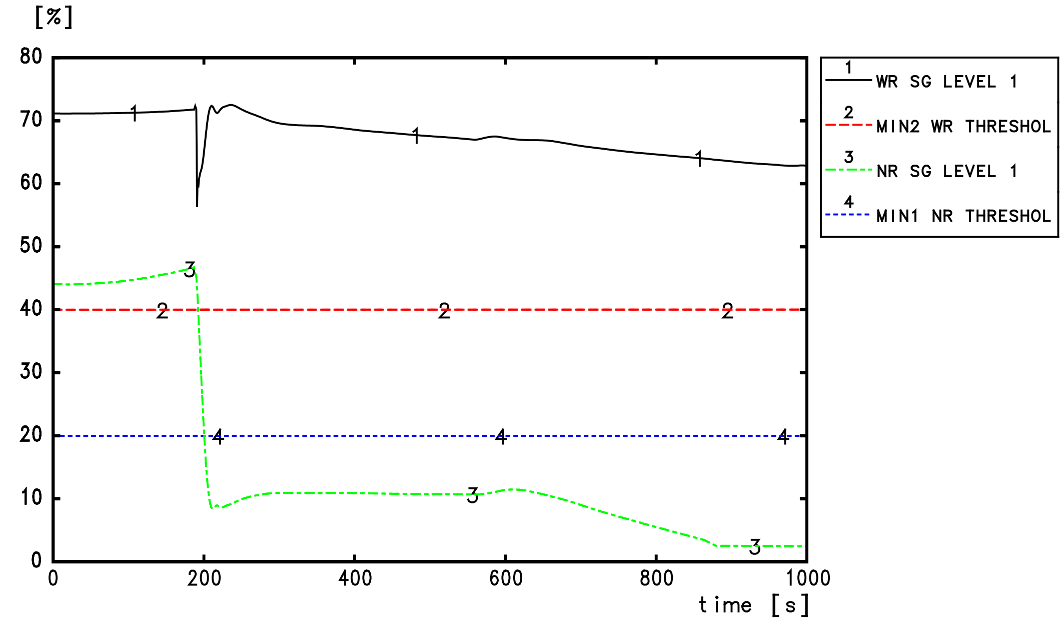


SUB-CHAPTER 16.5 – FIGURE 72

C1 - ATWS spray w/o RPR [PS] - 4: Pressuriser and SG levels



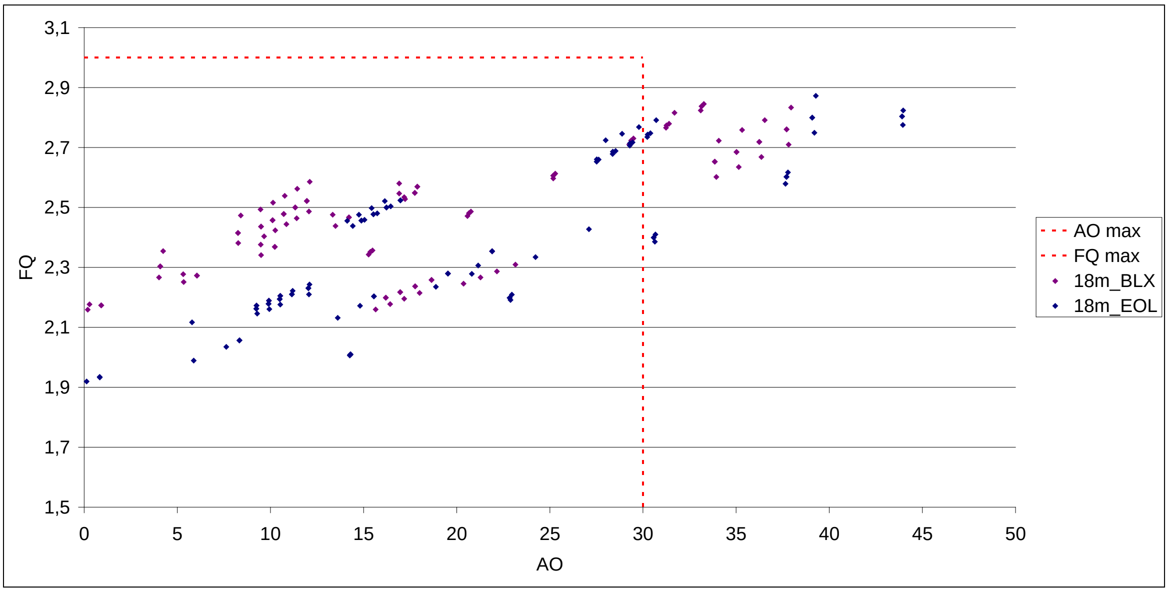
PZR LEVEL



WIDE RANGE AND NARROW RANGE SG LEVEL

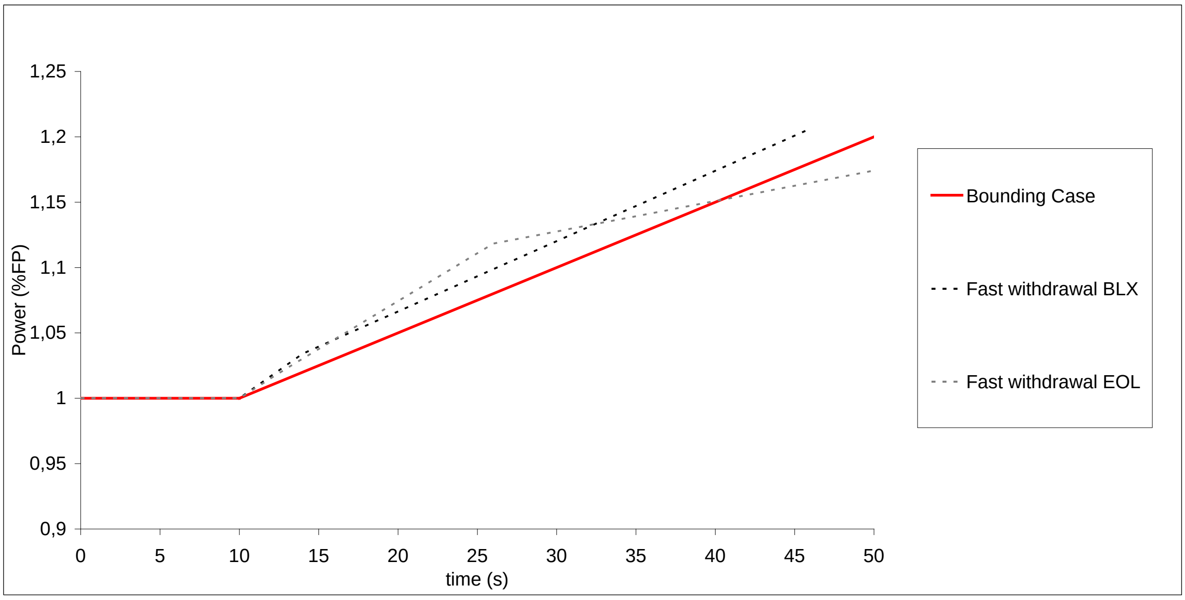
SUB-CHAPTER 16.5 – FIGURE 73

C1 - ATWS URBWP w/o TXS - 1: Flyspeck of Heat Flux Hot Channel factor function of AO



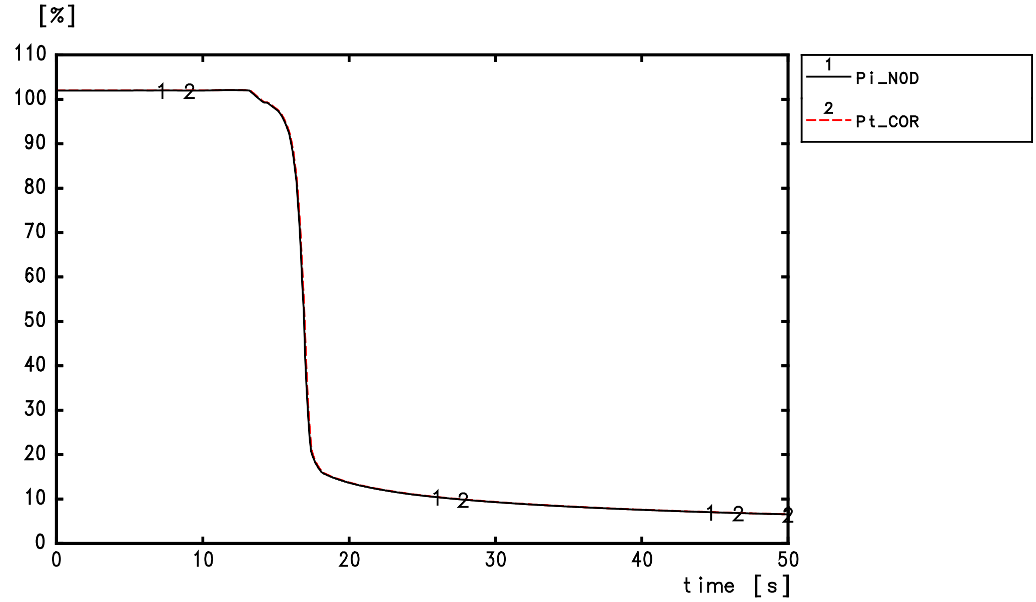
SUB-CHAPTER 16.5 – FIGURE 74

C1 - ATWS URBWP w/o TXS - 2: Bounding case nuclear power

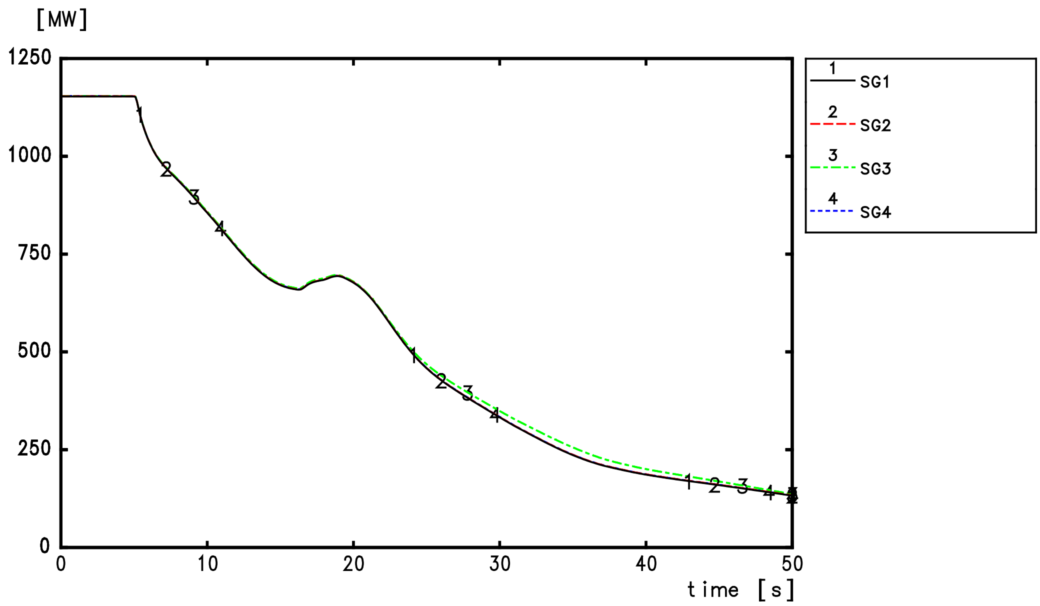


SUB-CHAPTER 16.5 – FIGURE 75

C2 - 4 VIVs [MSIV]s w/o PSVs - 1:
Core power and primary/secondary power exchange – with normal spray



