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02	Integration co-applicant and INSA review comments.	30-04-2008
03	Addition of sections relative to SGTR (2 tubes) and Non isolable small break or isolable SIS break in RHR mode, spent fuel pool drainage aspects	29-06-2008
04	Remarking with RESTRICTED classification	16-01-2009
05	June 2009 update including: - Text clarifications - Addition of references - Technical update to account for December 2009 Design Freeze notably partial cooldown rate, pressuriser safety valves and normal spray classification (sections 2, 6 and 8) and FPCS/FPPS characteristics (section 15)	28-06-2009
06	Removal of RESTRICTED classification	10-06-2010
07	Consolidated Step 4 PCSR update: - Minor editorial changes - Update of references - Update of Main Steam Line Break analysis (§2) - Update of Rod Ejection Accident analysis (§5) - Update of IB/LB Loss of Coolant Accident analysis (§6) - Update of Spent Fuel Pool analysis (§15)	29-03-2011

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REVISION HISTORY (Cont'd)

Issue	Description	Date
08	Consolidated PCSR update: - References listed under each numbered section or sub-section heading numbered [Ref-1], [Ref-2], [Ref-3], etc - Minor editorial changes - Update of references (English translations) - Update of SGTR analyses taking into account new assumptions regarding the CVCS (§10)	16-11-2012

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SUB-CHAPTER 14.5 – ANALYSES OF THE PCC-4 EVENTS

1. LONG-TERM LOSS OF OFFSITE POWER IN STATE C (>2 HOURS)

1.1. INTRODUCTION

The Loss of Off-site Power (LOOP) causes the loss of main and auxiliary grid connections, including the loss of transformers, and the failure of the house load operation.

The response of the plant to long-term LOOP is typically demonstrated by assessing the sizing of ASG [EFWS] water supplies. The strategy of maintaining the plant for 24 hours or some specific time frame is used to demonstrate that the plant can either be maintained at hot standby or cooldown to RIS/RRA [SIS/RHRS] connection with equipment powered from the Emergency Diesel Generators (EDGs).

Two types of LOOP in state C are identified based on their frequency of occurrence:

- Short-term LOOP with a recovery time for off-site power within 2 hours is classified as a PCC-2 event;

- Long-term LOOP with a recovery time for off-site power between 2 and 24 hours is classified as a PCC-4 event.

The short-term LOOP in state C is covered by the event in state A and its sequence of events is analysed in section 6 of Sub-chapter 14.3. This sub-chapter assesses the consequences of the long-term LOOP.

1.2. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

Reactor state C, which is attained no earlier than 10 hours after Reactor Trip (RT) covers the RCP [RCS] temperature range between 120°C and 55°C. Three different sub-states C1, C2 and C3 are defined. These depend on RCP [RCS] inventory, operating status of reactor coolant pumps and LHSI/RHR pumps, and SG availability for heat removal. These states [Ref-1] are defined in figures in Sub-chapter 14.0

The LOOP leads to the loss of any reactor coolant pumps in operation in states C1 and C2. In all states it leads to the temporary loss of the heat removal via the LHSI/RHR trains and any operational secondary side feedwater supply from the AAD [SSS]. The automatic start-up of the EDGs following a low voltage signal restores power supply to the LHSI/RHR trains within a short delay. This results in the RIS/RRA [SIS/RHRS] function being re-established within 40 seconds. The operational SG feedwater supply remains unavailable as it is not supplied by the EDG. However, the long-term secondary side heat removal capability is provided by the initial SG water inventory, the EDG supplied ASG [EFWS] pumps and heat removal to the atmosphere via the VDA [MSRT].

1.2.1. Relevant Cases and event mitigation

The two plant states C1 and C3 are of the main interest as they cover two different limiting conditions for the loss of heat removal. In state C1 the decay power is highest but the RCP [RCS] has the normal inventory appropriate to hot zero power conditions. In state C3 the decay power is lower but the RCP [RCS] inventory may correspond to that for low loop level operation. For the further conservative evaluation, only state C3 is considered. As the difference in the decay power is negligible compared to the difference in RCP [RCS] inventory between the two states, only state C3 is considered for the further simplified conservative evaluation. However, the power level corresponding to the beginning of state C (C1) is used in this evaluation.

The loss of RCP [RCS] cooling caused by the run-down of the LHSI/RHR pumps causes a heat-up of the RCP [RCS] water due to the decay heat. A bounding decay heat of 30 MW is assumed. This is the decay heat at 23 hours after RT and corresponds to 0.66% of nominal core power of 4500 MW_{th}. This is the highest possible value on entry to state C1. At this power level, the resulting average heat-up rate of the RCP [RCS] water mass at 3/4 loop level, is 0.035°C/s. This is calculated assuming the RCP [RCS] water mass equal to 124 te and including the heat-up of the 600 te structural mass [Ref-1].

The restart of the LHSI/RHR pumps following the connection to the EDG is conservatively assumed to occur 40 seconds after LOOP. Consequently, the heat-up of the RCP [RCS] water during the time that LHSI/RHR trains are not available is approximately 1.4°C. Assuming an initial temperature of 55°C, the RCP [RCS] temperature rises to approximately 56.4°C.

<u>Single Failure (SF) and Preventive Maintenance (PM)</u>

In State C, PM is not implemented except for PM on two SGs and their associated systems (VDA [MSRT] and ASG [EFWS]). The most significant SF in this state is one EDG which fails to start on demand. This SF affects the operation of one LHSI/RHR pump and one of the two remaining ASG [EFWS] pumps. In addition, another LHSI/RHR train is assumed to be unavailable.

This loss of heat removal capability is not significant for the following reasons:

In states C1 and C2, the loss of two RIS/RRA [SIS/RHRS] trains can be compensated for by the two SGs on standby and their corresponding VDA [MSRT] setpoints set at a maximum of 5 bar. The VDA [MSRT] would automatically open if the operator had not manually actuated it earlier. The SG mass inventory, without further ASG [EFWS] injection, is sufficient to provide heat removal for more than 1 hour.

In state C3, the availability of only one LHSI/RHR train does not lead to an unacceptable RCP [RCS] heat-up above the acceptance criterion of 95°C. The LHSI/RHR train on an average removes about 20 MW for suction temperatures between 55°C and 95°C. Thus 10 MW remains for RCP [RCS] heat-up.

This heat load of 10 MW causes a temperature increase of the water and structural masses of 0.0125°C/s. Consequently, the heat-up starting form 56.4°C leads to a temperature of about 78.9°C after 30 minutes. Thus significant margin remains to the acceptance criterion of 95°C. After 30 minutes the LHSI/RHR train in standby is assumed to be activated manually. Once this train is activated the capacity of two LHSI/RHR trains at an RCP [RCS] temperature of 75°C to 80°C is about 50 MW. This exceeds the 30 MW of decay heat and thus the RCP [RCS] temperature drops [Ref-1].

1.2.2. Conclusions

The evaluation shows that, even with the availability of only one LHSI/RHR train for 30 minutes, the LOOP (short or long term) in the most limiting state C3 is adequately mitigated. Starting of the LHSI/RHR train in standby after 30 minutes finally establishes the safe shutdown state. In states C1 and C2 the heat removal can be ensured by the two SGs on standby.

1.3. SYSTEM SIZING

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

2. MAIN STEAM LINE BREAK

2.1. MAIN STEAM LINE BREAK (STATE A)

A main steam line break in state A with a break size greater than the area equivalent to a diameter of 50 mm (20 cm²), is classified as a PCC-4 event. This accident is analysed in this section by carrying out sensitivity studies.

The first part of this study covers the 2A steam line break event using sensitivity studies based on a reference case calculated at hot shutdown. The sensitivities considered are the following:

- Failure of one MHSI pump
- Reactor Coolant Pump trip
- Xenon free state
- RBS [EBS] automatic actuation

An additional calculation is performed for a steam line break equivalent to an open MSRT and assuming two stuck rods.

The last part of this study considers the consequences of a steam line break both at hot shutdown conditions and at full power, for the whole break spectrum. The break spectrum covers the break size range from 20 cm² to the double ended guillotine break (2A-SLB). For each case analysed, the document shows that the calculated minimum Departure in Nucleate Boiling Ratio (DNBR) and maximum linear power density meet the safety criteria.

A main steam line break in state A with a break size lower than the area equivalent to a diameter of 50 mm (20 cm²), is classified as a PCC-3 event. This accident is analysed in section 1 of Sub-chapter 14.4.

2.1.1. Identification of causes and accident description

2.1.1.1. General concern

The EPR main steam lines (MSL) are High Integrity Components, designed according to the break preclusion concept, for the length extending from the SG outlet to the MSL fixed point downstream of the VIV [MSIV]. As a consequence, the "2A-SLB upstream VIV [MSIV]" break does not need to be considered as a PCC-4 event. However, despite the break preclusion concept, the whole break spectrum (up to the 2A break) is considered in the present analysis for core behaviour analysis. This is a conservative approach which combines the largest steam line break size and the most onerous location. This sequence is addressed in sub-section 2.1.4 of this document.

The range of analysed break sizes encompasses the opening of the main steam safety valves (MSSV) and opening of the VDA [MSRT]. There are no larger nozzles connected to the MSL inside the containment.

2.1.1.2. Typical sequence of events

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

Once the break occurs, the secondary system depressurises. The SG pressure drop or low pressure signals actuate the reactor trip signal. These signals also:

- actuate the turbine trip (if reactor power is higher than 25% NP)
- close all VIV [MSIV],
- close all ARE [MFWS] high-load lines, if they are initially open for operation above approximately 20% FP, and
- isolate the ARE [MFWS] low-load lines of the affected SG.

For a steam line break at full power the reactor trip signal can also be generated following a high core thermal power, low SG Level or low DNBR signal which close all ARE [MFWS] high-load lines and initiate the turbine trip.

All these actions are automatic and are classified F1A.

After the above isolations, only the affected steam generator, which experiences a non-isolatable SLB, continues to depressurise. As a bounding assumption, all of the SGs are supplied with emergency feedwater ASG [EFWS] from the beginning of the transient.

The energy removed from the RCP [RCS] causes a reduction of coolant temperature and pressure, with actuation of MHSI and partial cooldown on a "Pressuriser pressure < MIN3" safety injection (SI) signal, which is classified F1A.

With a negative moderator coefficient, the RCP [RCS] cooldown results in an insertion of positive reactivity. The reactor becomes critical with a resultant power excursion. The Doppler Effect limits this power increase.

Once the affected steam generator is empty, the power is rapidly reduced to a level corresponding to the steam discharge of the ASG [EFWS]. This removes approximately 4% Full Power.

The operator acts to isolate the ASG [EFWS] line connected to the affected SG, and actuate the RBS [EBS], or the RCV [CVCS] if available. This initiates an RCP [RCS] boration with 7000 ppm of enriched boron. Subsequently the core rapidly returns to subcritical conditions:

- in the case of a large break, isolation of the ASG [EFWS] is sufficient to rapidly reach subcriticality, due to complete emptying of the affected SG with no further heat removal,
- in the case of a small break, actuation of the RBS [EBS], or the RCV [CVCS] if available, accelerates the return to subcriticality without requiring water inventory depletion in the affected SG. If the closing of the VIV [MSIV] and ARE [MFWS] line(s) associated with the affected SG is not performed automatically, the operator must initiate closure to ensure the complete isolation of the affected SG.

The controlled state is then reached with:

- the core subcritical,
- the residual core power removed by the unaffected SGs, via their ASG [EFWS], or the ARE/AAD [MFWS/SSS] if available, and VDA [MSRT]. For smaller breaks, where there is no automatic closing of the VIV [MSIV] , the GCT [MSB] may be used,
- the RCP [RCS] coolant inventory having been stabilised.

2.1.1.2.2. ***From the controlled state to the safe shutdown state***

The safe shutdown state is defined as a state where the core is subcritical, the RIS/RRA [SIS/RHRS] operating conditions are reached, the RCS pressure below 30 bar and hot leg temperature below 180°C, and the affected steam generator is isolated.

In this state, the heat removal function is performed by the LHSI pump(s) operating in residual heat removal mode (LHSI/RHR).

The sequence of actions to be performed by the operator to reach the RIS/RRA [SIS/RHRS] operating conditions are the following:

Confirmation of the isolation of the affected steam generator

At the controlled state, the affected steam generator should already be isolated. The first operator action is to confirm this isolation for the steam discharge and water supply. The objectives of these actions, other than the prevention of the uncontrolled RCP [RCS] cooldown, already addressed in the achievement of the controlled state, are to:

- prevent an unacceptable increase of containment pressure and temperature, if the steam pipe break is located inside containment,
- prevent a complete draining via the break of the associated ASG [EFWS] tank.

Subsequently, the corresponding ASG [EFWS] flow can be re-aligned to another SG via the ASG [EFWS] header, if necessary. This action is classified F1B.

After the isolation of the affected steam generator, the RCP [RCS] temperature increases and stabilises at a value which corresponds to that reached at the end of a partial cooldown, being controlled by the steam relief setpoint in the unaffected SGs.

RCP [RCS] boration

RCP [RCS] boration initiated at the controlled state must be continued when transferring the plant to the RIS/RRA [SIS/RHRS] operating conditions.

During the cooldown, the RCP [RCS] boration is performed via the RBS [EBS], classified F1 or the RCV [CVCS] if it is available. However, this latter system is not classified F1.

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

RCP [RCS] cooldown

The RCP [RCS] cooldown to the RIS/RRA [SIS/RHRS] connection temperature of 180°C is performed using the secondary side. This is achieved by decreasing the VDA [MSRT] setpoints on the unaffected SGs. This action is classified F1B. Note, the GCT [MSB] is unavailable following VIV [MSIV] closure.

The EPR cooling rate is consistent with the ASG [EFWS] tanks capacity. Consequently the RIS/RRA [SIS/RHRS] operating conditions are reached before the ASG [EFWS] tanks are emptied.

The EPR design cooling rate is 50°C/h if two RBS [EBS] trains are available, or 25°C/h if only one RBS [EBS] train is available, provided this is not limited by the VDA [MSRT] capacity.

RCP [RCS] depressurisation

After RCP [RCS] cooldown, if the RCP [RCS] pressure is greater than the RIS/RRA [SIS/RHRS] connection pressure of 30 bar, the operator will briefly open the Pressuriser Safety Valves (PSV). This occurs in some cases with reactor coolant pumps off where the pressuriser sprays are unavailable to depressurise the RCP [RCS]. This action is classified F1B.

During this depressurisation phase, the LHSI maintains a minimum RCP [RCS] pressure of about 20 bar so that sufficient RCP [RCS] subcooled margin remains. This system is classified F1A.

The RIS/RRA [SIS/RHRS] connection conditions are thus met.

2.1.1.2.3. *Precautions limiting the event consequences*

Each steam generator is equipped with integral multi-nozzle flow limiters. These limiters restrict the steam flow at the steam generator outlet in the event of a main steam line break whatever the location of the break.

2.1.2. Acceptance criteria

The safety criteria are the radiological limits for PCC-4 (see Sub-chapter 14.0).

The analysis is performed to demonstrate that the amount of fuel rods experiencing DNB remains below 10%.

Nevertheless, as a decoupling criterion for studies at hot shutdown, the present analysis confirms that no core damage occurs by showing that:

- the linear power density remains lower than 590 W/cm,
- the DNBR value remains higher than 1.12.

Consistent with the safety analysis rules defined in Sub-chapter 14.0, the controlled state is reached relying only on F1A means. The safe shutdown state is reached relying on F1A and F1B means.

Other acceptance criteria:

- Reactor coolant pressure boundary: RCP [RCS] cooldown can cause thermal shocks on the reactor vessel. The effects of cooldown on the reactor coolant system are analysed in Sub-chapter 3.4 related to "Mechanical systems and components"

- Reactor containment: The effect of a steam pipe break on containment pressure and temperature is analysed in Sub-chapter 6.2 related to "Containment systems"

- Radiological effects: The safety criteria to be met are the dose equivalent limits for release to the atmosphere as discussed in Sub-chapter 14.6. The bounding PCC-4 transient, for the radiological releases, is the SGTR (two tubes) discussed in section 10 of Sub-chapter 14.6.

2.1.3. Methods and assumptions

2.1.3.1. Methods of analysis

2.1.3.1.1. *Studies at hot shutdown*

The analysis of a steam pipe break is performed using the internal coupling of:

- the MANTA V3.7 code (see Appendix 14A) for the overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SGs),

- the SMART V4.5 and FLICA III-F V3 codes (see Appendix 14A) for neutronic and thermal-hydraulic behaviour of the core.

Initial neutronic parameters are calculated using the SMART code.

For each time step, the thermal-hydraulic conditions of the RCP [RCS] and SGs are calculated by the MANTA code. Then the thermal-hydraulic conditions at the core inlet, temperature distribution, flow, pressure and boron concentration, are transferred to the FLICA code which calculates the initial core thermal-hydraulics.

From the core thermal-hydraulic conditions, the SMART code calculates the neutronic parameters and transfers them to the FLICA code, repeating the process until convergence is achieved. From the neutronic parameters, the FLICA code calculates core thermal-hydraulics, and transfers them back to SMART.

For each time step the DNBR is calculated, with a radial mesh at the scale of the assembly, thereby determining the time for minimum DNBR. At the time of minimum DNBR a fine calculation is performed by FLICA with the boundary conditions of the transient simulation calculated by MANTA/SMART/FLICA and with the fine power distribution calculated by the coupled code SMART/FLICA. The DNB is calculated using the FC2000 correlation.

SMART returns the power that has been generated in each one of the 241 core assemblies to MANTA. MANTA then redistributes it in the four core quadrants modelled, with one corresponding to each primary loop.

2.1.3.1.2. *Study at full power*

The analysis of a steam line break is performed using the following codes:

- the MANTA V3.7 code (see Appendix 14A) for the overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SGs),
- the FLICA III-F V3 code (see Appendix 14A) for DNBR and percentage of fuel rods entering DNB calculations. The DNBR is calculated using the FC correlation.

2.1.3.2. Specific assumptions

2.1.3.2.1. *Neutronic data and decay heat*

2.1.3.2.1.1. At hot shutdown

It is assumed that the reactor is operating under the following conditions:

- The fuel management considered for the analysis is the 22 month cycle with UO_2 fuel in its equilibrium cycle, which has a high moderator feedback coefficient.
- End of life (EOL) shutdown margin, at hot full power equilibrium xenon conditions. When the single failure is assumed to be a stuck rod at reactor trip, the rod cluster control assembly having the highest worth is identified.

For all studies at hot shutdown (except the sensitivity to two stuck rods), the initial shutdown margin is equal to 4000 pcm. Unlike the initial shutdown margin in previous analysis [Ref-1] which used preliminary conservative neutronic parameters, the present data is consistent with EPR operating conditions. Therefore, the DNBR calculated in the 2A-SLB case is higher than in the previous analysis [Ref-1].

The shutdown margin used in the two stuck rods study is 2380 pcm.

2.1.3.2.1.2. At full power

The following conservative fuel cycle conditions are assumed:

- The fuel management considered for the analysis is the 22 month cycle with UO_2 fuel rods in its equilibrium cycle, which has a high moderator feedback coefficient.
- End of life (EOL), at full power.

The heat transfer coefficient from pellet to clad is assumed to be at its maximum value. This assumption minimises the pellet temperature and consequently the Doppler Effect. This maximises the transfer of energy to the clad, hence pessimising the DNBR calculation.

2.1.3.2.2. *Choice of single failure and preventive maintenance*

The most limiting single failure considered in each case studied is summarised as follows:

- For all studies except the sensitivity to two stuck rods: one rod cluster control assembly stuck in its fully withdrawn position at RT (the highest-worth RCCA). This is the most conservative single failure for a main steam line break in State A as described in Appendix 14C.

- For the sensitivity to two stuck RCCAs: the two highest worth RCCAs stuck in the fully withdrawn position.

Preventive maintenance is not assumed as it has no adverse impact on the transient.

2.1.3.2.3. ***Initial state***

2.1.3.2.3.1. At hot shutdown

For assessing DNBR after the reactor trip, a steam line break is more onerous when the plant is initially at hot shutdown.

If the reactor is at full power, the RCP [RCS] contains more energy than at hot shutdown as there is additional energy stored in the fuel, acting as inertia against the cooldown.

In addition, since the initial mass of secondary fluid and SG pressure are greater at hot shutdown, the magnitude and duration of the reactor coolant system cooldown are greater.

The transient analyses therefore model an initial state at hot shutdown, just after a RT.

A small initial nuclear power is pessimistic for the insertion of positive reactivity. A conservative value of 10^{-8} of nominal power is assumed.

The initial RCP [RCS] temperature corresponds to the hot shutdown state, without uncertainties. These uncertainties are included in the initial shutdown margin calculation.

RCP [RCS] boron concentration is assumed to be zero. This maximises the reactivity insertion during the RCP [RCS] cooldown.

The initial conditions, identical for all the cases, and appropriate for EPR, are presented in Section 14.5.2 - Table 1.

2.1.3.2.3.2. At full power

The initial conditions correspond to 100% Full Power, without fouling and plugging of the SG tubes. Uncertainties are taken into account in a conservative manner when calculating the DNBR. The plant is considered with degraded conditions for this break spectrum.

The initial conditions are summarised in Section 14.5.2 - Table 2.

The DNBR is assumed to be initially at the Limiting Condition Operating (LCO) value (1.32).

2.1.3.2.4. ***Assumptions related to non-F1 systems***

2.1.3.2.4.1. Main Feedwater

- Cases at hot shutdown:

A minimum ARE [MFWS] temperature of 111.4°C is assumed.

In the three cases, the high load ARE [MFWS] injection is modelled, although the high load ARE [MFWS] lines should have been isolated following the reactor trip. This bounding assumption covers the different initial plant states.

Reference case and sensitivity studies (except sensitivity to two stuck RCCAs)

During these studies, it is assumed that a conservative feedwater flow of 670 kg/s is delivered into all SGs. ARE [MFWS] flow rate is taken at a conservative value to cover each power level.

Break spectrum

For the study at hot shutdown, it is assumed that the feedwater flow delivered to each SG is at a conservative value for the hot shutdown state, i.e. 20% of maximum flow rate. This is the maximum possible ARE [MFWS] flow rate at 0% Power. This value is different from the one considered in the others studies in which the whole power range from 0% to 100% Nominal Power was covered. This assumption is also taken into account for the study ‘two stuck RCCAs’.

- For the calculations at full power, it is assumed that the feedwater flow delivered into each SG is the nominal value (638 kg/s) at power.

2.1.3.2.5. ***Assumptions related to F1 systems***

2.1.3.2.5.1. VIV [MSIV] (F1A)

All VIV [MSIV] are closed on a "SG pressure drop > MAX1" signal with a setpoint of 5 bar/min. The setpoint of this signal is adjusted at 7 bar below the initial SG pressure. The delay for steam lines isolation consists of 0.9 seconds of channel delay, plus 5 seconds for the valve closure time which is modelled as a step. [Ref-1]

2.1.3.2.5.2. ARE [MFWS] high-load line isolation (F1A)

The ARE [MFWS] high-load line to the affected steam generator is isolated following all reactor trip signals. In the case of a large break, ARE [MFWS] high-load line is isolated following a "SG pressure drop > MAX1" signal with a setpoint of 5 bar/min. The setpoint of this signal is adjusted at 7 bar below the initial SG pressure. The delay for feedwater isolation consists of 0.9 seconds of channel delay, plus 20 seconds for the valve closure time, which is modelled as a step. [Ref-1]

2.1.3.2.5.3. ARE [MFWS] low-load line isolation (F1A):

The ARE [MFWS] low-load line to the affected steam generator is isolated on a "SG pressure drop > MAX2" signal with a setpoint of 5 bar/min. The setpoint of this signal is adjusted at 17 bar below the initial SG pressure. The delay for feedwater isolation consists of 0.9 seconds of channel delay, plus 20 seconds for the valve closure time which is modelled as a step.

The ARE [MFWS] low-load lines to the unaffected steam generators are isolated on a "SG level > MAX1" signal with a setpoint of 89% + 5% uncertainty on the wide range. The delay for feedwater isolation consists of 1.5 seconds of channel delay, plus 20 seconds for the valve closure time which is modelled as a step. [Ref-1]

2.1.3.2.5.4. MHSI (F1A)

A minimum safety injection capacity is assumed, corresponding to the minimum MHSI characteristics with a minimum delivery pressure of 85 bar.

The temperature of the MHSI is the minimum IRWST temperature of 15°C. [Ref-1]

The boron concentration in the in-containment refuelling water storage tank is assumed to be 2305 ppm for all the cases.

The delay for MHSI injection consists of:

- the time to generate the safety injection signal, which consists of the time to reach the safety injection setpoint "Pressuriser pressure < MIN3" of 110 bar (115 - 5 bar), plus the channel delay of 0.9 seconds to generate the signal,
- the time to start the MHSI pumps, assumed to be 10 seconds,
- the time for the MHSI pumps to reach full flow, assumed to be 5 seconds. [Ref-1]

The accumulators are modelled with a minimal injection pressure of 45 bar [Ref-1].

2.1.3.2.5.5. ASG [EFWS] (F1A)

ASG [EFWS] supplies all the SGs. The ASG [EFWS] flow rate is isolated on “high SG level MAX1”..

A maximum flow rate of 120 t/h to the SGs is assumed.

A minimum ASG [EFWS] temperature of 10°C is assumed. [Ref-1].

<u>At 0% NP</u>

It is conservatively assumed that the ASG [EFWS] is actuated at the start of the transient.

<u>At 100% NP</u>

The ASG [EFWS] is automatically actuated on SG level < MIN2.

2.1.3.2.6. ***Other assumptions***

2.1.3.2.6.1. Assumptions related to reactor coolant pump trip

If a loss of offsite power occurred at the start of the event, the accident would be less severe. When offsite power is lost, reactor coolant pumps stop and the coolant flow in the RCP [RCS] decreases which reduces the power generation in the core.

There is no dedicated automatic reactor coolant pump trip implemented to mitigate the steam line break event. However, for a SLB inside containment, the containment isolation stage 2 actuation on a "Containment pressure > MAX2" signal might occur. The reactor coolant pumps are then tripped by this signal as they are no longer cooled by the RRI [CCWS].

2.1.3.2.6.2. Break flow (or stuck-open valve flow) correlation

Steam flow through the break, or a stuck open valve, is computed at each calculation step. A perfect moisture separation in the steam generator secondary side is assumed, with a discharge quality of 1 at the steam generator outlet). Pure steam flow at the break maximises the RCP [RCS] cooling. The back pressure is assumed to remain at atmospheric pressure to maximise the break flow rate.

During the first seconds of the transient prior to main steam line isolation:

- the steam flow from the affected steam generator is limited by the flow limiter, with a cross-section area of 0.13 m^2, [Ref-1]
- the total steam flow from the three unaffected steam generators is limited by the VIV [MSIV] located on the steam line of the affected steam generator, with a cross-section area of 0.32 m^2. [Ref-2]

2.1.3.2.6.3. Other assumptions maximising core cooling

A maximum SG tube heat transfer coefficient is used in the analyses.

It is assumed in the MANTA code that for core inlet mixing a maximum of 73% of the flow entering through an inlet nozzle remains in the associated core quadrant. This minimum loop flow mixing within the reactor pressure vessel is conservative for the core power transient. [Ref-1]

The main assumptions are given in Section 14.5.2 - Table 1.

2.1.3.3. Protection and mitigation actions

The following F1A I&C functions provide protection following a steam pipe break assessed against the DNBR criterion.

- Reactor trip signal on:
  - SG pressure drop > MAX1,
  - SG pressure < MIN1,
  - pressuriser pressure < MIN2,
  - core power level > MAX3,
  - DNBR < MIN3.
- Closure of all main steam isolation valves on:
  - SG pressure drop > MAX1,
  - SG pressure < MIN1.
- Isolation of all main feedwater high-load lines and turbine trip on:
  - any reactor trip signal

- Isolation of the main feedwater low-load lines to the unaffected SGs (SG related) on:
  - o SG level > MAX1.
- Isolation of the main feedwater low load line to the affected SG on:
  - o SG pressure drop > MAX2,
  - o SG pressure < MIN2,
  - o SG level > MAX1.
- Automatic actuation of the ASG [EFWS] of the affected SG on SG water level < MIN2 signal. This assumption is considered only for the break spectrum study at full power.

Each signal is pessimised to delay the reactor trip and thereby minimise the DNBR value and maximise the percentage of fuel rods entering DNB. The assumptions applied comply with the uncertainties defined in Sub-chapter 14.1.

2.1.4. Reference case at hot shutdown

The transient calculation and description are split into two phases, the short term phase lasting until the power peak and minimum DNBR have been reached, and the long term phase lasting until the controlled state has been reached with the core sub-critical.

2.1.4.1. Short term phase

The double-ended guillotine rupture of the MSL at the SG outlet combined with the stuck rod bounds all other breaks and leaks when assessing the core behaviour. It results in the largest core overcooling, as this is the largest non-isolatable break size, and the highest reactivity impact due to the assumption of the SF of a "stuck rod".

On the basis of this reference case, sensitivity studies are analysed. These cases are calculated with data appropriate to the EPR.

The transient calculation and description are presented for the first phase lasting until the peak power and minimum DNBR have been reached.

The double-ended guillotine break of the main steam line leads to a rapid depressurisation of the secondary side.

The "SG pressure drop > MAX1" signal setpoint is reached almost immediately after the break occurs. This signal leads to isolation of the steam lines at 6 seconds.

After VIV [MSIV] closure, only the affected SG continues to depressurise.

The "SG pressure drop > MAX2" signal setpoint is reached at 1 second in the affected SG. This signal leads to the complete isolation of ARE [MFWS] to the affected SG 20 seconds later. This SG is then only fed by the ASG [EFWS].

The reactor becomes critical and hence the core thermal power increases.

The Doppler feedback limits the return to power. During the power increase, boiling occurs in the upper part of the fuel assembly with the stuck rod. This also limits the power increase. After 400 seconds the thermal power has stabilised.

The thermal power at the time of minimum DNBR is calculated to be 4.36% NP

The minimum value of DNBR is reached at 217.1 seconds and is equal to 2.98. This shows that the decoupling criterion of no core DNB is achieved as the criterion of DNBR > 1.12 is met with a significant margin.

During the final phase of SG draining, the thermal power decreases and the fast secondary transient ends. Consequently, the RCP [RCS] overcooling ceases.

Section 14.5.2 - Table 3 gives the sequence of events.

Section 14.5.2 - Table 4 gives the main thermal-hydraulic parameters at the time of minimum DNBR.

Section 14.5.2 - Figures 1 to 8 show the development of the main parameters during the transient.

2.1.4.2. Long term phase

The long term phase calculation is detailed in section 5.1.2 of Appendix 14C.

The controlled state is reached:

- the core is sub-critical,
- the core power is fully removed by the unaffected SG,
- the RCP [RCS] coolant inventory is stable,
- the affected SG is empty and isolated from the environment.

The controlled state has been reached using only F1A means:

- VIV [MSIV] and MFWS valves for isolation of the affected SG (automatic actions),
- ASG [EFWS] and VDA [MSRT] for RCP [RCS] heat removal from the unaffected SG (automatic actions),
- MHSI actuation on SI signal, and temporary injection from the accumulators (automatic actions),
- ASG [EFWS] isolation in the affected SG, and RBS [EBS] actuation (manual actions).

The development of the main parameters is shown in Appendix 14C – Figures 12 to 19.

2.1.5. Sensitivity study: SLB at hot shutdown with failure of one MHSI pump

The second study is a 2A Steam Line Break considering a failure on one MHSI pump. The failure on one MHSI pump on the affected loop has been chosen as a conservative assumption.

The assumptions used in this sensitivity study are exactly the same as those in the reference case, except for the failure to start of one MHSI pump is assumed.

The transient calculation and description are presented for the first phase lasting until the peak power and minimum DNBR have been reached.

The boron concentration has just started to increase when the minimum DNBR is reached. Thus the differences between this case and the reference case are not significant.

The thermal power at the time of minimum DNBR is calculated to be 4.49% NP

The minimum value of DNBR is reached at 258.4 seconds and is equal to 2.94. This shows that the decoupling criterion of no core DNB is achieved as the criterion of DNBR > 1.12 is met with a significant margin.

During the last phase of SG draining, the thermal power decreases and the fast secondary transient ends. Consequently, the RCP [RCS] overcooling ceases.

Section 14.5.2 - Table 6 gives the sequence of events.

Section 14.5.2 - Table 7 gives the main thermal-hydraulic parameters at the time of minimum DNBR.

Section 14.5.2 - Figures 9 to 16 show the development of the main parameters during the transient.

2.1.6. Sensitivity study: SLB at hot shutdown with Reactor Coolant Pump trip

In this sensitivity study, three cases are analysed. Reactor Coolant Pump trip occurs when:

- core reactivity reaches a maximum value (case 1),
- the DNBR reaches a minimum value (case 2),
- core power reaches a maximum value (case 3).

The transients calculations and description are presented for the first phase lasting until the minimum DNBR has been reached.

The assumptions used in these sensitivity studies are exactly the same as those in the reference case, except that the trip of the Reactor Coolant Pumps is considered.

2.1.6.1. Reactor Coolant Pump trip at the time of maximum reactivity

The results show that the primary flow rate decrease due to the Reactor Coolant Pump trip (occurring at 91 seconds) leads to a lower cooling of the core than for case 1. Therefore the core power remains lower than for case 1 and the minimum value of DNBR remains far above the criterion.

The minimum value of DNBR is reached at 143.8 seconds and is equal to 6.35. This shows that the decoupling criterion of no core DNB is achieved as the criterion of DNBR > 1.12 is met with a significant margin.

Section 14.5.2 - Table 9 gives the sequence of events.

Section 14.5.2 - Table 10 gives the main thermal-hydraulic parameters at the time of minimum DNBR.

Section 14.5.2 - Figures 17 to 25 show the development of the main parameters during the transient.

2.1.6.2. Reactor Coolant Pump trip at the time of minimum DNBR

Reactor Coolant Pump trip occurs at the time of minimum DNBR in the reference case, i.e. at 217.1 seconds.

The minimum value of DNBR is reached at 218.1 seconds and is equal to 2.98. It is the same value as in the reference case because the minimum value has already been reached when the Reactor Coolant Pump trip occurs and subsequently the DNBR increases. This shows that the decoupling criterion of no core DNB is achieved as the criterion of DNBR > 1.12 is met with a significant margin.

Section 14.5.2 - Table 12 gives the sequence of events.

Section 14.5.2 - Table 13 gives the main thermal-hydraulic parameters at the time of minimum DNBR.

Section 14.5.2 - Figures 26 to 34 show the development of the main parameters during the transient.

2.1.6.3. Reactor Coolant Pump trip at the time of maximum core power

Reactor Coolant Pump trip occurs at the time maximum core power in the reference case, i.e. at 216.5 seconds.

The minimum value of DNBR is reached at 217.6 seconds and is equal to 3.00. This shows that the decoupling criterion of no core DNB is achieved as the criterion of DNBR > 1.12 is met with a significant margin.

The results presented in sub-section 2.1.6.2 and are almost the same as those presented in the reference case. The Reactor Coolant Pump trip occurs at the time of maximum core power in the reference case and leads to an increase in the DNBR value. Thus, the minimum DNBR occurs at the same time and almost in same overall conditions as those in the reference case.

Section 14.5.2 - Table 15 gives the sequence of events.

Section 14.5.2 - Table 16 gives the main thermal-hydraulic parameters at the time of minimum DNBR.

Section 14.5.2 - Figures 35 to 44 show the development of the main parameters during the transient.

2.1.6.4. Conclusion on sensitivities related to Reactor Coolant Pump trip time

The sensitivity study shows that the value of the DNBR is almost the same as in the reference case. The sensitivity on the Reactor Coolant Pump trip shows that this failure is beneficial with regard to the DNBR criterion because there is almost no cooling of the core. Thus, it is more beneficial if Reactor Coolant Pump trip occurs early in the transient. During this transient the DNBR remains high. The DNBR criterion is met in the three cases.

2.1.7. Sensitivity study: SLB at hot shutdown in xenon free state

A sensitivity study assuming zero xenon concentration at the beginning of the transient is performed. The case studied has the same initial conditions as the reference case, i.e. a 2A-SLB at hot shutdown conditions.

The xenon concentration is set to zero at the beginning of the transient, all other assumptions being identical to those of the reference case.

The transient calculation and description are presented for the first phase lasting until the peak power and minimum DNBR have been reached.

The thermal power at the time of minimum DNBR is calculated to be 6.01% NP

The minimum value of DNBR is reached at 217 seconds and is equal to 4.02. This shows that the decoupling criterion of no core DNB is achieved as the criterion of DNBR > 1.12 is met with a significant margin.

Section 14.5.2 - Table 18 gives the sequence of events.

Section 14.5.2 - Table 19 gives the main thermal-hydraulic parameters at the time of minimum DNBR.

Section 14.5.2 - Figures 44 to 51 show the development of the main parameters during the transient.

When assessing DNB, the sensitivity considering zero xenon concentration at the beginning of the transient is less onerous than the case presented in sub-section 2.1.4. This is because the axial power distribution is less peaked when compared to the reference case.

2.1.8. Sensitivity study: SLB at hot shutdown with RBS [EBS] automatic actuation

A sensitivity calculation is performed assessing an automatic actuation of the Extra Boration System. The RBS [EBS] is automatically actuated on SG pressure < MIN4.

The assumptions used in this sensitivity study are the same as those in the reference case, except that the automatic actuation of the RBS [EBS] is considered.

The transient calculation and description are presented for the first phase lasting until the peak power and minimum DNBR have been reached.

The thermal power at the time of minimum DNBR is 3.76% NP.

The minimum value of DNBR is reached at 215 seconds and is equal to 3.34. This shows that the decoupling criterion of no core DNB is achieved as the criterion of DNBR > 1.12 is met with a significant margin.

Section 14.5.2 - Table 21 gives the sequence of events.

Section 14.5.2 - Table 22 gives the main thermal-hydraulic parameters at the time of minimum DNBR.

Section 14.5.2 - Figures 52 to 68 show the development of the main parameters during the transient.

This sensitivity shows that RBS [EBS] actuation is beneficial for the transient. This is due the arrival of boron in the core earlier in the transient.

2.1.9. Sensitivity study: spurious opening of one MSRT at hot shutdown with two stuck RCCAS

A spurious opening of one MSRT is assessed with two stuck rods initiated from the hot shutdown state. The two worst stuck rods have been chosen following a sensitivity study. A specific shutdown margin is thus calculated for this study. It is equal to 2380 pcm, all other assumptions used in this sensitivity study are presented in Section 14.5.2 - Table 1.

The transient calculation and description are presented for the first phase lasting until the peak power and minimum DNBR have been reached.

The thermal power at the time of minimum DNBR is calculated to be 3.37% NP

The minimum value of DNBR is reached at 1446 seconds and is equal to 2.71. This shows that the decoupling criterion of no core DNB is achieved as the criterion of DNBR > 1.12 is met with a significant margin.

Section 14.5.2 - Table 24 gives the sequence of events.

Section 14.5.2 - Table 25 gives the main thermal-hydraulic parameters at the time of minimum DNBR.

Section 14.5.2 - Figures 61 to 59 show the development of the main parameters during the transient.

This sensitivity study is very conservative because of the assumed neutronic data and break size. Modelling, the two stuck rods with a lower shutdown margin leads to smaller margins against the relevant criteria. In addition, this break size delays the arrival of boron in the core by maintaining primary pressure close to the safety injection minimum injection pressure. This is the result of the slow pressure decrease and thus of the low safety injection flow rate.

2.1.10. Break spectrum at hot shutdown

The aim of these calculations is to analyse the consequences of the main steam line break at hot shutdown conditions, for the whole break spectrum. The break spectrum covers the break size range from 20 cm² to the double ended guillotine break (2A-SLB).

For each case analysed, the calculated minimum DNBR and maximum linear power density meet the safety criteria.

The assumptions used in this sensitivity study are the same as those in the reference case, except for the break size and MFW flow rate (see Section 14.5.2 - Table 1).

A break spectrum assessment is performed to analyse the DNBR variation as a function of steam line break size at hot shutdown. The following breaks are analysed:

- 20 cm²

- 158 cm² (corresponding to a spurious opening of one MSSV)
- 253 cm² (corresponding to spurious opening of one VDA [MSRT])
- 500 cm², 600 cm², 800 cm², 1000 cm², 1200 cm²
- 1300 cm² (corresponding to the steam flow limiter)
- 2A (double ended guillotine break)

When assessing the minimum DNBR, the most onerous break size is the 2A (double ended guillotine break) due to the rapid depressurisation of the primary side. The large breaks are therefore worst.

Section 14.5.2 - Figure 69 presents the results for breaks higher than 500 cm². The smaller breaks results are not presented because in these cases the increase of core thermal power is very low and thus the DNBR decrease and linear power density increase are not significant.