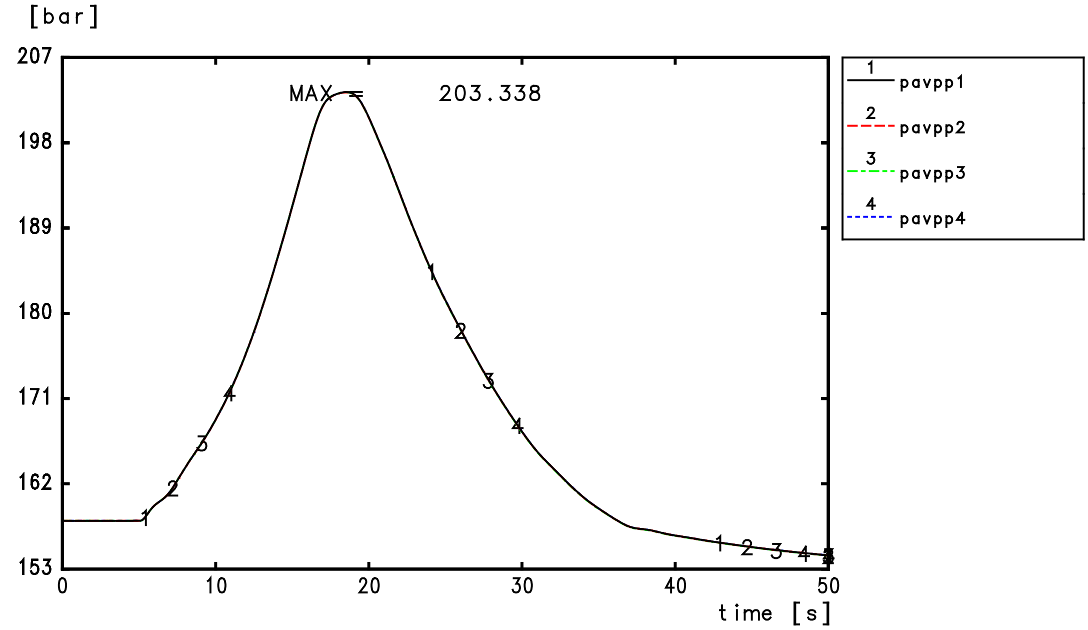
CORE POWER



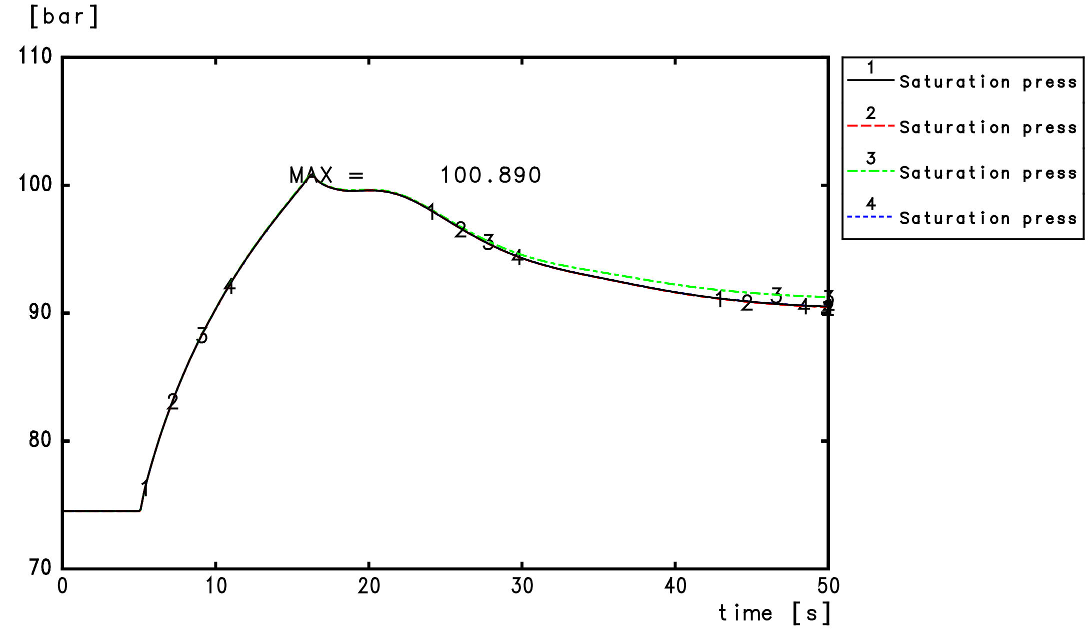
PRIMARY/SECONDARY POWER EXCHANGE

SUB-CHAPTER 16.5 – FIGURE 76

C2 - 4 VIVs [MSIV]s w/o PSVs - 2:
Reactor coolant pump outlets and secondary pressures – with normal spray



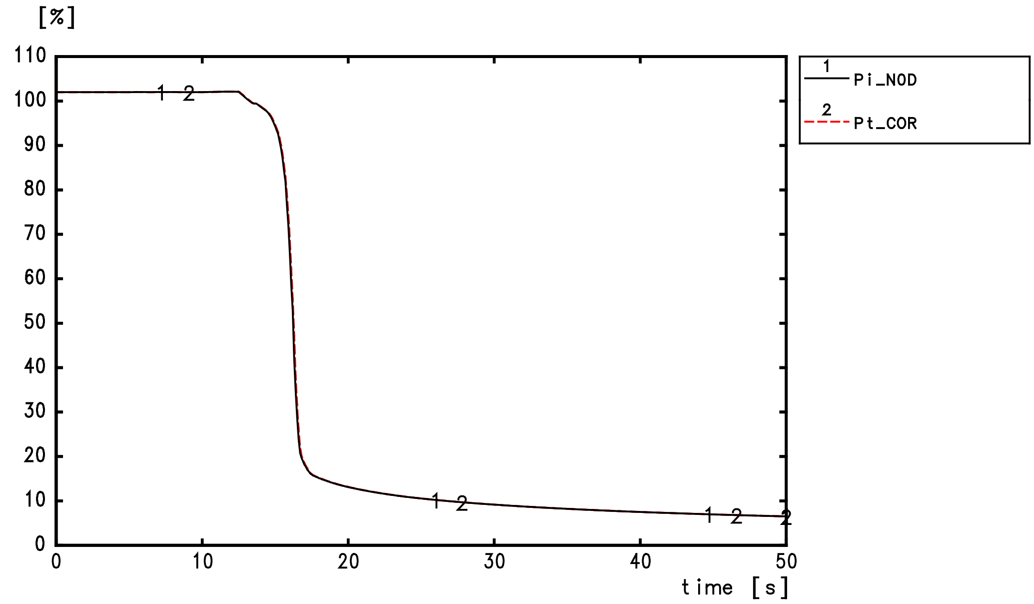
PRESSURE AT PUMP OUTLET IN RCP



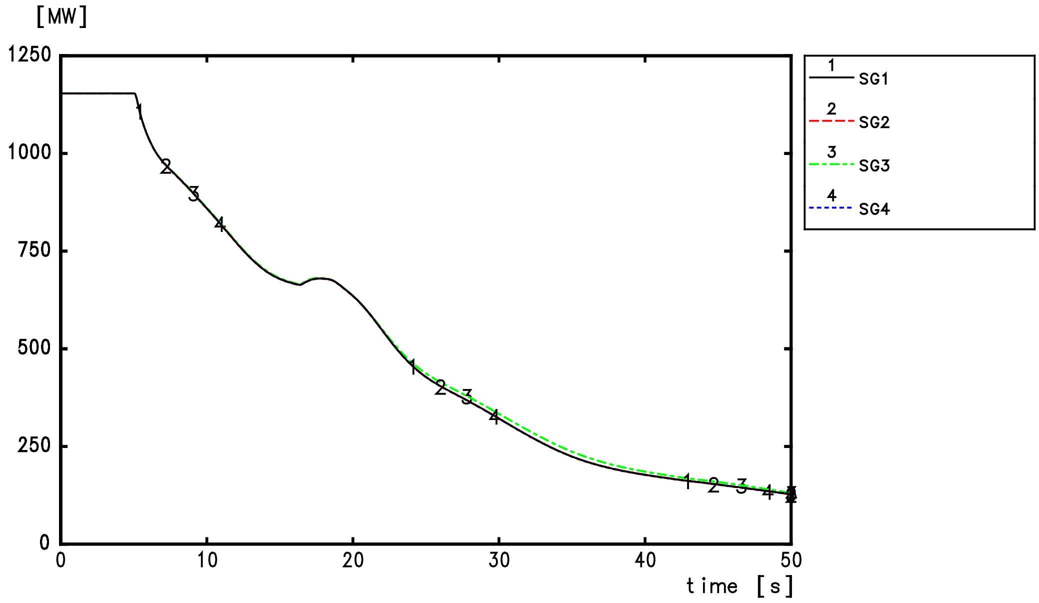
Psat in steam generators

SUB-CHAPTER 16.5 – FIGURE 77

C2 - 4 VIVs [MSIV]s w/o PSVs - 3:
Core power and primary/secondary power exchange – without normal spray



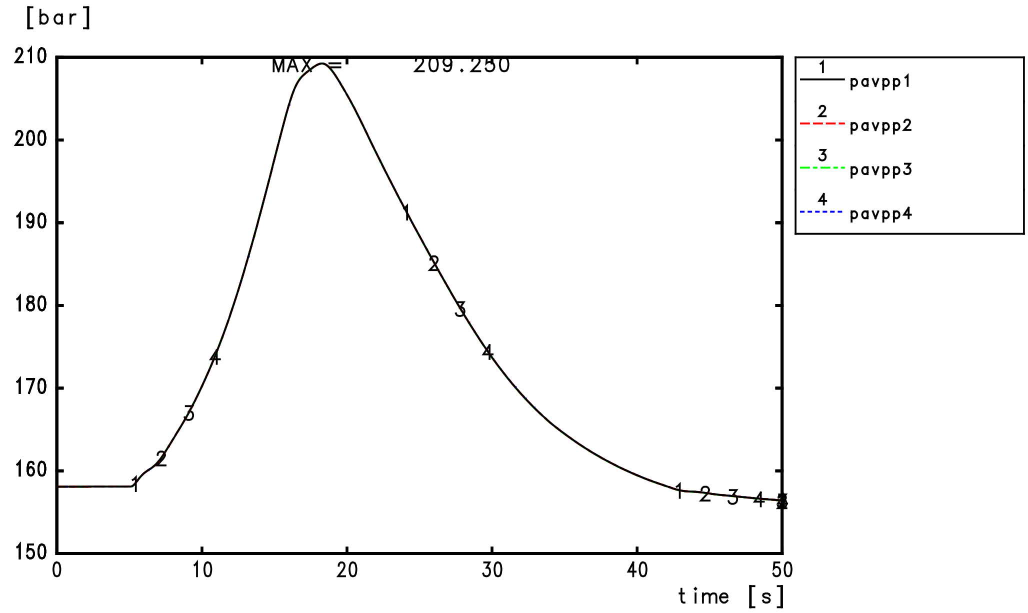
CORE POWER



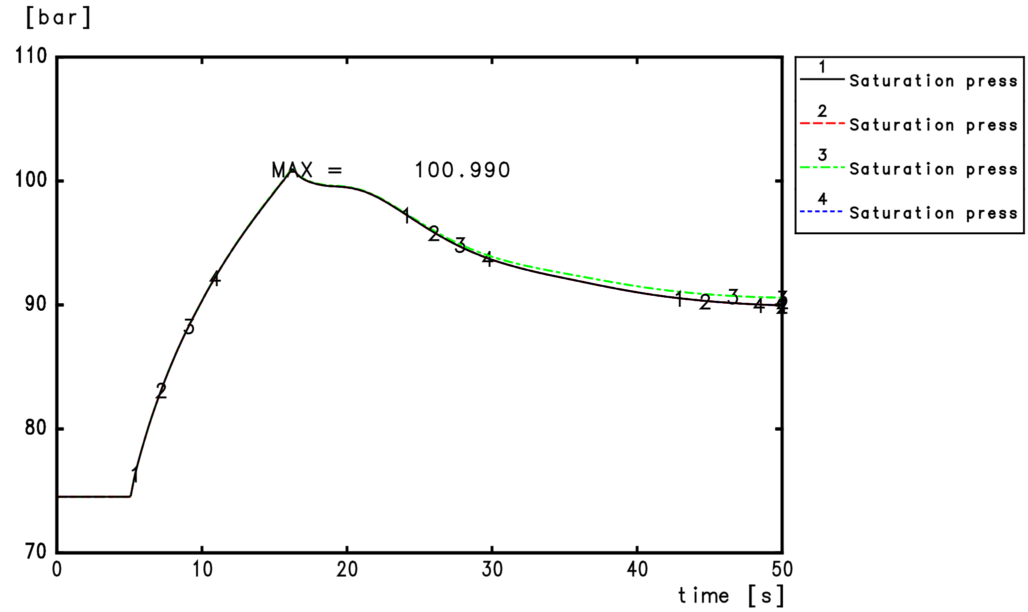
PRIMARY/SECONDARY POWER EXCHANGE

SUB-CHAPTER 16.5 – FIGURE 78

C2 - 4 VIVs [MSIV]s w/o PSVs - 4:
Reactor coolant pump outlets and secondary pressures – without normal spray



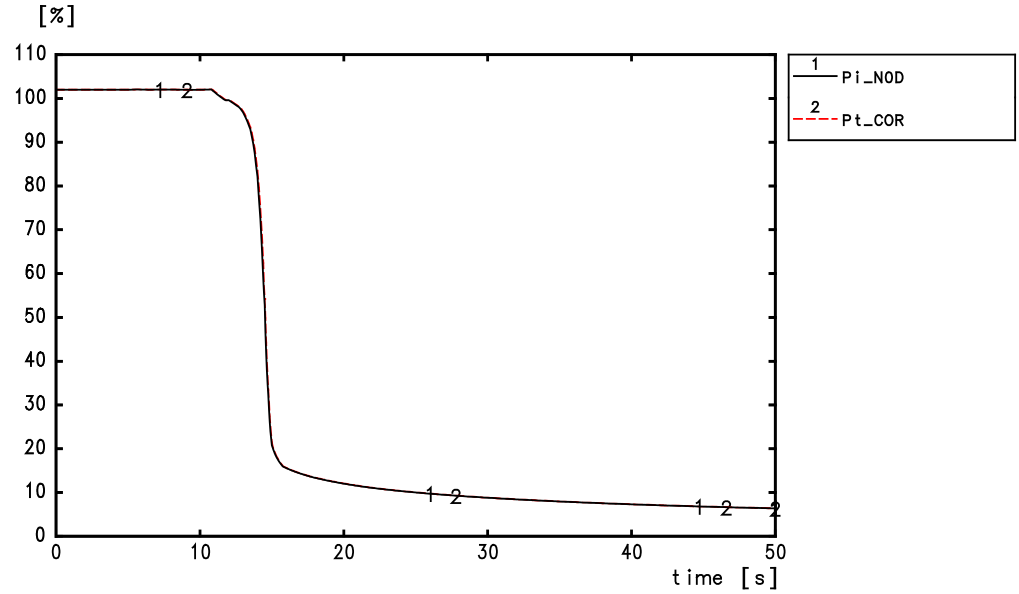
PRESSURE AT PUMP OUTLET IN RCP



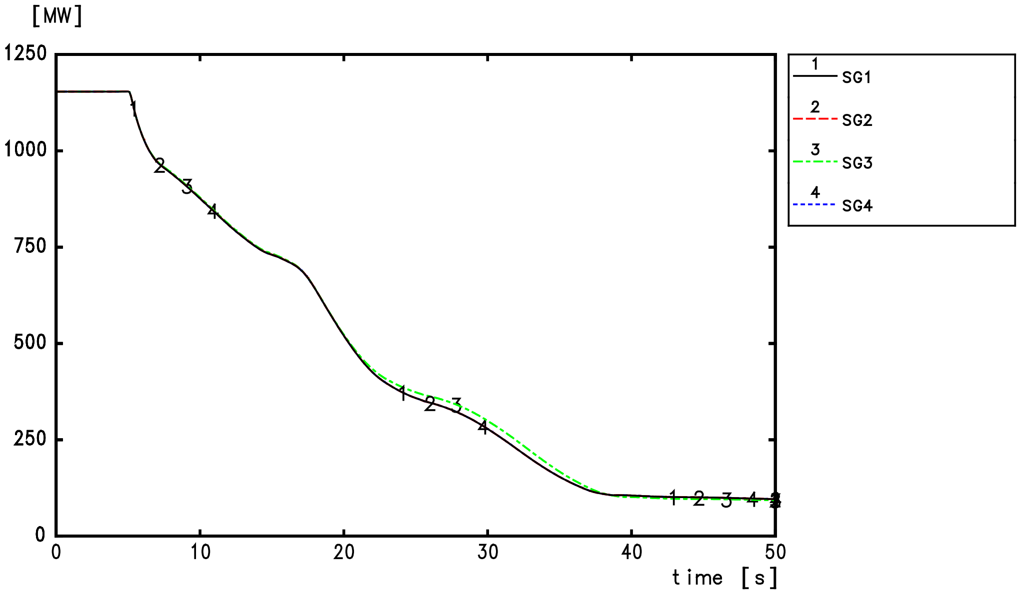
Psat in steam generators

SUB-CHAPTER 16.5 – FIGURE 79

O1 - 4 VIVs [MSIV]s w/o VDAs [MSRT]s - 1: Core Power and primary/secondary power exchange with operational spray



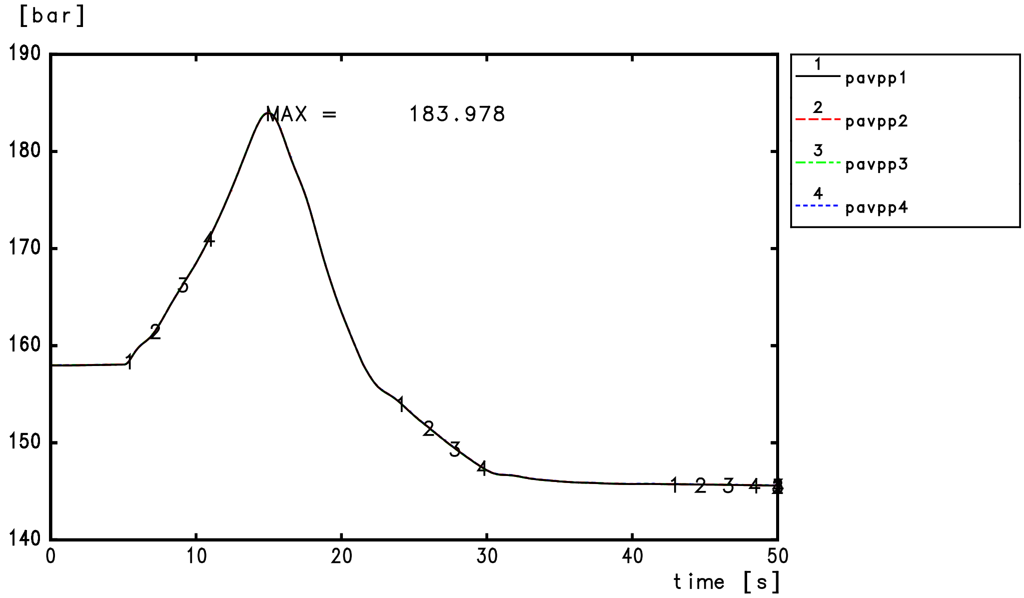
CORE POWER



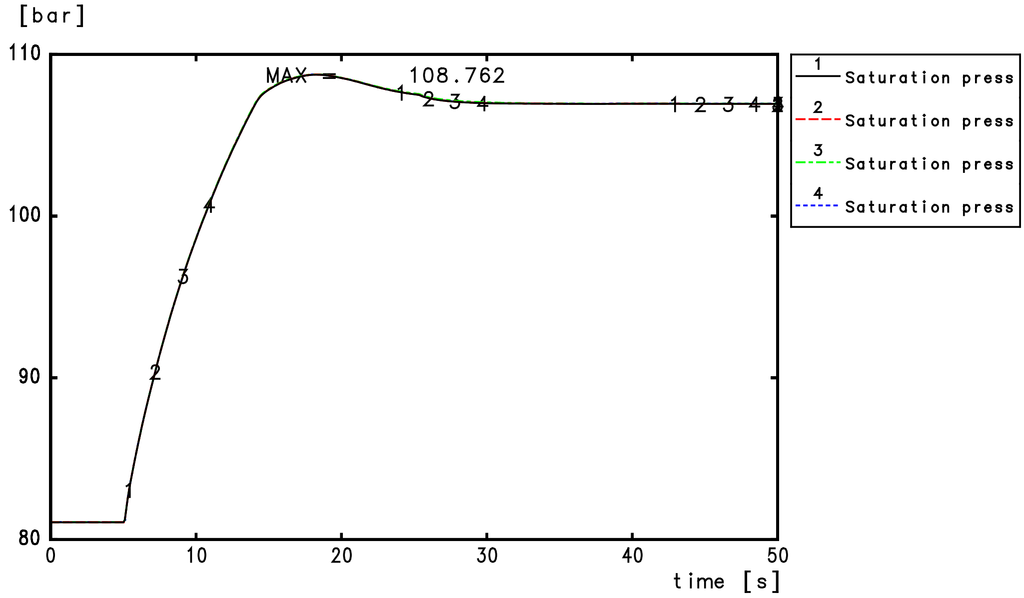
PRIMARY/SECONDARY POWER EXCHANGE

SUB-CHAPTER 16.5 – FIGURE 80

O1 - 4 VIVs [MSIV]s w/o VDAs [MSRT]s - 2:
Reactor coolant pumps outlet pressure and SG pressure with operational spray



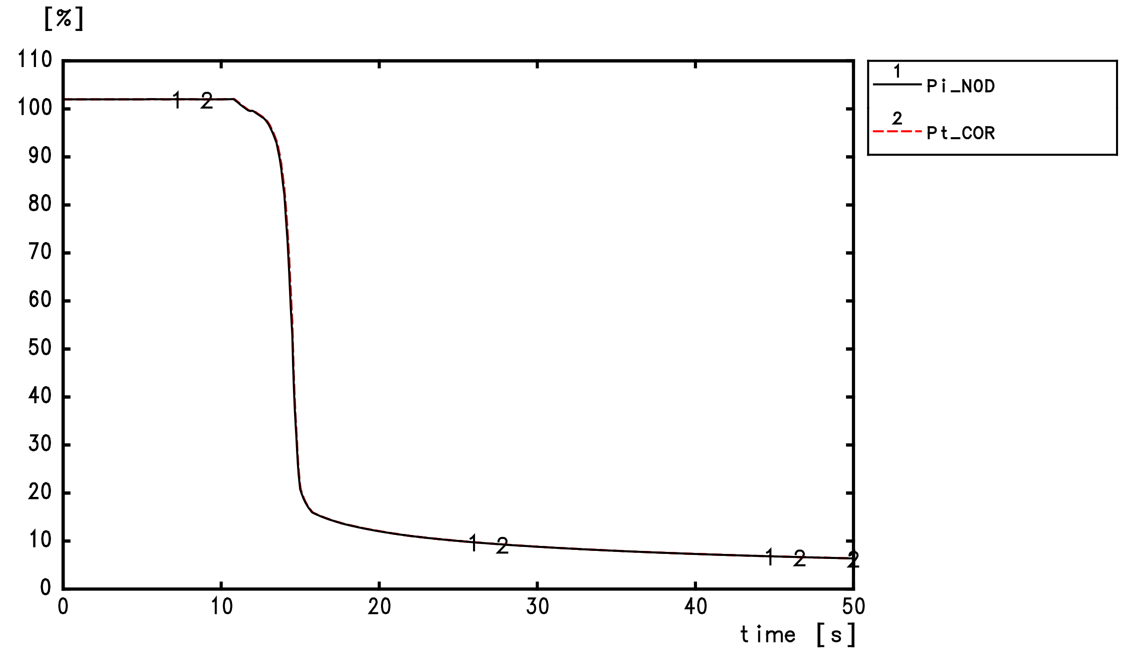
PRESSURE AT PUMP OUTLET IN RCP



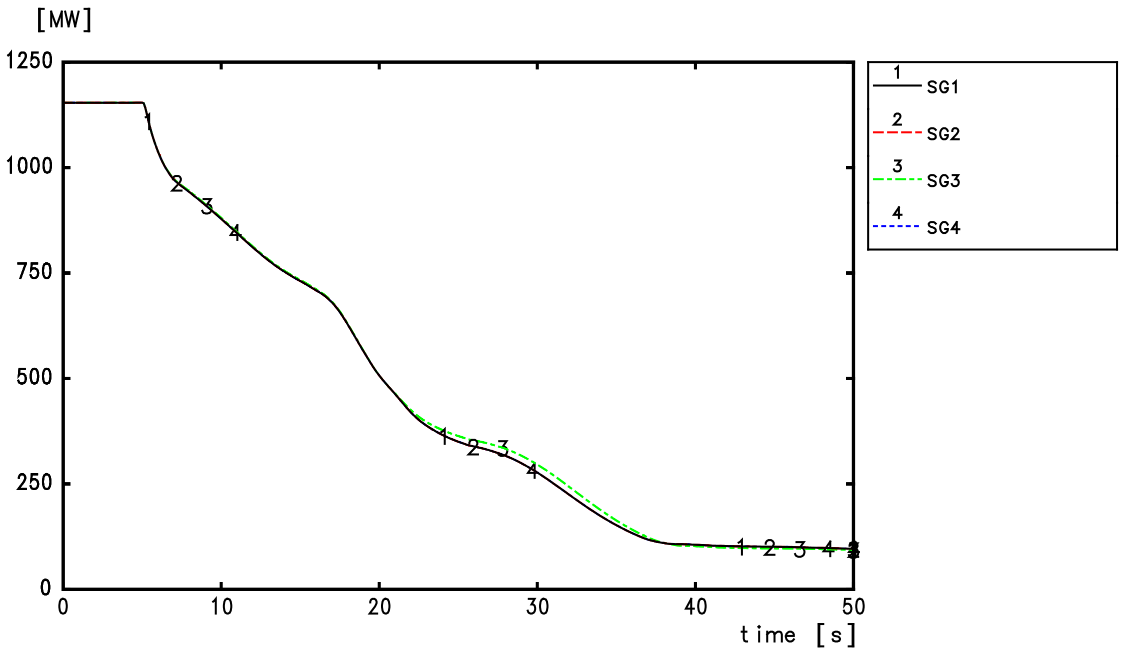
Psat in steam generators

SUB-CHAPTER 16.5 – FIGURE 81

O1 - 4 VIVs [MSIV]s w/o VDAs [MSRT]s - 3: Core Power and primary/secondary power exchange without operational spray