There is an intermediate break for which isolation of main steam lines is not calculated to occur. This break size is lower than the breaks presented, i.e. smaller than 500 cm². In that case the core power is close to 0% NP. Thus the DNBR decrease and the linear power density increase are not significant. The speed of the transient is very slow for this break size and consequently the operator can mitigate the transient 30 minutes after the initiating event.

2.1.10.1. Results of break size sensitivity

State A

The sequence of events is given in Section 14.5.2 - Table 27.

The most representative parameters are presented in the following figures:

- Section 14.5.2 - Figure 69: Break Spectrum at Hot Shutdown - Minimum DNBR as a Function of Steam Line Break Size
- Section 14.5.2 - Figure 70: Break Spectrum – Double-Ended Guillotine Break – Reactivity and Core Power
- Section 14.5.2 - Figure 71: Break Spectrum –Double-Ended Guillotine Break – Pressuriser Level and Pressuriser Pressure
- Section 14.5.2 - Figure 72: Break Spectrum –Double-Ended Guillotine Break – Hot Leg Temperature and Cold Leg Temperature
- Section 14.5.2 - Figure 73: , Break Spectrum –Double-Ended Guillotine Break – Total Accumulators Flow rate and Total MHSI Flow rate
- Section 14.5.2 - Figure 74: Break Spectrum – Double-Ended Guillotine Break – Vapour Mass Flow rate and SG Pressure
- Section 14.5.2 - Figure 75: Break Spectrum – Double-Ended Guillotine Break – Boron Concentration at Core Active Part Inlet and Emergency Feedwater Flow rate

- Section 14.5.2 - Figure 76: Break Spectrum – Double-Ended Guillotine Break – Main Feedwater Flow rate and SG Liquid Mass

- Section 14.5.2 - Figure 77: Break Spectrum – Double-Ended Guillotine Break – SG Narrow Range Level and SG Power Exchanged

The maximum core thermal power is calculated to be 4.15% of nominal power and the minimum DNBR value is 3.07 at 246 seconds. The maximum linear power density is 506 W/cm at 240 seconds.

2.1.10.2. Conclusion on break spectrum at hot shutdown calculations

The main steam line break spectrum analysis at hot shutdown shows that the acceptance criteria are met for all break sizes, i.e. minimum DNBR > 1.12 and maximum linear power density < 590 W/cm. The break spectrum at hot shutdown conditions shows that the most onerous break is the double-ended guillotine break and even in this case the margins are significant.

2.1.11. Break spectrum at power

The purpose of this part of the assessment is to analyse the consequences of the steam line break occurring at power conditions, covering the whole break spectrum. The break spectrum covers the break size range from 20 cm² to the double ended guillotine break (2A-SLB).

The analysis is performed to demonstrate that the number of fuel rods experiencing DNB remains below 10%.

The assumptions used in this sensitivity study are summarised in Section 14.5.2 - Table 2.

A break spectrum assessment is performed to variation of DNBR as a function of steam line break size at 100% NP. The following breaks are analysed:

- 20 cm²
- 158 cm² (corresponding to a spurious opening of one MSSV)
- 253 cm² (corresponding to a spurious opening of one VDA [MSRT])
- 800 cm²
- 1300 cm² (corresponding to the steam flow limiter)
- 1400 cm² (corresponding to the worst break)
- 1575 cm²
- 2A (double ended guillotine break)

When assessing the minimum DNBR, the worst break size is the 1400 cm² break.

Section 14.5.2 - Table 31 presents the DNBR results for each break size.

Others results are summarised for each break in Section 14.5.2 - Table 32.

2.1.11.1. Results for the worst break size – 1400 cm²

This break is the most onerous when assessing the safety criteria.

This break causes a large pressure decrease in the SGs. The core power increases and the DNBR decreases.

The reactor trip is actuated at 9.5 seconds simultaneously on high SG pressure drop and low DNBR. The maximum core thermal power is calculated to be 118.9% NP.

The sequence of events is given in Section 14.5.2 - Table 33.

The minimum DNBR is 0.91 and the total number of fuels rods entering DNB is 2%, significantly below the 10% limit. The PCC-4 criterion is therefore met.

The most representative parameters are presented in the following figures:

- Section 14.5.2 - Figure 93: Break Spectrum at Full Power, 1400 cm² - Penalising break - Reactivity and Nuclear Power
- Section 14.5.2 - Figure 94: Break Spectrum at Full Power, 1400 cm² - Penalising break - Pressuriser Level and Pressuriser Pressure
- Section 14.5.2 - Figure 95: Break Spectrum at Full Power, 1400 cm² - Penalising break - Hot and Cold Leg Temperature
- Section 14.5.2 - Figure 96: Break Spectrum at Full Power, 1400 cm² - Penalising break - SG Mass Flow rate and SG Pressure
- Section 14.5.2 - Figure 97: Break Spectrum at Full Power, 1400 cm² - Penalising break - Main Feed Water Flow Rate and SG Liquid Mass
- Section 14.5.2 - Figure 98: Break Spectrum at Full Power, 1400 cm² - Penalising break - Boron Concentration and ASG [EFWS] Flow rate
- Section 14.5.2 - Figure 99: Break Spectrum at Full Power, 1400 cm² - Penalising break - SG Narrow Range Level and SG Power Exchanged

2.1.11.2. Conclusion on break spectrum at full power calculations

The main steam line break spectrum analysis at full power shows that the acceptance criterion is met for all break sizes with significant margins.

This analyse demonstrates that the most onerous break occurs at the crossover between protection being provided by the high SG pressure drop and low DNBR reactor trip signals. The corresponding break area is 1400 cm², and for this worst case, the total number of fuel rods entering into DNB is 2%, below the PCC-4 criterion of 10%.

2.1.12. Conclusion

During this study, numerous cases of steam line break have been assessed.

At zero power, a 2A steam line break has been calculated with sensitivities to failure of one MHSI pump, Reactor Coolant Pump trip, zero xenon concentration and automatic RBS [EBS] actuation. Sensitivities on Reactor Coolant Pump trip, zero xenon concentration and automatic RBS [EBS] actuation are shown to result in an increase in the minimum DNBR and are therefore beneficial to the transient. The case considering the failure of one MHSI pump is almost the same as the reference case when comparing the minimum DNBR.

A case has been performed assessing the effect of two stuck rods following the spurious opening of one MSRT. This case meets the relevant criteria.

A break spectrum analysis has been carried out at zero power. The 2A break has been identified as the most onerous.

A break spectrum analysis has also been carried out at full power. In this case, the most onerous break is one of 1400 cm^2.

For all the analysed cases, the safety criteria are met with significant margins.

SECTION 14.5.2 - TABLE 1 (1/2)

Main Assumptions at Hot Shutdown

Main Steam Line Break

Parameters		Values used
Initial conditions		
- Reactor power	(FP)	10^{-8}
- Shutdown margin	(pcm)	- 4000
- Shutdown margin (in the case of two stuck rods)	(pcm)	- 2380
- RCP [RCS] boron concentration	(ppm)	0
- RCP [RCS] flow rate	(kg/s)	T/H design flow rate
- Average RCP [RCS] temperature	(°C)	303.3
- Pressuriser pressure	(bara)	152.5
- Pressuriser level	(% MR)	39 + 8.5 = 47.5
- SG level	(% NR)	49 + 6 = 55
- Flow limiter cross-section (per SG)	(m²)	0.13
- MSIV cross-section	(m²)	0.32
Studies at hot shutdown (except the break spectrum)		
- Initial ARE [MFWS] flow rate in the affected SG	(kg/s)	670
- Initial ARE [MFWS] flow rate in the unaffected SGs	(kg/s)	670
Break spectrum at hot shutdown		
- Initial ARE [MFWS] flow rate in the affected SG	(kg/s)	134
- Initial ARE [MFWS] flow rate in the unaffected SGs	(kg/s)	134
- ASG [EFWS] flow rate to all SGs	(kg/s)	33.3
MSRT		
- Opening setpoint	(bar abs)	95.5 - 5 = 90.5
- Partial cooldown gradient	(°C/h)	- 250
Safety injection:		
- Time to open valves and start pumps	(s)	10
- Time to reach full flow	(s)	5
- Concentration of borated water		
in IRWST	(ppm)	2305
in accumulators	(ppm)	2305

SECTION 14.5.2 - TABLE 1 (2/2)

Main Assumptions at Hot Shutdown

Main Steam Line Break

Parameters		Values used
Extra Boration System - Concentration of borated water	(ppm)	11200
RBS [EBS] actuation setpoint, “SG pressure < MIN4”	(bar)	35
RBS [EBS] flow rate per train (two trains)	(kg/s)	2.8
RBS [EBS] actuation time	(s)	15
"SG pressure drop > MAX1" setpoint	(bar/min)	5 Setpoint adjusted 8.5 bar below the initial value (7 bar)
"SG pressure drop > MAX2" setpoint	(bar/min)	5 Setpoint adjusted 18.5 bar below the initial value (17 bar)
"Pressuriser pressure < MIN3" setpoint	(bara)	115 - 5 = 110
Steam line isolation delay	(s)	0.9 + 5 = 5.9
Main feedwater low-load line isolation delay	(s)	20

SECTION 14.5.2 - TABLE 2

Main Assumptions at Full Power

Main Steam Line Break

Parameters		Values used
Initial conditions		
- Reactor power	(FP)	102%
- RCP [RCS] flow rate	(kg/s)	T/H design flow rate
- Average RCP [RCS] temperature	(°C)	312.7 + 2.5 = 315.2
- Pressuriser pressure	(bara)	155 – 2.5 = 152.5
- Pressuriser level	(% MR)	56
- SG level	(% NR)	49 + 5 = 54
- Flow limiter cross-section (per SG)	(m^2)	0.13
- MSIV cross-section	(m^2)	0.32
Initial ARE [MFWS] temperature	(°C)	230
- Initial ARE [MFWS] flow rate in the affected SG	(kg/s)	638
- Initial ARE [MFWS] flow rate in the unaffected SG	(kg/s)	638
SAFETY SIGNALS		
"SG pressure drop > MAX1" setpoint	(bar/min)	5 Setpoint adjusted 8.5 bar below the initial value (7 bar)
"SG pressure drop > MAX2" setpoint	(bar/min)	5 Setpoint adjusted 18.5 bar below the initial value (17 bar)
"Pressuriser pressure < MIN3" setpoint	(bar)	115 - 5 = 110
Steam line isolation delay	(s)	0.9 + 5 = 5.9
Main feedwater low-load line isolation delay	(s)	20

SECTION 14.5.2 - TABLE 3

Sequence of Events – 2A SLB – Reference Case

EVENT	TIME (seconds)
Main steam line break	0
"SG pressure drop > MAX1" setpoint is reached (RT/MSIV isolation[1])	1
"SG pressure drop > MAX2" setpoint is reached in affected SG1	1.2
Main feedwater low-load line isolation signal in the affected SG on high SG pressure drop	1.2
Main feedwater high-load line isolation signal in all the SGs on high SG pressure drop	1.9
Steam lines isolation	6.0
ARE [MFWS] low-load line isolation signal in SG 2 , SG 3 and SG 4	11
ARE [MFWS] high-load line isolation in all the SGs	22
Pressuriser is empty (Pressuriser level = 0)	28
ARE [MFWS] low-load line isolation in SG 2, SG 3,and SG 4	32
"Pressuriser pressure < MIN3" setpoint is reached (SIS actuation and partial cooldown actuation)	51
Reactor becomes critical	61
MSRT opening (SG 3)	63
"SG pressure drop > MAX2" setpoint is reached in SG 3	68
MHSI begins to inject	99
MSRT opening (SG 4)	102
MSRT opening (SG 2)	117
Boron concentration - first arrival of boron at core inlet	208
Minimum DNBR is reached (2.98)	**217.1**
Maximum linear power density is reached (518.3 W/cm)	**255**

[1] The SLB being initiated at 0% NP, the RT has already occurred, and thus only the VIV [MSIV] closure is performed.

SECTION 14.5.2 - TABLE 4

Conditions at the Time of Minimum DNBR – 2A SLB – Reference Case

Time	217.1 s
Thermal power	4.36% NP
Concentration of boron in the core	4.9 ppm
Average core pressure	62.3 bar
Cold leg temperature in affected loop	201.9°C
Cold leg temperature in unaffected loops (loop 2)	237.9°C
Core flow rate (fraction of nominal)	1
Minimum DNBR	**2.98**

SECTION 14.5.2 - TABLE 5

Conditions at the Time of Maximum Linear Power Density -2A SLB – Reference Case

Time	255 s
Thermal power	4.28% NP
Concentration of boron in the core	32.0 ppm
Average core pressure	57.8 bar
Cold leg temperature in affected loop	198.8°C
Cold leg temperature in unaffected loops	232.2°C
Core flow rate (fraction of nominal)	1
Maximum linear power density (W/cm)	**518.3**

SECTION 14.5.2 - TABLE 6

Sequence of Events - 2A SLB – failure of one MHSI pump

EVENT	TIME (seconds)
Main steam line break	0
"SG pressure drop > MAX1" setpoint is reached (RT/MSIV isolation[2])	1
"SG pressure drop > MAX2" setpoint is reached in affected SG1	1.2
Main feedwater low-load line isolation signal in the affected SG on high SG pressure drop	1.2
Main feedwater high-load line isolation signal in all the SGs on high SG pressure drop	1.9
Steam lines isolation	6.0
ARE [MFWS] low-load line isolation signal in SG 2 , SG 3 and SG 4	11
ARE [MFWS] high-load line isolation in all the SGs	22
Pressuriser empty (Pressuriser level = 0)	28
ARE [MFWS] low-load line isolation in SG 2, SG 3, SG 4	32
"Pressuriser pressure < MIN3" setpoint is reached (SIS actuation and partial cooldown actuation)	51
Reactor becomes critical	61
MSRT opening (SG 3)	63
"SG pressure drop > MAX2" setpoint is reached in SG 3	68
MHSI begins to inject	99
MSRT opening (SG 4)	102
MSRT opening (SG 2)	117
Boron concentration - first arrival of boron at core inlet	206
Minimum DNBR is reached (2.94)	**258.4**
Maximum linear power density is reached (531.4 W/cm)	**290**

[2] The SLB being initiated at 0% NP, the RT has already occurred, and thus only the VIV [MSIV] closure is performed.

SECTION 14.5.2 - TABLE 7

Conditions at the Time of Minimum DNBR –2A SLB – Failure of One MHSI Pump

Time	258.4 s
Thermal power	4.49% NP
Concentration of boron in the core	28.4 ppm
Average core pressure	55.7 bar
Cold leg temperature in affected loop	199.3°C
Cold leg temperature in unaffected loops (loop 2)	232.3°C
Core flow rate (fraction of nominal)	1
Minimum DNBR	**2.94**

SECTION 14.5.2 - TABLE 8

Conditions at the Time of Maximum Linear Power Density –2A SLB – Failure of One MHSI Pump

Time	290 s
Thermal power	4.46% NP
Concentration of boron in the core	48.5 ppm
Average core pressure	51.8 bar
Cold leg temperature in affected loop	196.7°C
Cold leg temperature in unaffected loops (loop 2)	228.0°C
Core flow rate (fraction of nominal)	1
Minimum linear power density (W/cm)	**531.4**

SECTION 14.5.2 - TABLE 9

Sequence of Events –2A SLB – Reactor Coolant Pump Trip at Time of Maximum Reactivity

EVENT	TIME (seconds)
Main steam line break	0
"SG pressure drop > MAX1" setpoint is reached (RT/MSIV isolation[3])	1
"SG pressure drop > MAX2" setpoint is reached in affected SG1	1.2
Main feedwater low-load line isolation signal in the affected SG on high SG pressure drop	1.2
Main feedwater high-load line isolation signal in all the SGs on high SG pressure drop	1.9
Steam lines isolation	6.0
ARE [MFWS] low-load line isolation signal in SG 2 , SG 3 and SG4	11
ARE [MFWS] high-load line isolation in all the SGs	22
Pressuriser is empty (Pressuriser level = 0)	28
ARE [MFWS] low-load line isolation in SG 2, SG 3, SG 4	32
"Pressuriser pressure < MIN3" setpoint is reached (SIS actuation and partial cooldown actuation)	51
Reactor becomes critical	61
MSRT opening (SG 3)	63
"SG pressure drop > MAX2" setpoint is reached in SG 3	68
Reactor Coolant Pump trip	91
MHSI begins to inject	93
MSRT opening (SG 4)	102
MSRT opening (SG 2)	115
Maximum linear power density is reached (161.1 W/cm)	**132**
Minimum DNBR is reached (6.35)	**143.8**

[3] The SLB being initiated at 0% NP, the RT has already occurred, and thus only the VIV [MSIV] closure is performed.

SECTION 14.5.2 - TABLE 10

Conditions at the Time of Minimum DNBR –2A SLB – Reactor Coolant Pump Trip at Time of Maximum Reactivity

Time	143.8 s
Thermal power	1.31% NP
Concentration of boron in the core	0 ppm
Average core pressure	77.5 bar
Cold leg temperature in affected loop	194.3°C
Cold leg temperature in unaffected loops (loop 2)	259.3°C
Core flow rate (fraction of nominal)	0.21
Minimum DNBR	**6.35**

SECTION 14.5.2 - TABLE 11

Conditions at the Time of Maximum Linear Power Density –2A SLB – Reactor Coolant Pump Trip at time of maximum Reactivity

Time	132 s
Thermal power	1.25% NP
Concentration of boron in the core	0 ppm
Average core pressure	78.0 bar
Cold leg temperature in affected loop	199.4°C
Cold leg temperature in unaffected loops (loop 2)	260.3°C
Core flow rate (fraction of nominal)	0.26
Maximum linear power density (W/cm)	**161.1**

SECTION 14.5.2 - TABLE 12

Sequence of Events –2A SLB – Reactor Coolant Pump Trip at time of minimum DNBR

EVENT	TIME (seconds)
Main steam line break	0
"SG pressure drop > MAX1" setpoint is reached (RT/MSIV isolation[4])	1
"SG pressure drop > MAX2" setpoint is reached in affected SG1	1.2
Main feedwater low-load line isolation signal in the affected SG on high SG pressure drop	1.2
Main feedwater high-load line isolation signal in all the SGs on high SG pressure drop	1.9
Steam lines isolation	6.0
ARE [MFWS] low-load line isolation signal in SG 2 , SG 3 and SG4	11
ARE [MFWS] high-load line isolation in all the SGs	22
Pressuriser is empty (Pressuriser level = 0)	28
ARE [MFWS] low-load line isolation in SG 2, SG 3, SG 4	32
"Pressuriser pressure < MIN3" setpoint is reached (SIS actuation and partial cooldown actuation)	51
Reactor becomes critical	61
MSRT opening (SG 3)	63
"SG pressure drop > MAX2" setpoint is reached in SG 3	68
MHSI begins to inject	99
MSRT opening (SG 4)	102
MSRT opening (SG 2)	117
Boron concentration - first arrival of boron at core inlet	208
Maximum linear power density is reached (520.0 W/cm)	**210**
Reactor Coolant Pump trip	217.1
Minimum DNBR is reached (2.98)	**218.1**

[4] The SLB being initiated at 0% NP, the RT has already occurred, and thus only the VIV [MSIV] closure is performed.

SECTION 14.5.2 - TABLE 13

Conditions at the Time of Minimum DNBR – 2A SLB – Reactor Coolant Pump Trip at time of minimum DNBR

Time	218.1 s
Thermal power	4.34% NP
Concentration of boron in the core	5.61 ppm
Average core pressure	62.2 bar
Cold leg temperature in affected loop	201.8°C
Cold leg temperature in unaffected loops (loop 2)	237.8°C
Core flow rate (fraction of nominal)	1
Minimum DNBR	**2.98**

SECTION 14.5.2 - TABLE 14

Conditions at the Time of Maximum Linear Power Density –2A SLB – Reactor Coolant Pump Trip at time of minimum DNBR

Time	210 s
Thermal power	4.3% NP
Concentration of boron in the core	1.6 ppm
Average core pressure	63.3 bar
Cold leg temperature in affected loop	202.4°C
Cold leg temperature in unaffected loops (loop 2)	239.0°C
Core flow rate (fraction of nominal)	1
Maximum linear power density	**520.0**

SECTION 14.5.2 - TABLE 15

Sequence of Events – 2A SLB – Reactor Coolant Pump Trip at time of peak Core Power

EVENT	TIME (seconds)
Main steam line break	0
"SG pressure drop > MAX1" setpoint is reached (RT/MSIV isolation[5])	1
"SG pressure drop > MAX2" setpoint is reached in affected SG1	1.2
Main feedwater low-load line isolation signal in the affected SG on high SG pressure drop	1.2
Main feedwater high-load line isolation signal in all the SGs on high SG pressure drop	1.9
Steam lines isolation	6.0
ARE [MFWS] low-load line isolation signal in SG 2 , SG 3 and SG4	11
ARE [MFWS] high-load line isolation in all the SGs	22
Pressuriser is empty (Pressuriser level = 0)	28
ARE [MFWS] low-load line isolation in SG 2, SG 3, SG 4	32
"Pressuriser pressure < MIN3" setpoint is reached (SIS actuation and partial cooldown actuation)	51
Reactor becomes critical	61
MSRT opening (SG 3)	63
"SG pressure drop > MAX2" setpoint is reached in SG 3	68
MHSI begins to inject	99
MSRT opening (SG 4)	102
MSRT opening (SG 2)	117
Boron concentration - first arrival of boron at core inlet	208
Maximum linear power density is reached (520.0 W/cm)	**210**
Reactor Coolant Pump trip	216.5
Minimum DNBR is reached (3.00)	**217.6**

[5] The SLB being initiated at 0% NP, the RT has already occurred, and thus only the VIV [MSIV] closure is performed.

SECTION 14.5.2 - TABLE 16

Conditions at the Time of Minimum DNBR – 2A SLB – Reactor Coolant Pump Trip at time of peak Core Power

Time	217.6 s
Thermal power	4.33% NP
Concentration of boron in the core	5.31 ppm
Average core pressure	62.2 bar
Cold leg temperature in affected loop	201.8°C
Cold leg temperature in unaffected loops (loop 2)	237.9°C
Core flow rate (fraction of nominal)	0.96
Minimum DNBR	**3.00**

SECTION 14.5.2 - TABLE 17

Conditions at the Time of Maximum Linear Power Density –2A SLB – Reactor Coolant Pump Trip at time of peak Core Power

Time	210 s
Thermal power	4.3% NP
Concentration of boron in the core	1.6 ppm
Average core pressure	63.3 bar
Cold leg temperature in affected loop	202.4°C
Cold leg temperature in unaffected loops (loop 2)	239.0°C
Core flow rate (fraction of nominal)	1
Maximum linear power density (W/cm)	**520.0**

SECTION 14.5.2 - TABLE 18

Sequence of Events – 2A SLB – Sensitivity to Xenon Concentration

EVENT	TIME (seconds)
Main steam line break	0
"SG pressure drop > MAX1" setpoint is reached (RT/MSIV isolation[6])	1
"SG pressure drop > MAX2" setpoint is reached in affected SG1	1.2
Main feedwater low-load line isolation signal in the affected SG on high SG pressure drop	1.2
Main feedwater high-load line isolation signal in all the SGs on high SG pressure drop	1.9
Steam lines isolation	6.0
ARE [MFWS] low-load line isolation signal in SG 2 , SG 3 and SG4	11
ARE [MFWS] high-load line isolation in all the SGs	22
Pressuriser is empty (Pressuriser level = 0)	28
ARE [MFWS] low-load line isolation in SG 2, SG 3, SG 4	32
"Pressuriser pressure < MIN3" setpoint is reached (SIS actuation and partial cooldown actuation)	51
Reactor becomes critical	57
MSRT opening (SG 3)	63
"SG pressure drop > MAX2" setpoint is reached in SG 3	68
MHSI begins to inject	100
MSRT opening (SG 4)	102
MSRT opening (SG 2)	117
Boron concentration - first arrival of boron at core inlet	210
Minimum DNBR is reached (4.02)	**217**
Maximum linear power density is reached (369.6 W/cm)	**280**

[6] The SLB being initiated at 0% NP, the RT has already occurred, and thus only the VIV [MSIV] closure is performed.

SECTION 14.5.2 - TABLE 19

Conditions at the Time of Minimum DNBR – 2A SLB – Sensitivity to Xenon Concentration

Time	217 s
Thermal power	6.01% NP
Concentration of boron in the core	3.30 ppm
Average core pressure	63.2 bar
Cold leg temperature in affected loop	204.1°C
Cold leg temperature in unaffected loops (loop 2)	240.1°C
Core flow rate (fraction of nominal)	1
Minimum DNBR	**4.02**

SECTION 14.5.2 - TABLE 20

Conditions at the Time of Maximum Linear Power Density – 2A SLB – Sensitivity to Xenon Concentration

Time	280 s
Thermal power	5.6% NP
Concentration of boron in the core	48.6 ppm
Average core pressure	56.2 bar
Cold leg temperature in affected loop	199.5°C
Cold leg temperature in unaffected loops (loop 2)	231.9°C
Core flow rate (fraction of nominal)	1
Maximum linear power density	**369.6**

SECTION 14.5.2 - TABLE 21

Sequence of Events – 2A SLB – Sensitivity to RBS [EBS] Actuation

EVENT	TIME (seconds)
Main steam line break	0
"SG pressure drop > MAX1" setpoint is reached (RT/MSIV isolation[7])	1
"SG pressure drop > MAX2" setpoint is reached in affected SG1	1.2
Main feedwater low-load line isolation signal in the affected SG on high SG pressure drop	1.2
Main feedwater high-load line isolation signal in all the SGs on high SG pressure drop	1.9
Steam lines isolation	6.0
ARE [MFWS] low-load line isolation signal in SG 2 , SG 3 and SG4	11
ARE [MFWS] high-load line isolation in all the SGs	22
Pressuriser is empty (Pressuriser level = 0)	28
“SG pressure > MIN4” setpoint is reached (RBS [EBS] actuation)	31
ARE [MFWS] low-load line isolation in SG 2, SG 3, SG 4	32
RBS [EBS] start to inject	46
"Pressuriser pressure < MIN3" setpoint is reached (SIS actuation and partial cooldown actuation)	52
Reactor becomes critical	61
MSRT opening (SG 3)	63
"SG pressure drop > MAX2" setpoint is reached in SG 3	68
MHSI begins to inject	100
MSRT opening (SG 4)	102
MSRT opening (SG 2)	117
Boron concentration - first arrival of boron at core inlet	160
Maximum linear power density is reached (470.4 W/cm)	**205**
Minimum DNBR is reached (3.34)	**215**

[7] The SLB being initiated at 0% NP, the RT has already occurred, and thus only the VIV [MSIV] closure is performed.

SECTION 14.5.2 - TABLE 22

Conditions at the Time of Minimum DNBR – 2A SLB – Sensitivity to RBS [EBS] Actuation

Time	215 s
Thermal power	3.76% NP
Concentration of boron in the core	24.34 ppm
Average core pressure	63.0 bar
Cold leg temperature in affected loop	201.4°C
Cold leg temperature in unaffected loops (loop 2)	237.7°C
Core flow rate (fraction of nominal)	1
Minimum DNBR	**3.34**

SECTION 14.5.2 - TABLE 23

Conditions at the Time of Maximum Linear Power Density – 2A SLB – Sensitivity to RBS [EBS] Actuation

Time	205 s
Thermal power	3.7% NP
Concentration of boron in the core	18.8 ppm
Average core pressure	64.3 bar
Cold leg temperature in affected loop	202.3°C
Cold leg temperature in unaffected loops (loop 2)	239.4°C
Core flow rate (fraction of nominal)	1
Maximum linear power density	**470.4**

SECTION 14.5.2 - TABLE 24

Sequence of Events – Open MSRT – Sensitivity to Two Stuck Rods

EVENT	TIME (seconds)
Main steam line break	0
"SG pressure drop > MAX1" setpoint is reached (RT/MSIV isolation[8])	112
"SG pressure drop > MAX2" setpoint is reached in affected SG1	123
Main feedwater low-load line isolation signal in the affected SG on high SG pressure drop	123
Steam lines isolation	118
Pressuriser is empty (Pressuriser level = 0)	132
Reactor becomes critical	143
ARE [MFWS] low-load line isolation in SG 1	143
ARE [MFWS] low-load line isolation signal in SG 2 and SG4	157
ARE [MFWS] low-load line isolation signal in SG 3	167
ARE [MFWS] low-load line isolation in SG 2 and SG 4	177
"Pressuriser pressure < MIN3" setpoint is reached (SIS actuation and partial cooldown actuation)	180
ARE [MFWS] low-load line isolation in SG 3	187
MHSI begins to inject	251
Boron concentration - first arrival of boron at core inlet	383
MSRT opening (SG 3)	393
MSRT opening (SG 2)	416
MSRT opening (SG 4)	417
Minimum DNBR is reached (2.71)	**1446**
Maximum linear power density is reached (552.1 W/cm)	**1600**

[8] The SLB being initiated at 0% NP, the RT has already occurred, and thus only the VIV [MSIV] closure is performed.

SECTION 14.5.2 - TABLE 25

Conditions at the Time of Minimum DNBR – Open MSRT – Sensitivity to Two Stuck Rods

Time	1446 s
Thermal power	3.37% NP
Concentration of boron in the core	88.0 ppm
Average core pressure	79.7 bar
Cold leg temperature in affected loop	227.4°C
Cold leg temperature in unaffected loops (loop 2)	234.7°C
Core flow rate (fraction of nominal)	1
Minimum DNBR	**2.71**

SECTION 14.5.2 - TABLE 26

Conditions at the Time of Maximum Linear Power Density – Open MSRT – Sensitivity to Two Stuck Rods

Time	1600 s
Thermal power	3.36% NP
Concentration of boron in the core	90.8 ppm
Average core pressure	80.1 bar
Cold leg temperature in affected loop	227.2°C
Cold leg temperature in unaffected loops (loop 2)	233.8°C
Core flow rate (fraction of nominal)	1
Maximum linear power density	**552.1**

SECTION 14.5.2 - TABLE 27

Sequence of Events – 2A SLB – Break Spectrum at Hot Shutdown

EVENT	TIME (seconds)
Main steam line break	0.1
"SG pressure drop > MAX1" setpoint is reached (RT/MSIV isolation[9])	1.2
"SG pressure drop > MAX2" setpoint is reached in affected SG1	1.2
Main feedwater low-load line isolation signal in the affected SG on high SG pressure drop	2.0
Steam lines isolation	7.5
Main feedwater low-load line isolation signal in the unaffected SGs on high SG pressure drop	12
ARE [MFWS] low-load line is isolated in affected SG1	22
ARE [MFWS] low-load line is isolated in SG2, SG3, SG4	32
Pressuriser is empty (Pressuriser level = 0)	58
Reactor becomes critical	77
"Pressuriser pressure < MIN3" setpoint is reached (SIS actuation and partial cooldown actuation)	92
MHSI begins to inject	130
MSRT opening (SG 3)	155
"SG pressure drop > MAX2" setpoint is reached in SG 3	159
MSRT opening (SG 4)	192
MSRT opening (SG 2)	199
Boron concentration - first arrival of boron at core inlet	239
Maximum linear power density is reached (506 W/cm)	**240**
Minimum DNBR is reached (3.07)	**246**

[9] The SLB being initiated at 0% NP, the RT has already occurred, and thus only the VIV [MSIV] closure is performed.

SECTION 14.5.2 - TABLE 28

Conditions at the Time of Minimum DNBR – 2A SLB – Break Spectrum at Hot Shutdown

Time	246 s
Thermal power	4.15% NP
Concentration of boron in the core	4 ppm
Average core pressure	62.2 bar
Cold leg temperature in affected loop	202.4°C
Cold leg temperature in unaffected loops (loop 2)	239.9°C
Core flow rate (fraction of nominal)	1
Minimum DNBR	**3.07**

SECTION 14.5.2 - TABLE 29

Conditions at the Time of Maximum Linear Power Density – 2A SLB – Break Spectrum at Hot Shutdown

Time	240 s
Thermal power	4.15% NP
Concentration of boron in the core	1.5 ppm
Average core pressure	62.9 bar
Cold leg temperature in affected loop	202.8°C
Cold leg temperature in unaffected loops (loop 2)	240.8°C
Core flow rate (fraction of nominal)	1
Maximum linear power density	**506 W/cm**

SECTION 14.5.2 - TABLE 30

Break Spectrum at Hot Shutdown, minimum DNBR as a function of steam line break size

Break size (cm²)	500	600	800	1000	1200	1300	2A
Minimum DNB	8.69	6.32	4.55	3.73	3.25	3.08	3.07
Maximum Linear power density (W/cm)	175.87	251.66	346.60	421.09	480.51	502.39	506.05

SECTION 14.5.2 - TABLE 31

Break Spectrum at Full Power, minimum DNBR value as a function of steam line break size

Break diameter	DNBR minimal value	DNB minimal value time [s]	Fuel rod fraction entering in DNB [%]
20 cm²	1,32	0,80	0,00
158 cm²	1,22	593,50	0,00
253 cm²	1,15	375,00	0,00
700 cm²	0,97	31,00	1,16
1300 cm²	0,93	13,00	1,55
1400 cm²	**0,91**	**12,50**	**1,96**
1575 cm²	1,00	10,90	0,74
2A	1,32	0,60	0,00

SECTION 14.5.2 - TABLE 32

Break spectrum at full power, break size spectrum results

Break diameter	Reactor Trip signal	Reactor Trip actuating time [s]	MSIV closure time [s]	DNBR minimal value	DNB minimal value time [s]	Fuel rod fraction entering in DNB [%]	Peak of core nuclear power [%] (Time)	Peak of core thermal power [%] (Time)
20 cm²	SG low level	1450	-	1,32	0,8	0,00	102,7 (1450 s)	102,7 (1451 s)
158 cm²	SG low level	593,5	-	1,22	593,5	0,00	106,5 (593,5 s)	106,6 (594,5 s)
253 cm²	SG low level	443	-	1,15	375,0	0,00	108,7 (443 s)	108,9 (443,5 s)
700 cm²	High core thermal power	30,9	93,5	0,97	31,0	1,16	114,6 (31 s)	115,8 (33,5 s)
1300 cm²	SG pressure drop	10,5	10,3	0,93	13,0	1,55	113,7 (13,3 s)	118,1 (15,6 s)
1400 cm²	**SG Pressure drop / Low DNBR**	**9,3**	**9,5**	**0,91**	**12,5**	**1,96**	**117,7 (12 s)**	**118,9 (15,3 s)**
1575 cm²	High SG Pressure drop	7,9	7,7	1,00	10,9	0,74	110,9 (11 s)	115,7 (13,8 s)
2A	High SG pressure drop	1,2	0,9	1,32	0,6	0,00	102 (0 s)	102 (0 s)

SECTION 14.5.2 - TABLE 33

Break spectrum at full power, worst break (1400 cm²) – sequence of events

EVENT	TIME (seconds)
Steady state	0.
Main steam line break	0.1
Reactor trip on high SG pressure drop and low DNBR value signals	9.3
MSIV closure	9.5
Beginning of rod drop	9.7
Turbine trip	11.5
Minimum DNBR value (0.91)	12.5
Peak of core nuclear power (117.7% NP)	12.6
Peak of core thermal power (118.9% NP)	15.2

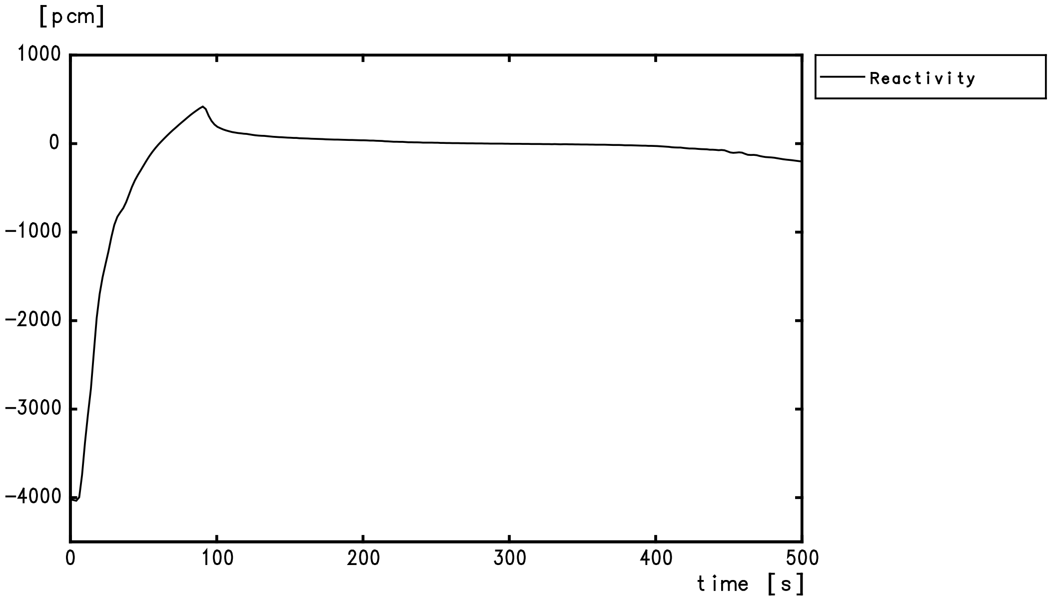
SECTION 14.5.2 - TABLE 34

Break Spectrum at Full Power, worst Break (1400 cm²) Conditions at the Time of Minimum DNBR

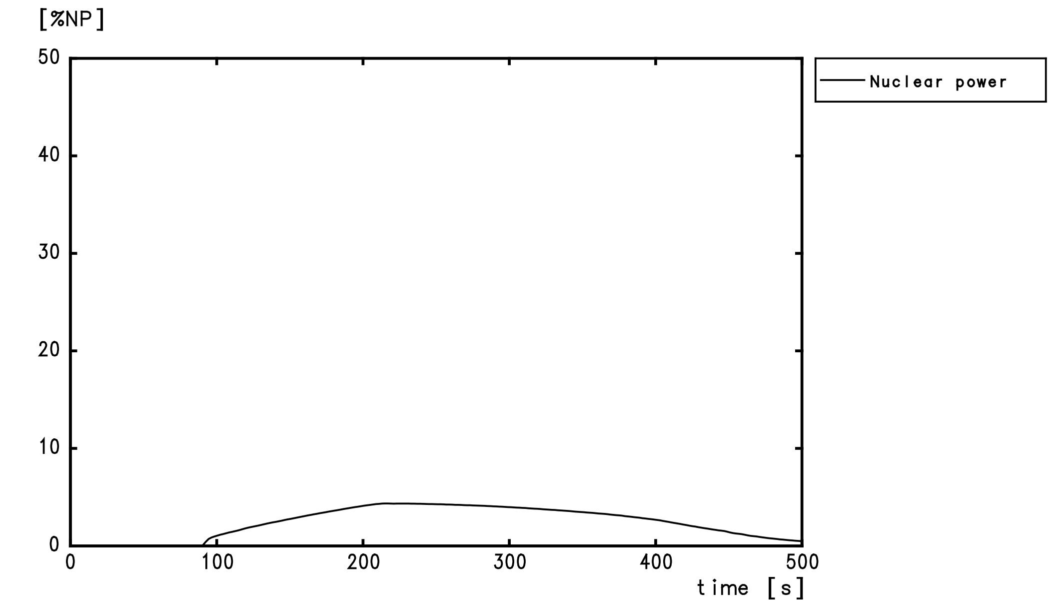
Time	12.5 s
Thermal power	112.0% NP
Primary pressure	148.9 bar
Cold leg temperature in affected loop	290.6°C
Cold leg temperature in unaffected loops (loop 2)	332°C
Minimum DNBR	**0.91**

SECTION 14.5.2 - FIGURE 1

Reference Case – Reactivity and Core Power



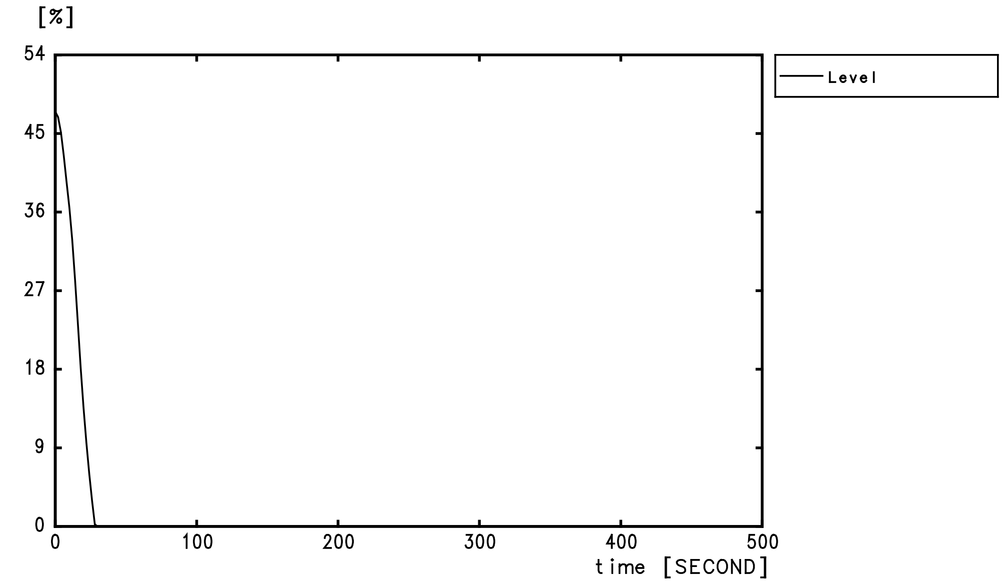
REACTIVITY



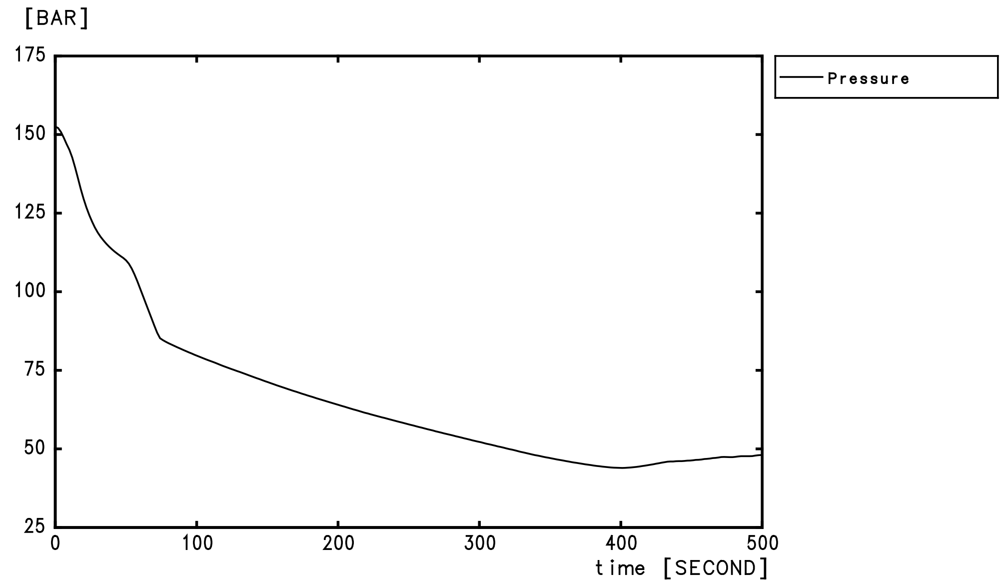
CORE POWER

SECTION 14.5.2 - FIGURE 2

Reference Case – Pressuriser Level and Pressuriser Pressure



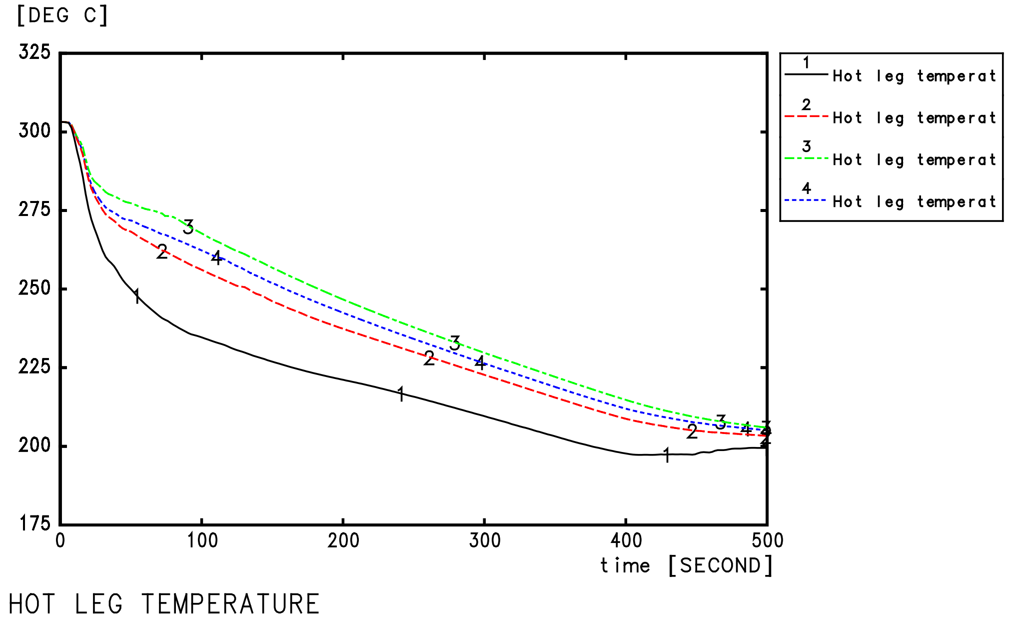
PRESSURIZER LEVEL

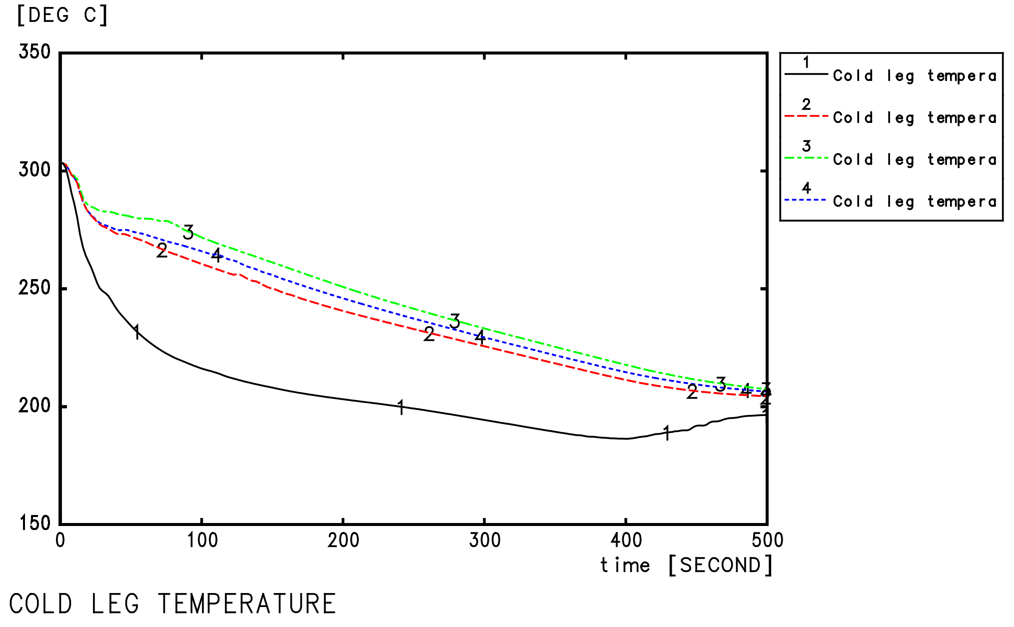


PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 3

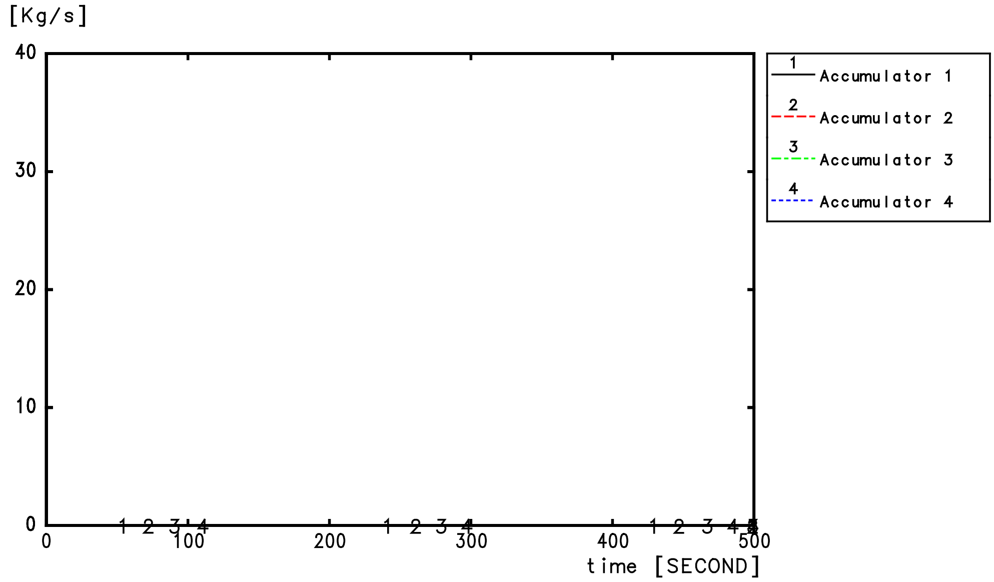
Reference Case – Hot Leg Temperature and Cold Leg Temperature



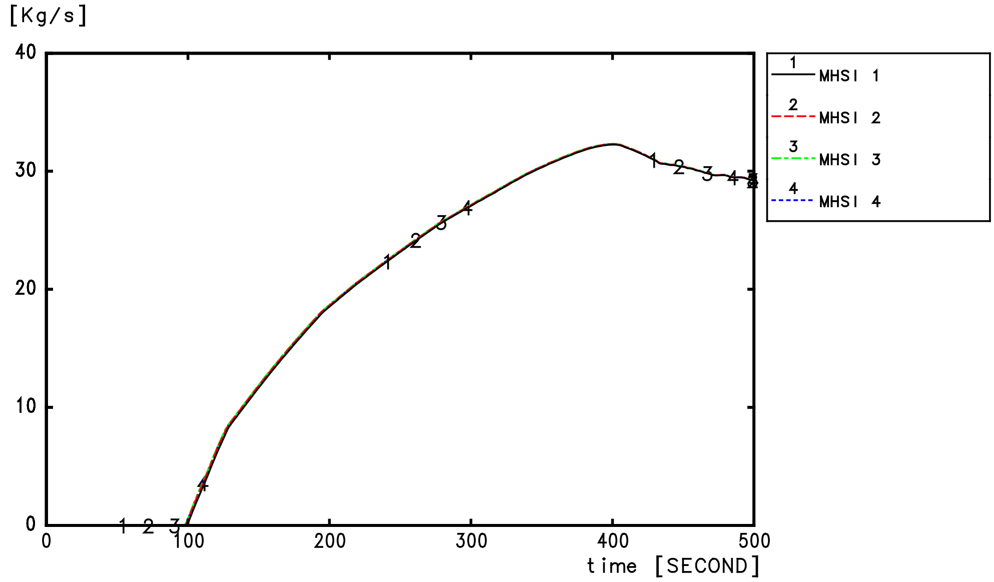


SECTION 14.5.2 - FIGURE 4

Reference Case – Total Accumulators Flow Rate and Total MHSI Flow Rate



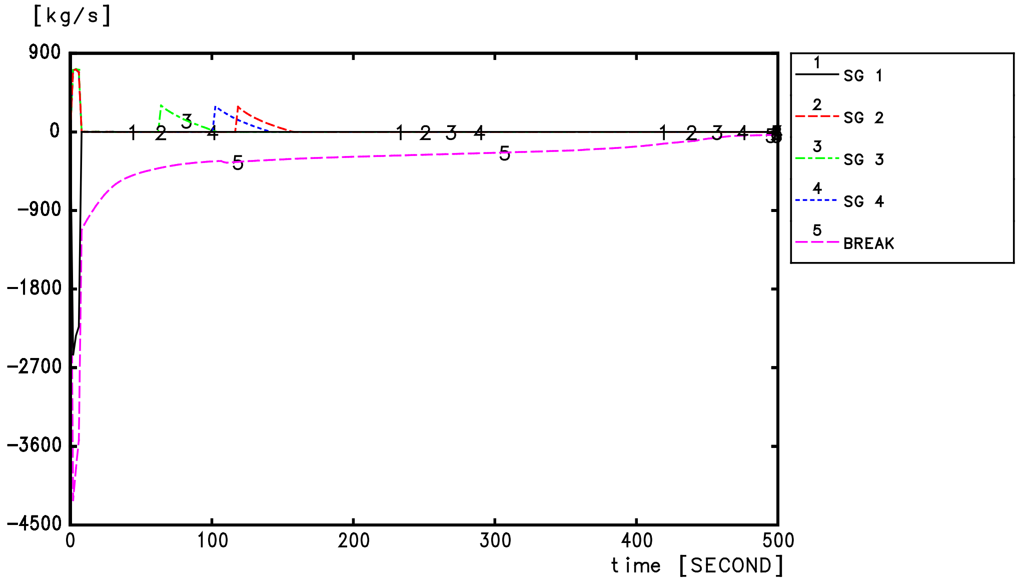
TOTAL ACCUMULATORS FLOW RATE



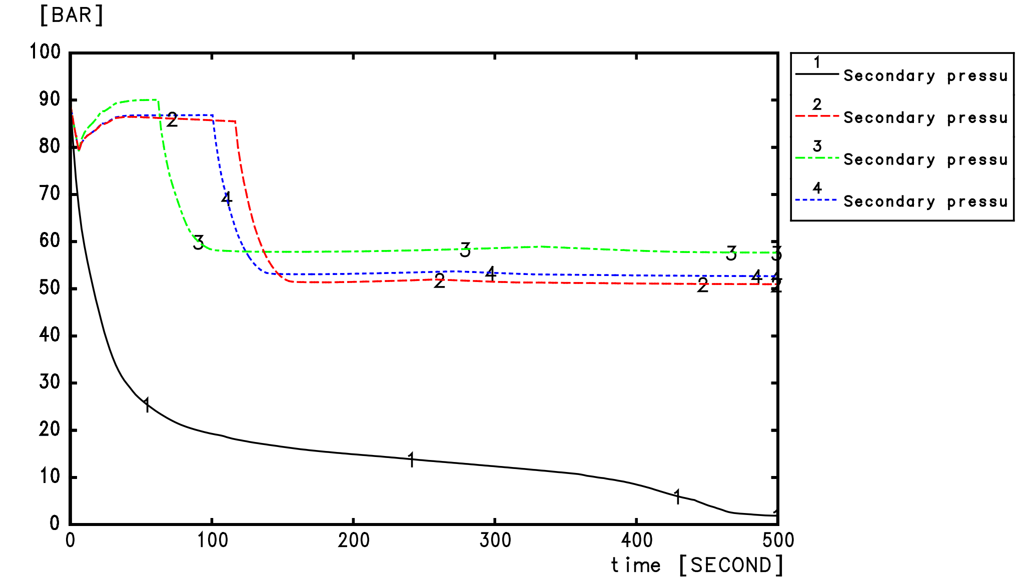
TOTAL MHSI FLOW RATE

SECTION 14.5.2 - FIGURE 5

Reference Case – Vapour Mass Flow Rate and SG Pressure



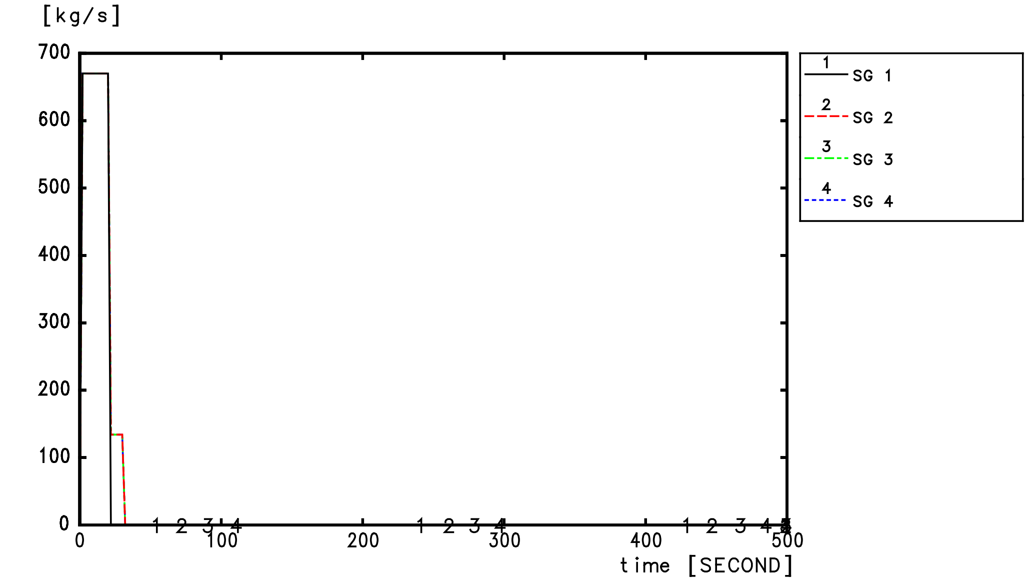
VAPOR MASS FLOWRATE



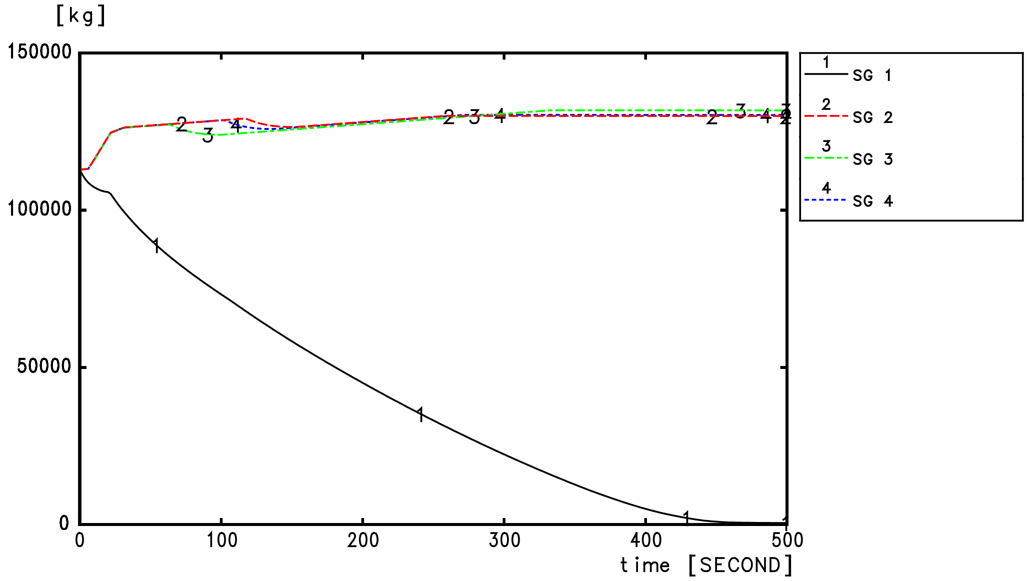
STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 6

Reference Case – Main Feedwater Flow Rate and SG Liquid Mass



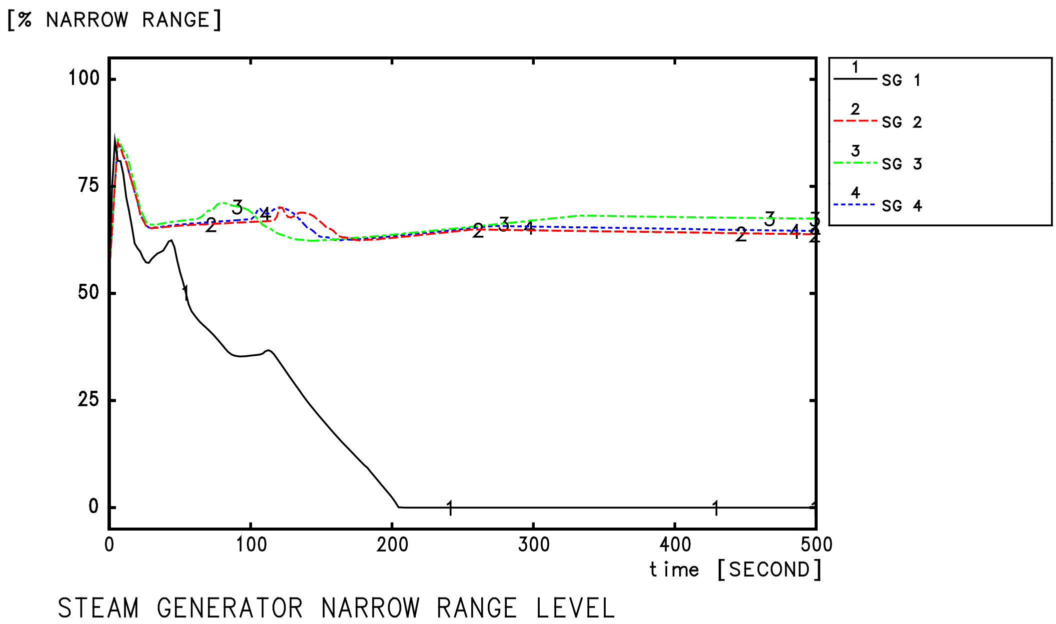
MFW FLOWRATE

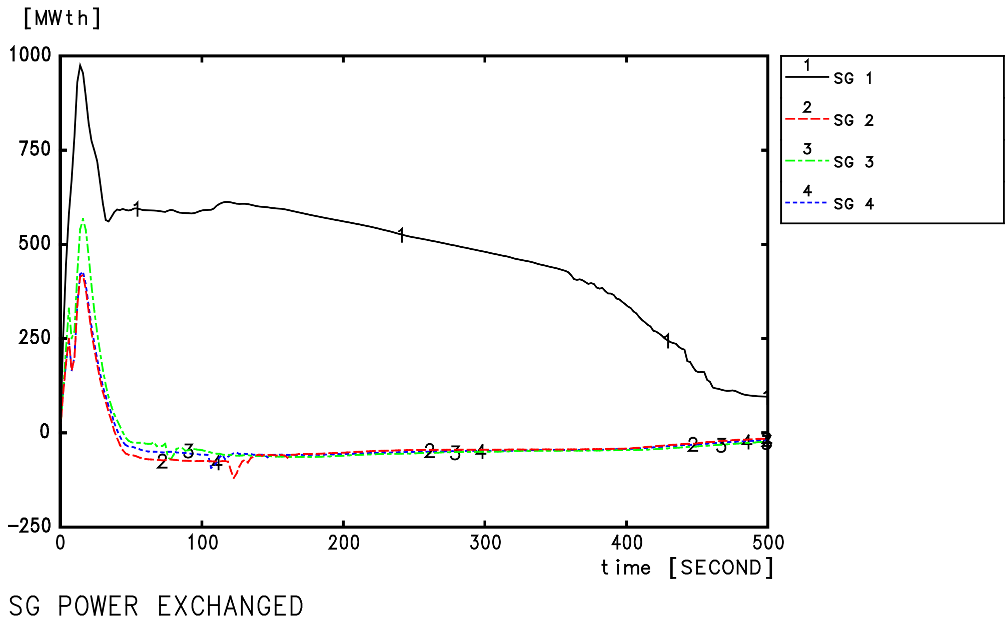


STEAM GENERATOR LIQUID MASS

SECTION 14.5.2 - FIGURE 7

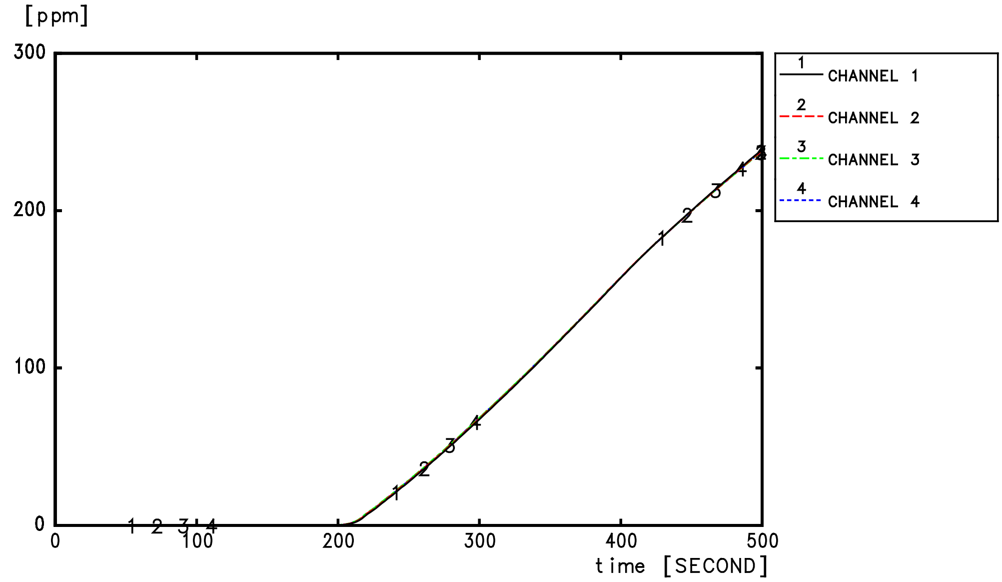
Reference Case – SG NR Level and SG Power Exchanged



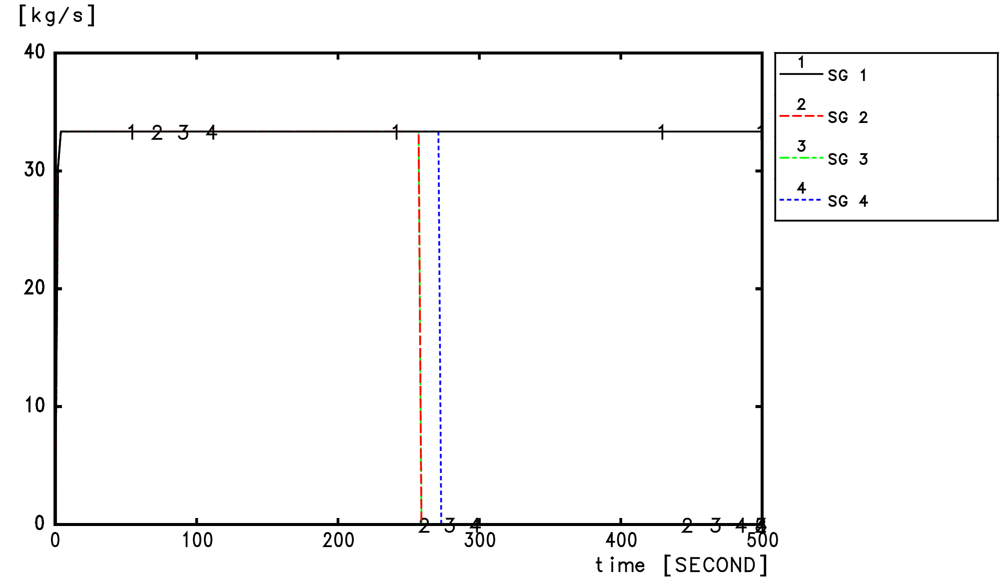


SECTION 14.5.2 - FIGURE 8

Reference Case – Boron Concentration and ASG [EFWS] Flow Rate



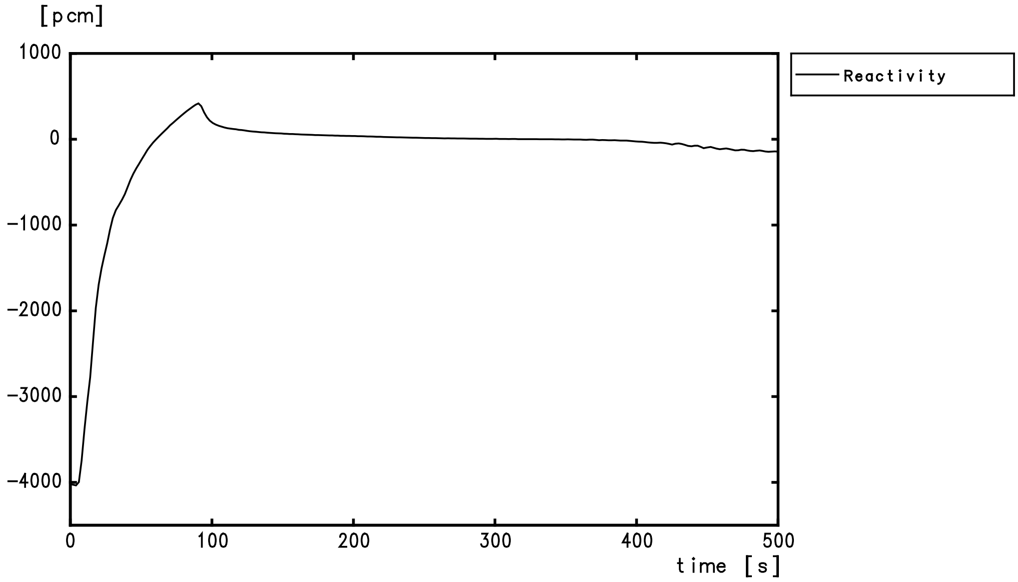
BORON CONCENTRATION AT CORE ACTIVE PART INLET



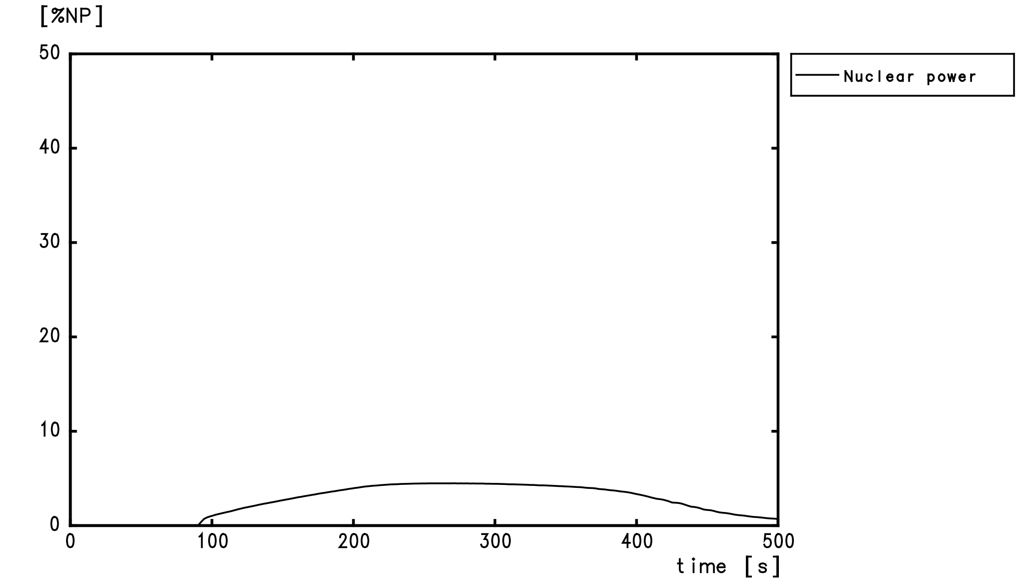
ASG FLOWRATE

SECTION 14.5.2 - FIGURE 9

SLB at Hot Shutdown with Failure of One MHSI – Reactivity and Core Power



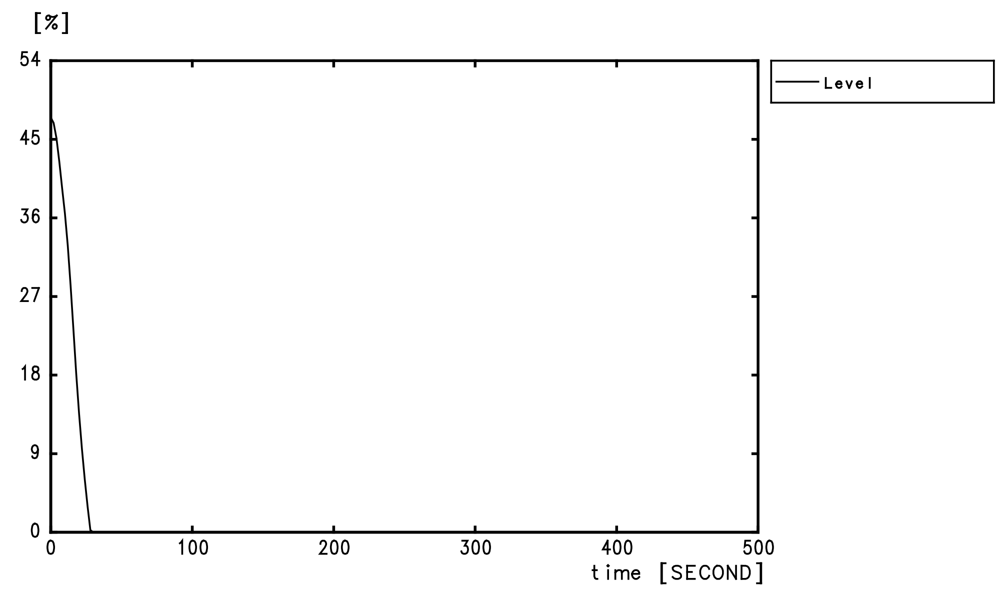
REACTIVITY



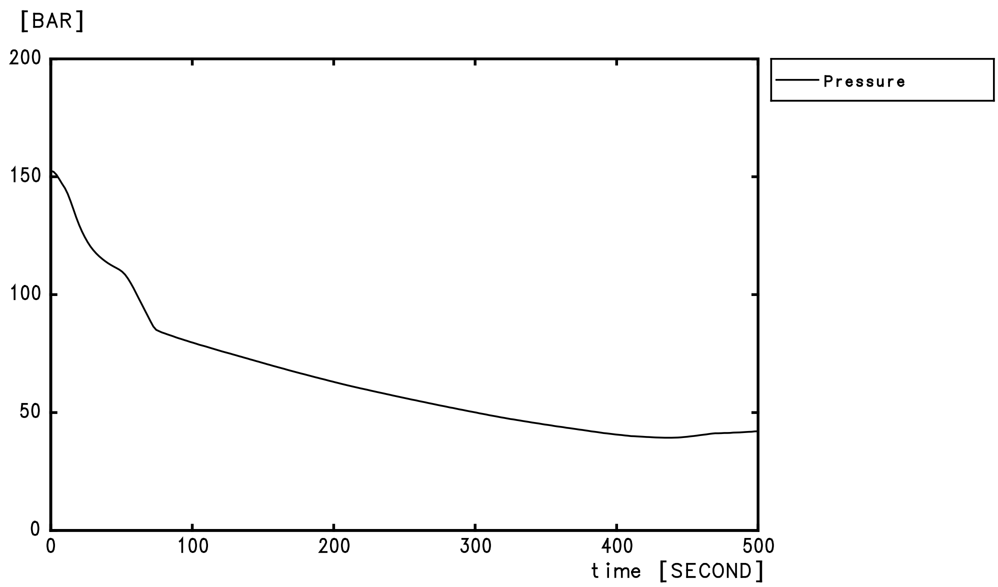
CORE POWER

SECTION 14.5.2 - FIGURE 10

SLB at Hot Shutdown with Failure of One MHSI – Pressuriser Level and Pressuriser Pressure



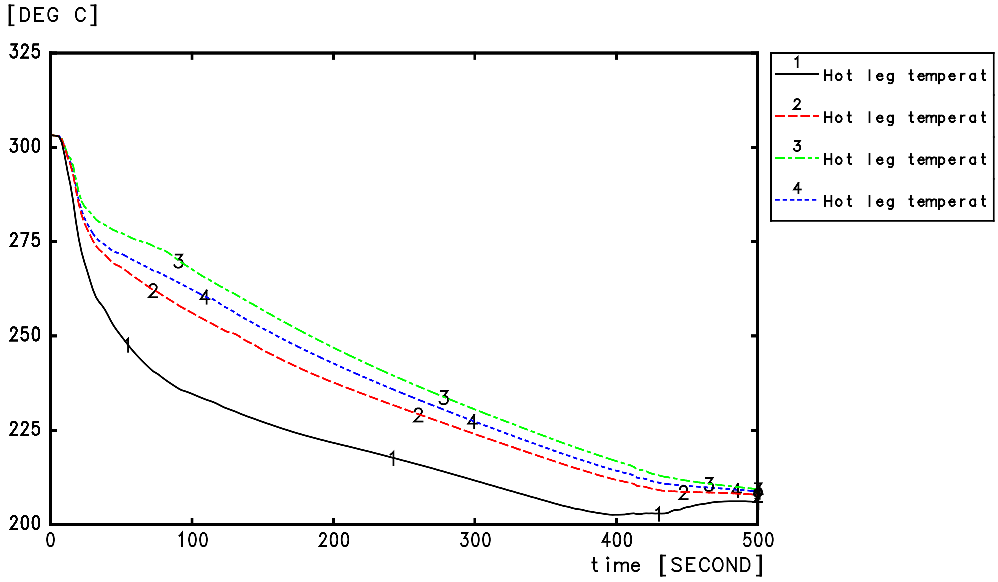
PRESSURIZER LEVEL



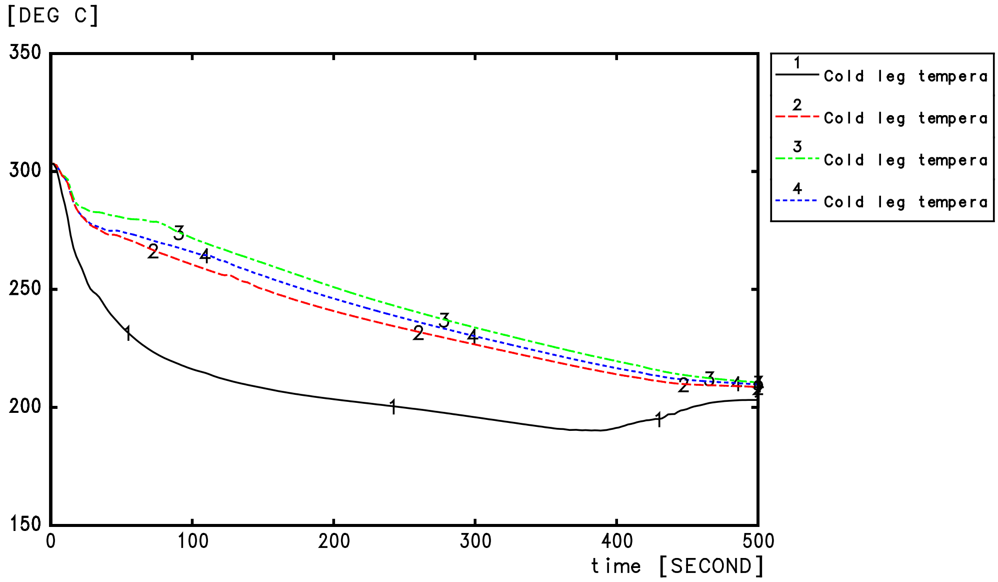
PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 11

SLB at Hot Shutdown with Failure of One MHSI – Hot Leg Temperature and Cold Leg Temperature



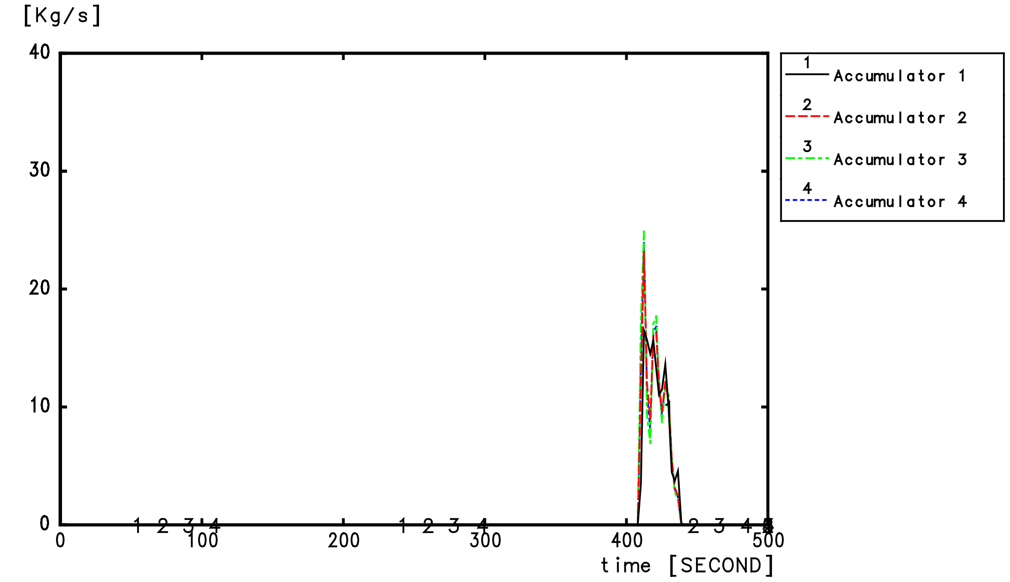
HOT LEG TEMPERATURE



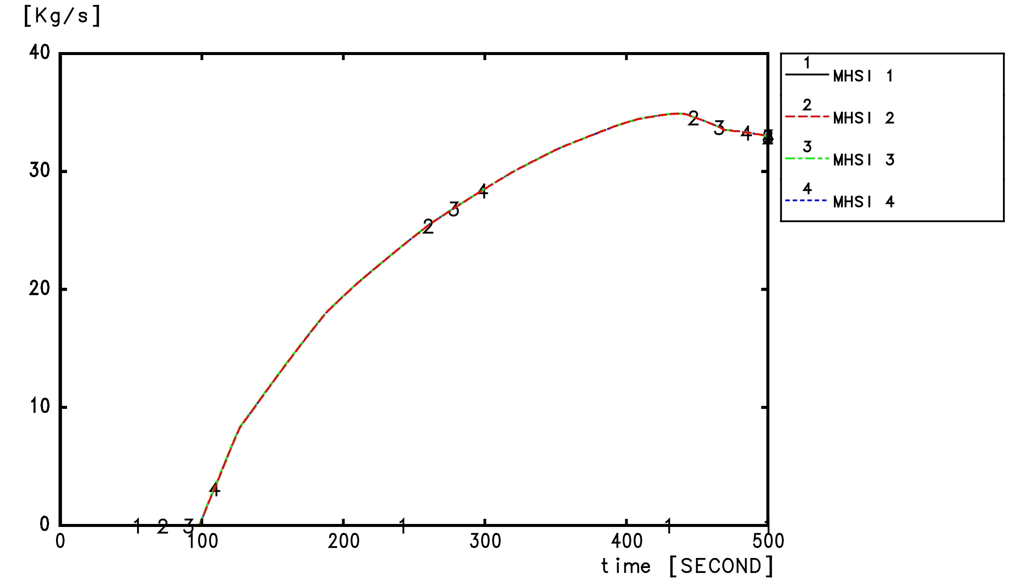
COLD LEG TEMPERATURE

SECTION 14.5.2 - FIGURE 12

SLB at Hot Shutdown with Failure of One MHSI – Total Accumulators Flow Rate and Total MHSI Flow Rate