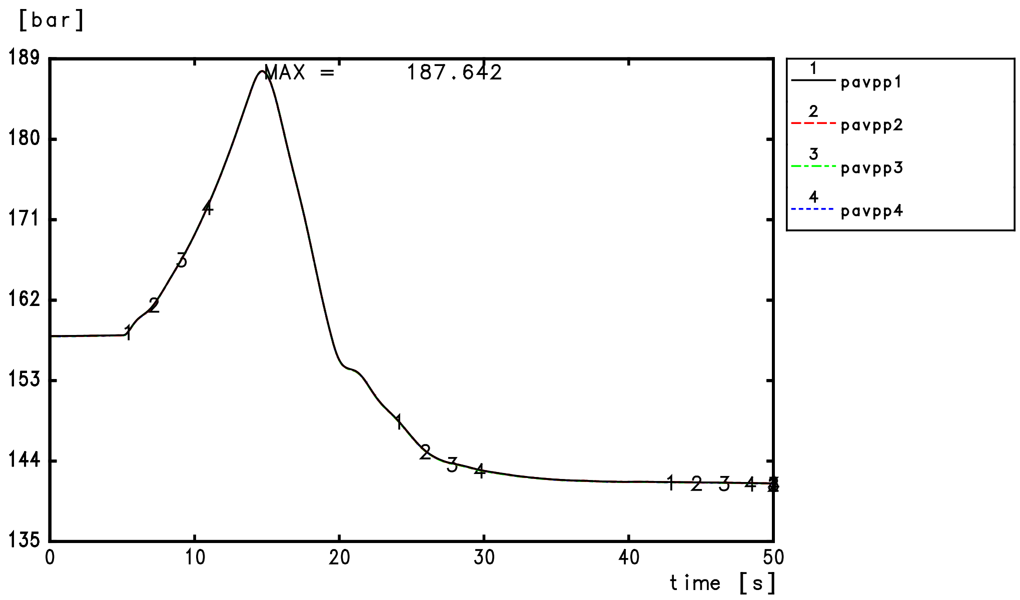
CORE POWER



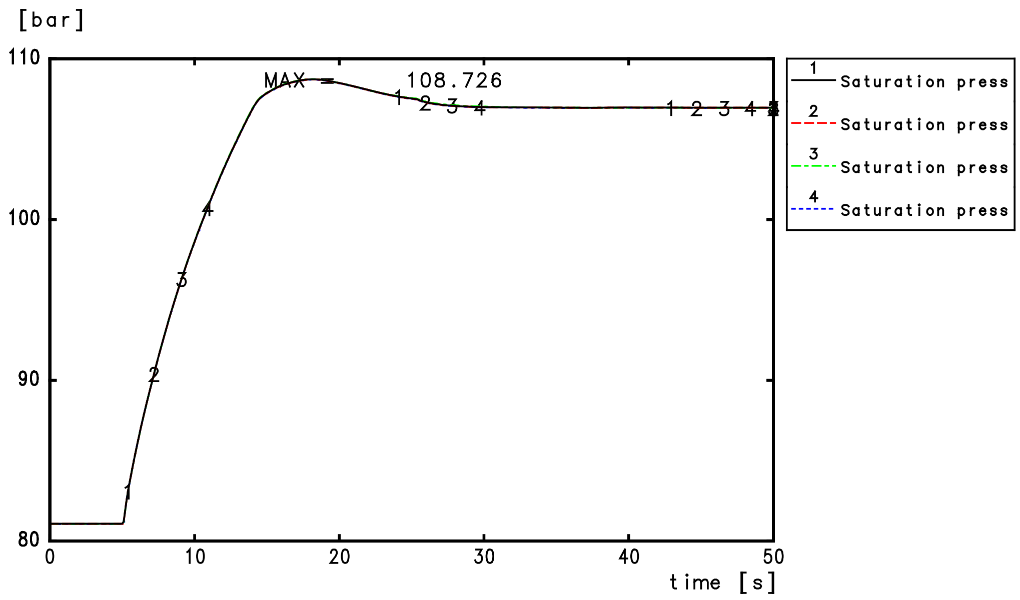
PRIMARY/SECONDARY POWER EXCHANGE

SUB-CHAPTER 16.5 – FIGURE 82

O1 - 4 VIVs [MSIV]s w/o VDAs [MSRT]s - 4:
Reactor coolant pump outlet pressure and SG pressure without operational spray