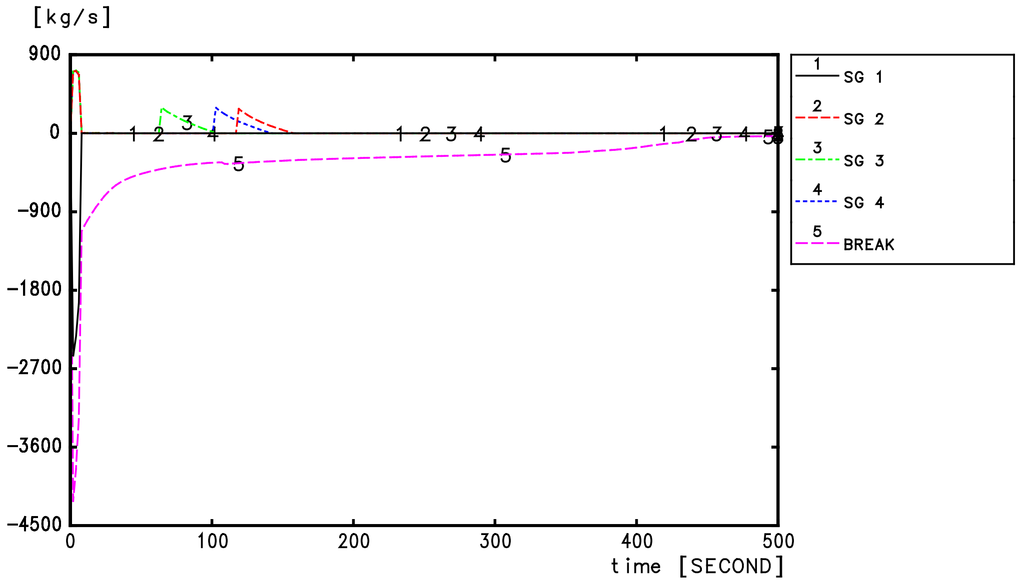
TOTAL ACCUMULATORS FLOW RATE



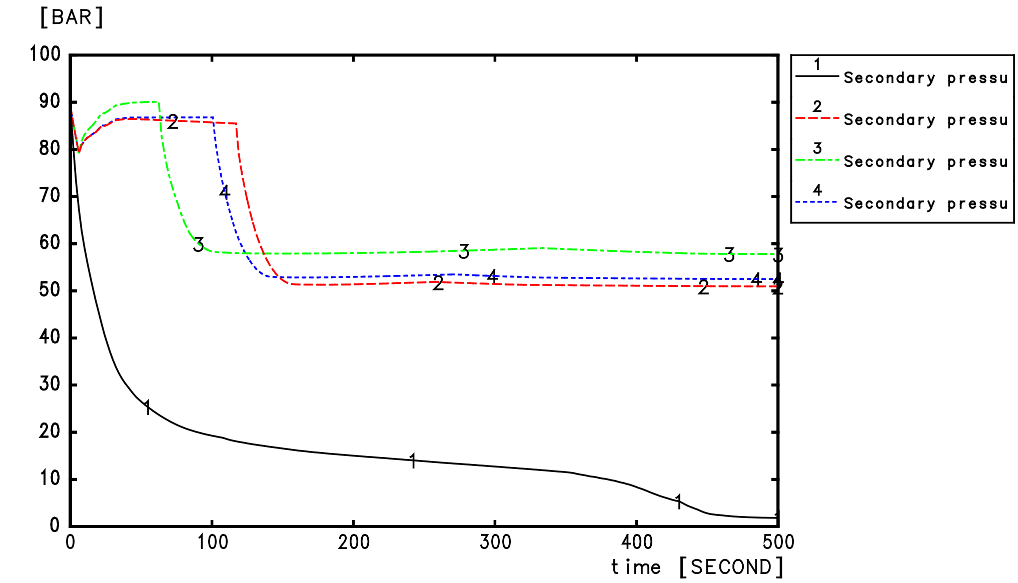
TOTAL MHSI FLOW RATE

SECTION 14.5.2 - FIGURE 13

SLB at Hot Shutdown with Failure of One MHSI – Vapour Mass Flow Rate and SG Pressure



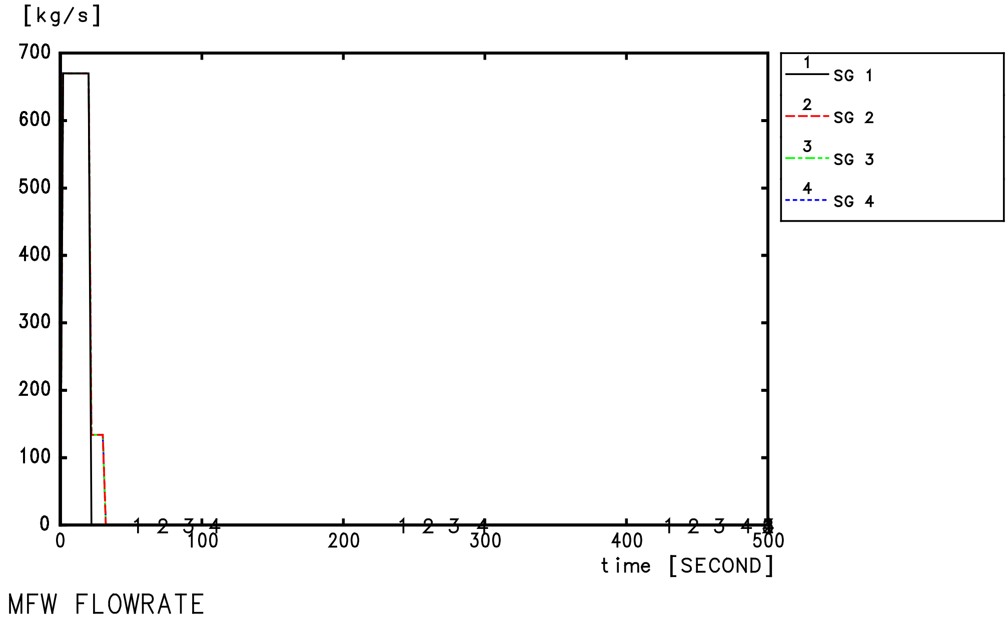
VAPOR MASS FLOWRATE

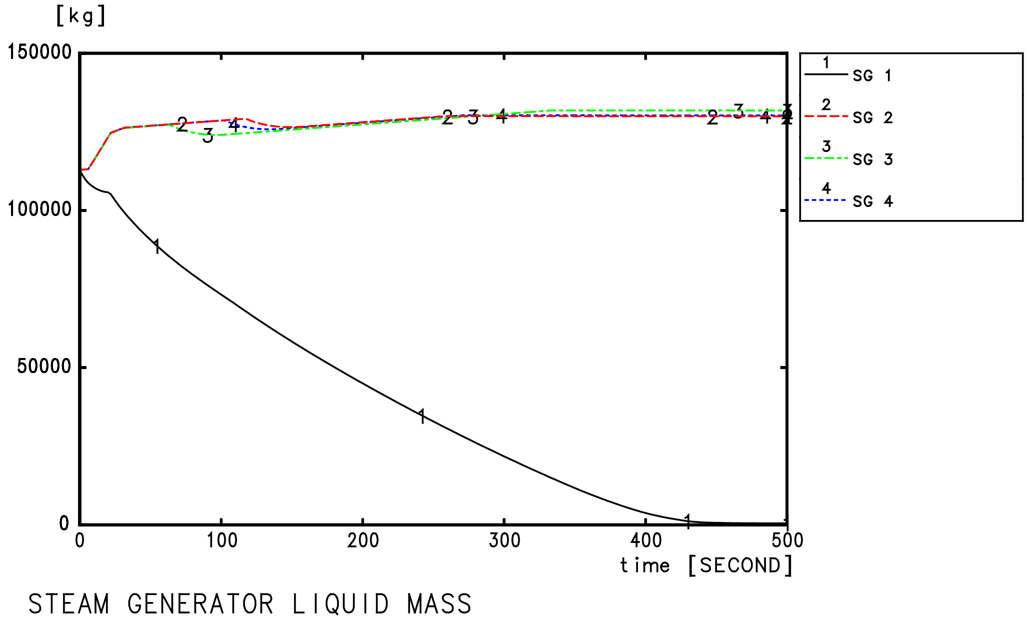


STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 14

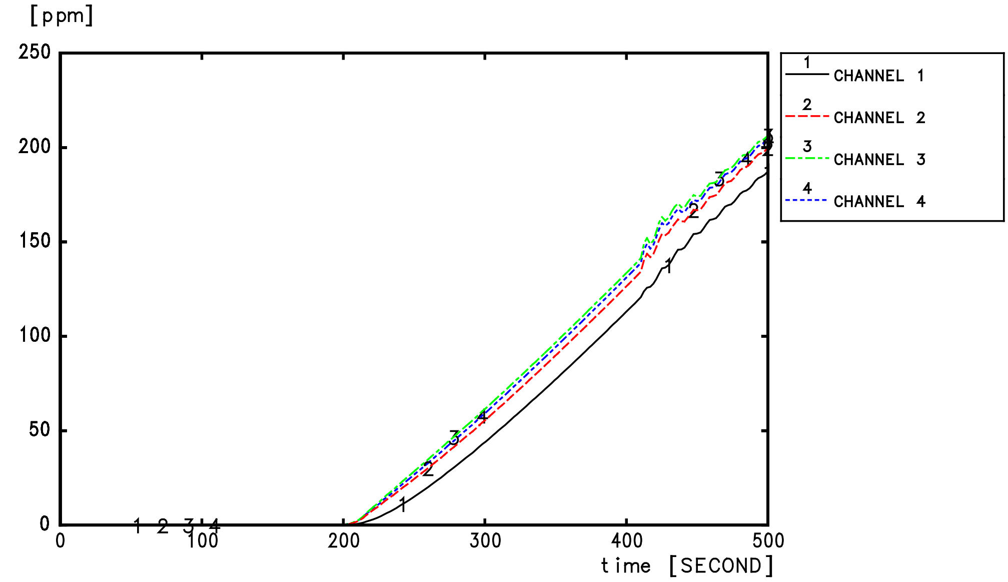
SLB at Hot Shutdown with Failure of One MHSI – Main Feedwater Flow Rate and SG Liquid Mass



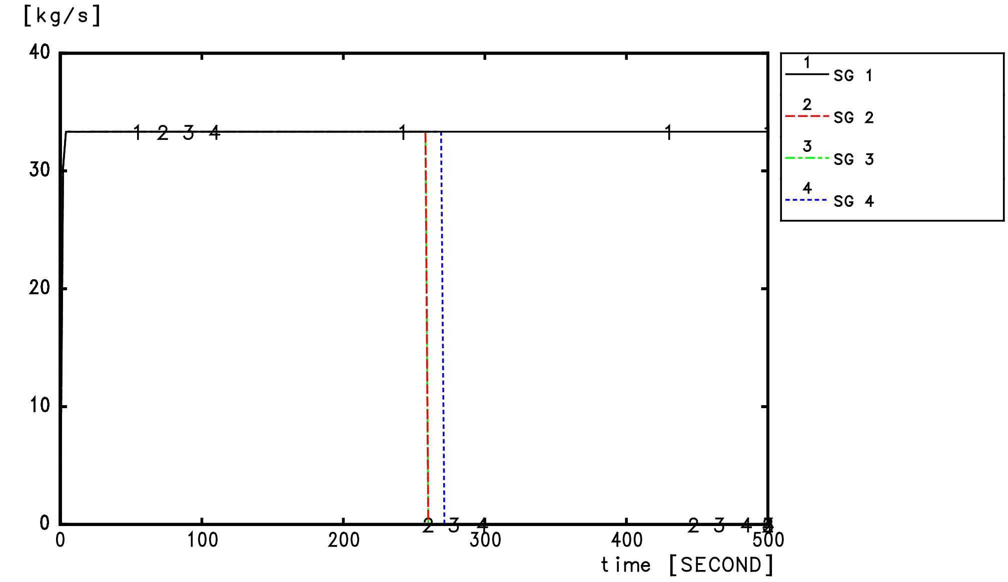


SECTION 14.5.2 - FIGURE 15

SLB at Hot Shutdown with Failure of One MHSI – Boron Concentration and ASG [EFWS] Flow Rate



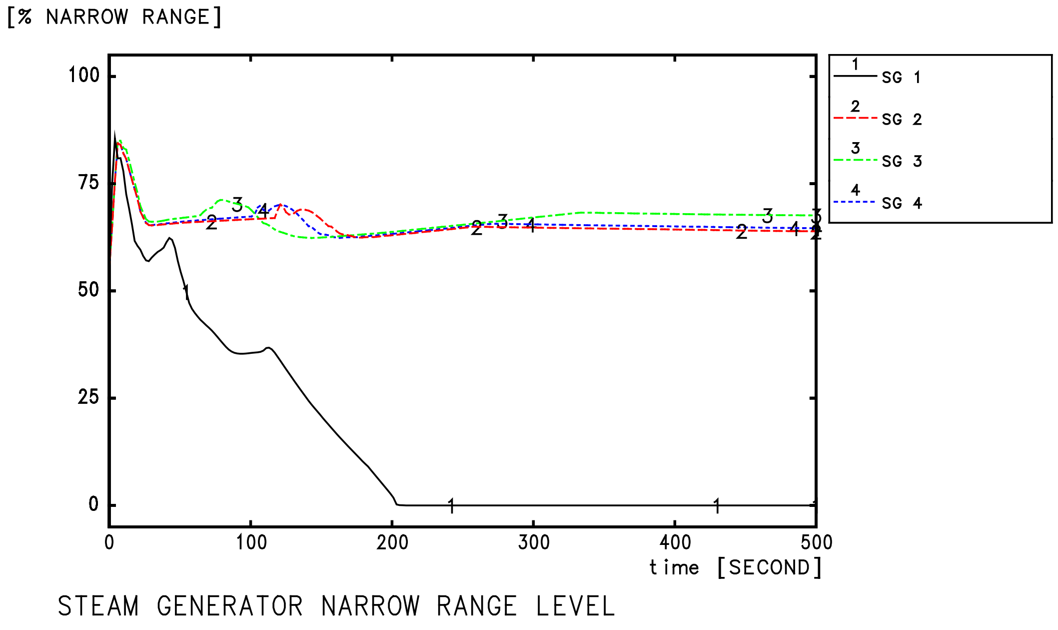
BORON CONCENTRATION AT CORE ACTIVE PART INLET

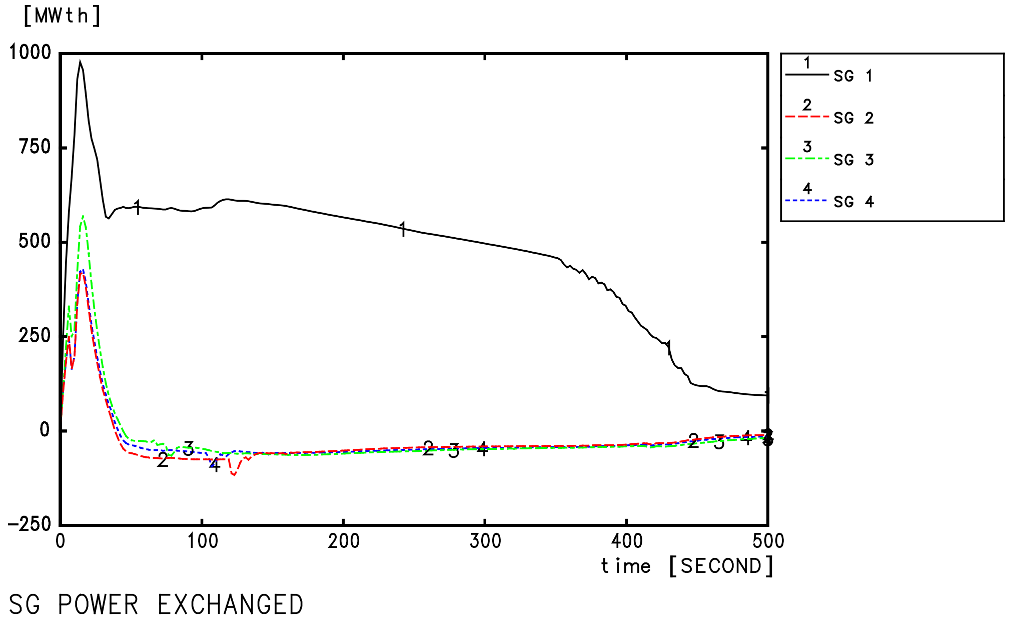


ASG FLOWRATE

SECTION 14.5.2 - FIGURE 16

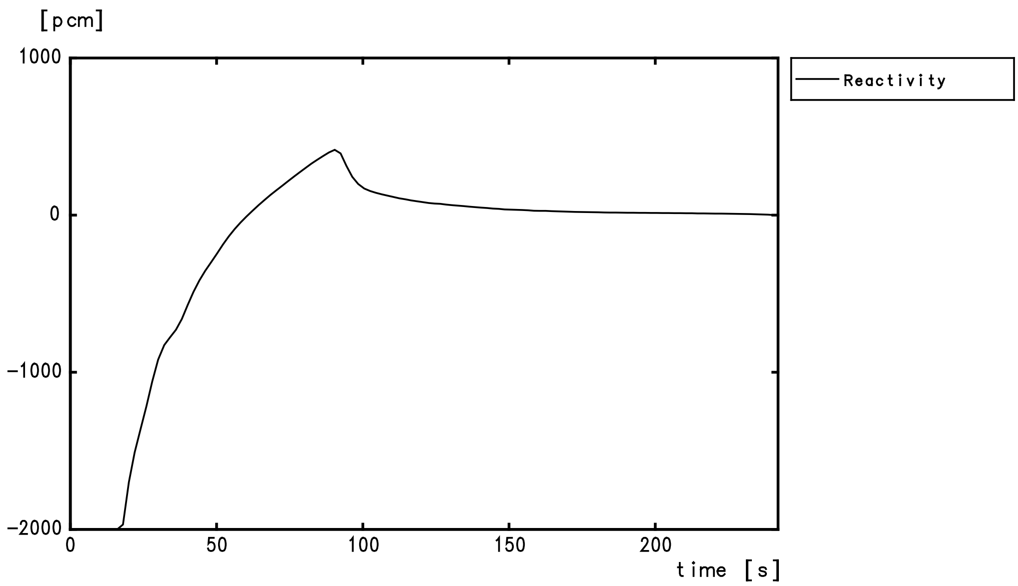
SLB at Hot Shutdown with Failure of One MHSI – SG NR Level and SG Power Exchanged



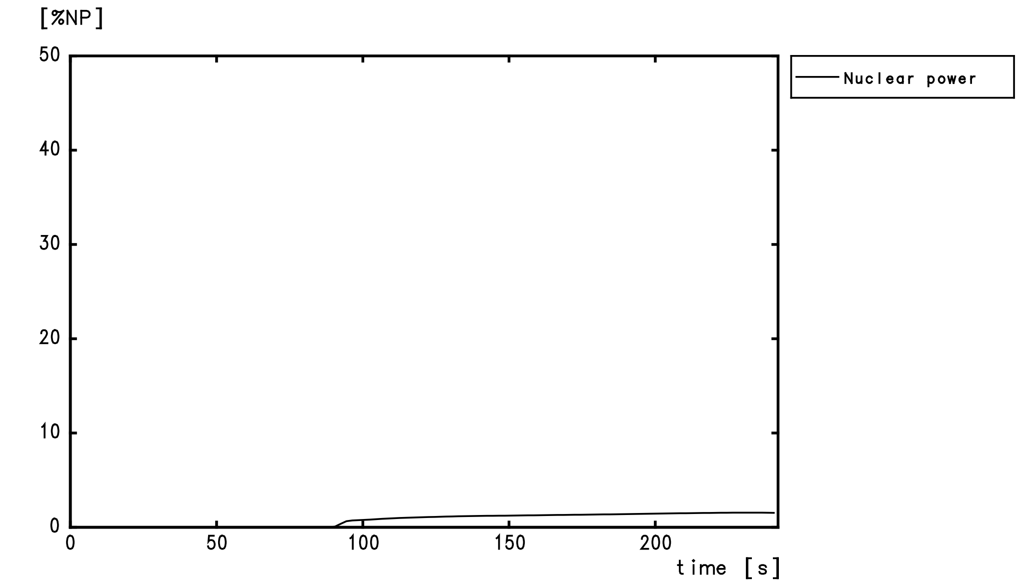


SECTION 14.5.2 - FIGURE 17

SLB at hot shutdown with Reactor Coolant Pump trip, Case 1 – Reactivity and Core Power



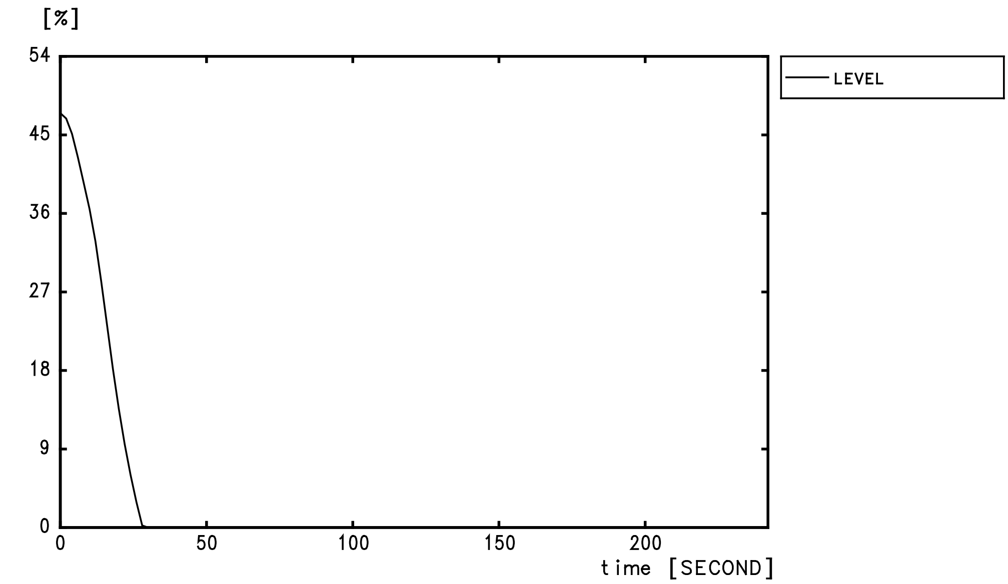
REACTIVITY



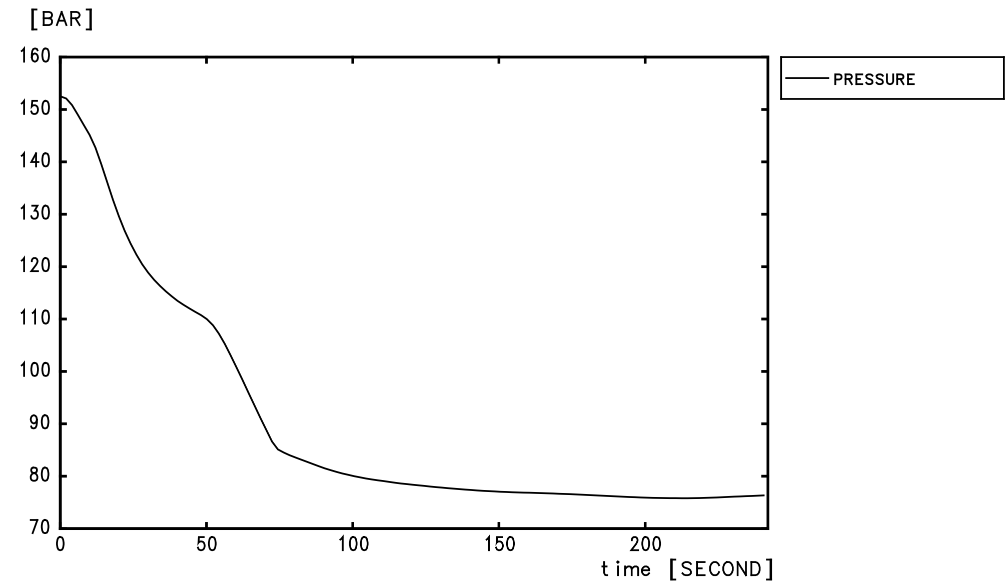
CORE POWER

SECTION 14.5.2 - FIGURE 18

SLB at hot shutdown with Reactor Coolant Pump trip, Case 1 – Pressuriser Level and Pressuriser Pressure



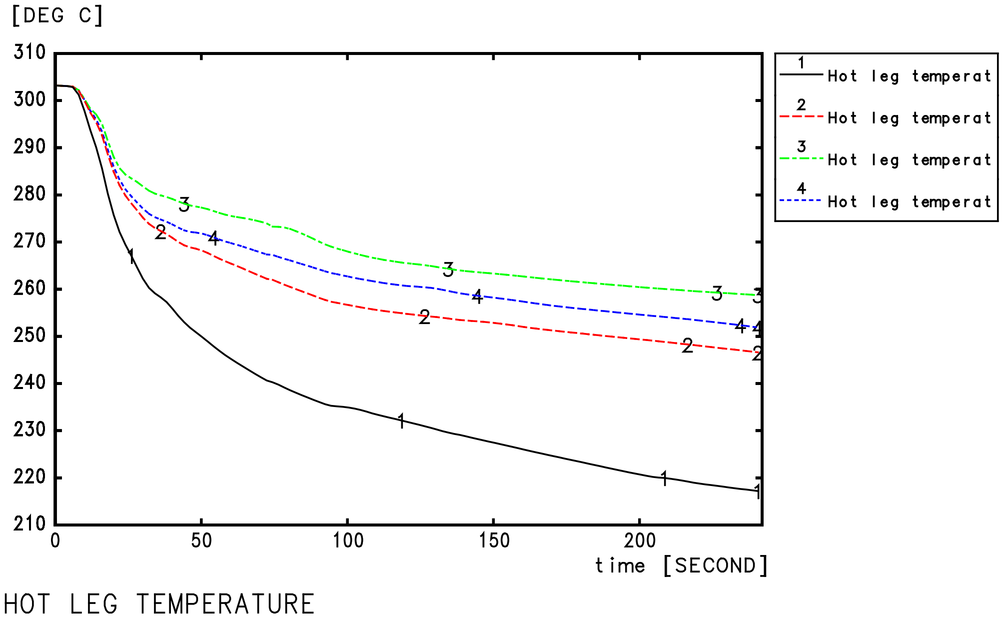
PRESSURIZER LEVEL

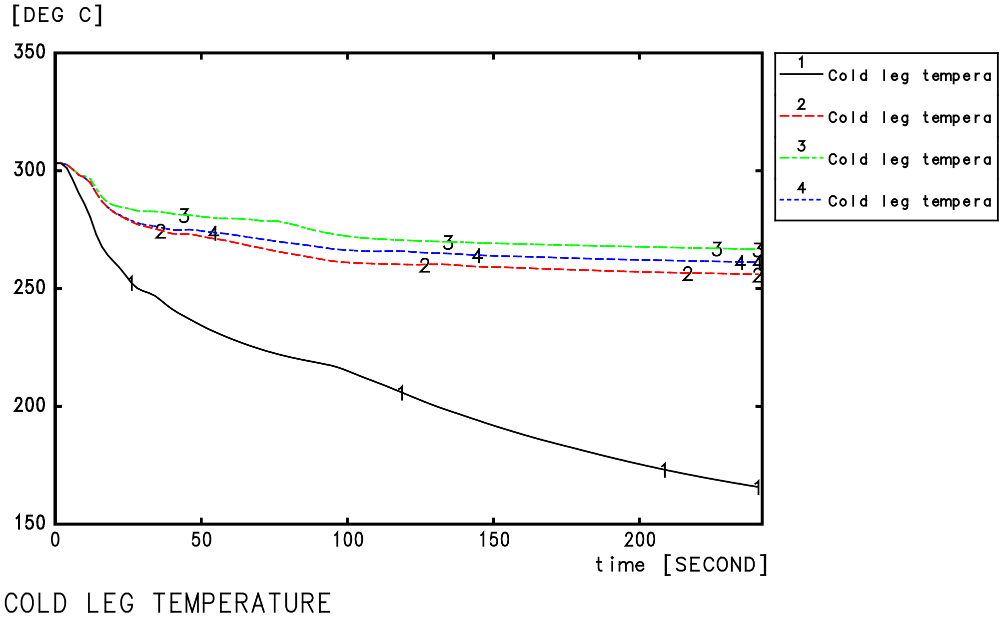


PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 19

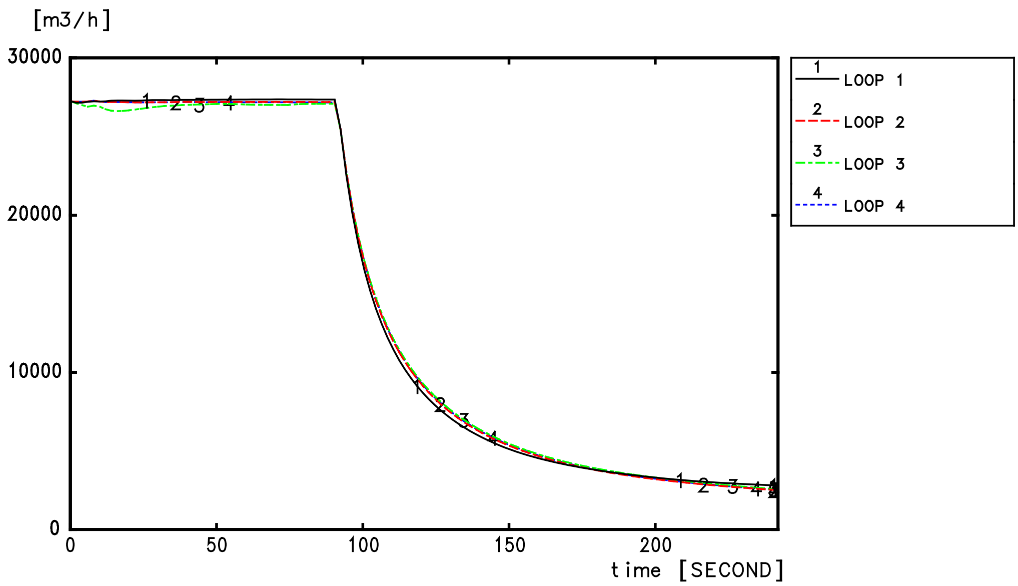
SLB at hot shutdown with Reactor Coolant Pump trip, Case 1 – Hot Leg Temperature and Cold Leg Temperature



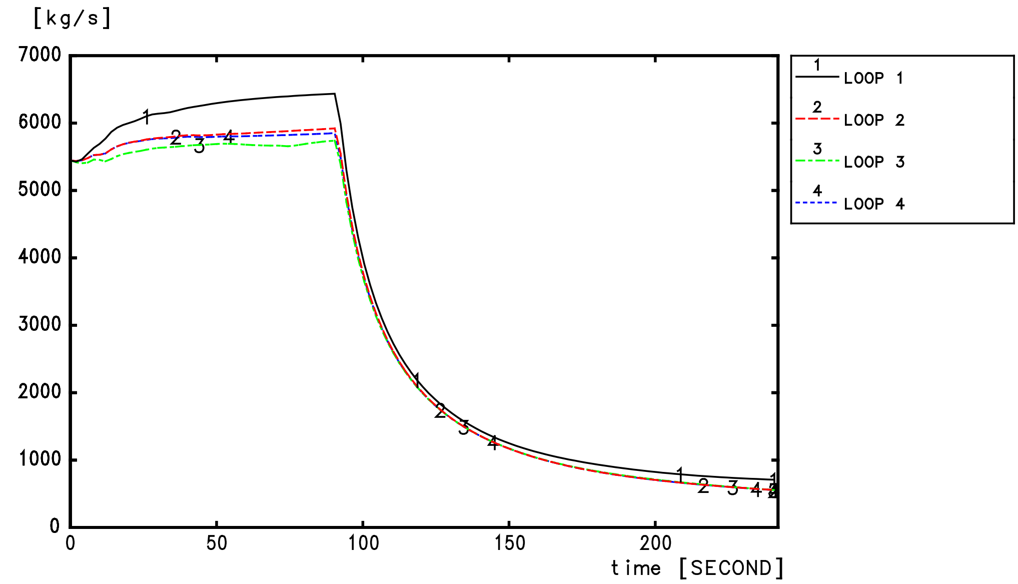


SECTION 14.5.2 - FIGURE 20

SLB at hot shutdown with Reactor Coolant Pump trip, Case 1 – Cold Leg Volumetric Flow Rate and Mass Flow Rate



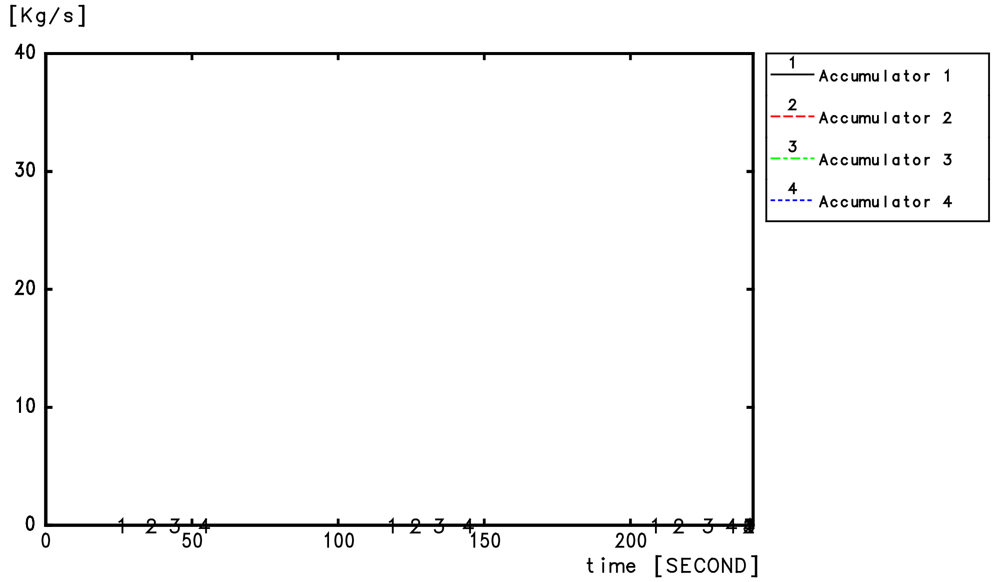
COLD LEG VOLUMETRIC FLOW RATE



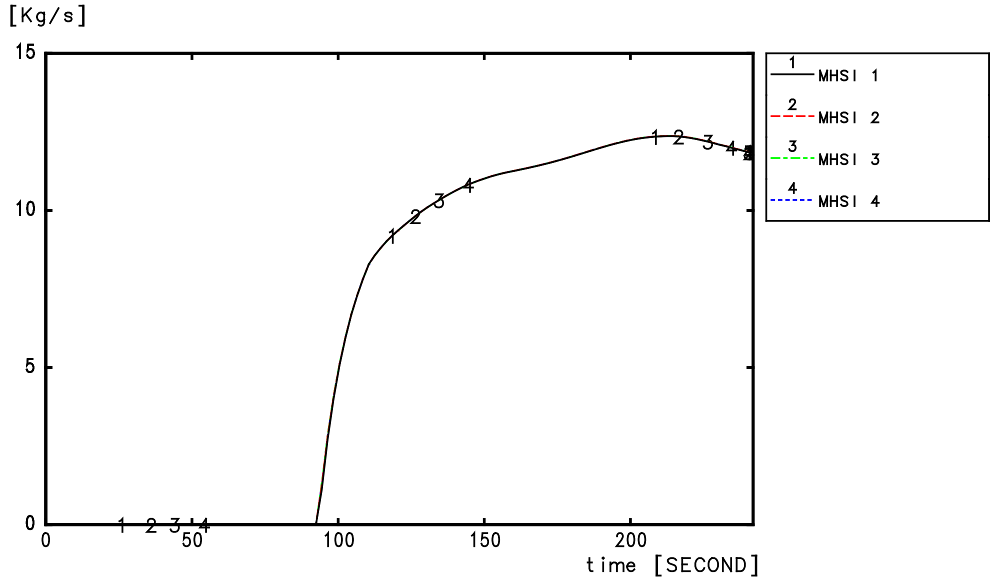
COLD LEG MASS FLOW RATE

SECTION 14.5.2 - FIGURE 21

SLB at hot shutdown with Reactor Coolant Pump trip, Case 1 – Total Accumulators Flow Rate and Total MHSI Flow Rate



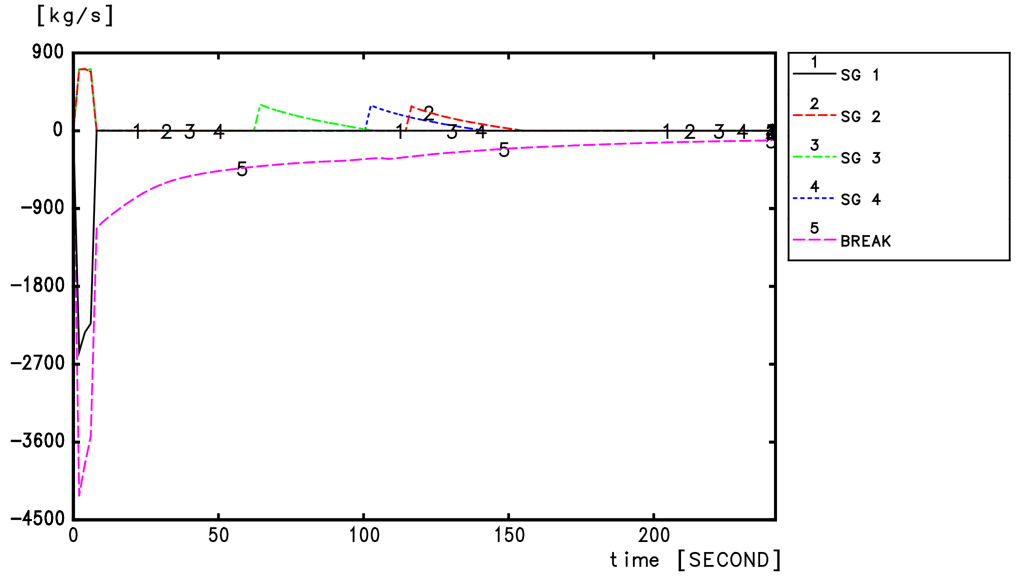
TOTAL ACCUMULATORS FLOW RATE



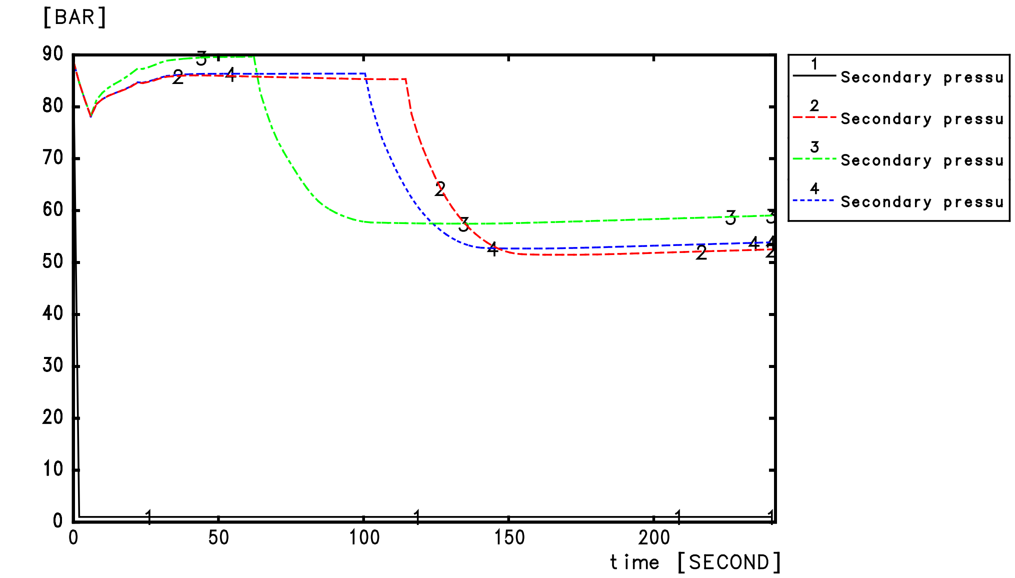
TOTAL MHSI FLOW RATE

SECTION 14.5.2 - FIGURE 22

SLB at hot shutdown with Reactor Coolant Pump trip, Case 1 – Vapour Mass Flow Rate and SG Pressure



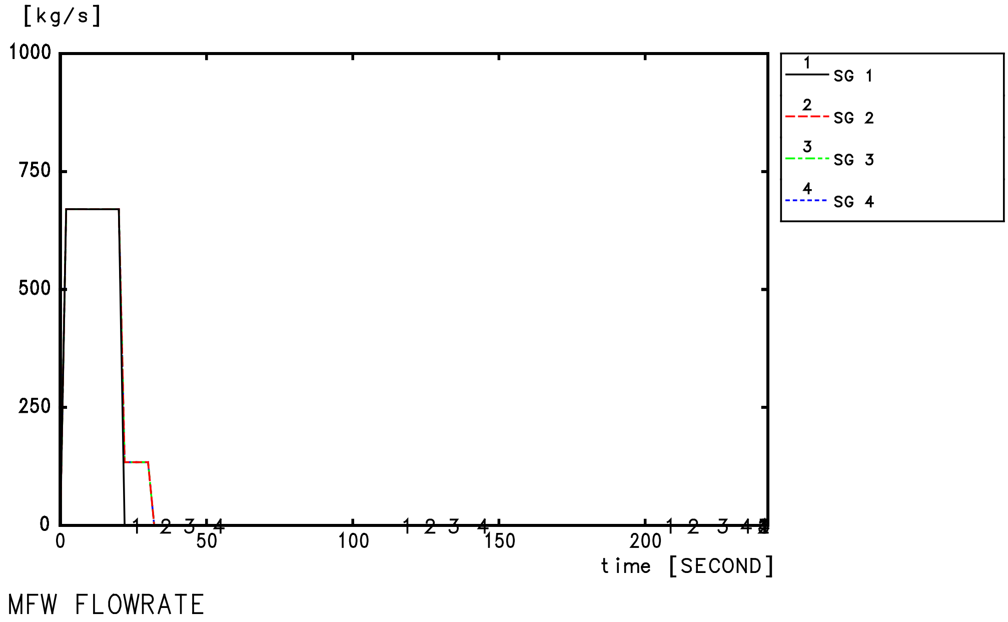
VAPOR MASS FLOW RATE

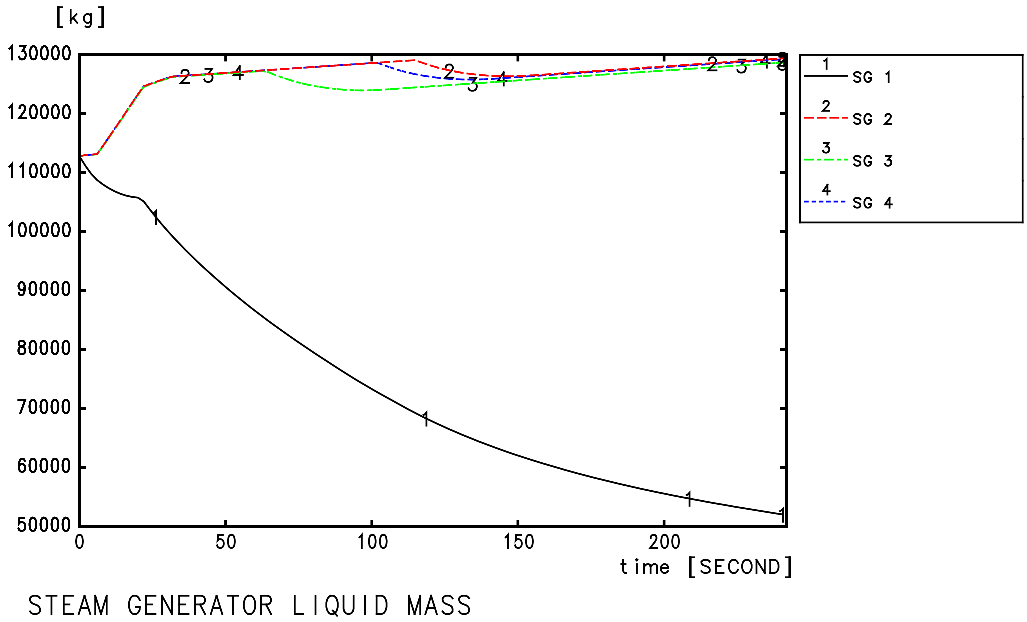


STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 23

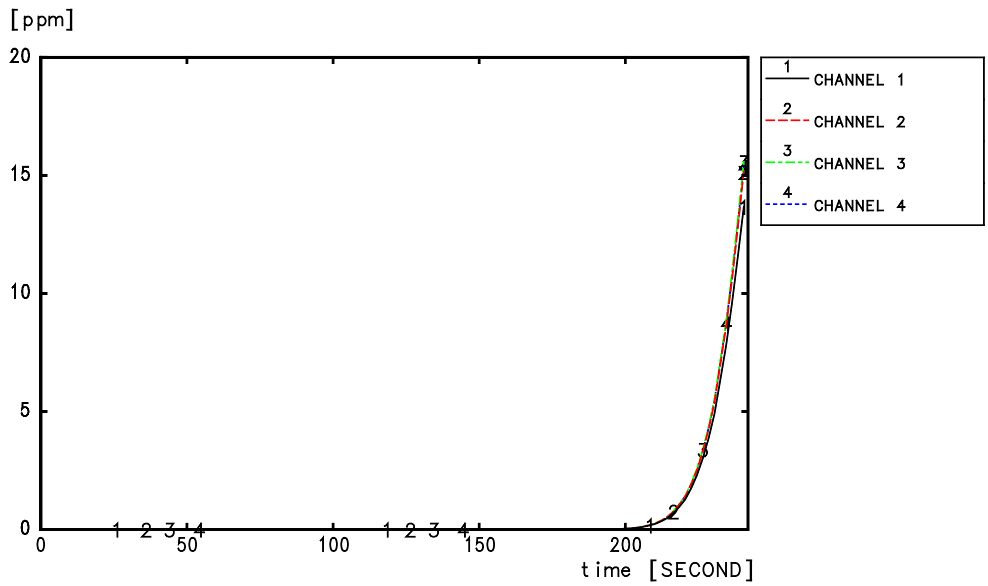
SLB at hot shutdown with Reactor Coolant Pump trip, Case 1 – Main Feedwater Flow Rate and SG Liquid Mass



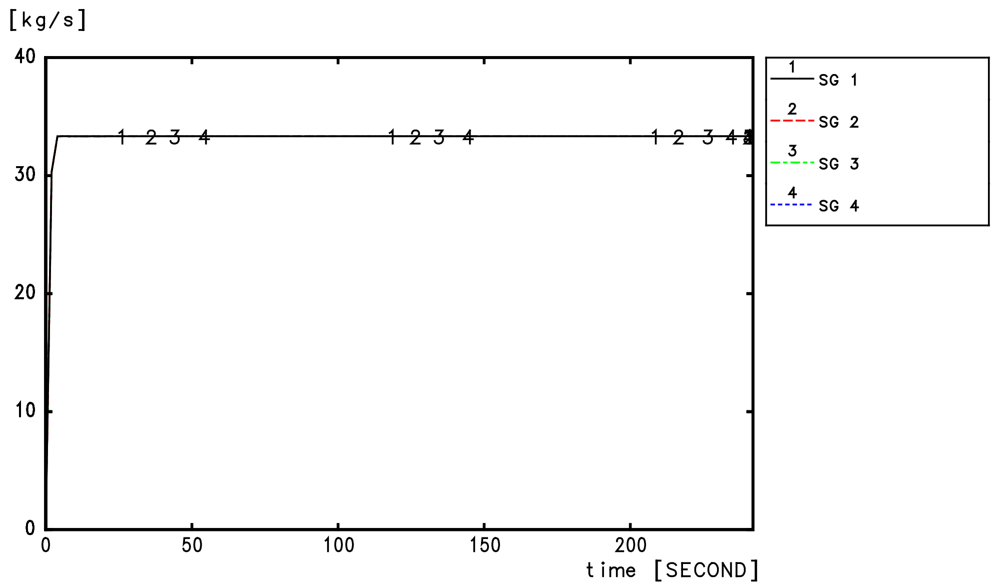


SECTION 14.5.2 - FIGURE 24

SLB at hot shutdown with Reactor Coolant Pump trip, Case 1 – Boron Concentration and ASG [EFWS] Flow Rate



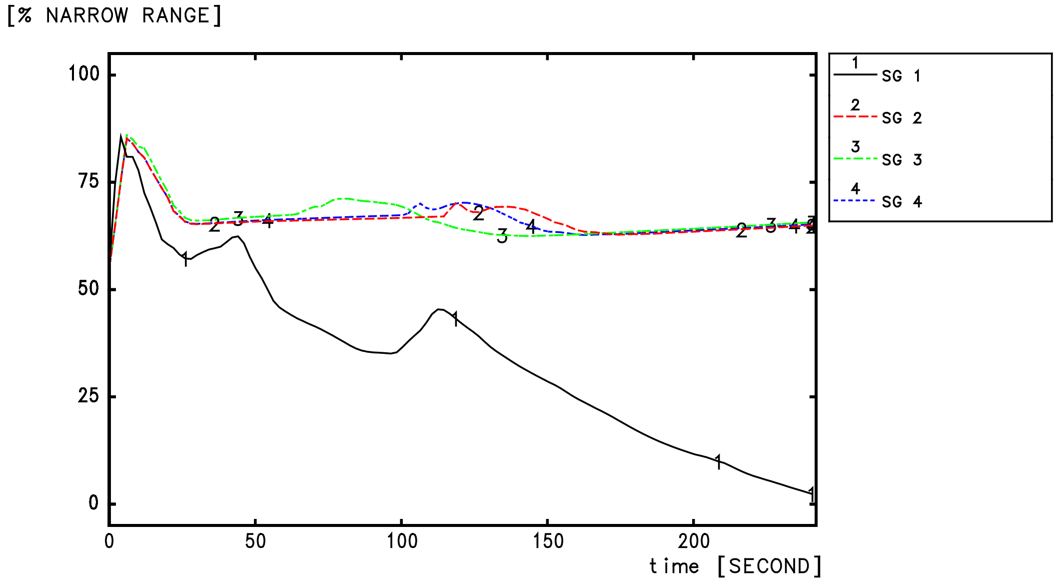
BORON CONCENTRATION AT CORE ACTIVE PART INLET



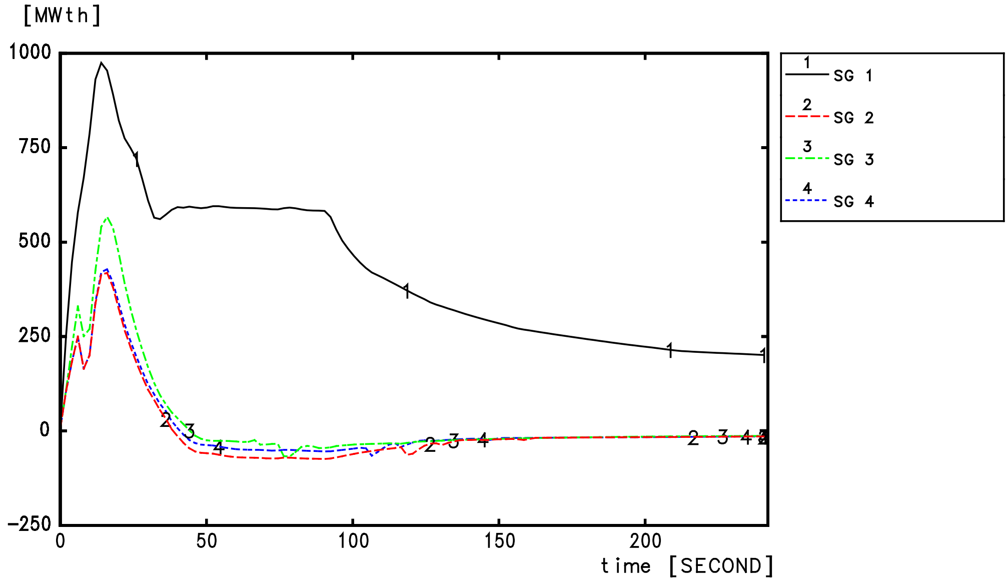
ASG FLOWRATE

SECTION 14.5.2 - FIGURE 25

SLB at hot shutdown with Reactor Coolant Pump trip, Case 1 – SG Narrow Range Level and SG Power Exchanged



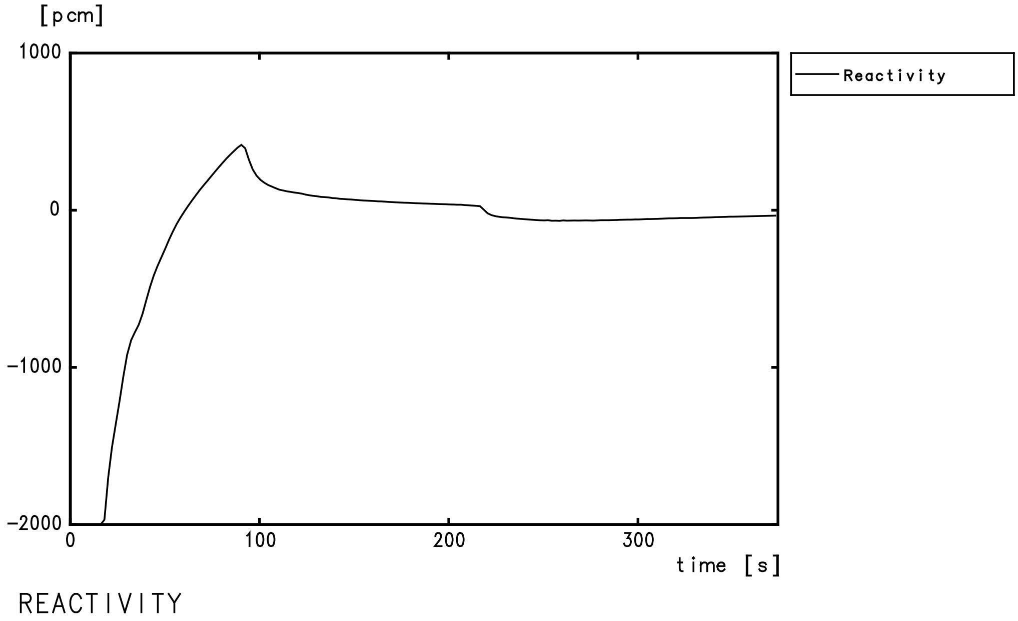
STEAM GENERATOR NARROW RANGE LEVEL

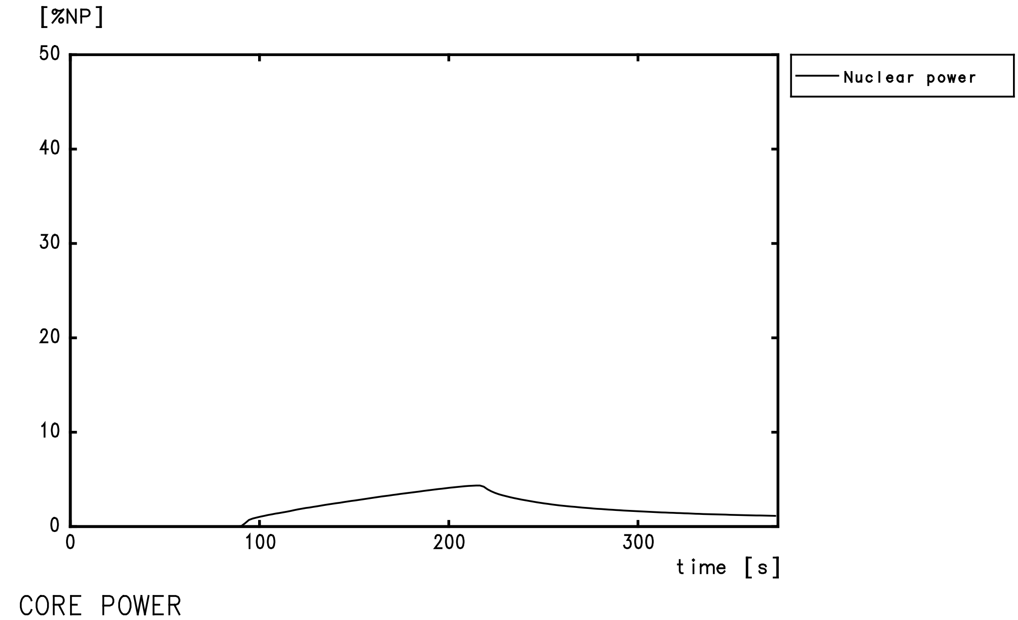


SG POWER EXCHANGED

SECTION 14.5.2 - FIGURE 26

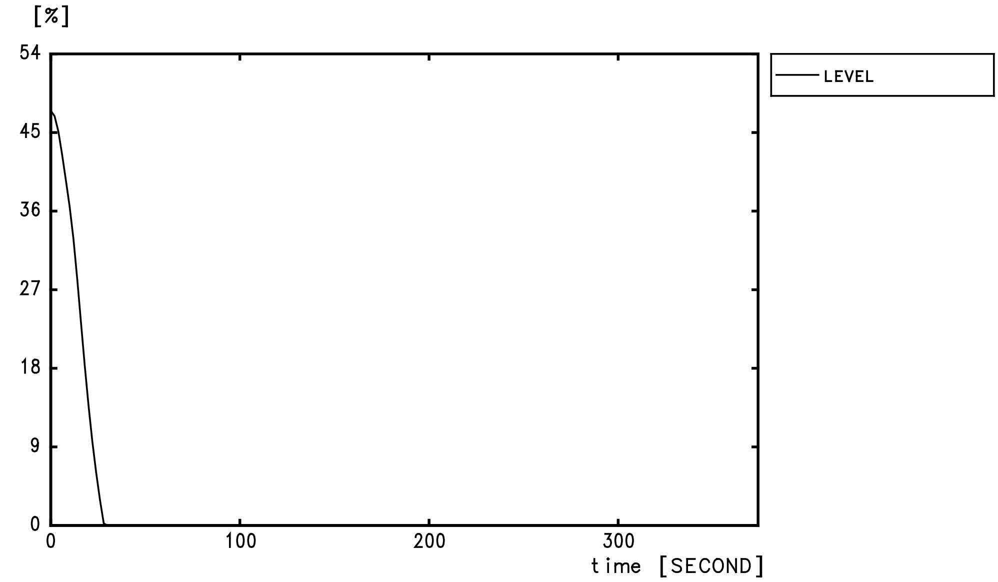
SLB at hot shutdown with Reactor Coolant Pump trip, Case 2 – Reactivity and Core Power



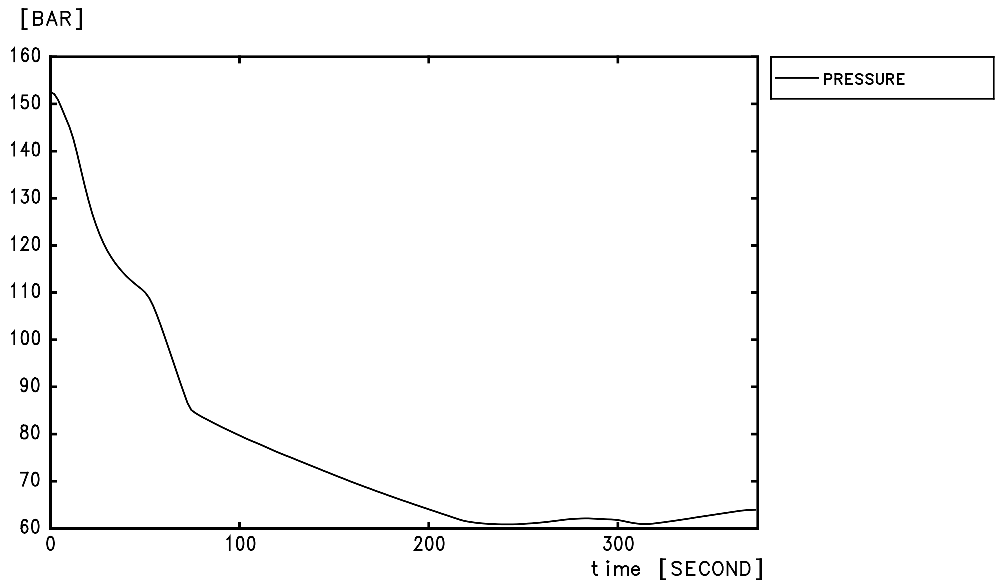


SECTION 14.5.2 - FIGURE 27

SLB at hot shutdown with Reactor Coolant Pump trip, Case 2 – Pressuriser Level and Pressuriser Pressure



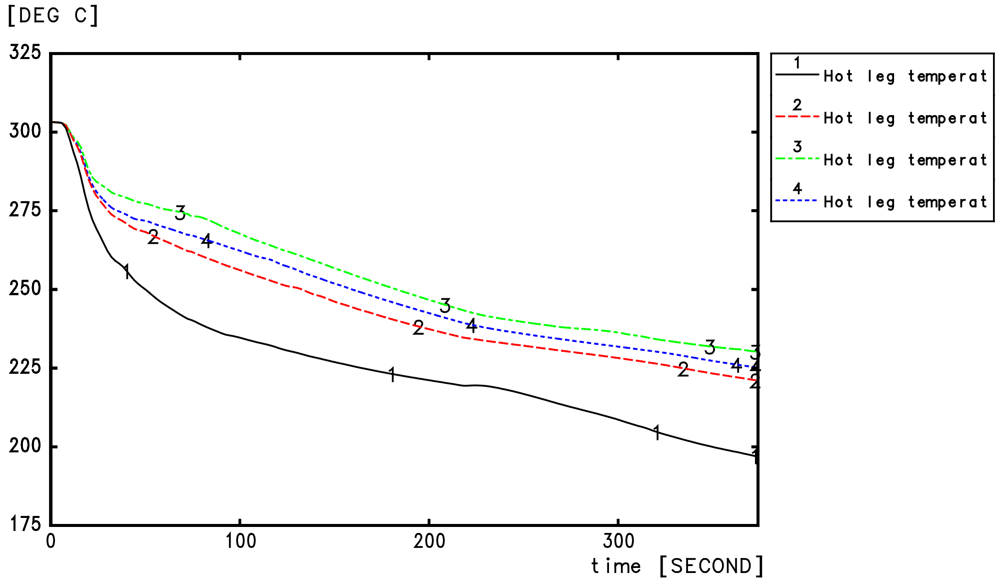
PRESSURIZER LEVEL



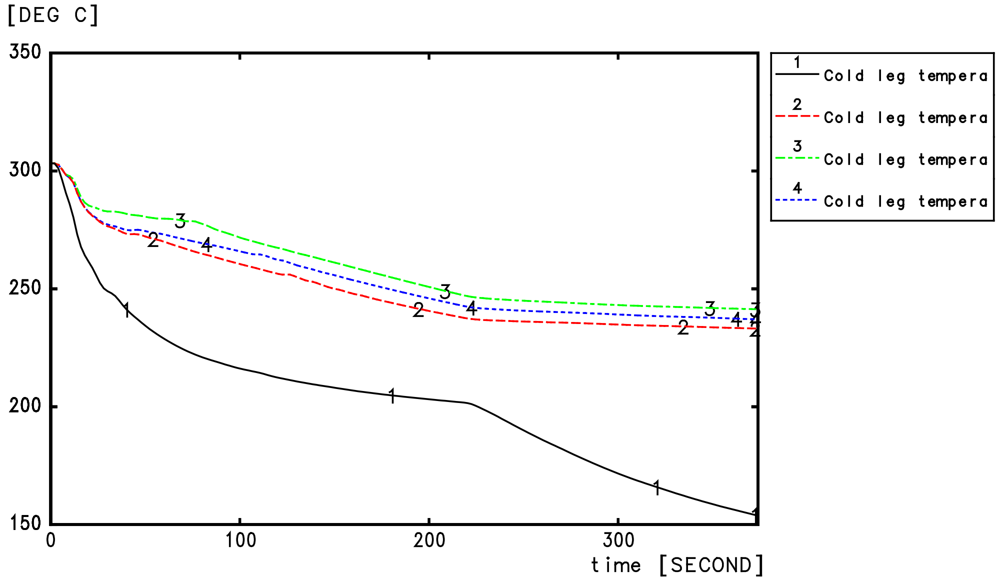
PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 28

SLB at hot shutdown with Reactor Coolant Pump trip, Case 2 – Hot Leg Temperature and Cold Leg Temperature



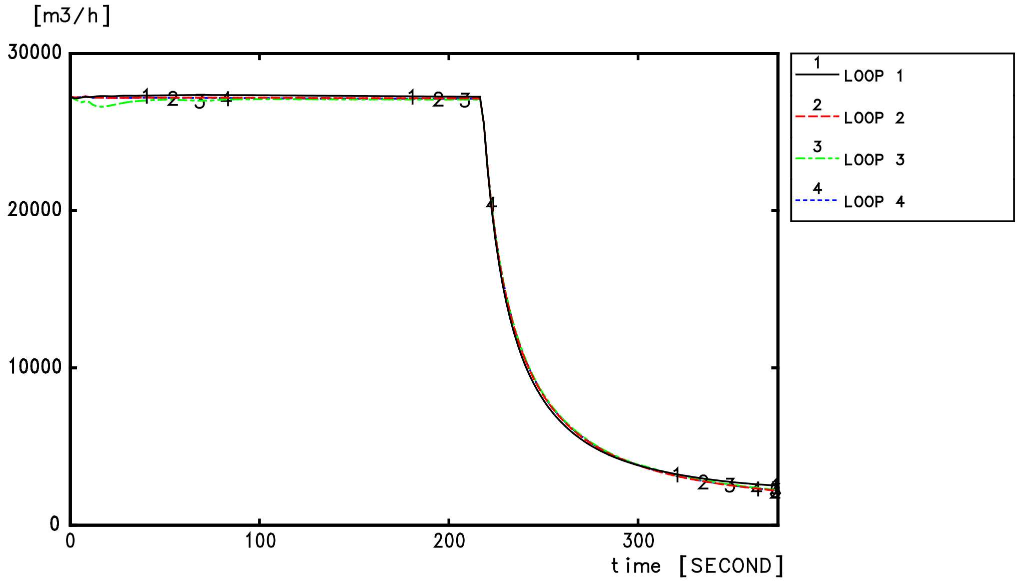
HOT LEG TEMPERATURE



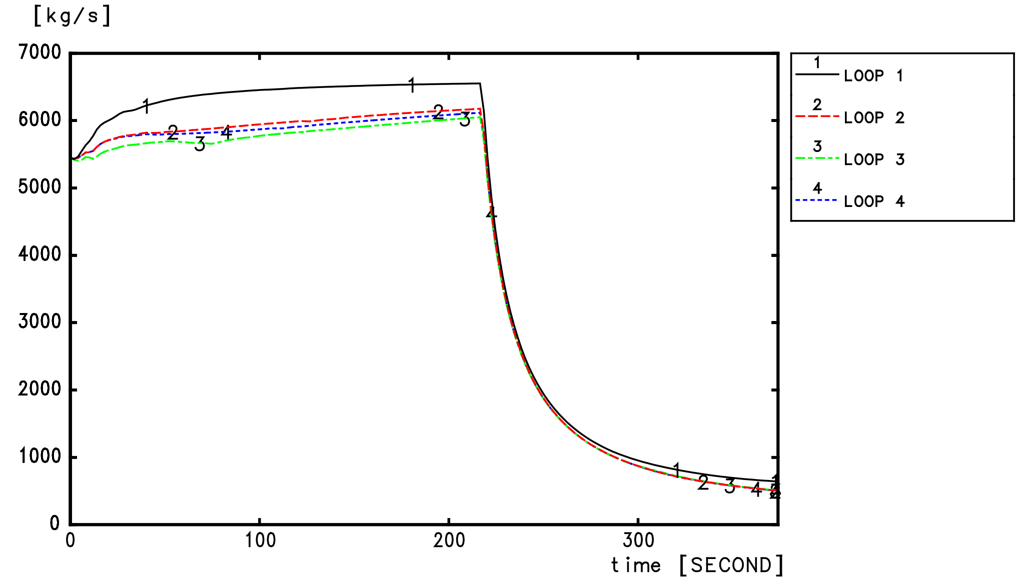
COLD LEG TEMPERATURE

SECTION 14.5.2 - FIGURE 29

SLB at hot shutdown with Reactor Coolant Pump trip, Case 2 – Cold Leg Volumetric Flow Rate and Mass Flow Rate



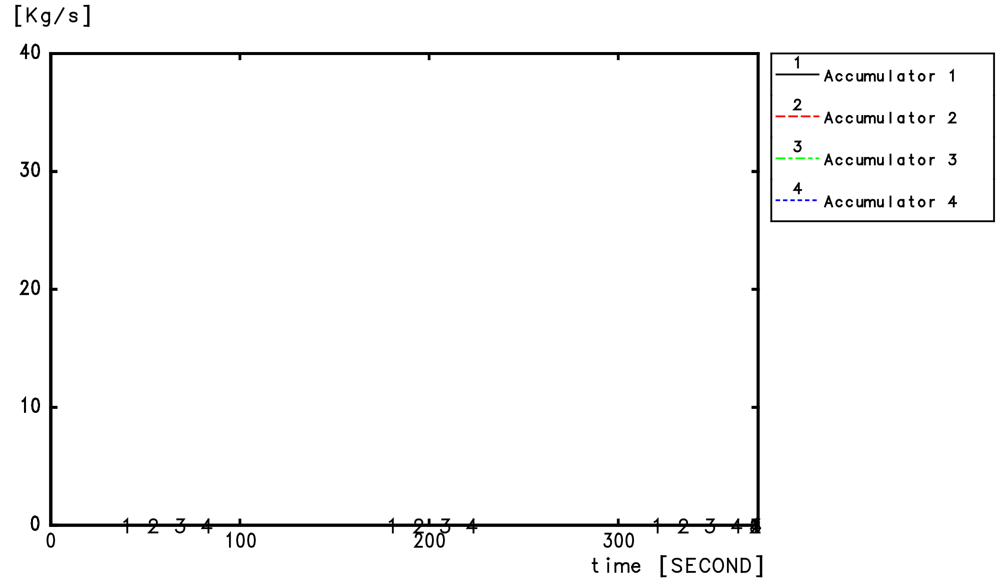
COLD LEG VOLUMETRIC FLOW RATE



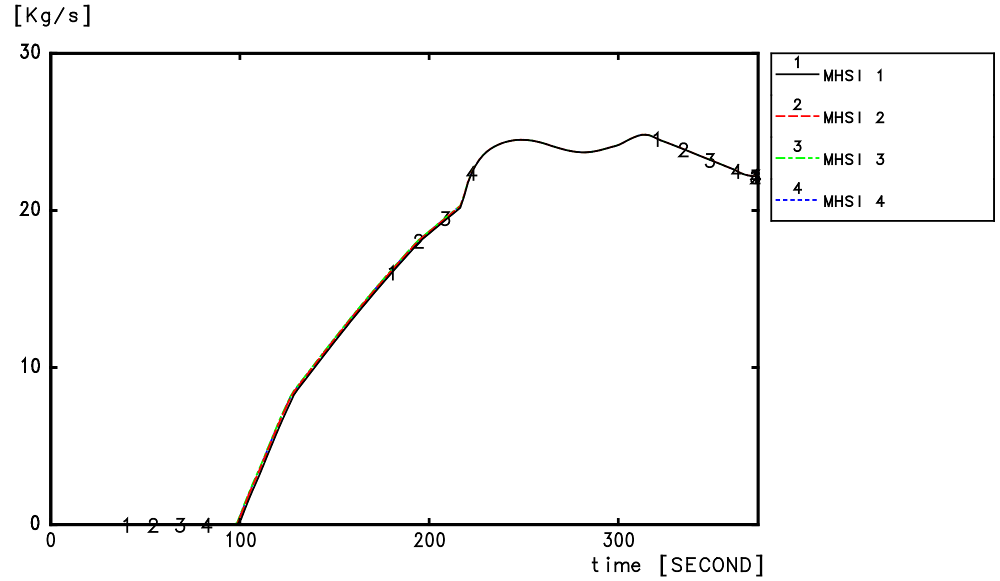
COLD LEG MASS FLOW RATE

SECTION 14.5.2 - FIGURE 30

SLB at hot shutdown with Reactor Coolant Pump trip, Case 2 – Total Accumulators Flow Rate and Total MHSI Flow Rate



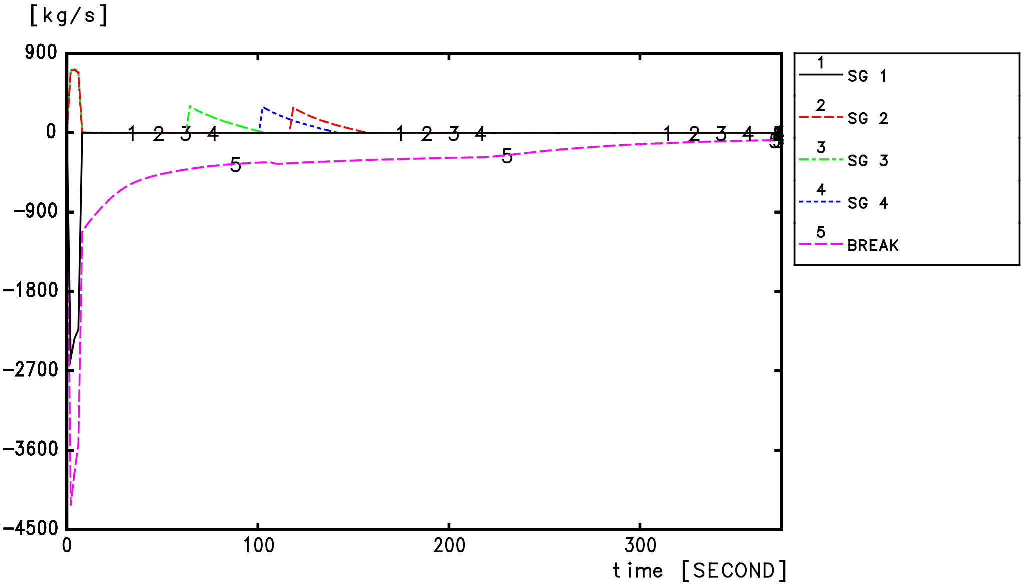
TOTAL ACCUMULATORS FLOW RATE



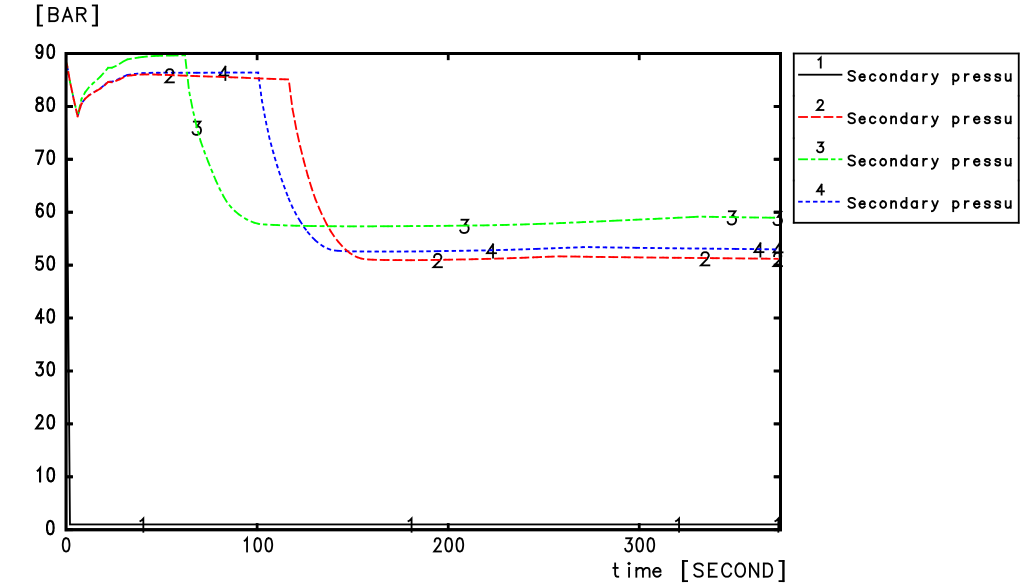
TOTAL MHSI FLOW RATE

SECTION 14.5.2 - FIGURE 31

SLB at hot shutdown with Reactor Coolant Pump trip, Case 2 – Vapour Mass Flow Rate and SG Pressure