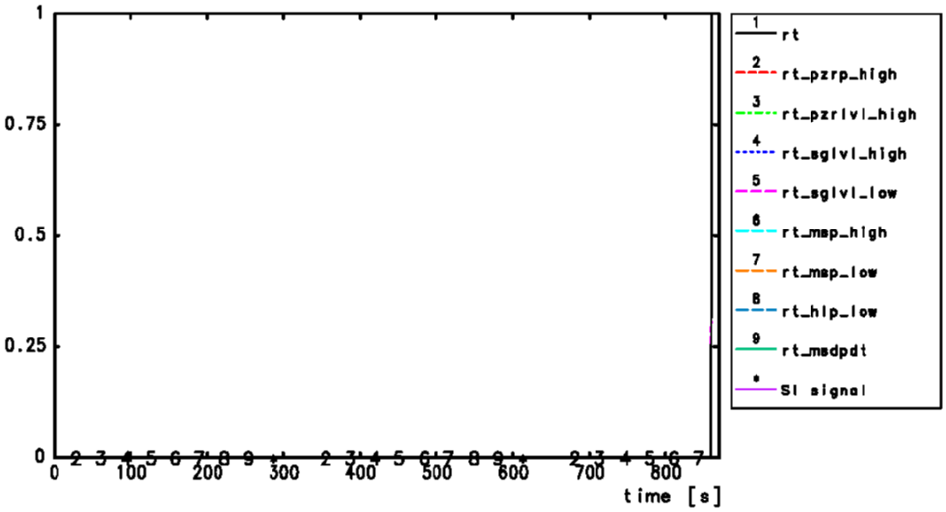
PRESSURE AT PUMP OUTLET IN RCP



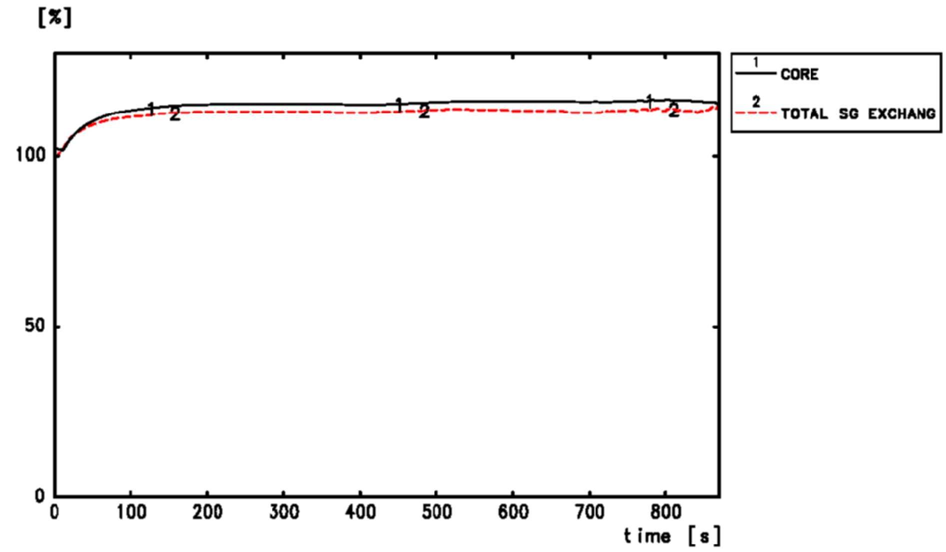
Psat in steam generators

SUB-CHAPTER 16.5 – FIGURE 83

R2 – ATWS EISF w/o TXS - 1: Core power and reactor coolant pump speed



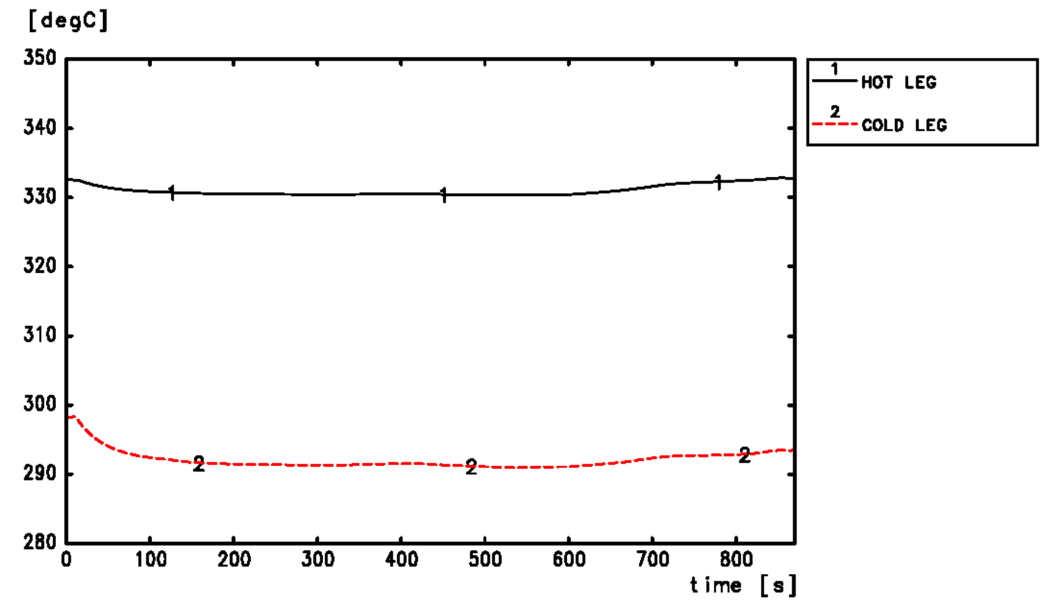
Signal RT



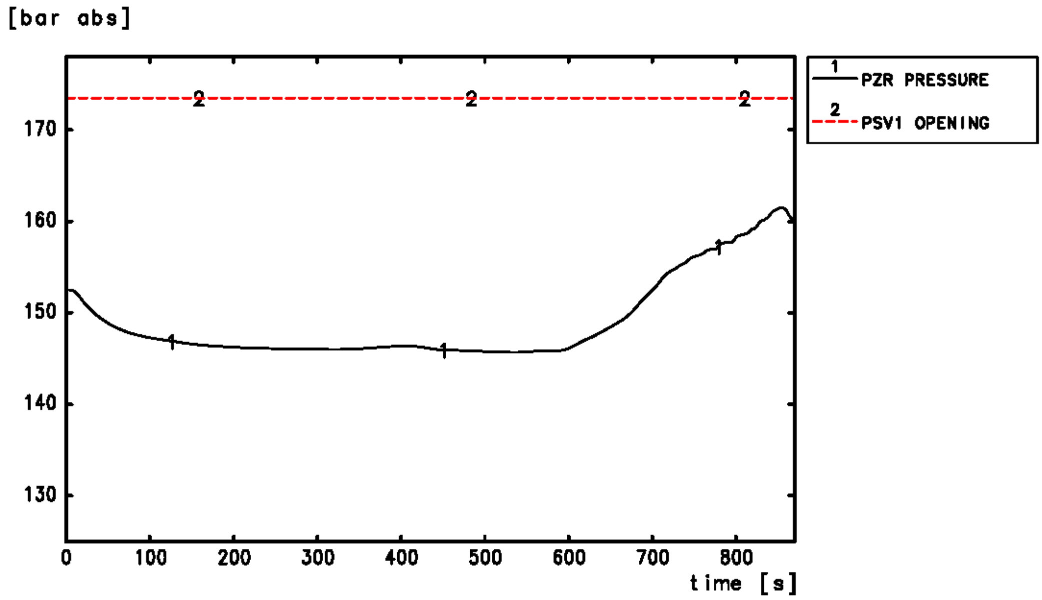
CORE POWER AND SG EXCHANGE

SUB-CHAPTER 16.5 – FIGURE 84

R2 – ATWS EISF w/o TXS - 2: RCP [RCS] temperatures and pressuriser pressure



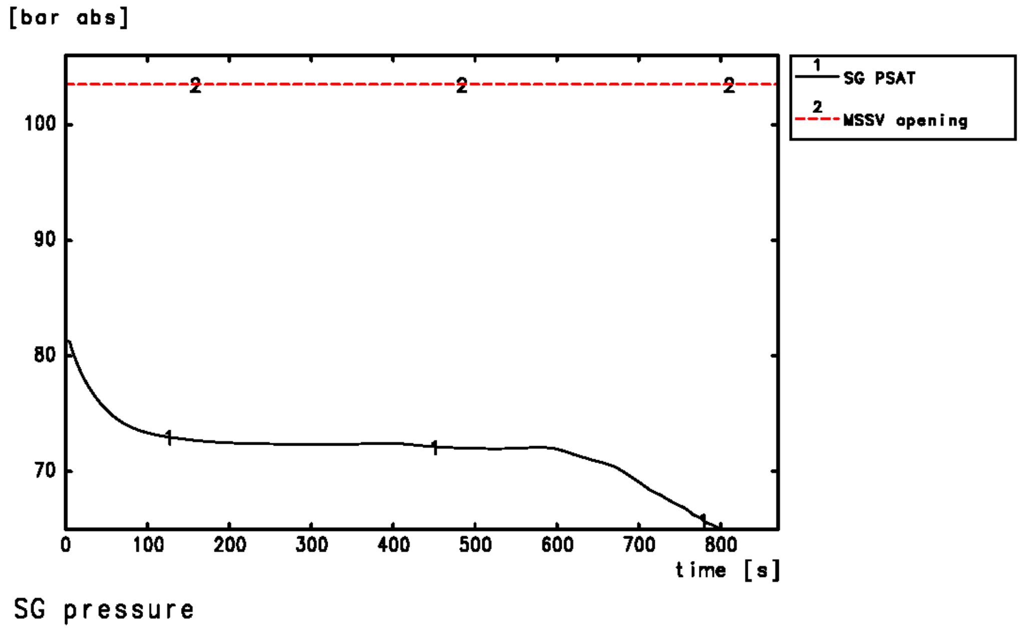
PRIMARY TEMPERATURES

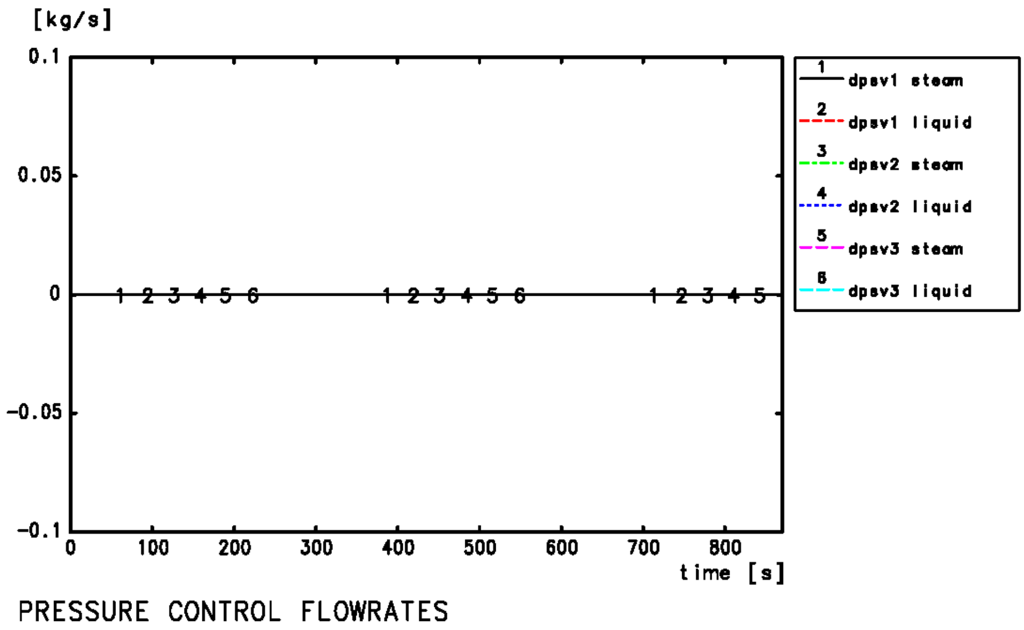


PRIMARY PRESSURE

SUB-CHAPTER 16.5 – FIGURE 85

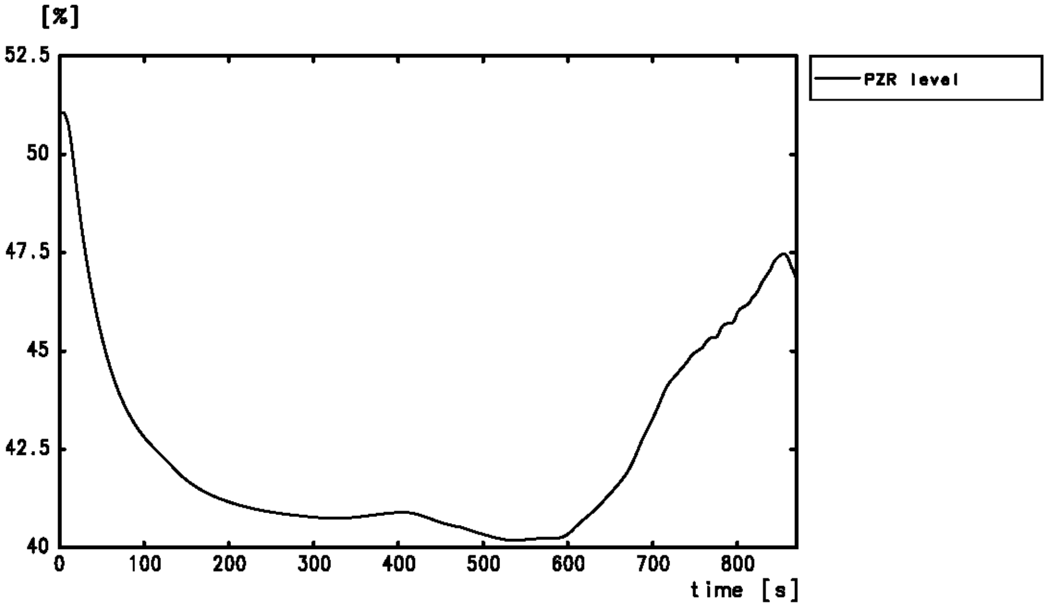
R2 – ATWS EISF w/o TXS - 3: SG pressure and PSV flow rate