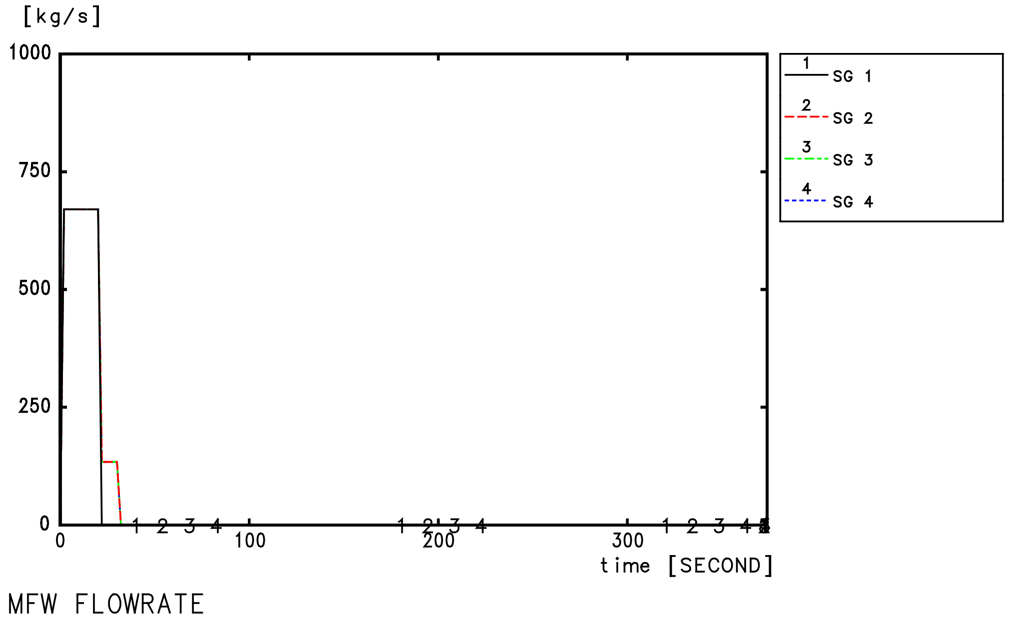
VAPOR MASS FLOW RATE

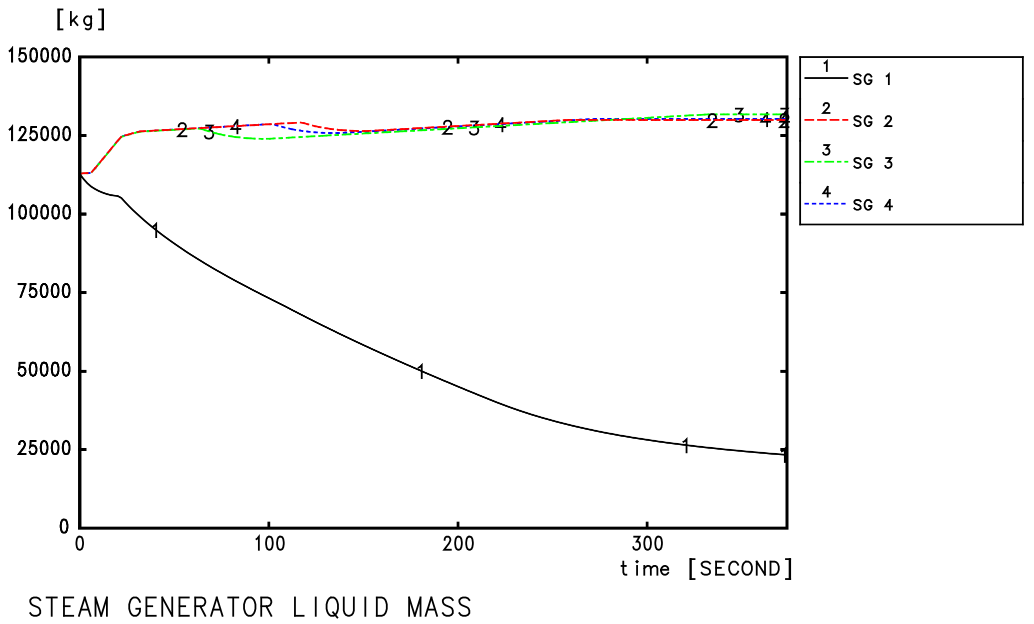


STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 32

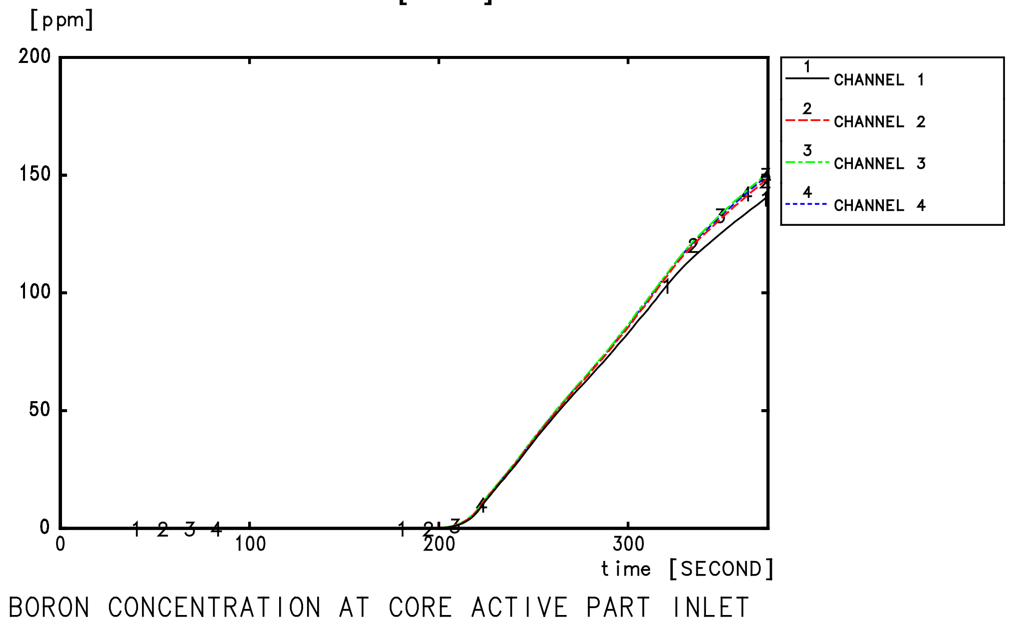
SLB at hot shutdown with Reactor Coolant Pump trip, Case 2 – Main Feedwater Flow Rate and SG Liquid Mass

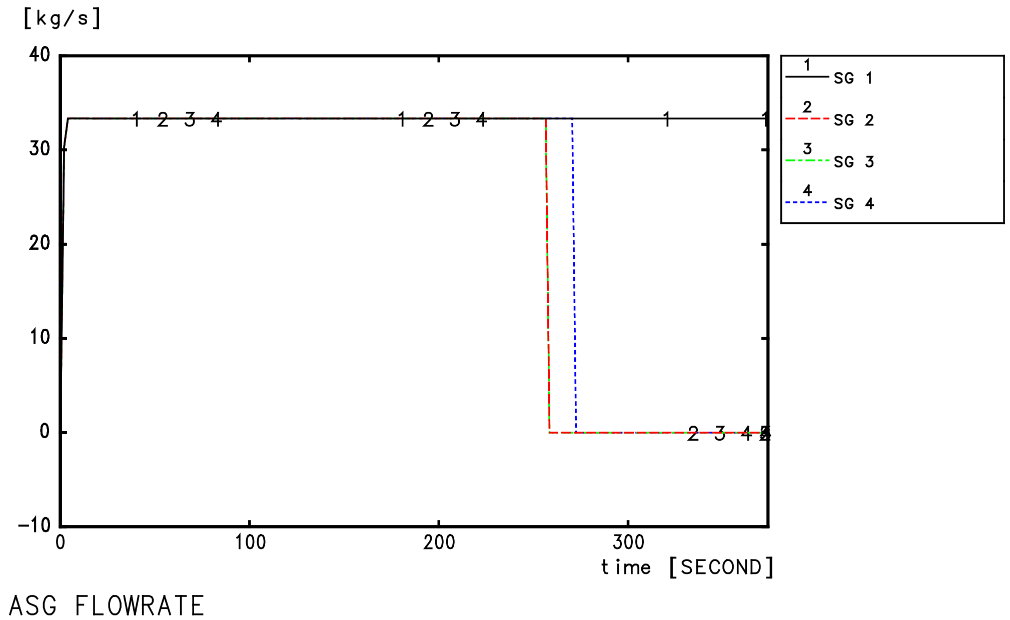




SECTION 14.5.2 - FIGURE 33

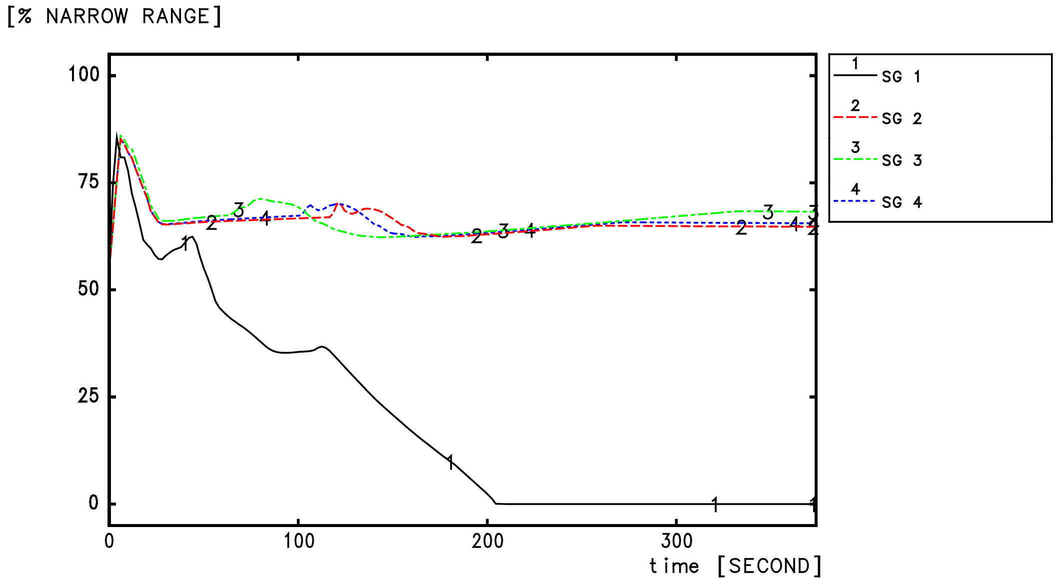
SLB at hot shutdown with Reactor Coolant Pump trip, Case 2 – Boron Concentration and ASG [EFWS] Flow Rate



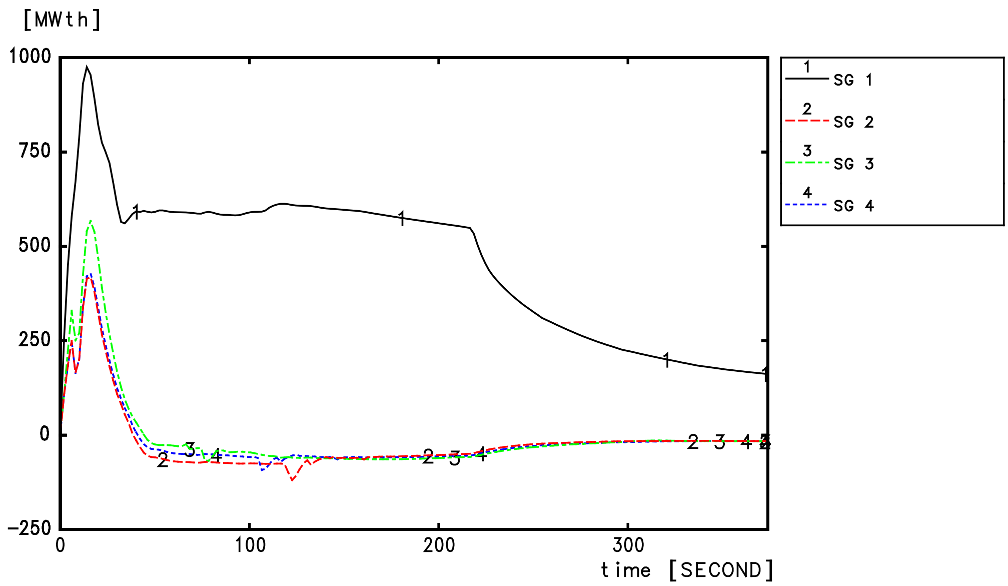


SECTION 14.5.2 - FIGURE 34

SLB at hot shutdown with Reactor Coolant Pump trip, Case 2 – SG Narrow Range Level and SG Power Exchanged



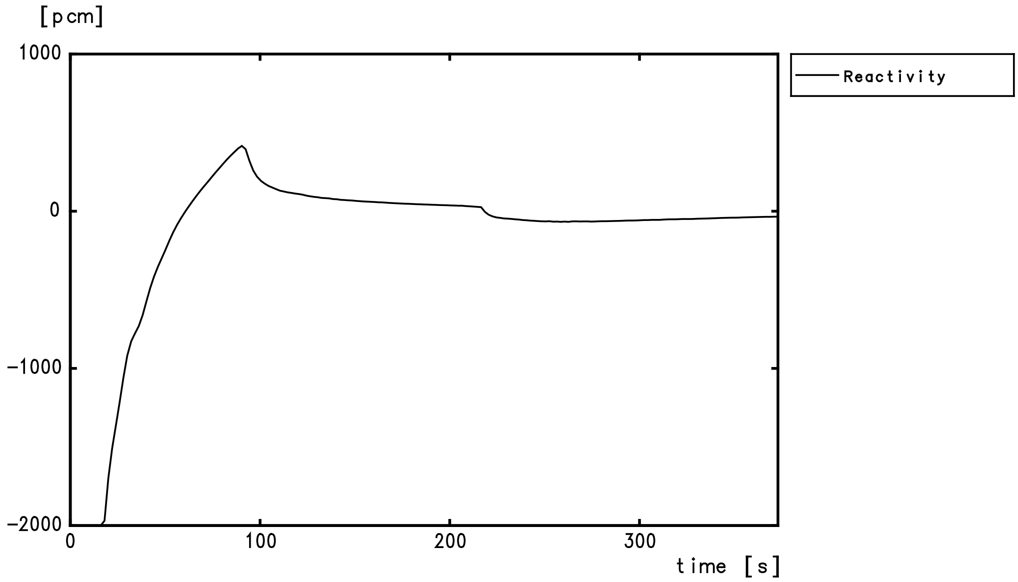
STEAM GENERATOR NARROW RANGE LEVEL



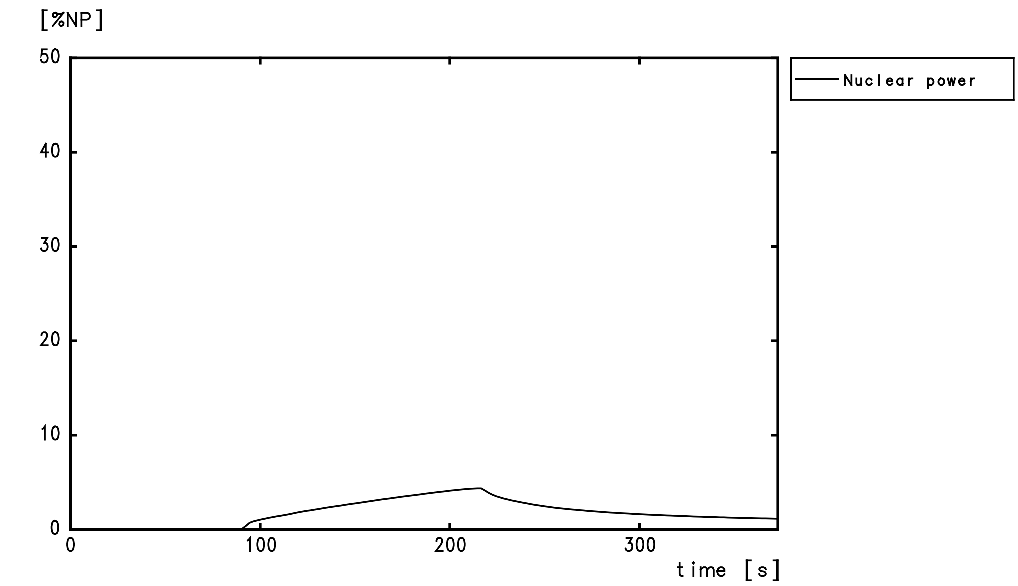
SG POWER EXCHANGED

SECTION 14.5.2 - FIGURE 35

SLB at hot shutdown with Reactor Coolant Pump trip, case 3 – Reactivity and Core Power



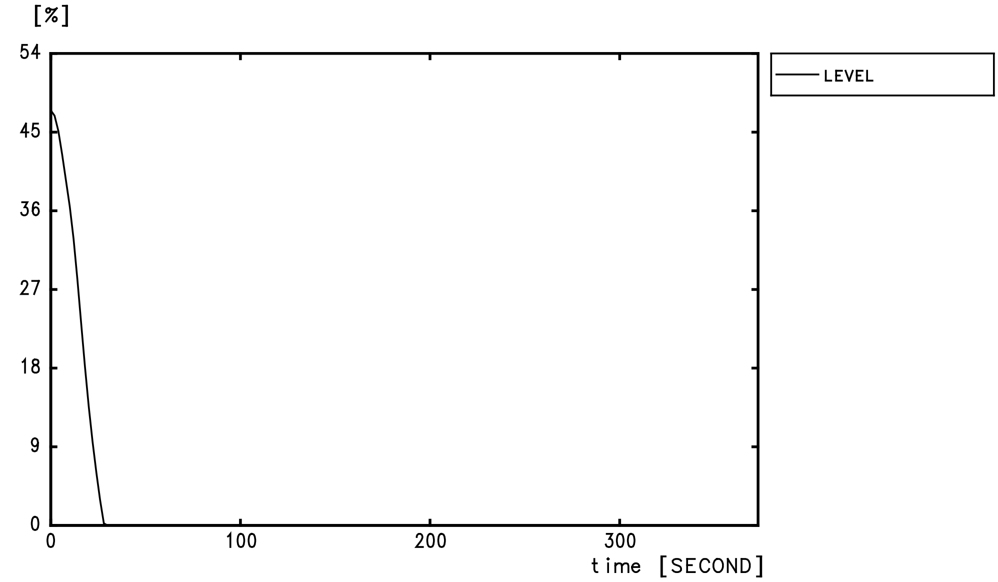
REACTIVITY



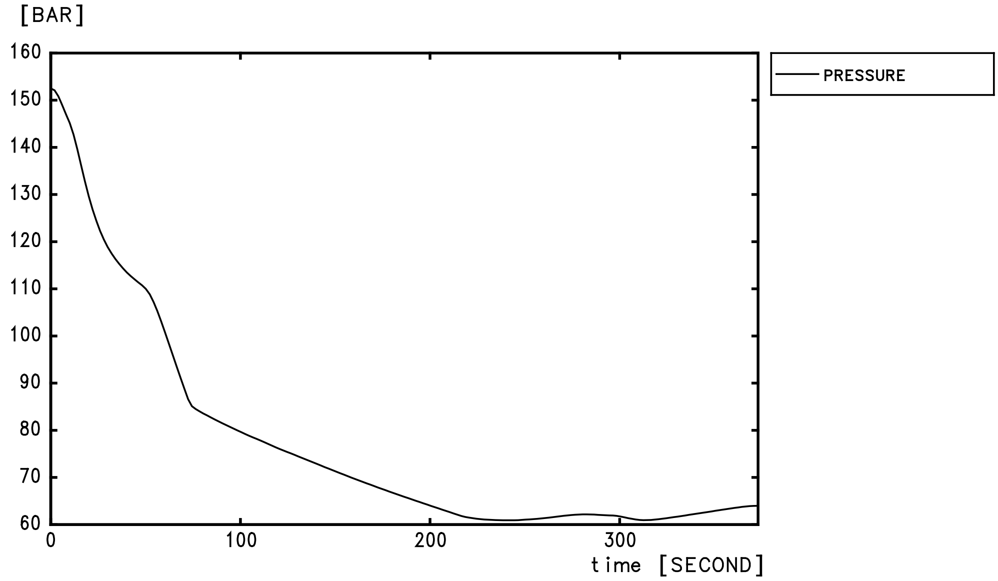
CORE POWER

SECTION 14.5.2 - FIGURE 36

SLB at hot shutdown with Reactor Coolant Pump trip, Case 3 – Pressuriser Level and Pressuriser Pressure



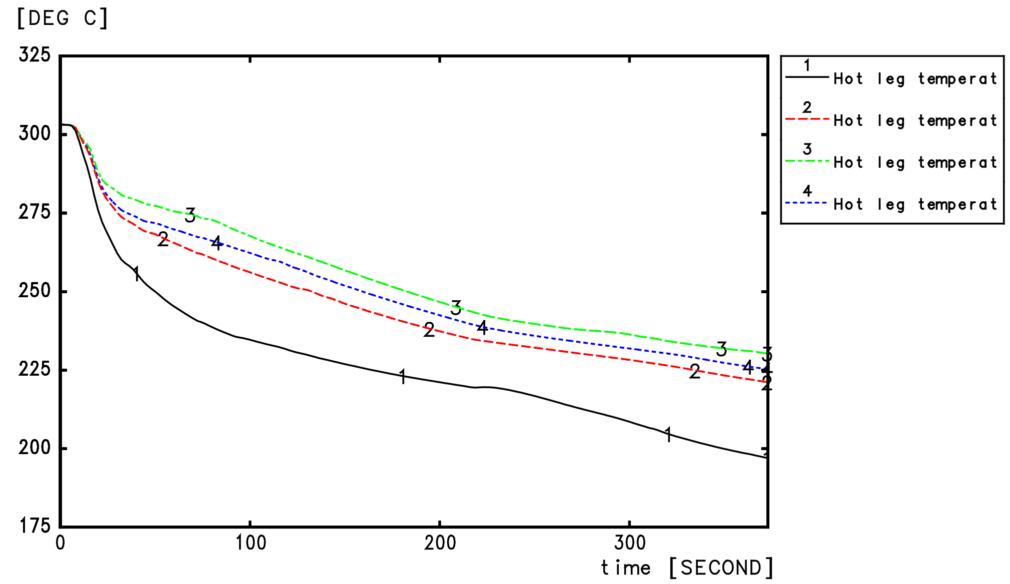
PRESSURIZER LEVEL



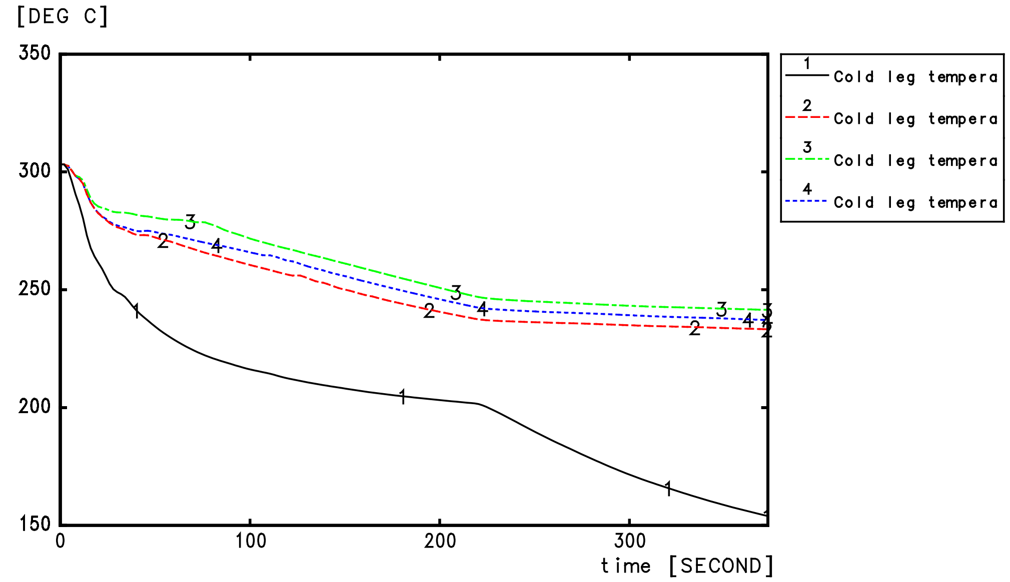
PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 37

SLB at hot shutdown with Reactor Coolant Pump trip, Case 3 – Hot Leg Temperature and Cold Leg Temperature



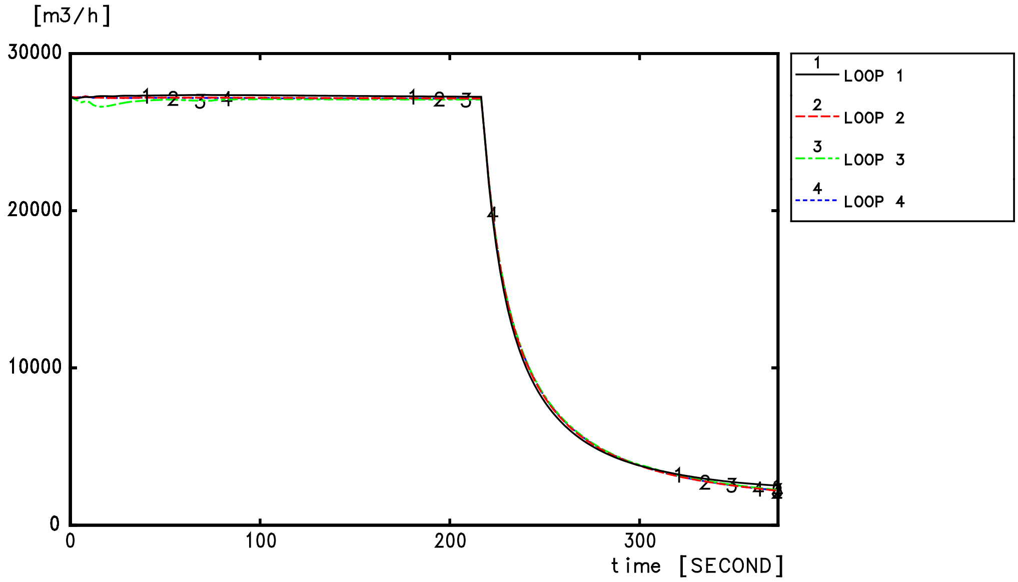
HOT LEG TEMPERATURE



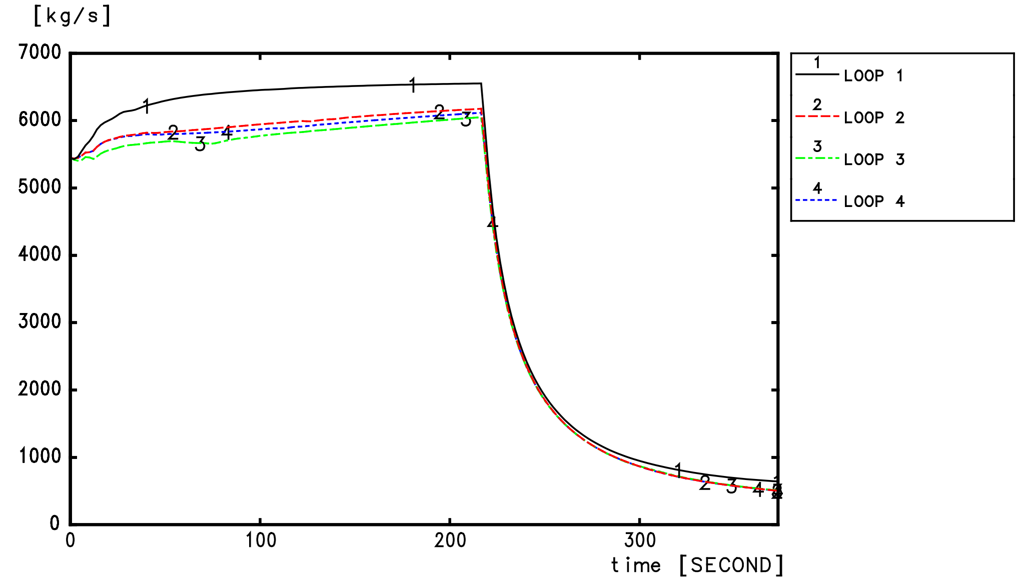
COLD LEG TEMPERATURE

SECTION 14.5.2 - FIGURE 38

SLB at hot shutdown with Reactor Coolant Pump trip, Case 3 – Cold Leg Volumetric Flow Rate and Mass Flow Rate



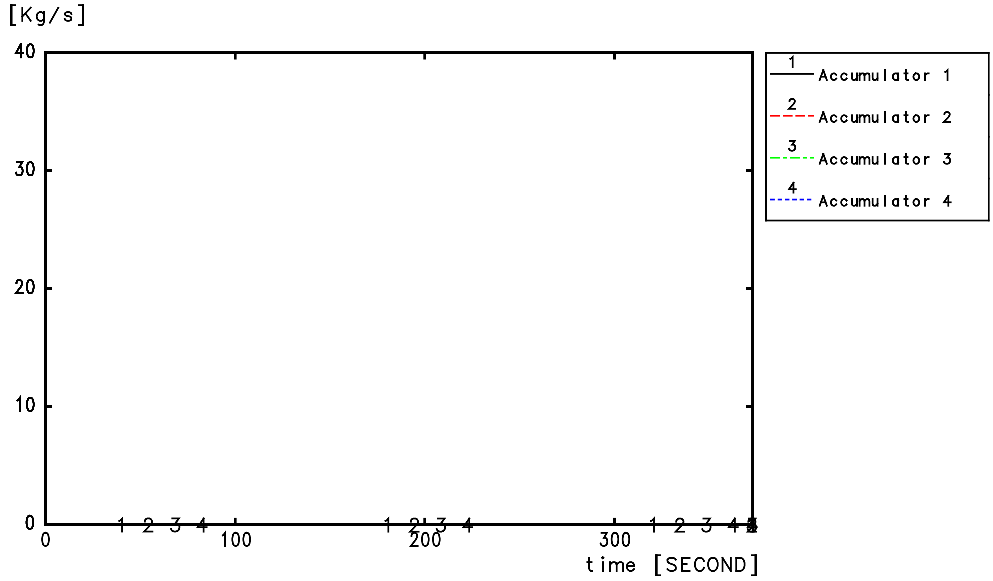
COLD LEG VOLUMETRIC FLOW RATE



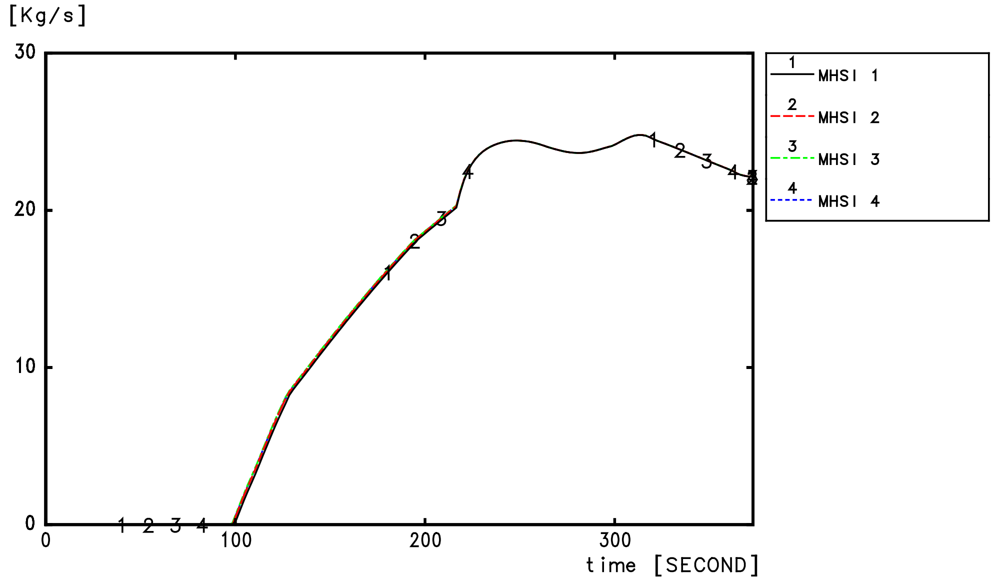
COLD LEG MASS FLOW RATE

SECTION 14.5.2 - FIGURE 39

SLB at hot shutdown with Reactor Coolant Pump trip, Case 3 – Total Accumulators Flow Rate and Total MHSI Flow Rate



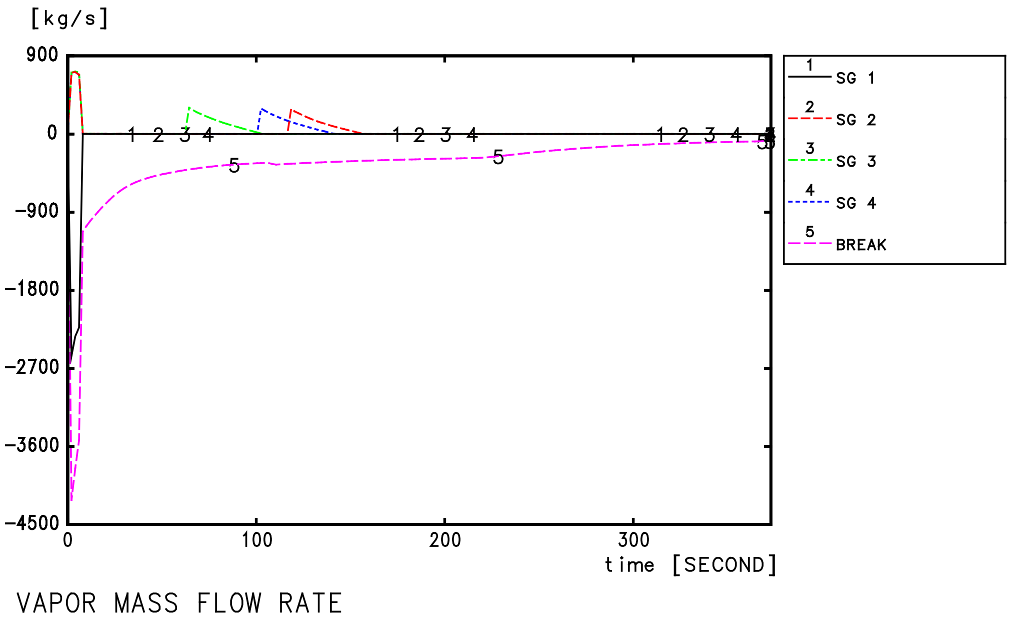
TOTAL ACCUMULATORS FLOW RATE

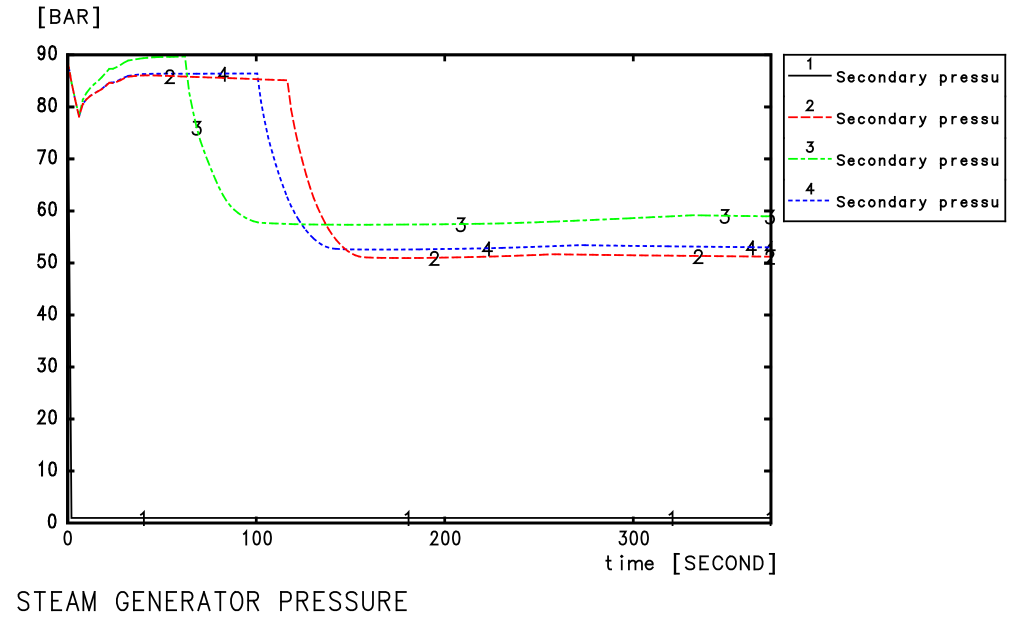


TOTAL MHSI FLOW RATE

SECTION 14.5.2 - FIGURE 40

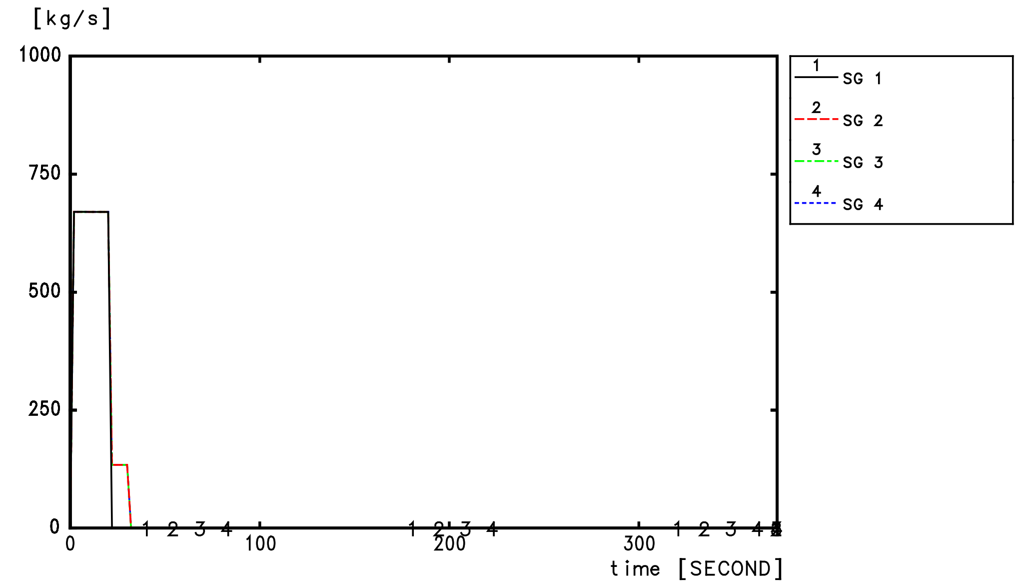
SLB at hot shutdown with Reactor Coolant Pump trip, Case 3 – Vapour Mass Flow Rate and SG Pressure



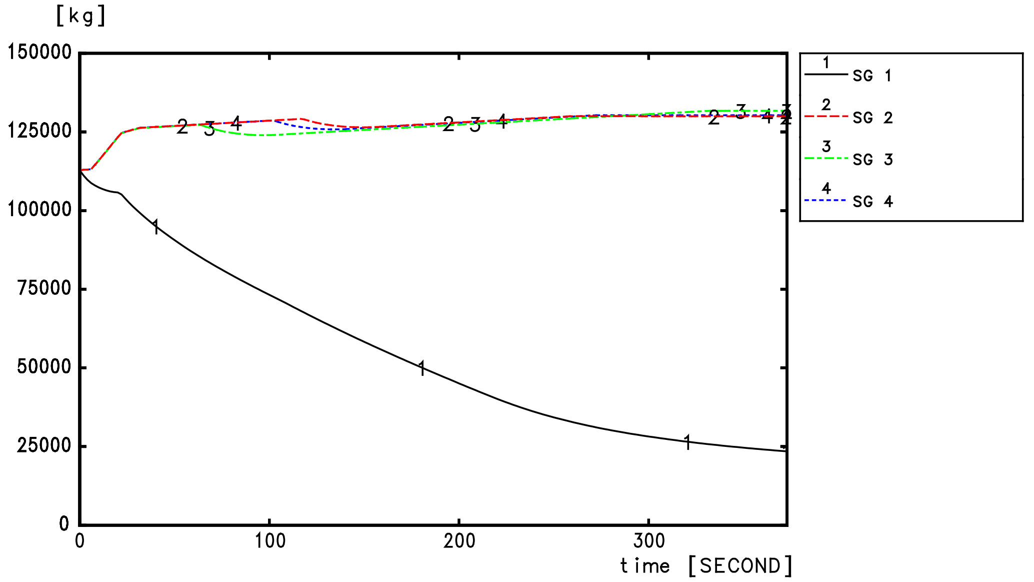


SECTION 14.5.2 - FIGURE 41

SLB at hot shutdown with Reactor Coolant Pump trip, Case 3 – Main Feedwater Flow Rate and SG Liquid Mass



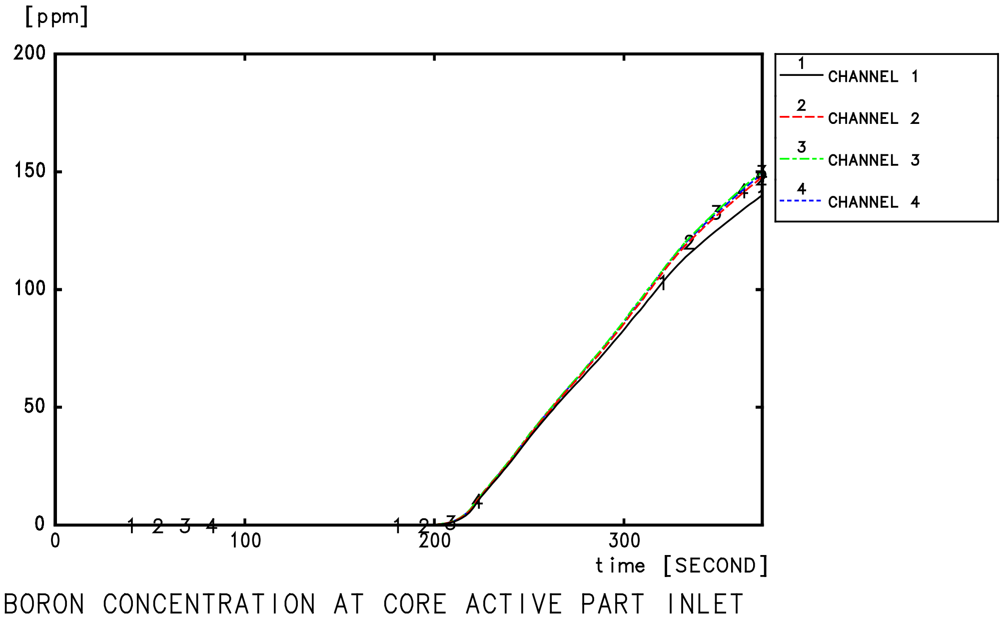
MFW FLOWRATE

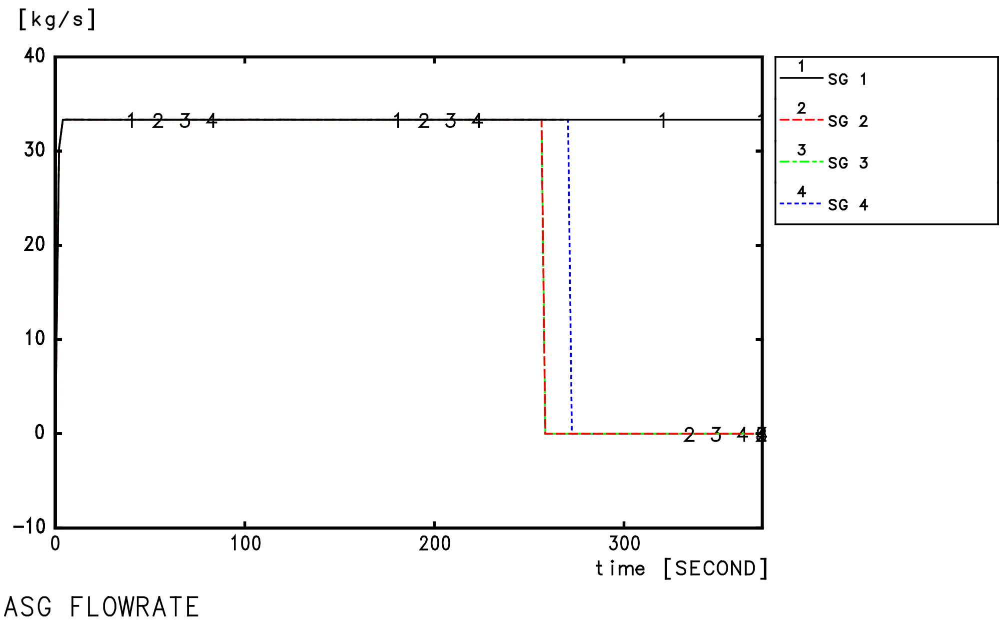


STEAM GENERATOR LIQUID MASS

SECTION 14.5.2 - FIGURE 42

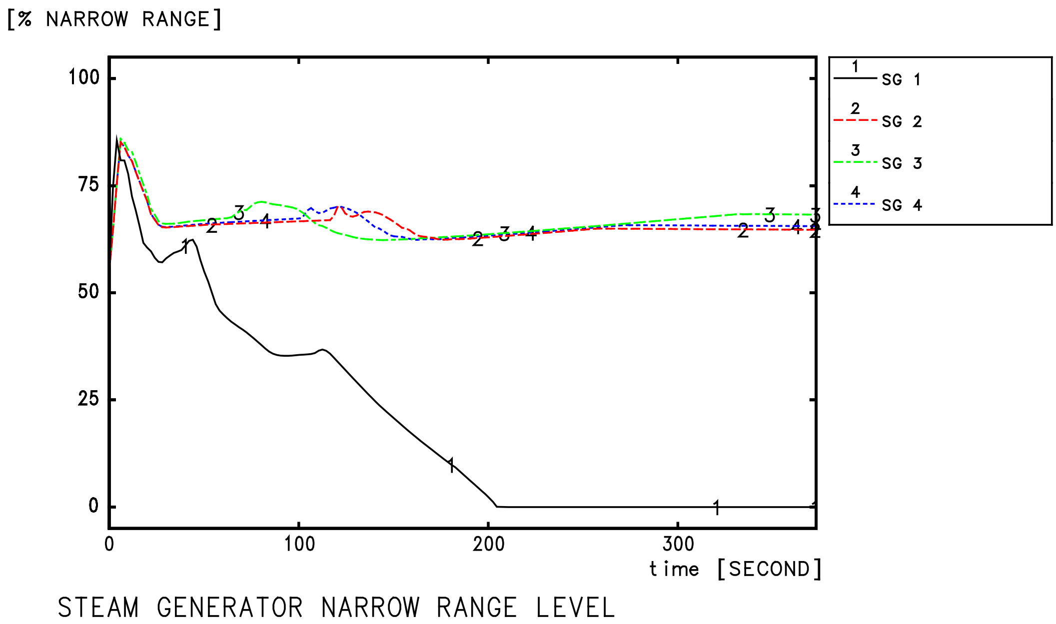
SLB at hot shutdown with Reactor Coolant Pump trip, Case 3 – Boron Concentration and ASG [EFWS] Flow Rate

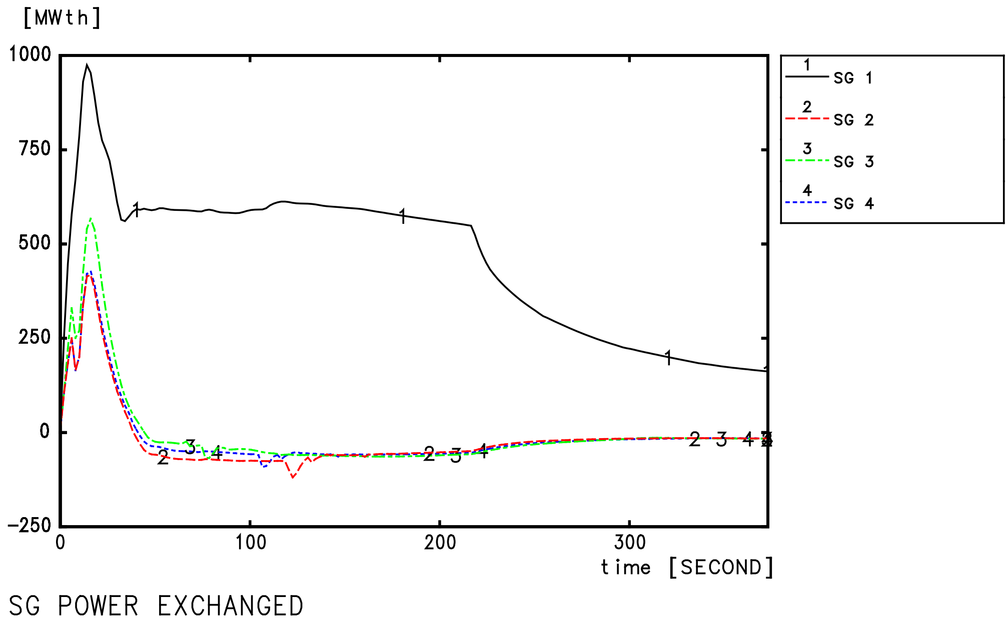




SECTION 14.5.2 - FIGURE 43

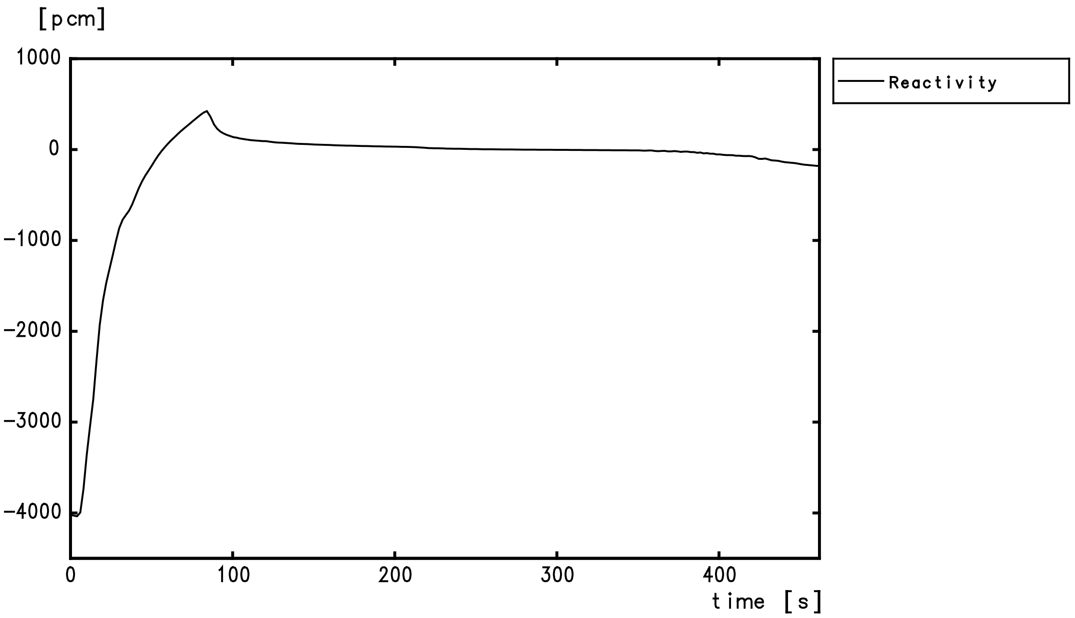
SLB at hot shutdown with Reactor Coolant Pump trip, Case 3 – SG Narrow Range Level and SG Power Exchanged



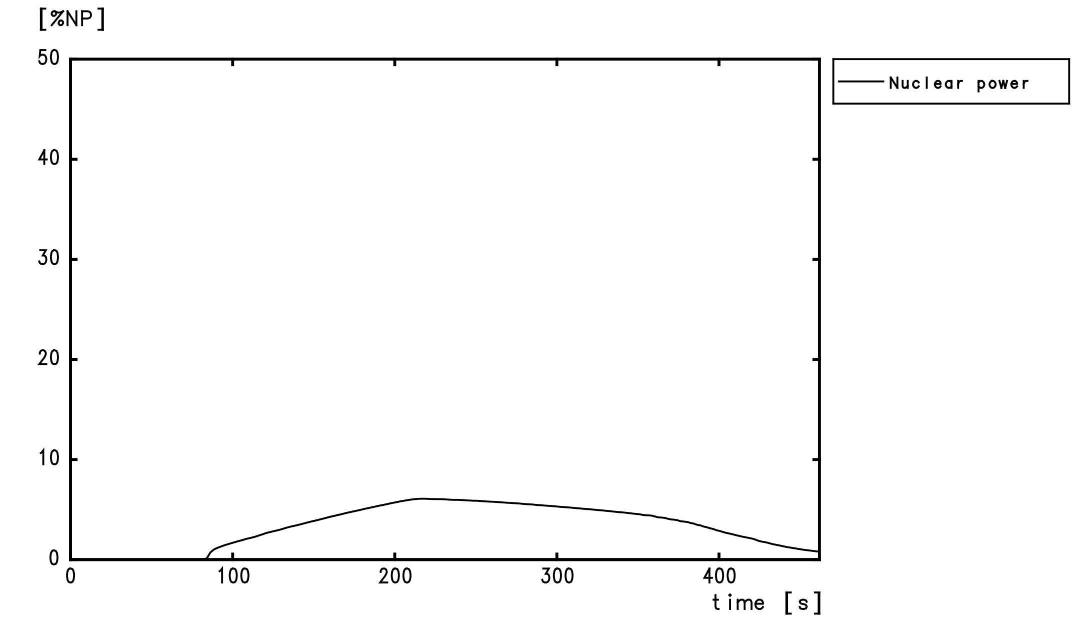


SECTION 14.5.2 - FIGURE 44

Zero Xenon Concentration Initial Conditions – Reactivity and Core Power



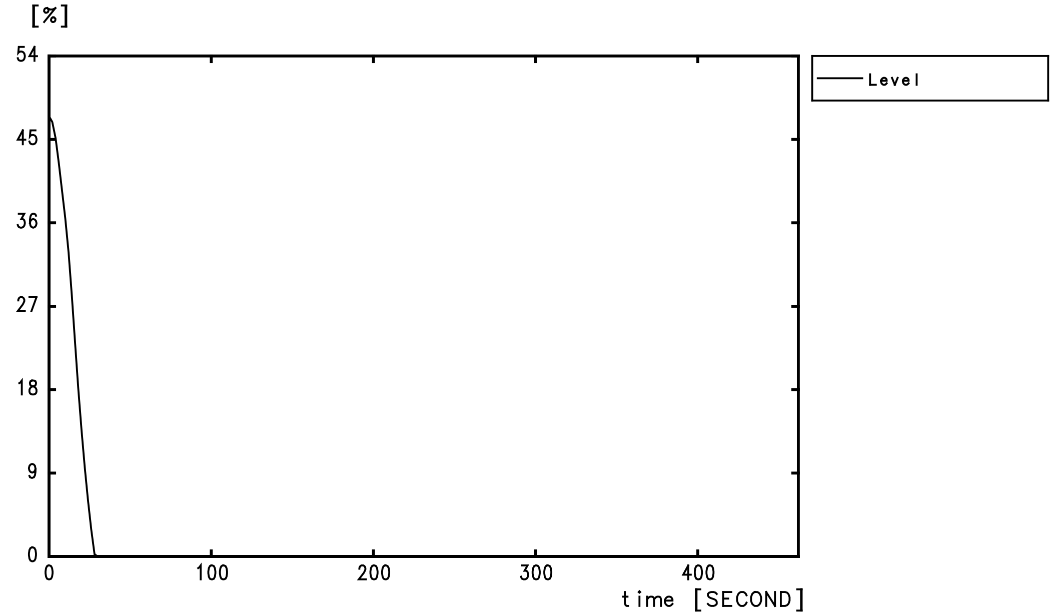
REACTIVITY



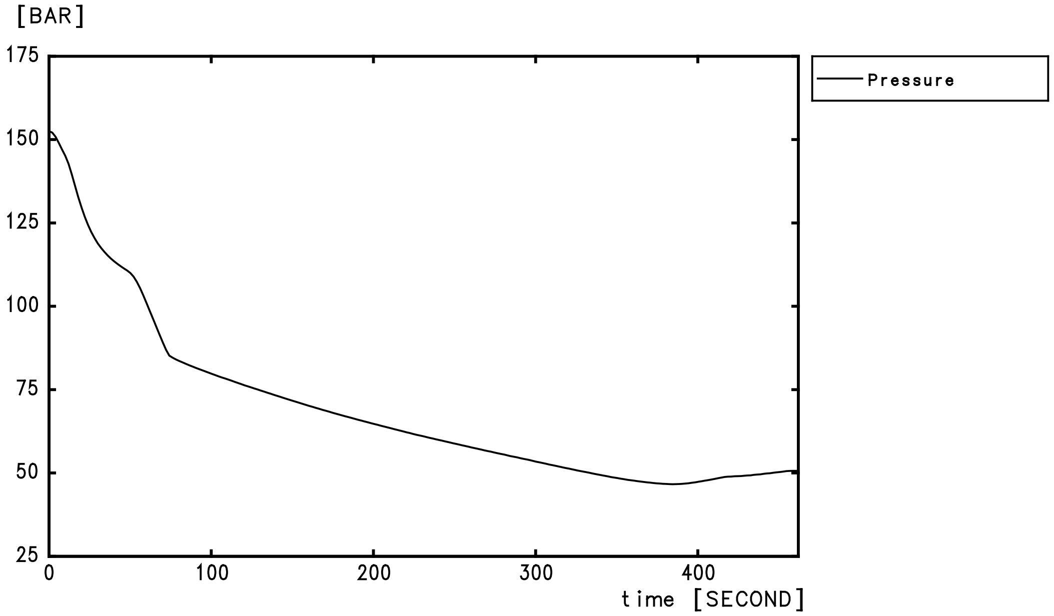
CORE POWER

SECTION 14.5.2 - FIGURE 45

Zero Xenon Concentration Initial Conditions – Pressuriser Level and Pressuriser Pressure



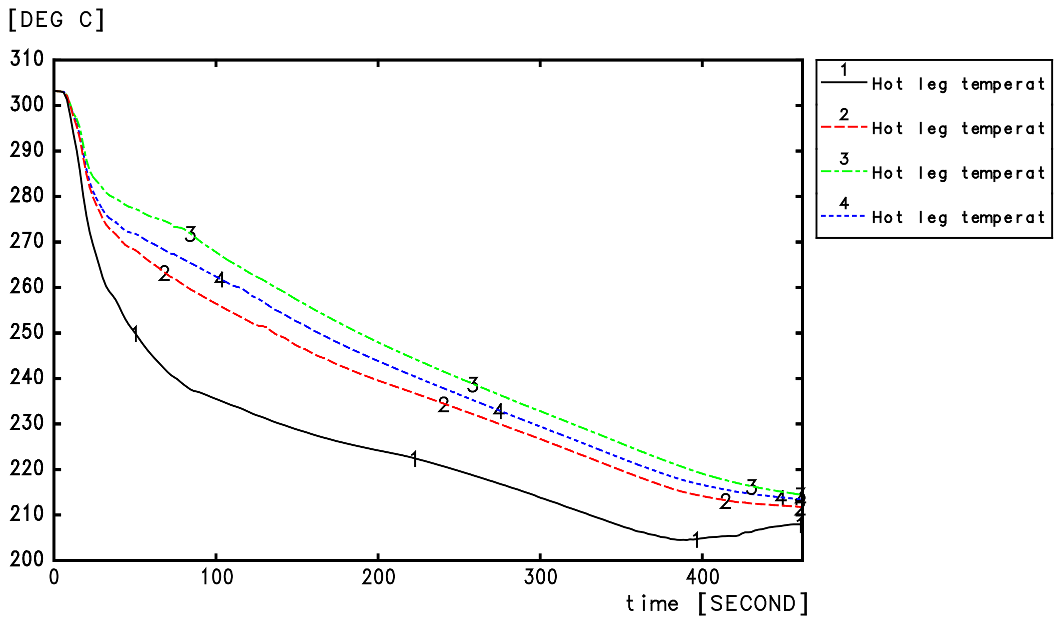
PRESSURIZER LEVEL



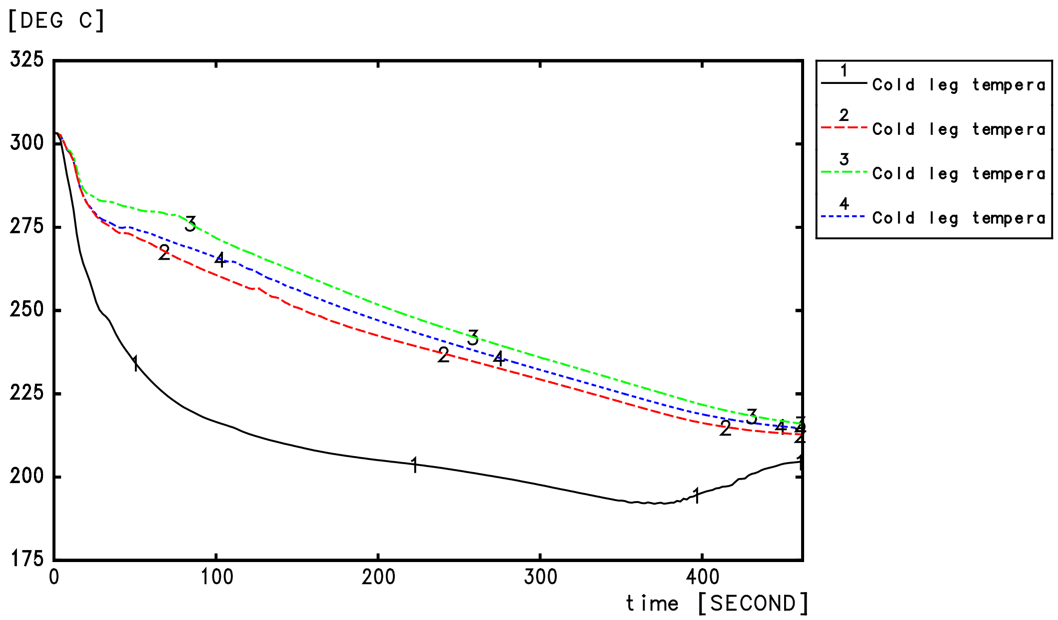
PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 46

Zero Xenon Concentration Initial Conditions – Hot Leg Temperature and Cold Leg Temperature



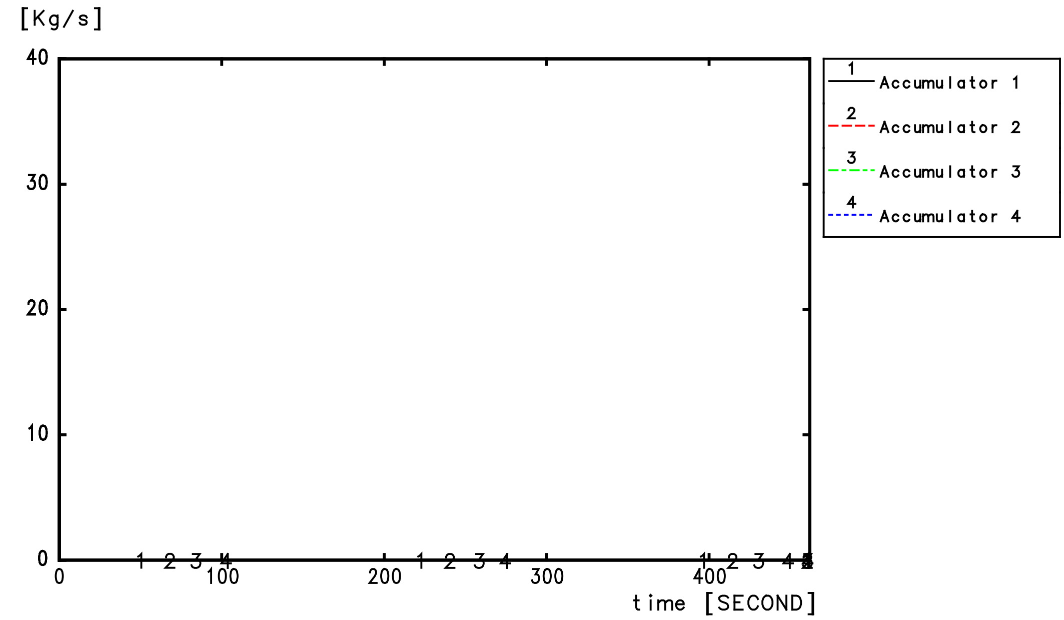
HOT LEG TEMPERATURE



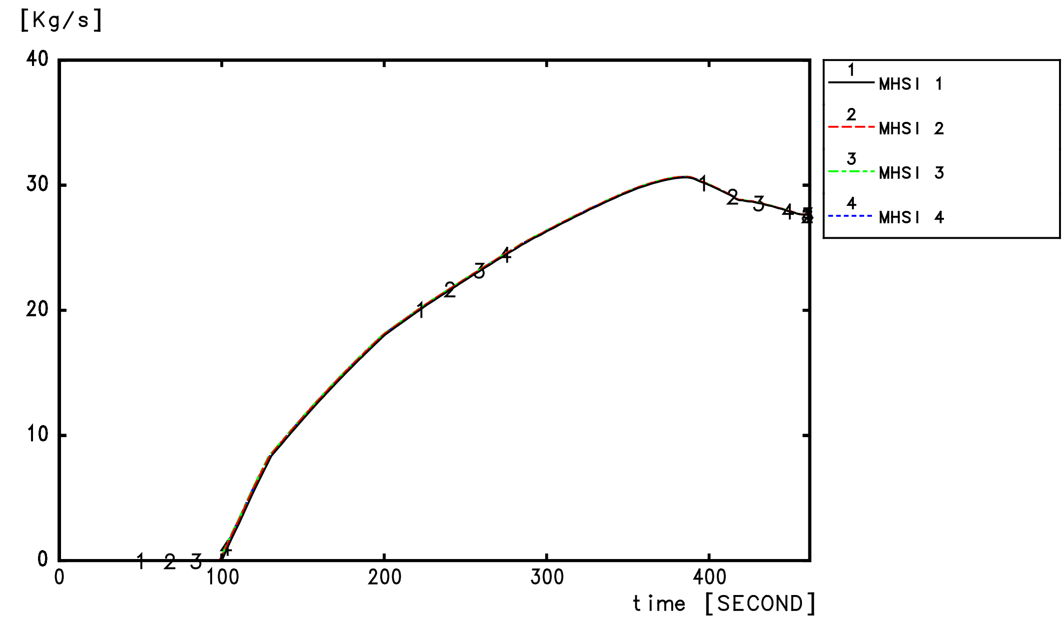
COLD LEG TEMPERATURE

SECTION 14.5.2 - FIGURE 47

Zero Xenon Concentration Initial Conditions – Total Accumulators Flow Rate and Total MHSI Flow Rate



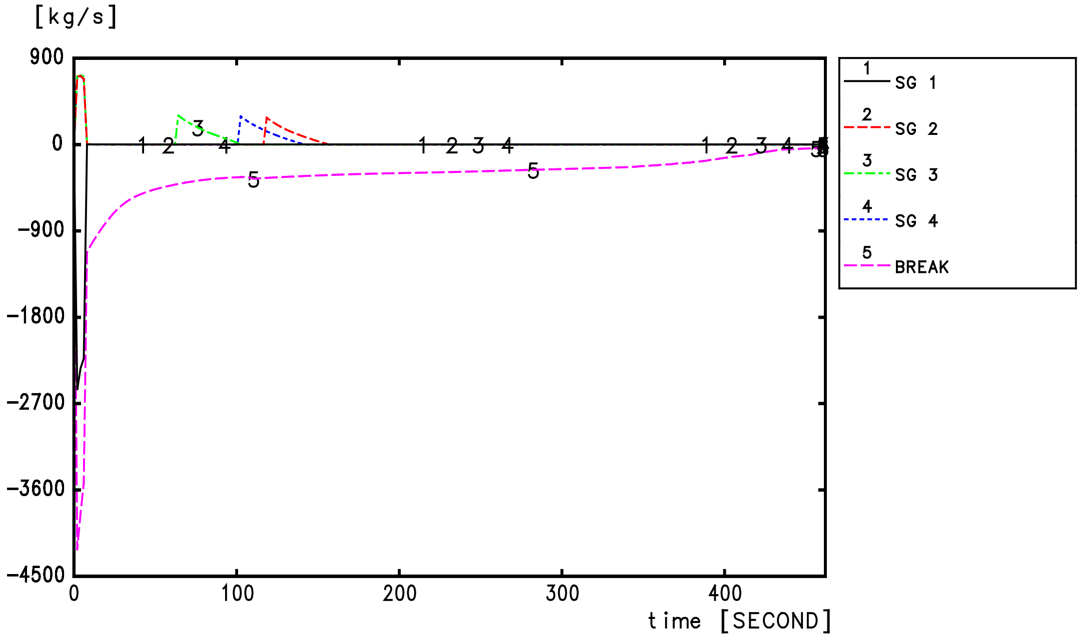
TOTAL ACCUMULATORS FLOW RATE



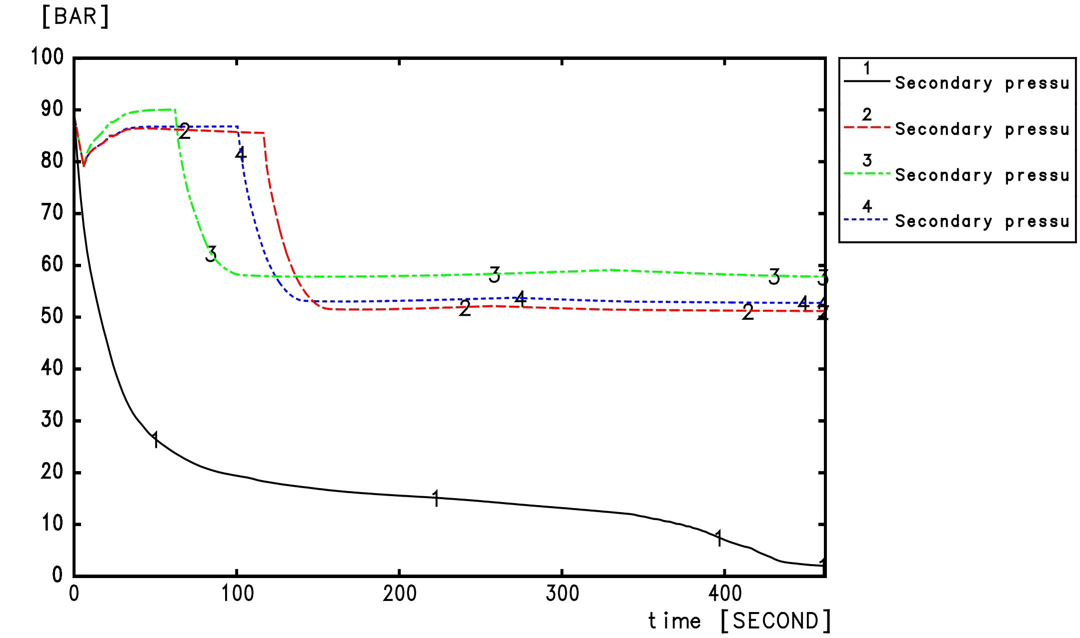
TOTAL MHSI FLOW RATE

SECTION 14.5.2 - FIGURE 48

Zero Xenon Concentration Initial Conditions – Vapour Mass Flow Rate and SG Pressure



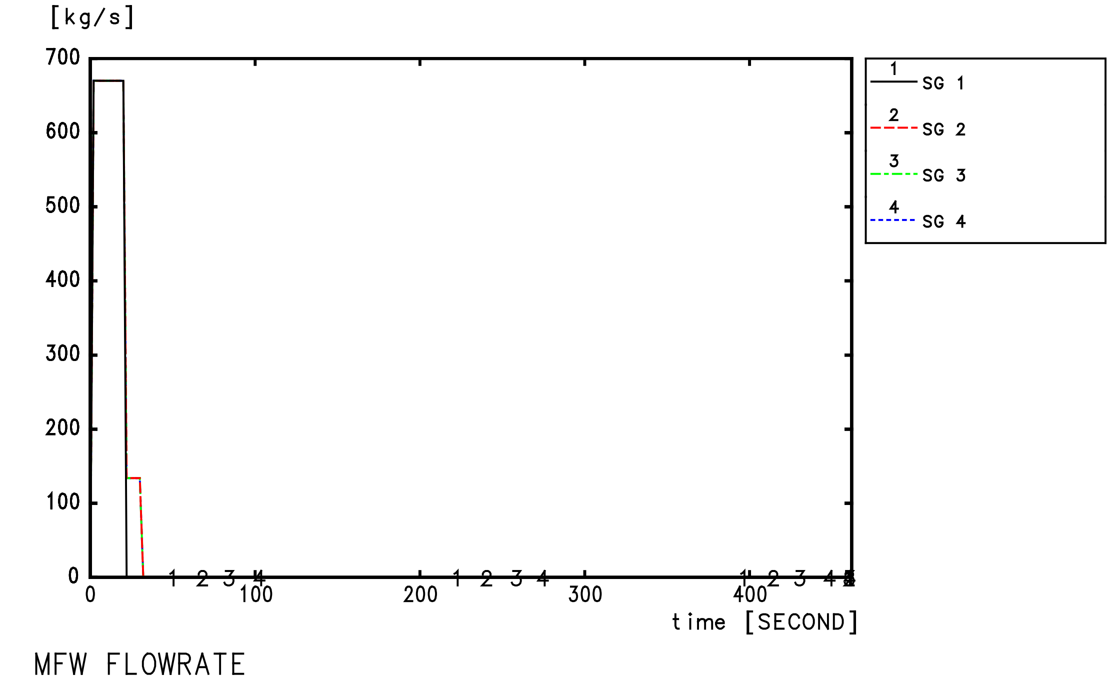
VAPOR MASS FLOWRATE

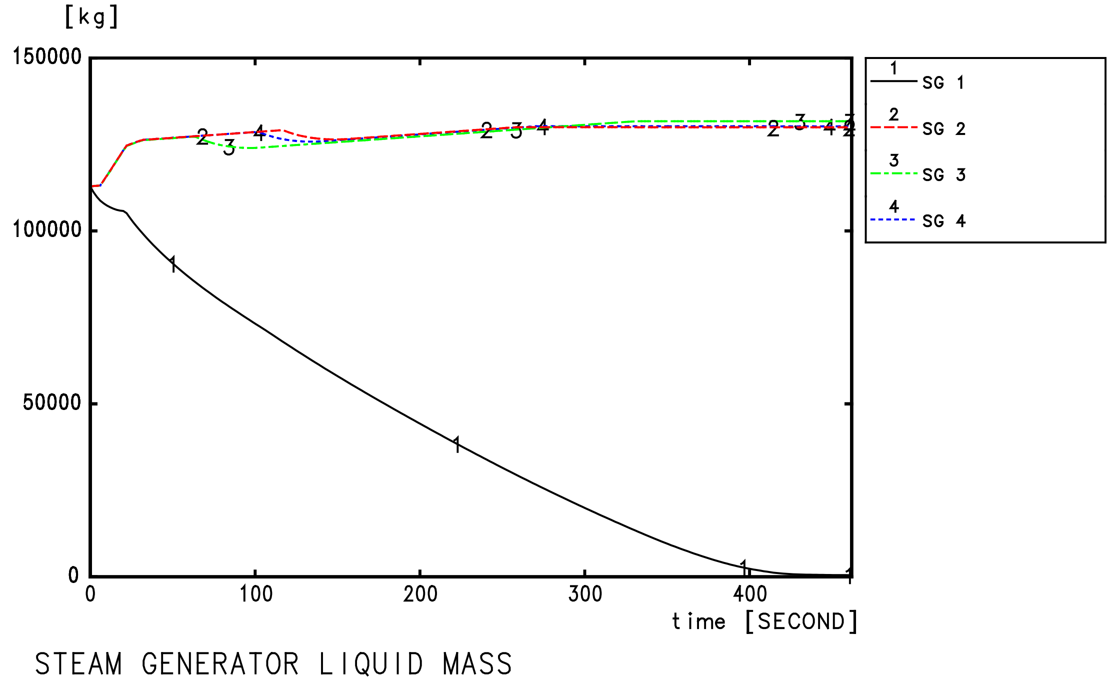


STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 49

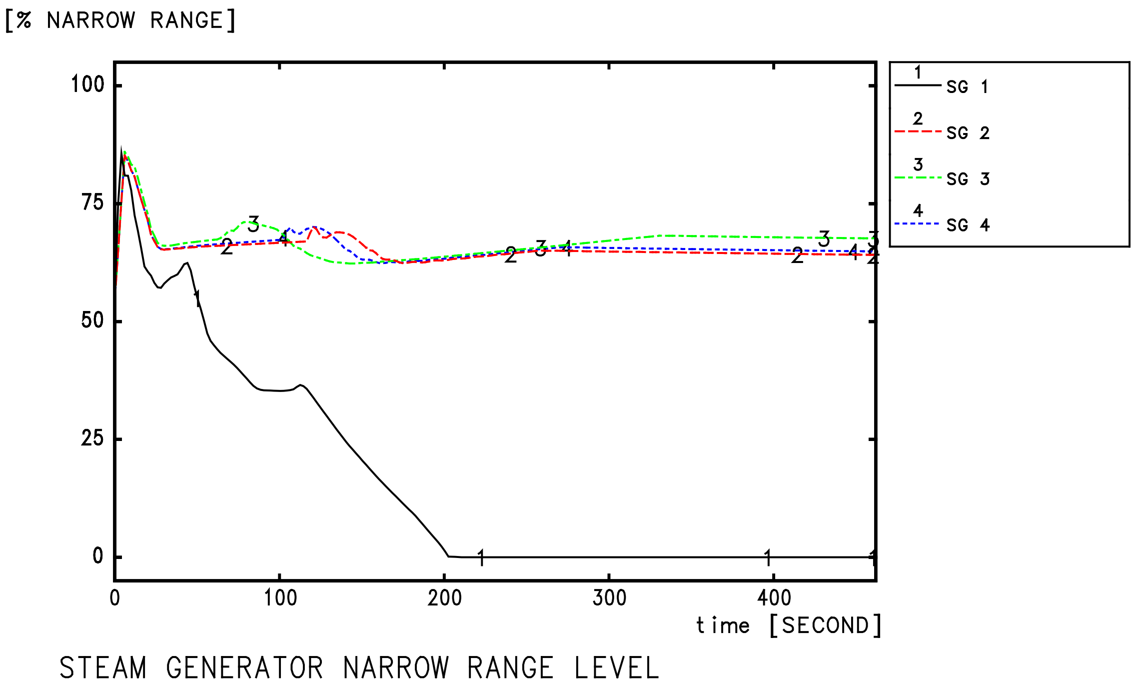
Zero Xenon Concentration Initial Conditions – Main Feedwater Flow Rate and SG Liquid Mass

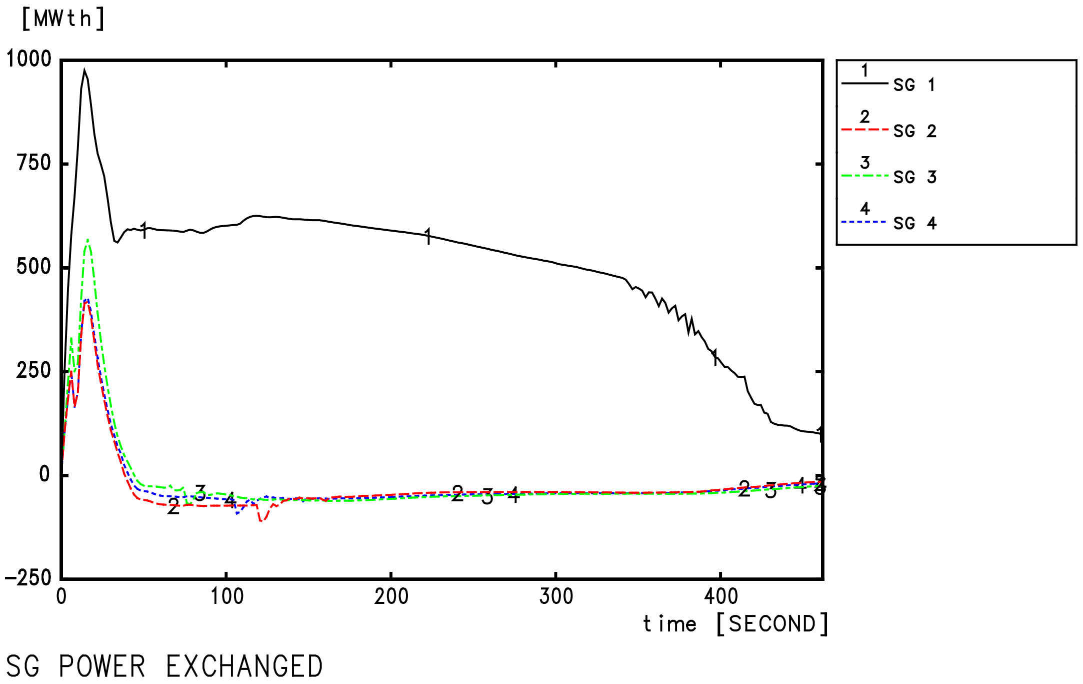




SECTION 14.5.2 - FIGURE 50

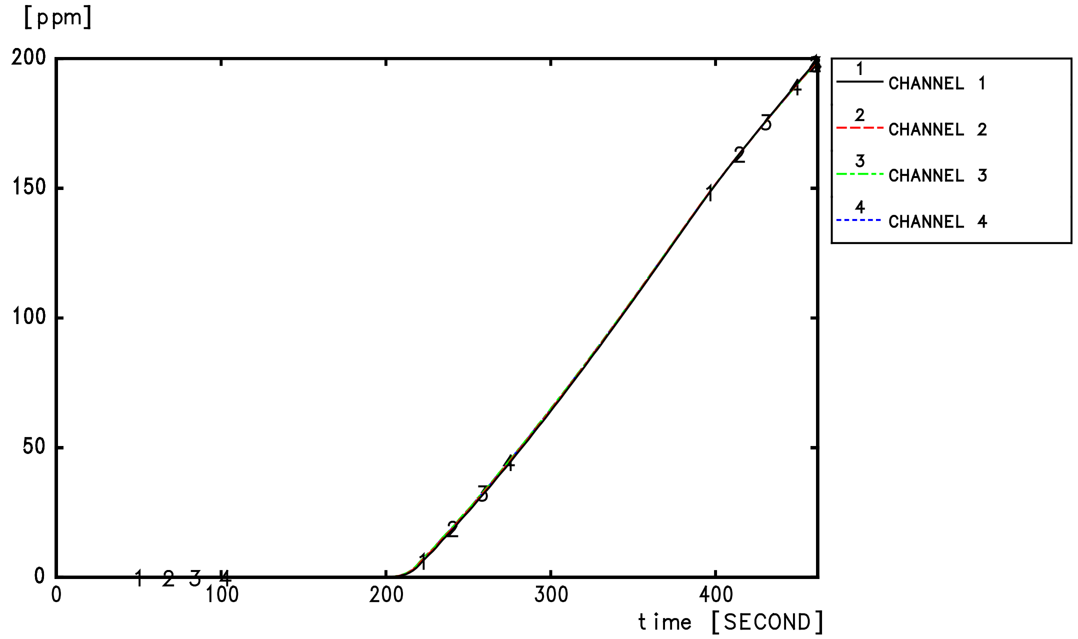
Zero Xenon Concentration Initial Conditions – SG NR Level and SG Power Exchanged