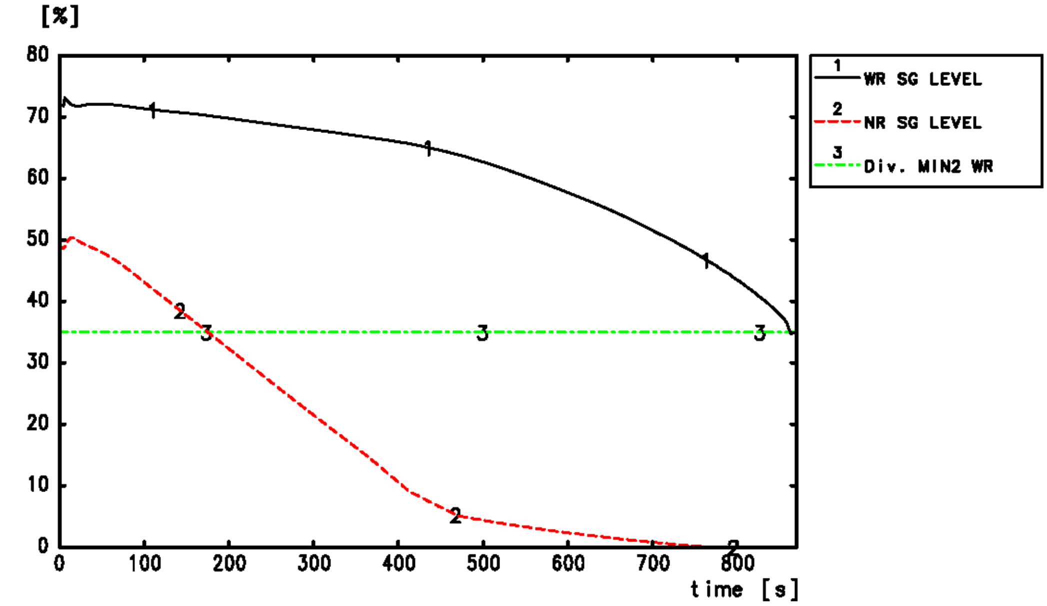


SUB-CHAPTER 16.5 – FIGURE 86

R2 – ATWS EISF w/o TXS - 4: Pressuriser and SG levels



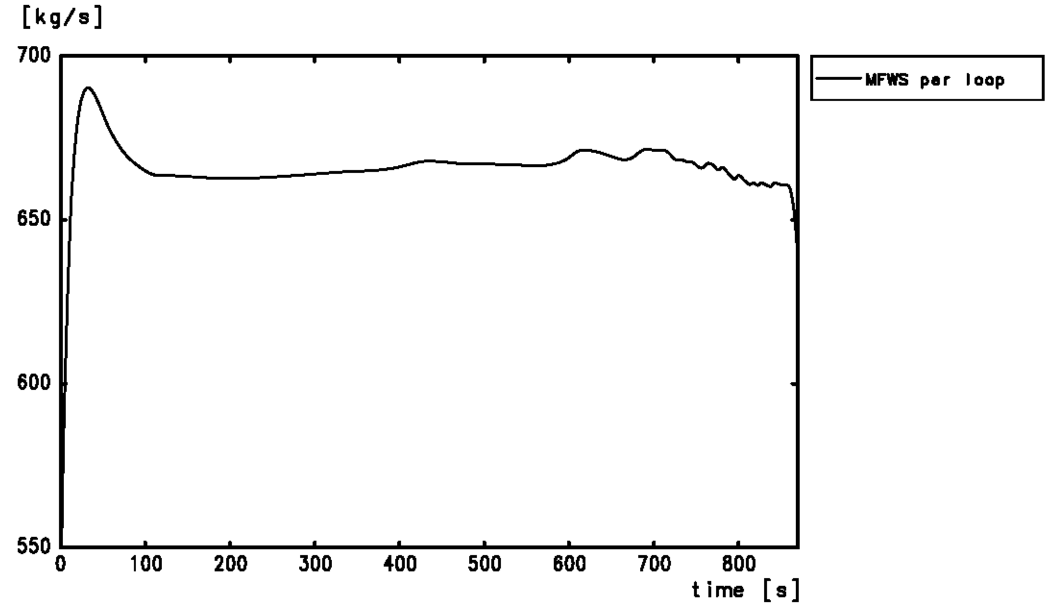
PZR LEVEL



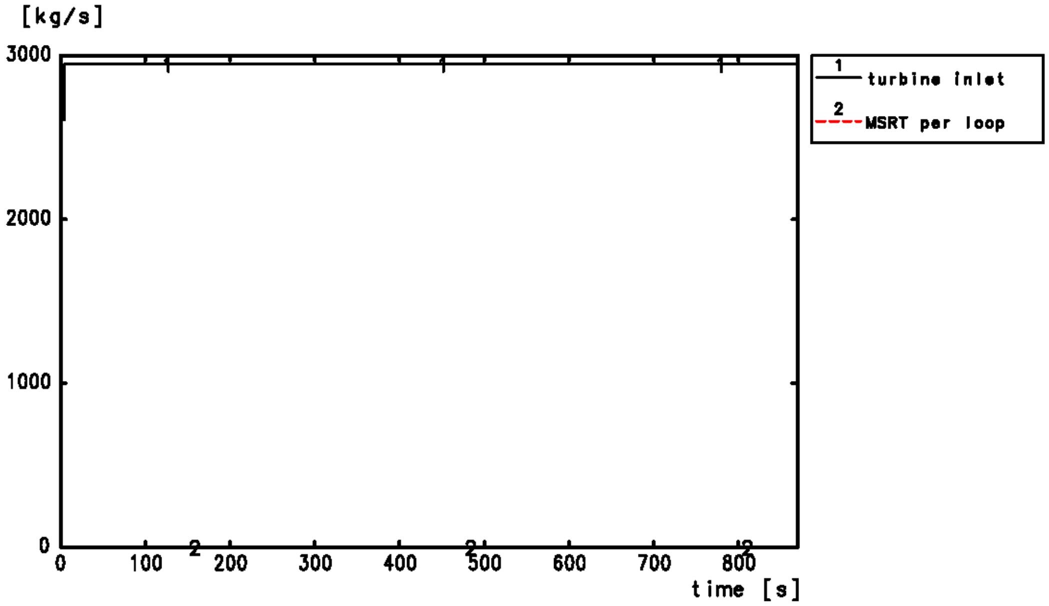
WIDE RANGE AND NARROW RANGE SG LEVELS

SUB-CHAPTER 16.5 – FIGURE 87

R2 – ATWS EISF w/o TXS - 5: Main feedwater flow rate and steam flow rate



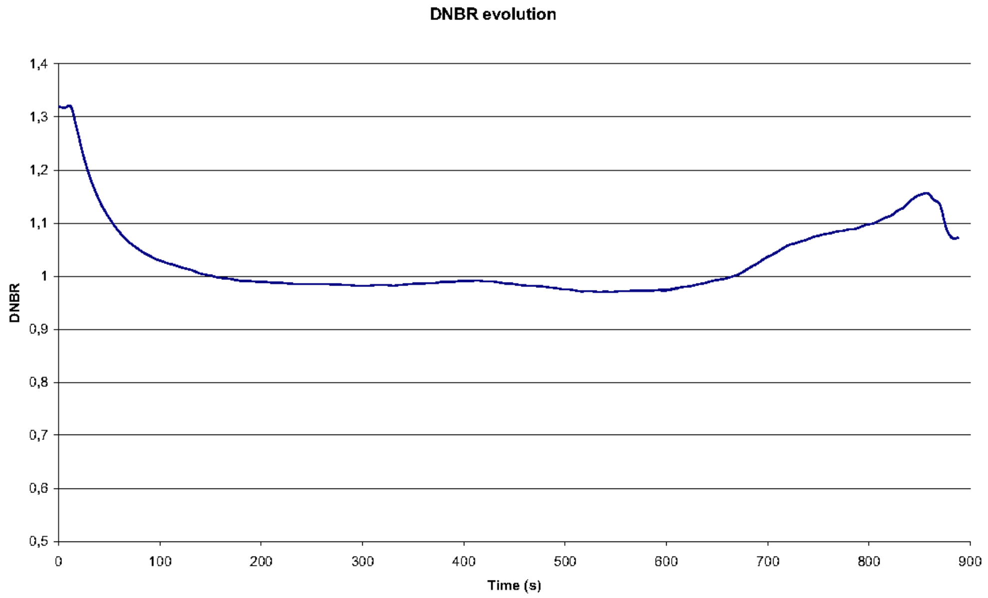
MAIN FEEDWATER FLOWRATE



STEAM FLOWRATE

SUB-CHAPTER 16.5 – FIGURE 88

R2 – ATWS EISF w/o TXS - 6: DNBR transient



SUB-CHAPTER 16.5 – FIGURE 89

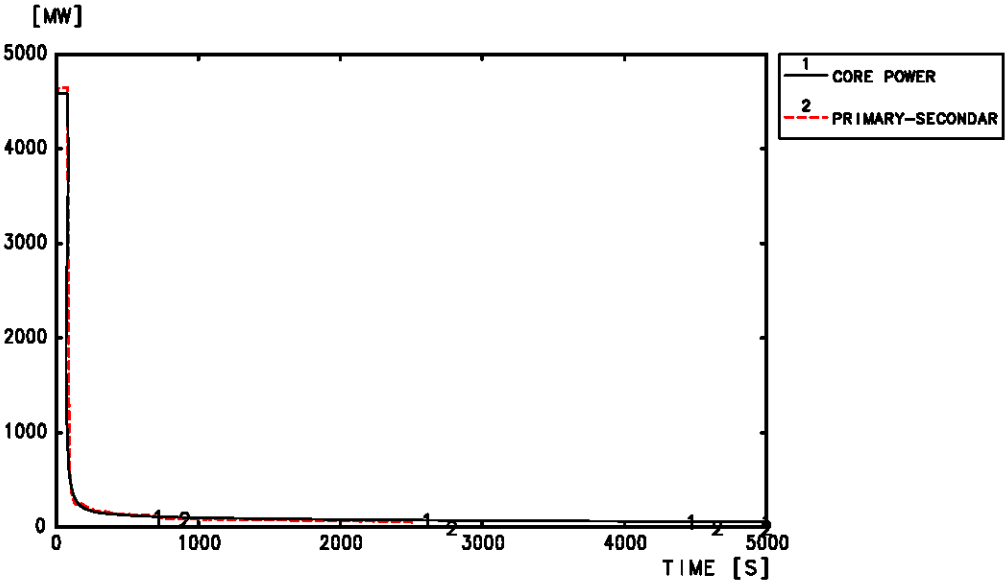
H1 – SB LOCA w/o VDAs [MSRT]s
1: Decay heat curve, A+B+C terms

{CCI Removed}

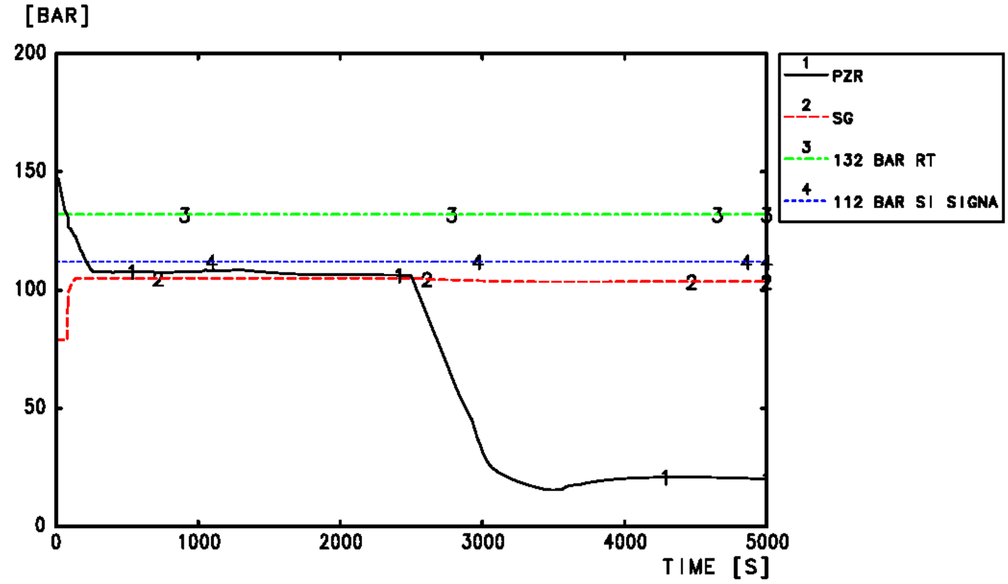
b

SUB-CHAPTER 16.5 – FIGURE 90

H1 –SB LOCA w/o VDAs [MSRT]s
2: Primary and secondary exchanged power, primary and secondary pressure



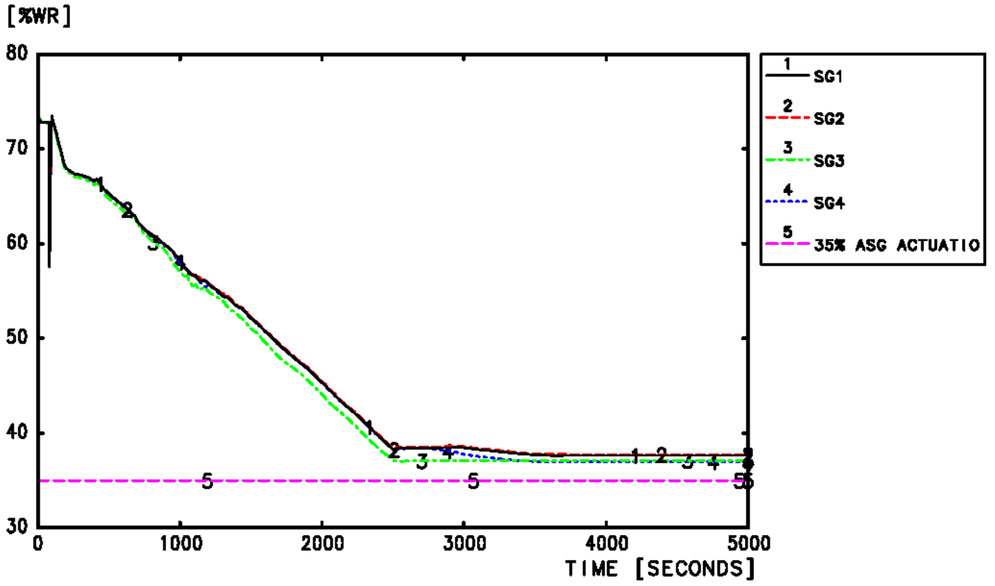
POWER



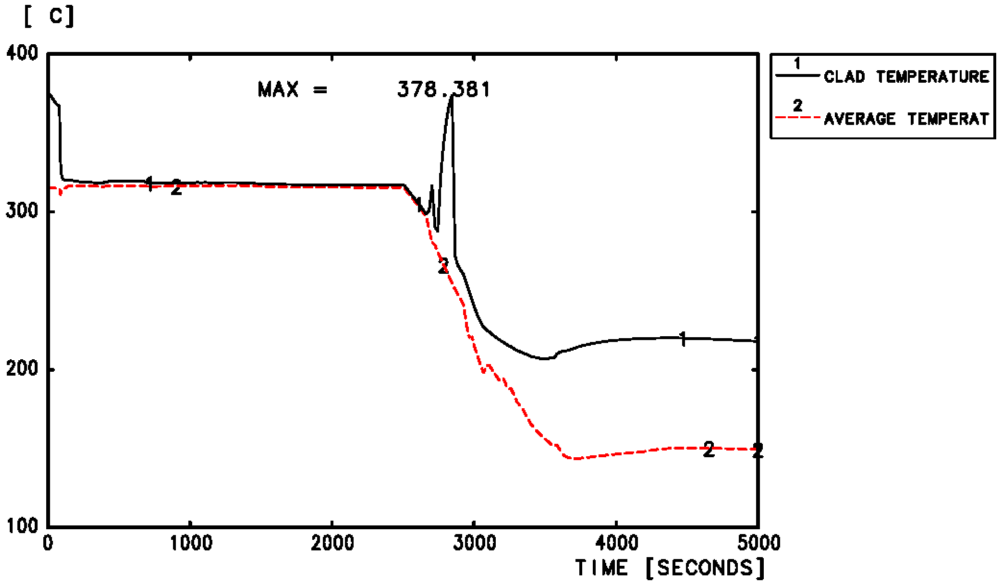
PRIMARY AND SECONDARY PRESSURES

SUB-CHAPTER 16.5 – FIGURE 91

H1 – SB LOCA w/o VDAs [MSRT]s
3: SG wide range levels, core temperatures



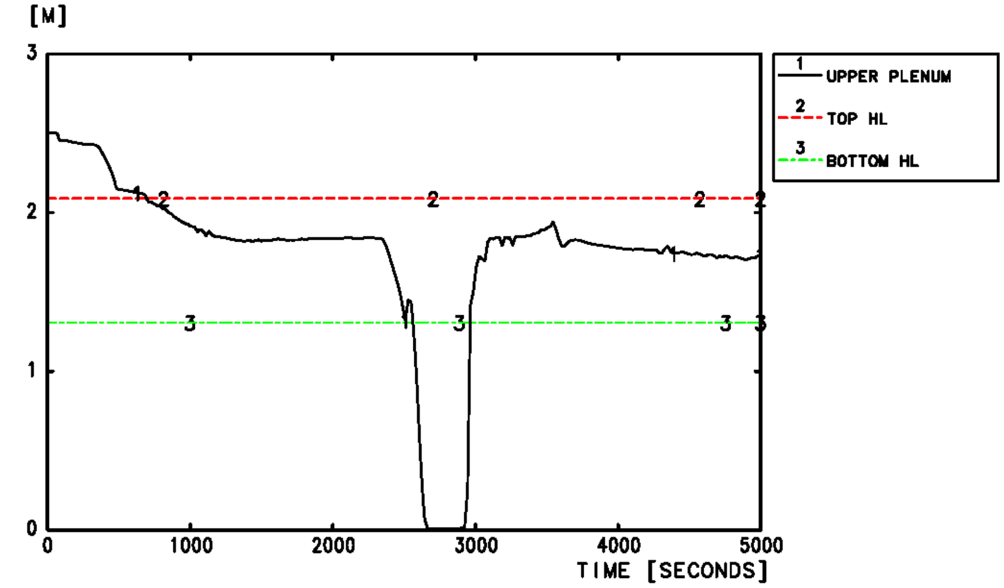
SG LEVEL



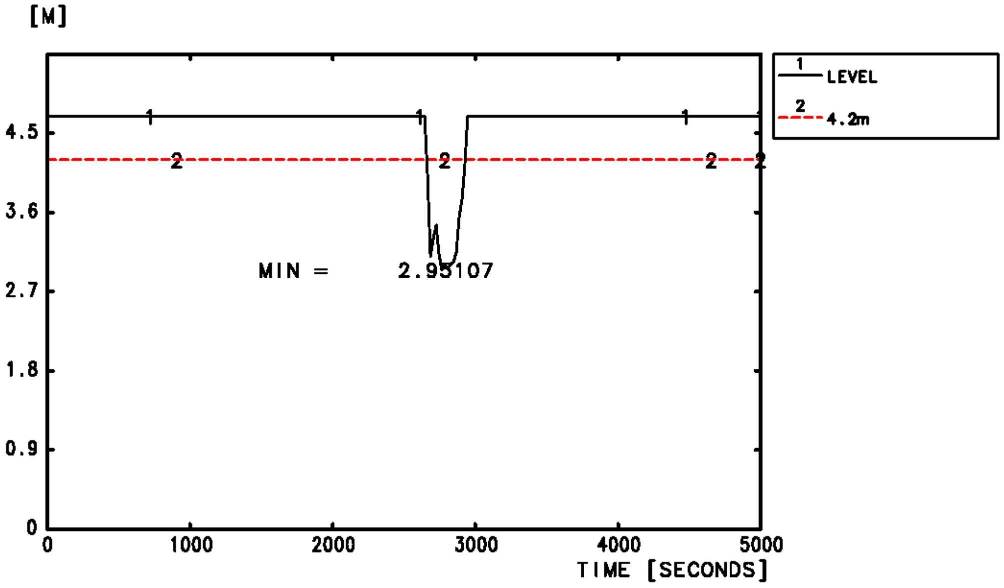
CORE TEMPERATURES

SUB-CHAPTER 16.5 – FIGURE 92

H1 – SB LOCA w/o VDAs [MSRT]s
4: Primary and secondary masses, upper plenum liquid level



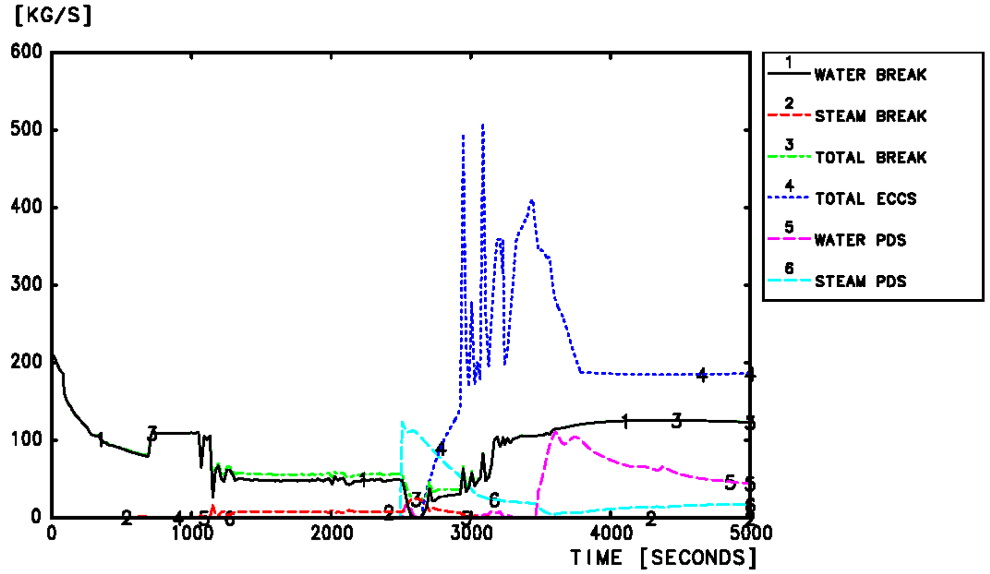
UPPER PLENUM LEVEL



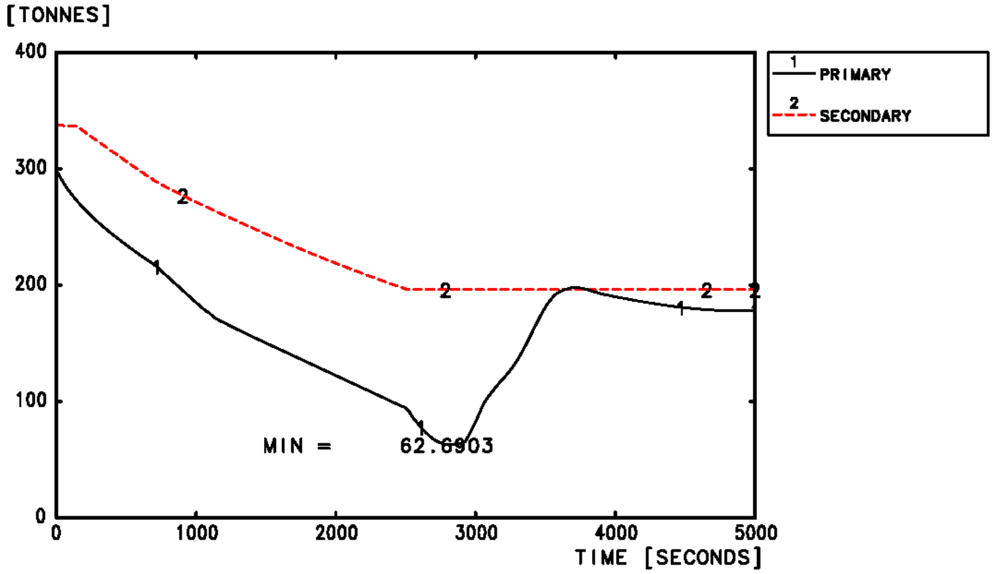
CORE LEVEL

SUB-CHAPTER 16.5 – FIGURE 93

H1 – SB LOCA w/o VDAs [MSRT]s
5: RIS [SIS], PDS and break flow rates, primary and secondary masses



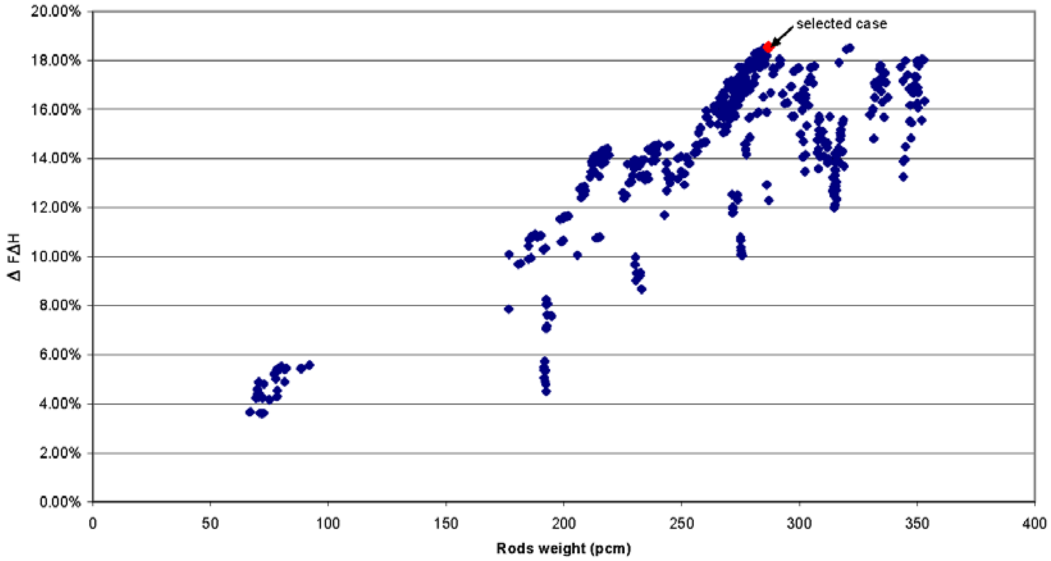
BREAK, ECCS, AND PDS FLOWRATES



PRIMARY AND SECONDARY MASSES

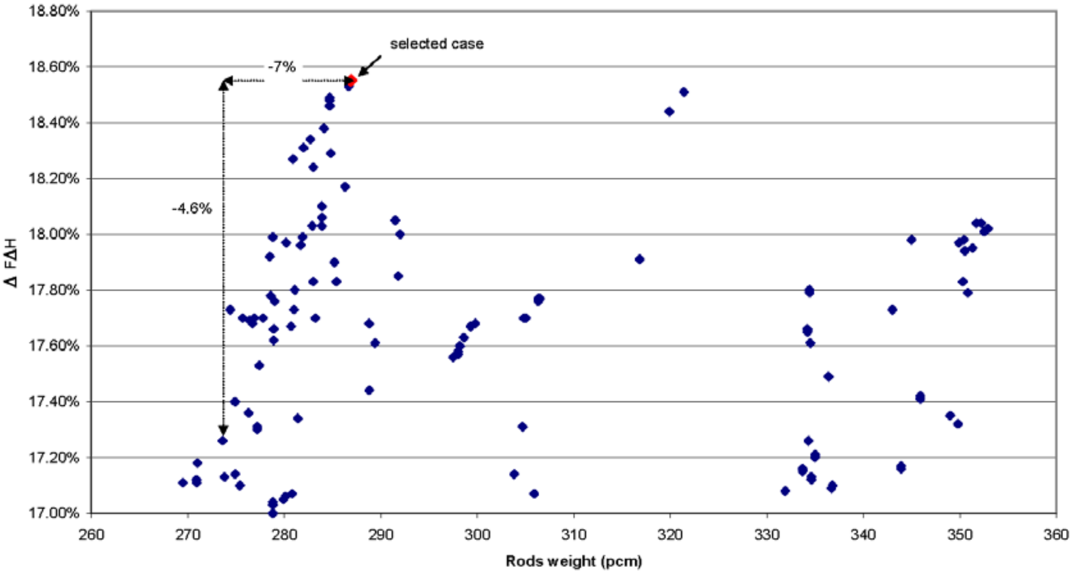
SUB-CHAPTER 16.5 – FIGURE 94

R2 – ATWS ROD DROP w/o TXS
1: Rod drop case at beginning of life (FΔH variation as a function of rod worth)



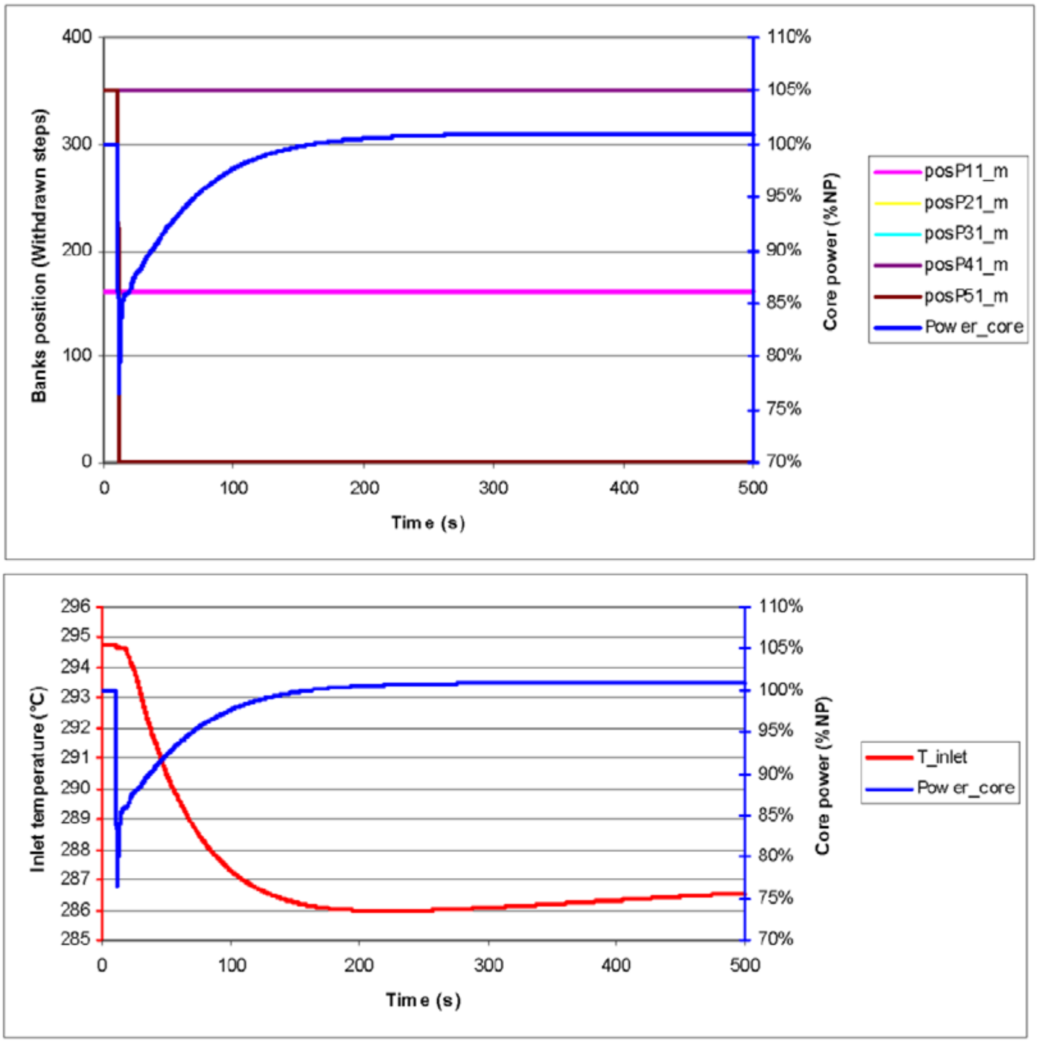
SUB-CHAPTER 16.5 – FIGURE 95

R2 – ATWS ROD DROP w/o TXS
2: Rod drop case at beginning of life (FΔH variation as a function of rod worth)
focus on high variations



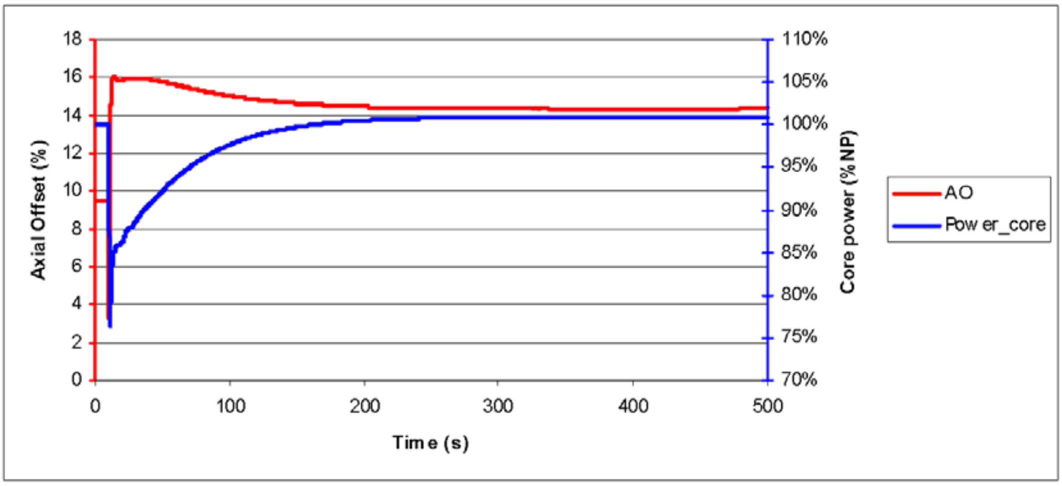
SUB-CHAPTER 16.5 – FIGURE 96

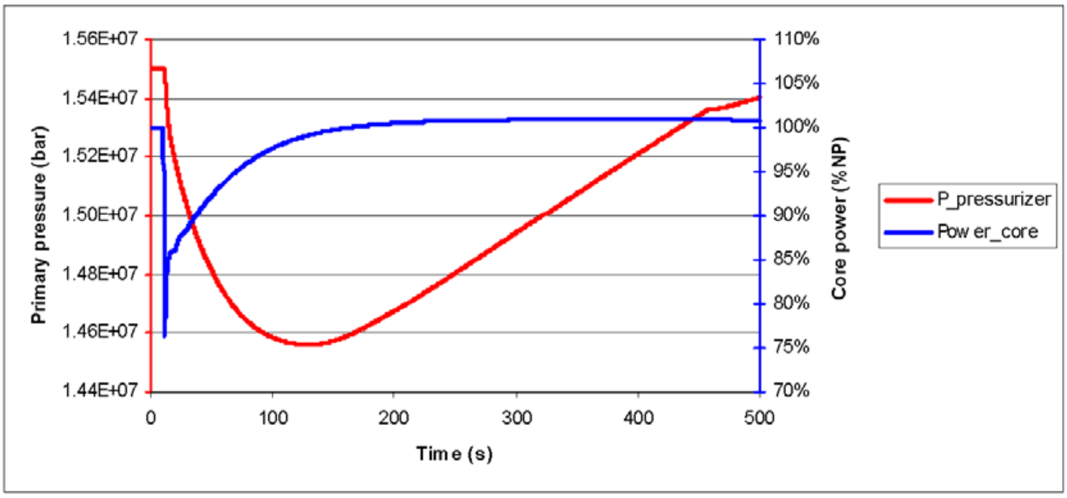
R2 – ATWS ROD DROP w/o TXS – 3: Variations of Core Power, Bank Position, Inlet Temperature, Axial Offset and Primary Pressure (1 of 2)



SUB-CHAPTER 16.5 – FIGURE 97

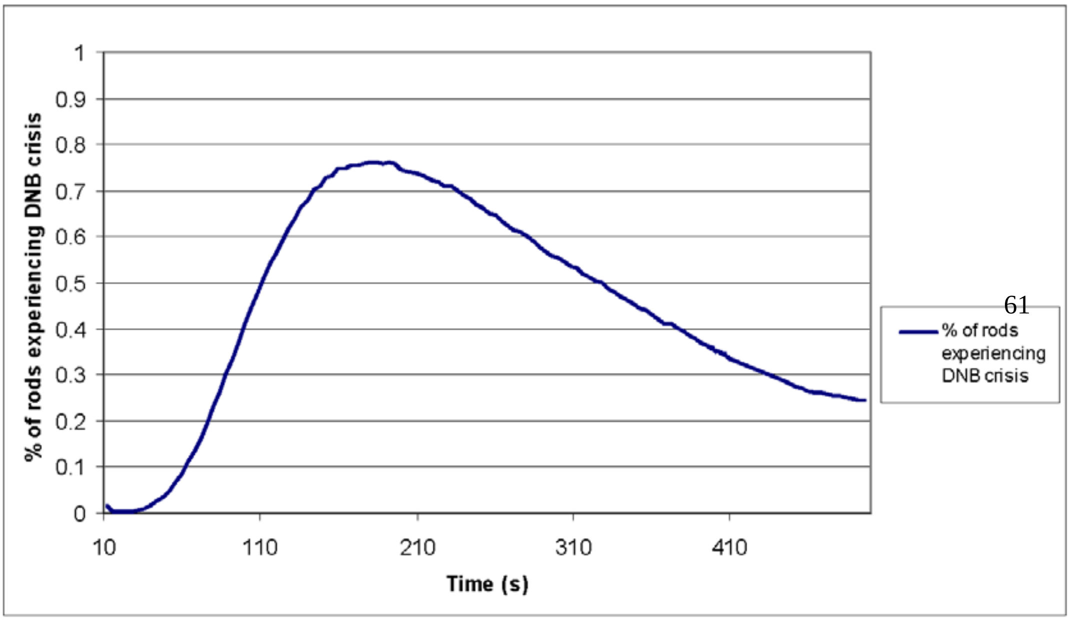
R2 – ATWS ROD DROP w/o TXS – 4: Variations of Core Power, Bank Position, Inlet Temperature, Axial Offset and Primary Pressure (2 of 2)





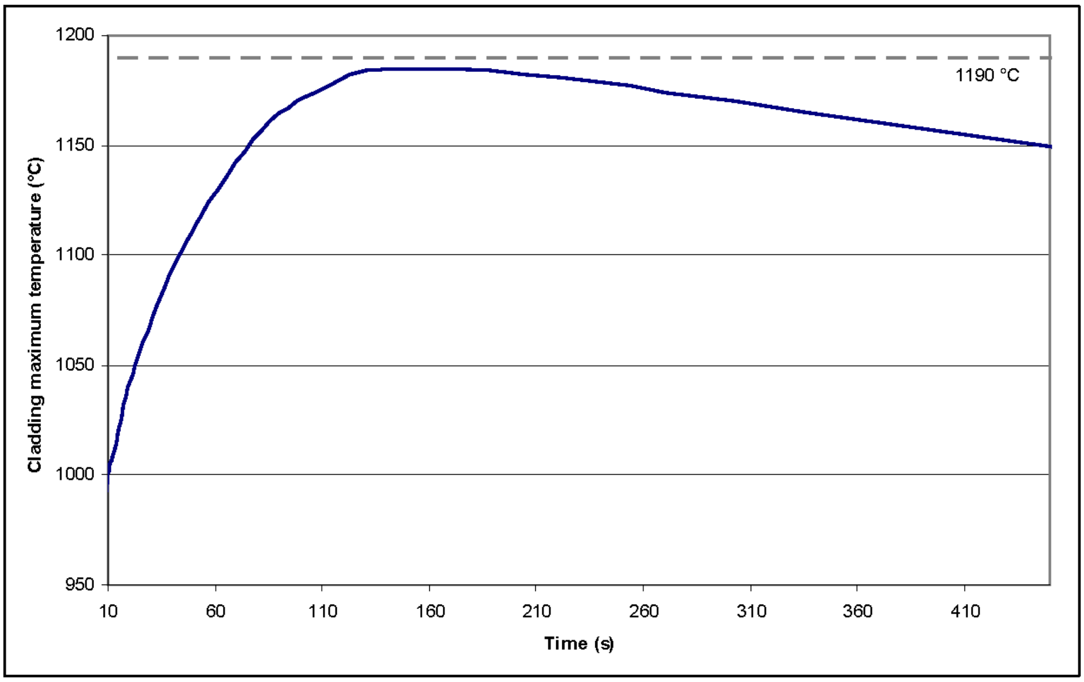
SUB-CHAPTER 16.5 – FIGURE 98

R2 – ATWS ROD DROP w/o TXS
5: Variation in the number of rods experiencing DNB



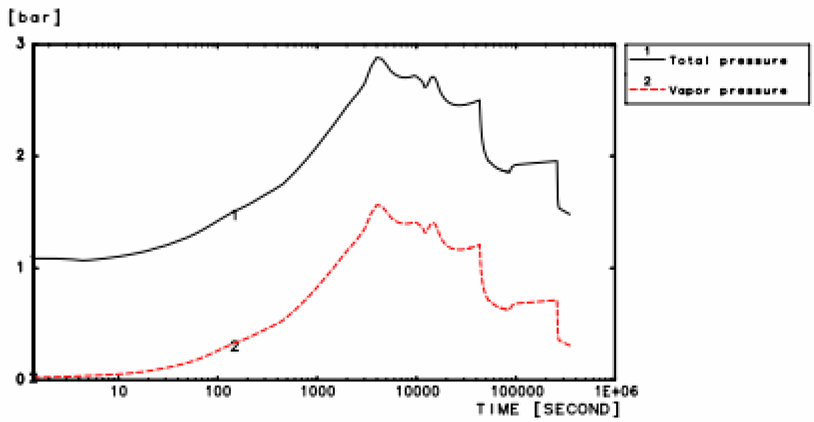
SUB-CHAPTER 16.5 – FIGURE 99

R2 –ATWS ROD drop w/o TXS
6: Cladding maximum temperature

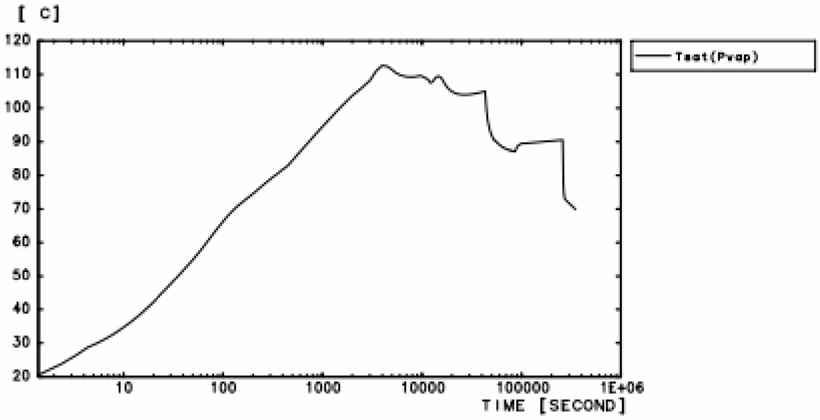


SUB-CHAPTER 16.5 – FIGURE 100

Other safety functions - Diversity to safe shutdown state
1: Pressure and temperature in reactor building in LOCA RRC-A
(4 days with activation of 2 EVU [CHRS] at 12 hours and 2 LHSI recovery at 3 days)



CONTAINMENT PRESSURE



CONTAINMENT TEMPERATURE

SUB-CHAPTER 16.5 – REFERENCES

External references are identified within this sub-chapter by the text [Ref-1], [Ref-2], etc at the appropriate point within the sub-chapter. These references are listed here under the heading of the section or sub-section in which they are quoted.

2. METHODOLOGY FOR THE DEMONSTRATION OF FUNCTIONAL DIVERSITY

2.1. INTRODUCTION

[Ref-1] UK EPR – Consistency between PSA list and PCC list. NEPR-F DC 584 Revision A. AREVA. July 2010. (E)

2.5. TRANSIENT SELECTION

2.5.9. Support systems

2.5.9.3. Instrumentation and Control

[Ref-1] EPR UK – Functional requirements on non-computerized safety I&C functions. NEPR-F DC 551 Revision C. AREVA. July 2012. (E)

2.5.9.4. Diversity of sensors

[Ref-1] Functional analysis for sensors’ Common Cause Failure. PEPR-F DC 83 Revision C. AREVA. October 2012. (E)

2.5.10. Loss of RCV [CVCS] faults

[Ref-1] EPR UK GDA – FS02A7 – Development of a diverse protection system for RCV [CVCS] homogeneous boron dilution events in shutdown states. PEPCF.12.0678 Revision 1. AREVA. July 2012. (E)

2.5.11. Emergency Operating Procedures

2.5.11.2. Frequent Postulated Initiating Events

2.5.11.2.3. Increase in heat removal

[Ref-1] UK EPR – Main steam isolation valves ALARP assessment regarding functional diversity and single failure criterion. PESS-F DC 27 Revision A. AREVA. November 2010. (E)

2.7. CONCLUSIONS

[Ref-1] ALARP demonstration for the design of the pressurizer safety valves regarding the passive single failure. PEPR-F DC 28 Revision A. AREVA. November 2010. (E)

[Ref-2] UK EPR – Main steam isolation valves ALARP assessment regarding functional diversity and single failure criterion. PESS-F DC 27 Revision A. AREVA. November 2010. (E)

[Ref-3] UK EPR – Containment Isolation Valve Diversity ALARP assessment. PESS-F DC 28 Revision A. AREVA. November 2010. (E)

[Ref-4] EPR UK – Diversity for frequent faults: ATWS LOOP cumulated with EDG start-up failure. ECESN120274 Revision A. EDF. May 2012. (E)

[Ref-5] UK EPR – Consistency between PSA list and PCC list. NEPR-F DC 584 Revision A. AREVA. July 2010. (E)

3. FUNCTIONAL DIVERSITY TRANSIENT ANALYSES

3.5. CONTAINMENT SAFETY FUNCTION

3.5.2. C2 - Maintain integrity of the Reactor Coolant Pressure Boundary

[Ref-1] ALARP demonstration for the design of the pressurizer safety valves regarding the passive single failure. PEPR-F DC 28 Revision A. AREVA. November 2010. (E)

3.6. OTHER SAFETY FUNCTIONS

3.6.1. O1 – Prevent the failure or limit the consequences of failure of a structure, system or component whose failure could cause the impairment of a safety function

3.6.1.2. Diversity to safe shutdown state

3.6.1.2.2. Decrease in RCP [RCS] inventory faults with failure of the LHSI

[Ref-1] In-containment pressure and temperature in PCC and RRC-A accidents. NFPSR DC 1055 Revision F. AREVA. June 2008. (E)

This study is referenced for information only, to provide an example of the study carried out for the Flamanville detailed design.