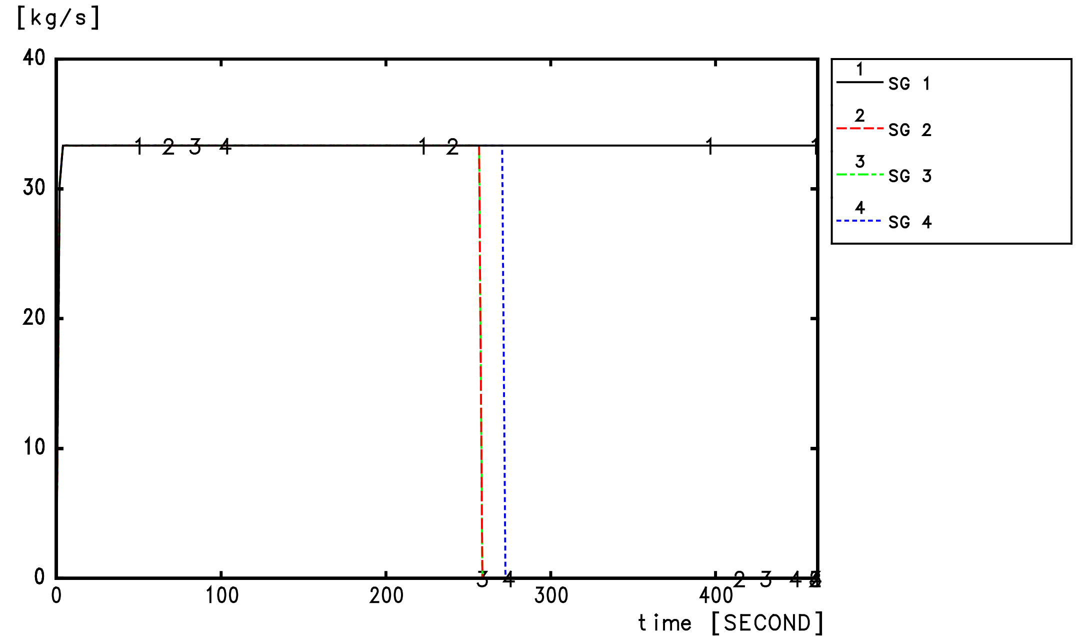


SECTION 14.5.2 - FIGURE 51

Zero Xenon Concentration Initial Conditions – Boron Concentration and ASG [EFWS] Flow Rate



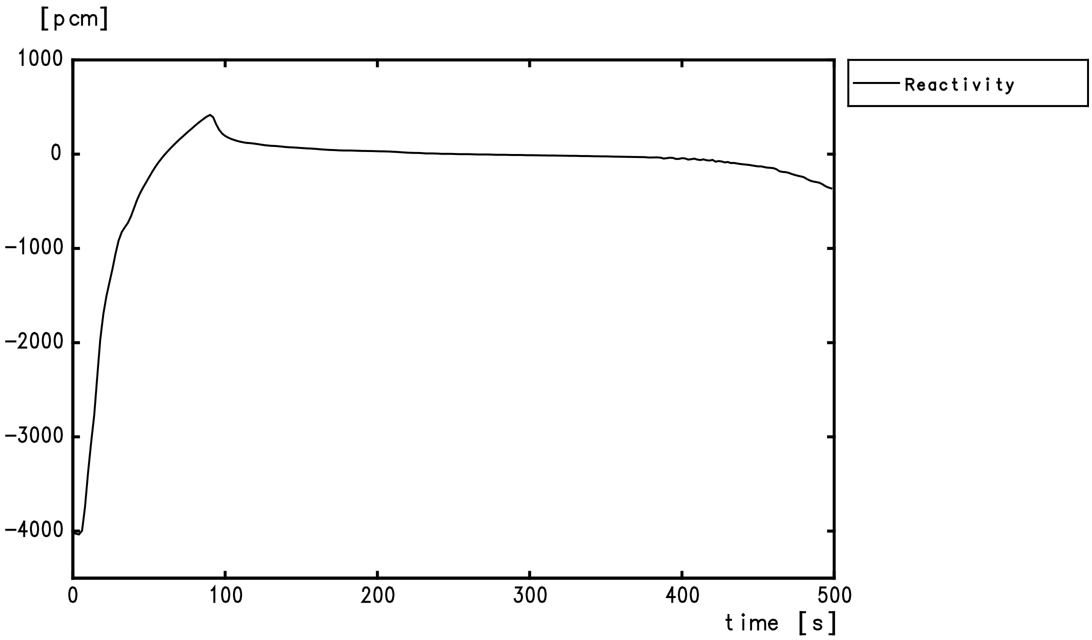
BORON CONCENTRATION AT CORE ACTIVE PART INLET



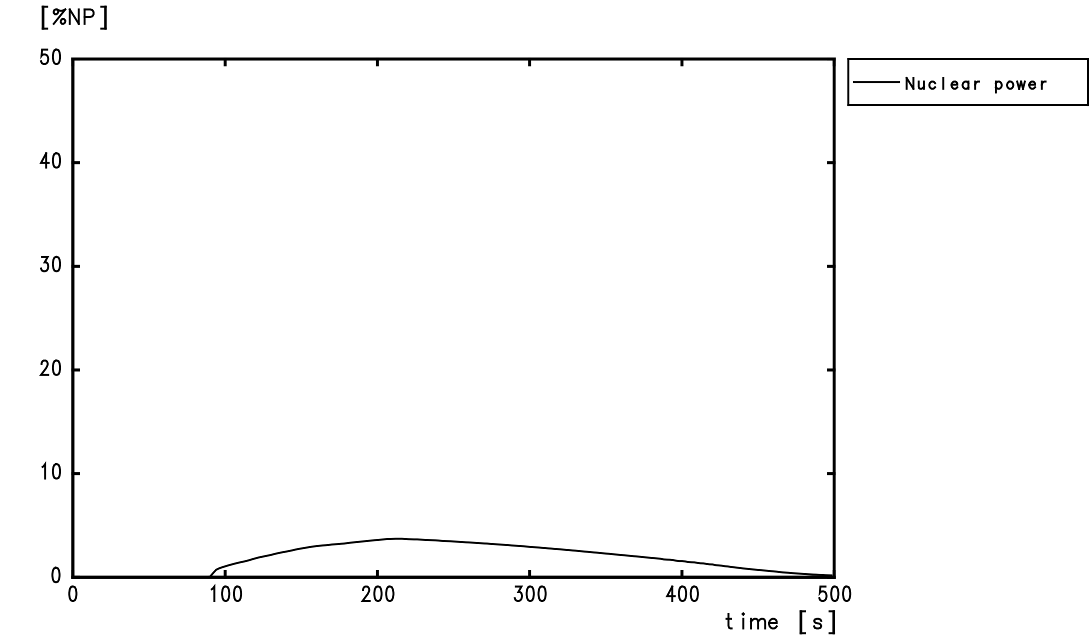
ASG FLOWRATE

SECTION 14.5.2 - FIGURE 52

RBS [EBS] Automatic Actuation – Reactivity and Core Power



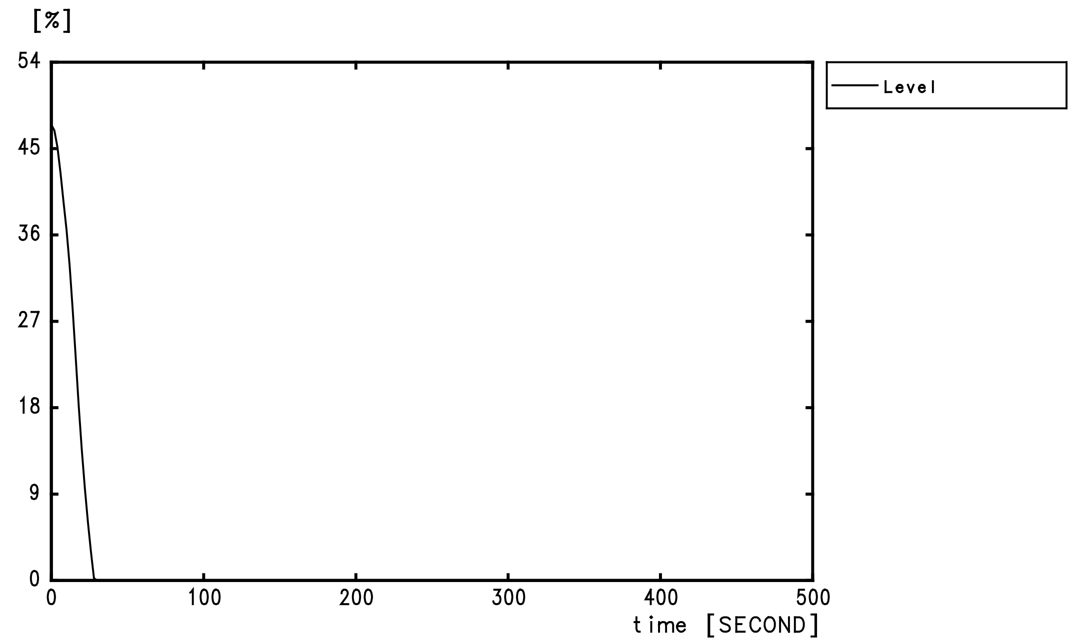
REACTIVITY



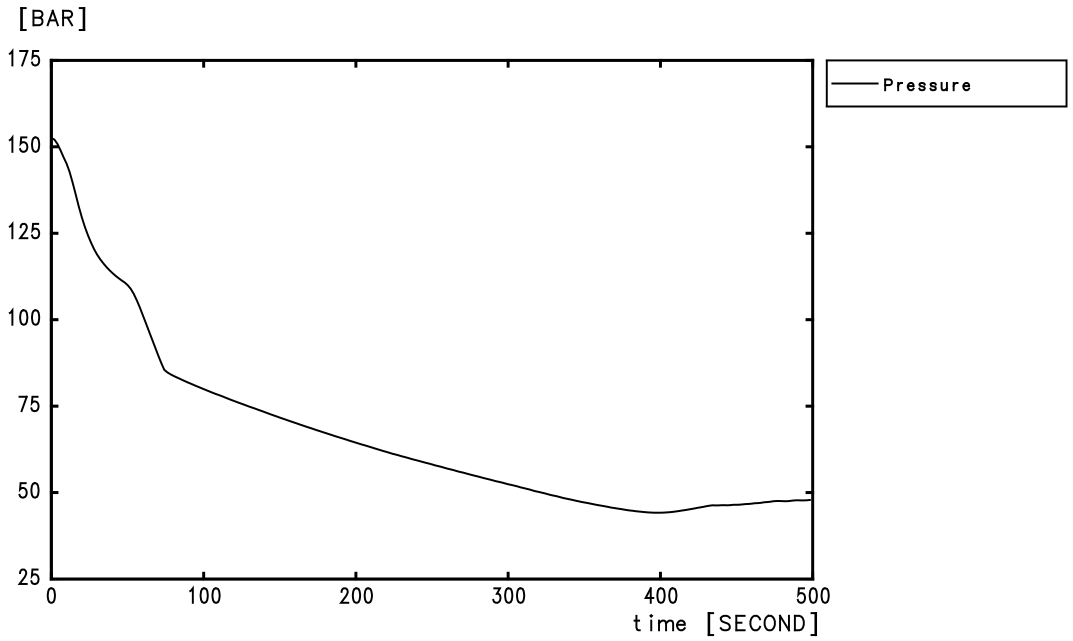
CORE POWER

SECTION 14.5.2 - FIGURE 53

RBS [EBS] Automatic Actuation – Pressuriser Level and Pressuriser Pressure



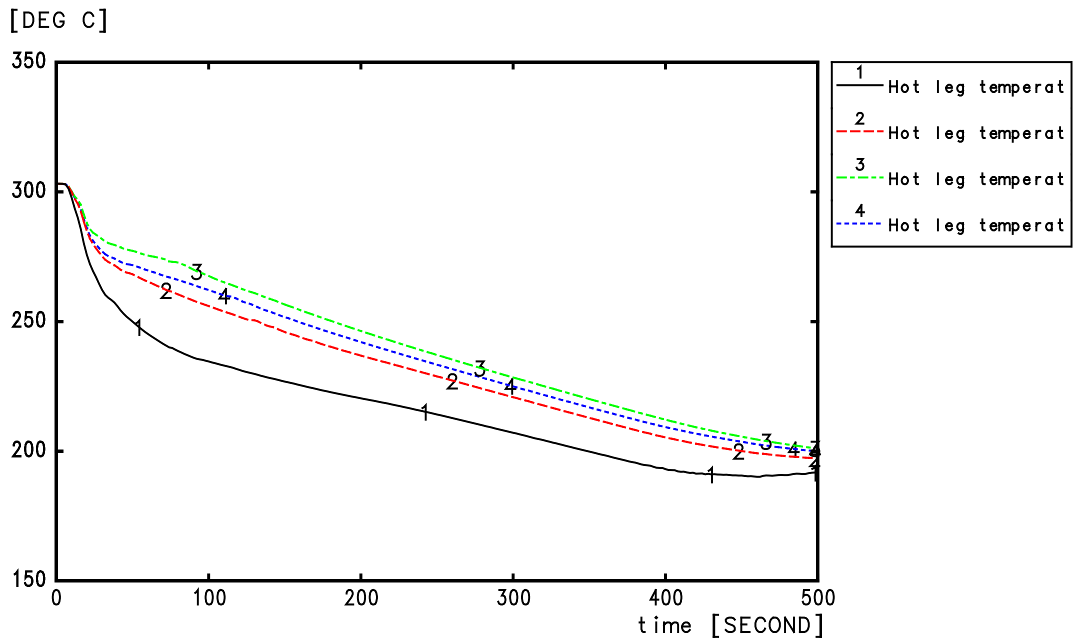
PRESSURIZER LEVEL



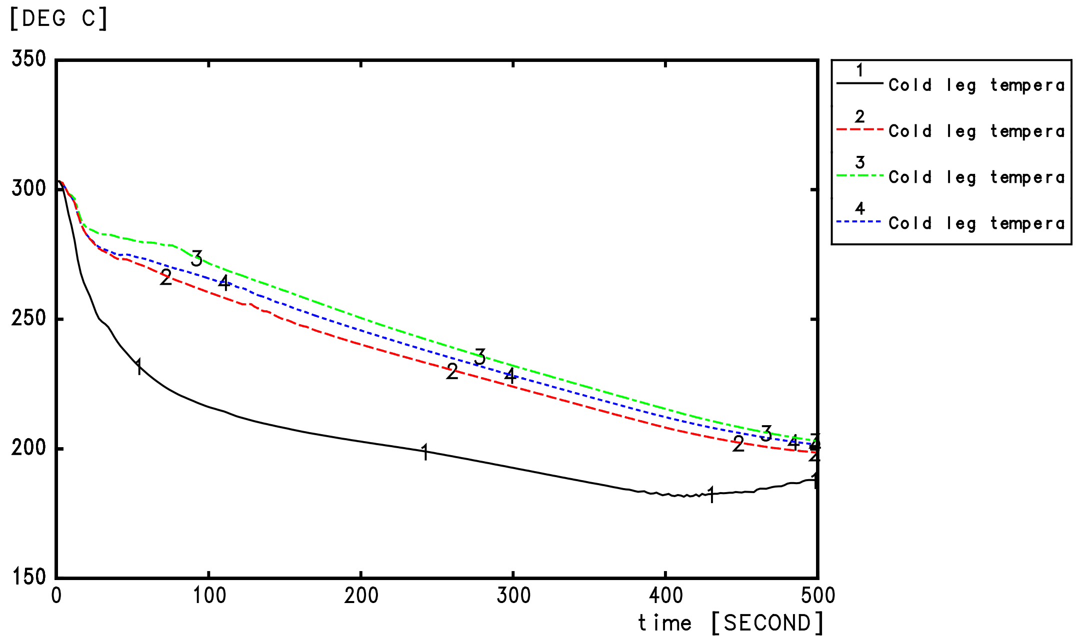
PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 54

RBS [EBS] Automatic Actuation – Hot Leg Temperature and Cold Leg Temperature



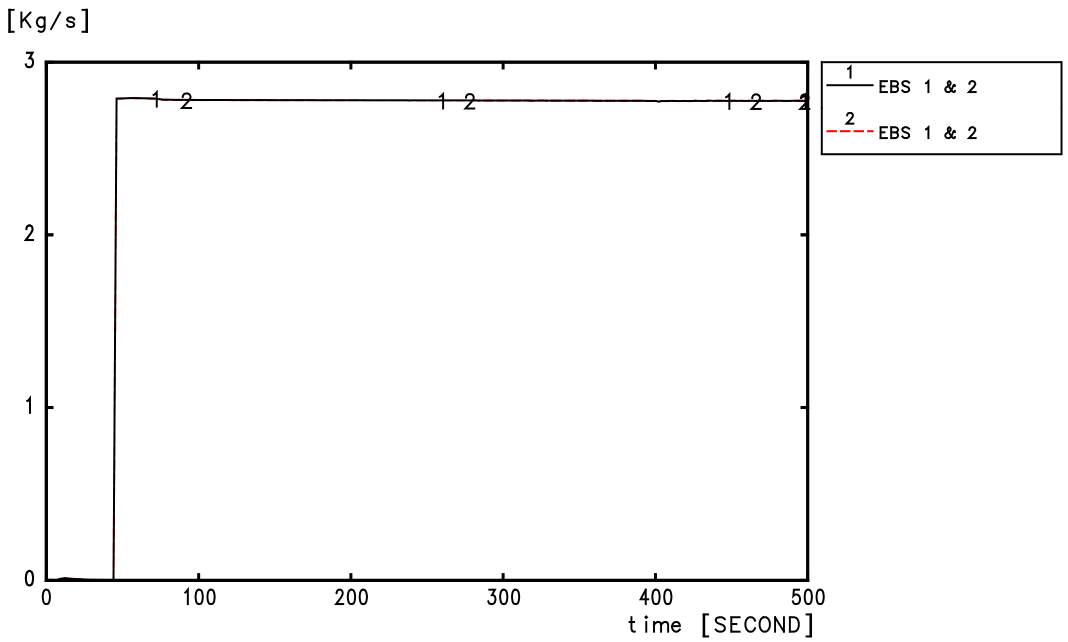
HOT LEG TEMPERATURE



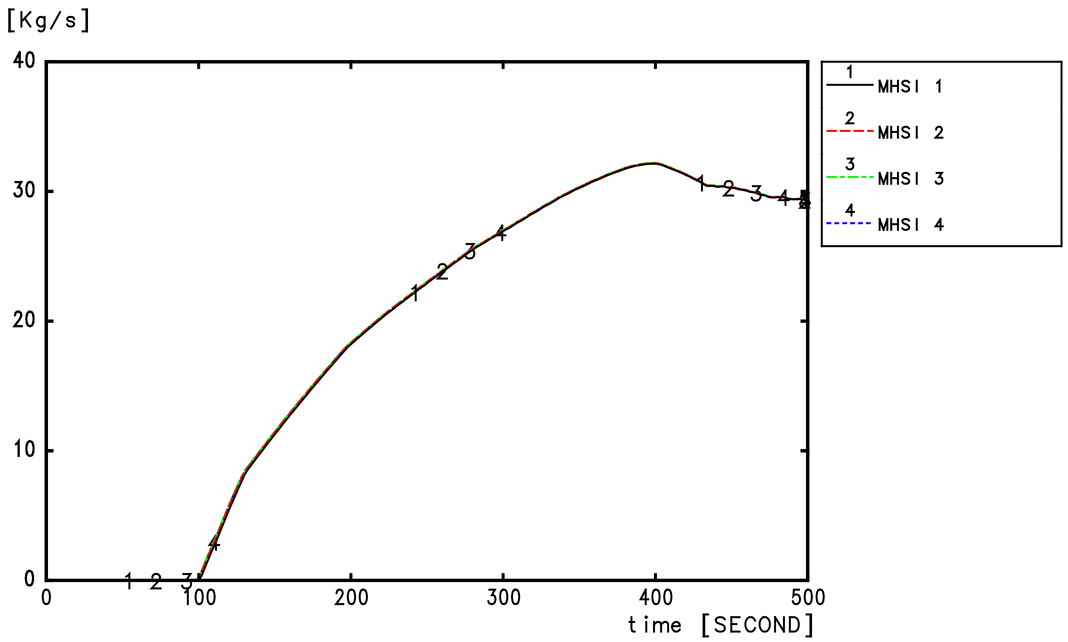
COLD LEG TEMPERATURE

SECTION 14.5.2 - FIGURE 55

RBS [EBS] Automatic Actuation – Total RBS [EBS] Flow Rate and Total MHSI Flow Rate



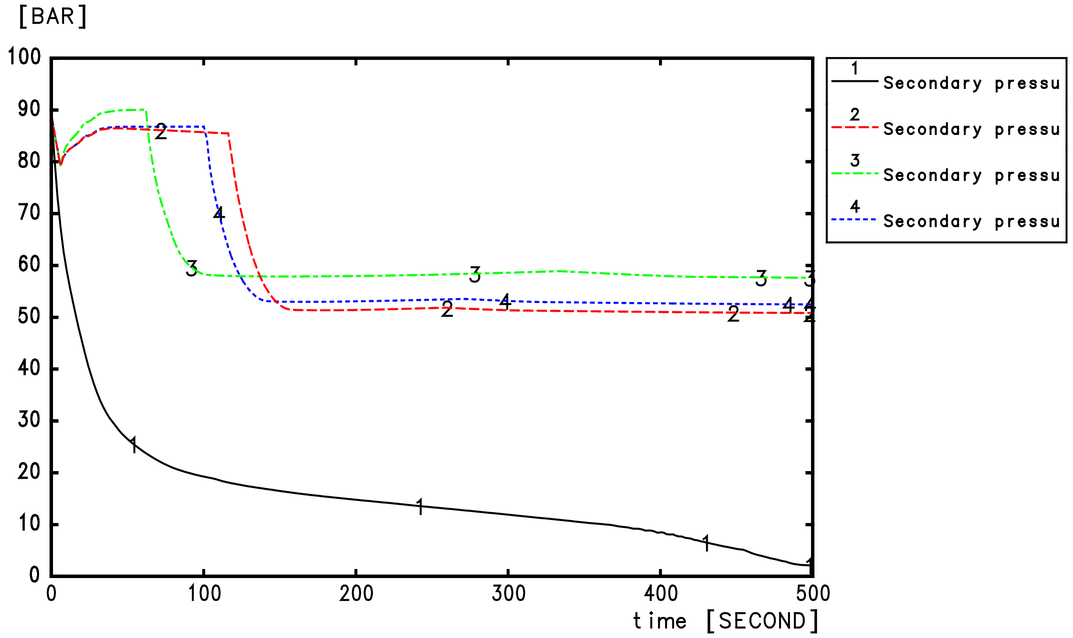
TOTAL EBS FLOW RATE



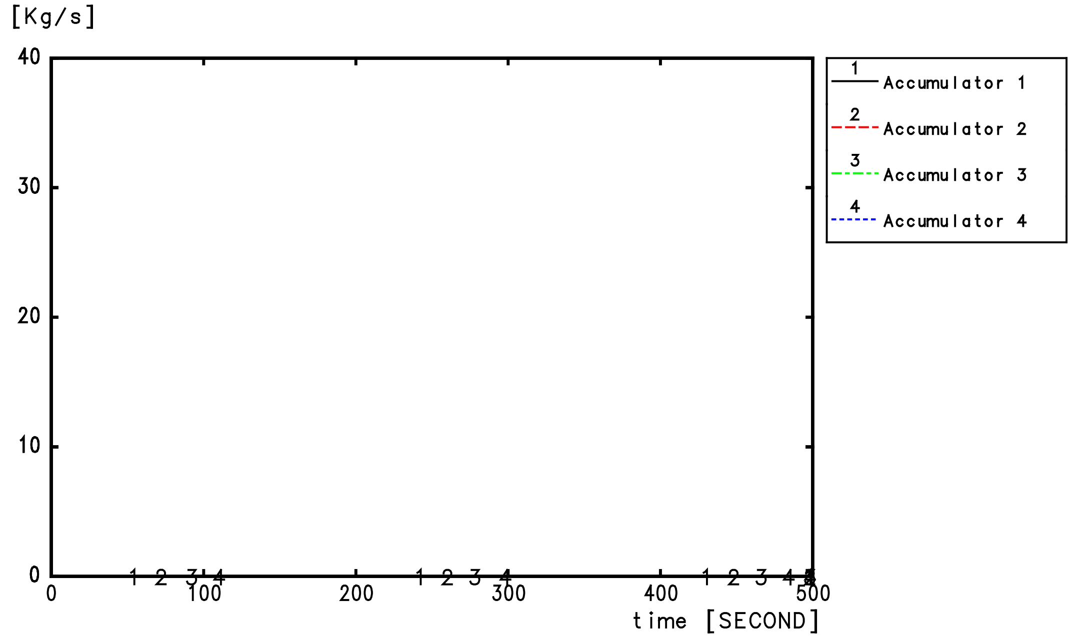
TOTAL MHSI FLOW RATE

SECTION 14.5.2 - FIGURE 56

RBS [EBS] Automatic Actuation – SG Pressure and Total Accumulators Flow Rate



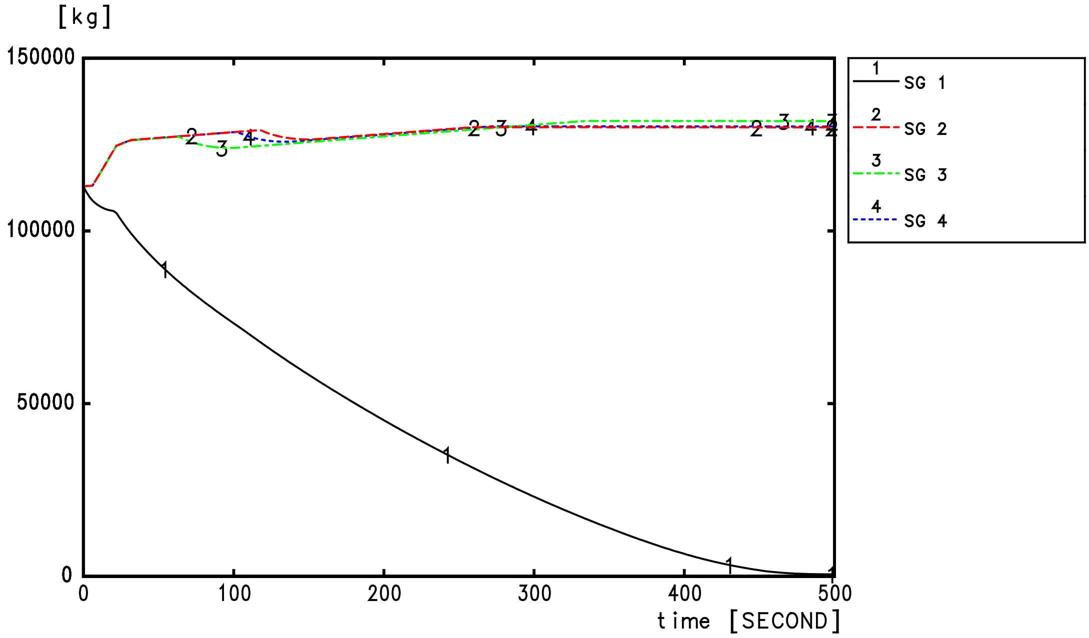
STEAM GENERATOR PRESSURE



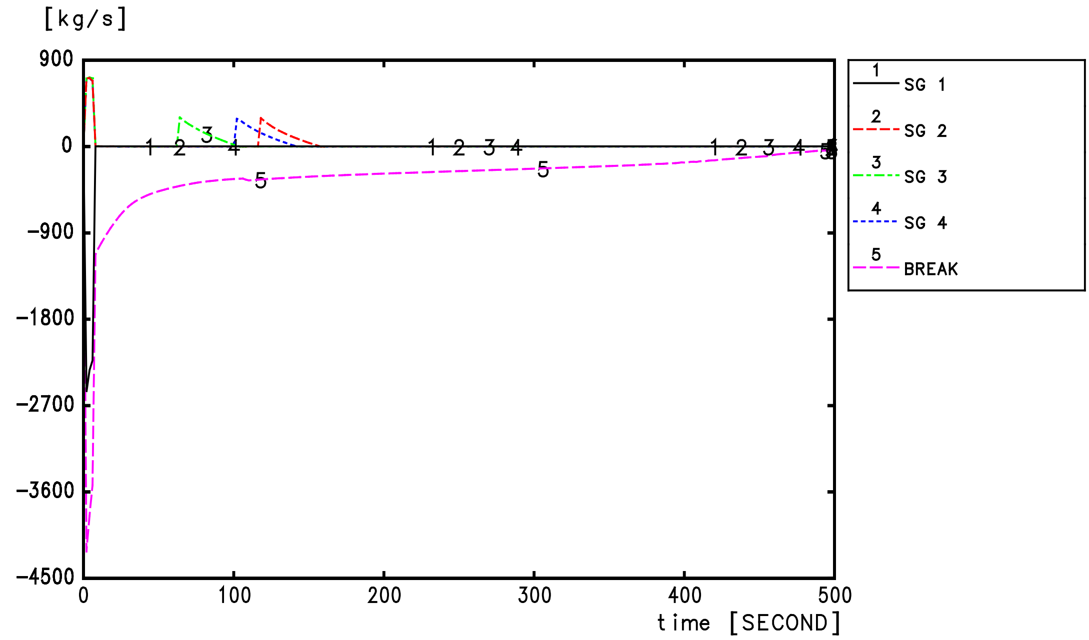
TOTAL ACCUMULATORS FLOW RATE

SECTION 14.5.2 - FIGURE 57

RBS [EBS] Automatic Actuation – SG Liquid Mass and Vapour Mass Flow Rate



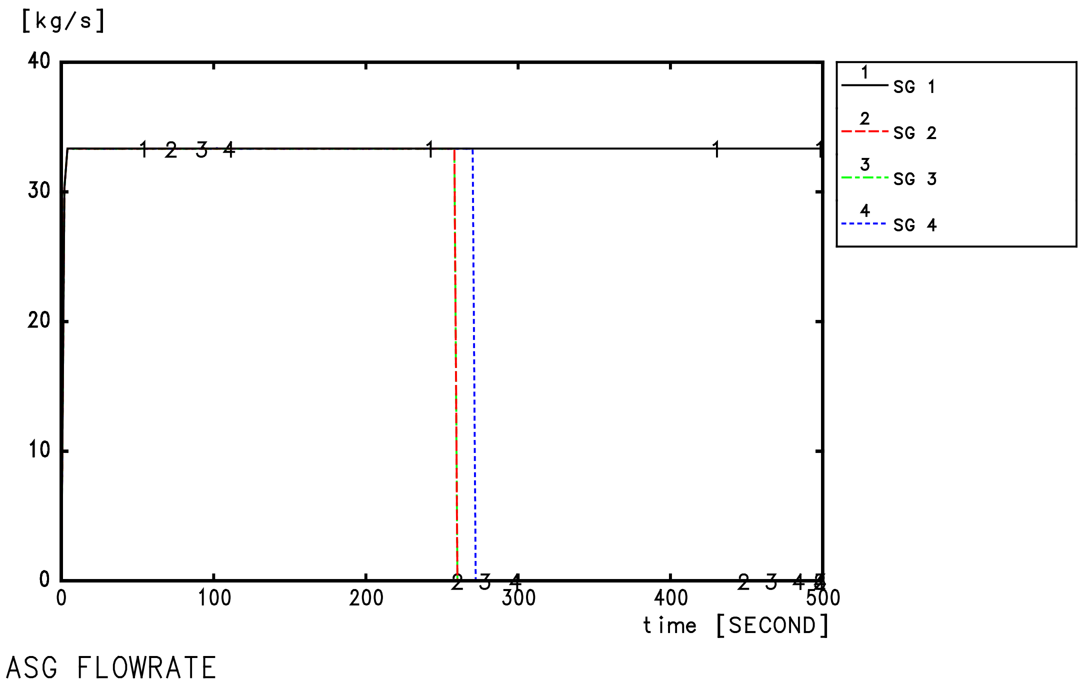
STEAM GENERATOR LIQUID MASS

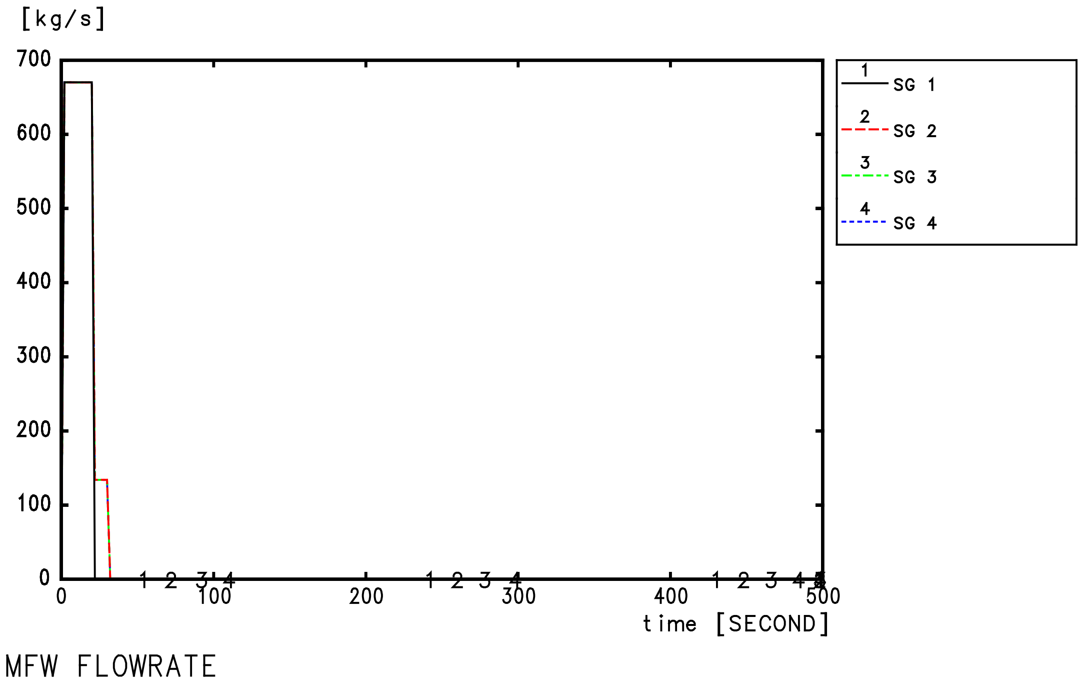


VAPOR MASS FLOWRATE

SECTION 14.5.2 - FIGURE 58

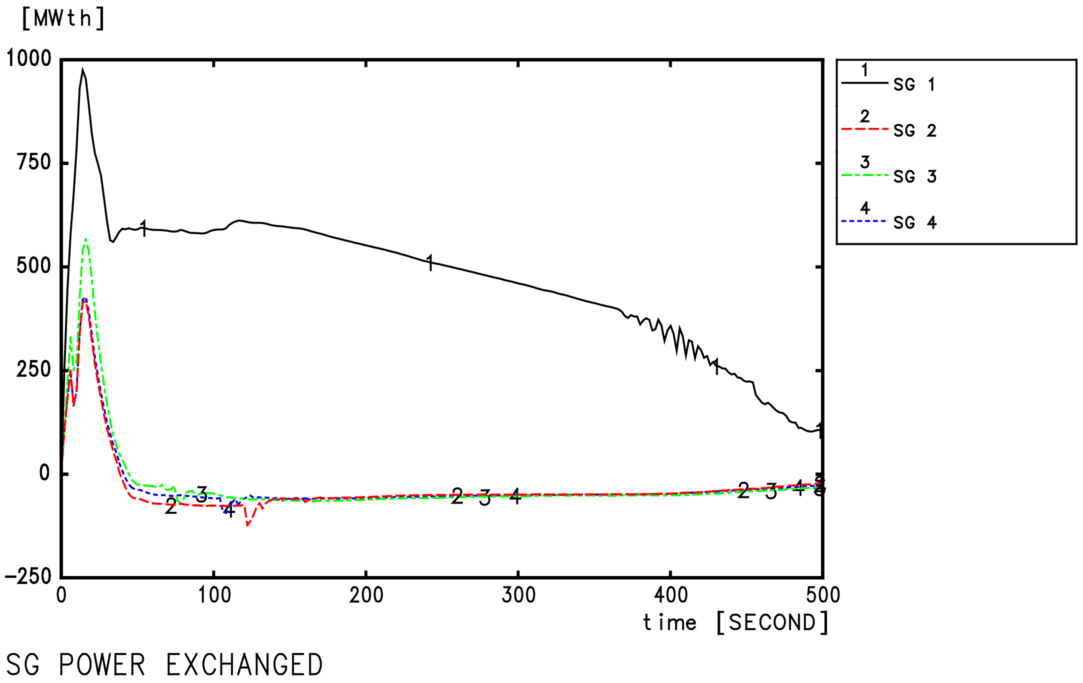
RBS [EBS] Automatic Actuation – ASG [EFWS] Flow Rate and MFW Flow Rate

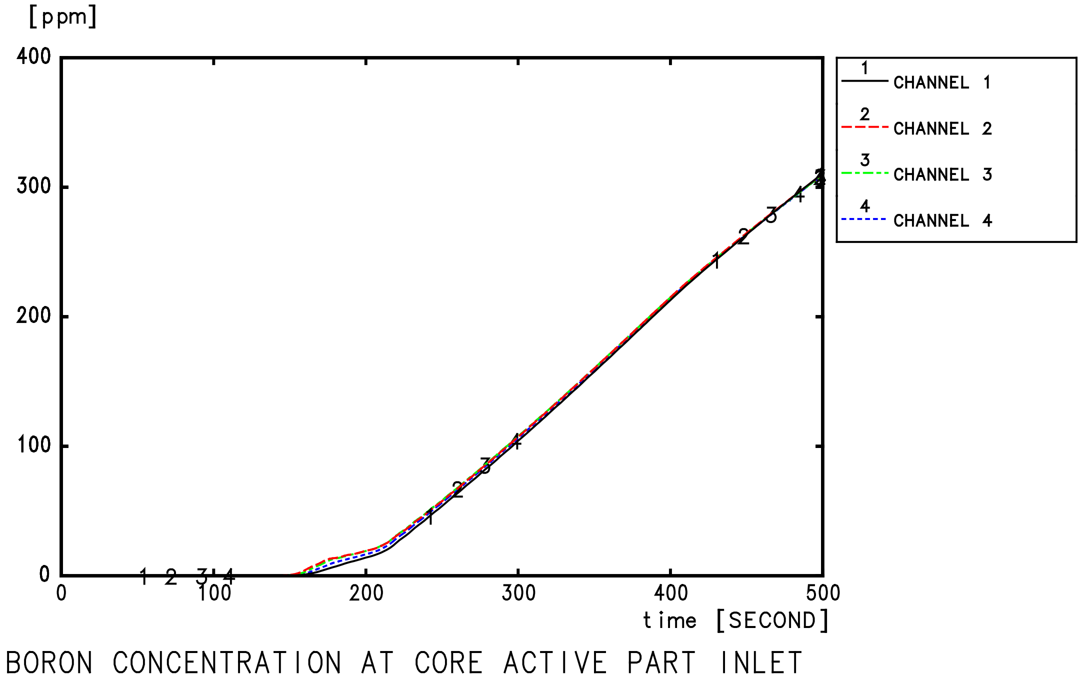




SECTION 14.5.2 - FIGURE 59

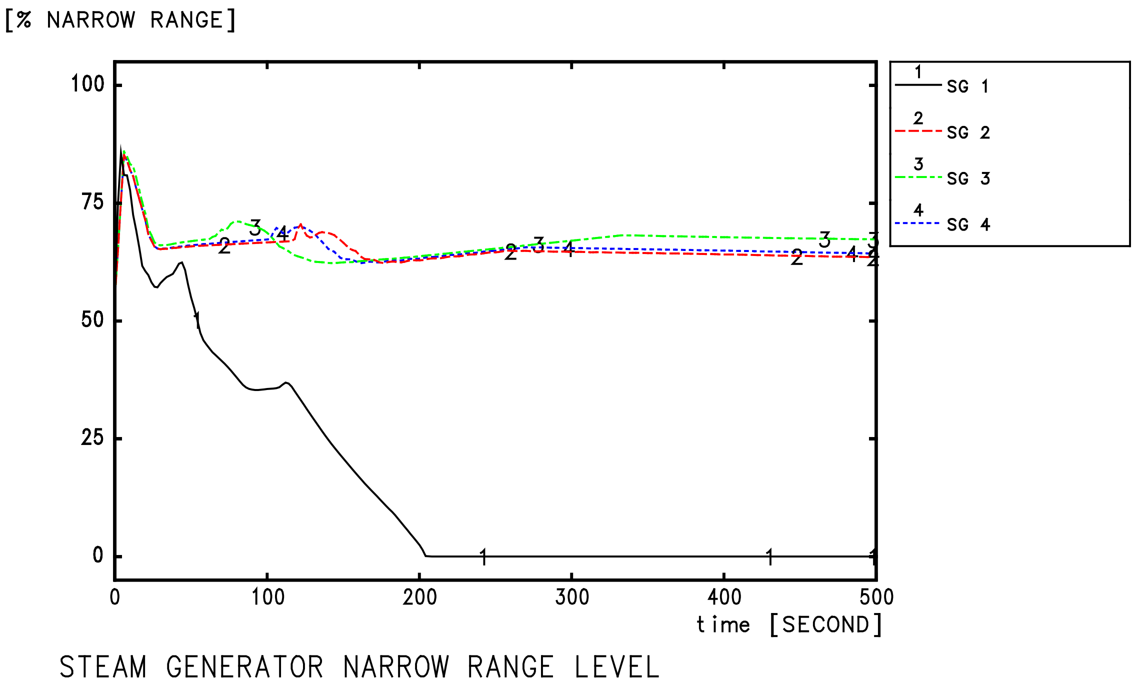
RBS [EBS] Automatic Actuation – Boron Concentration and SG Power Exchanged





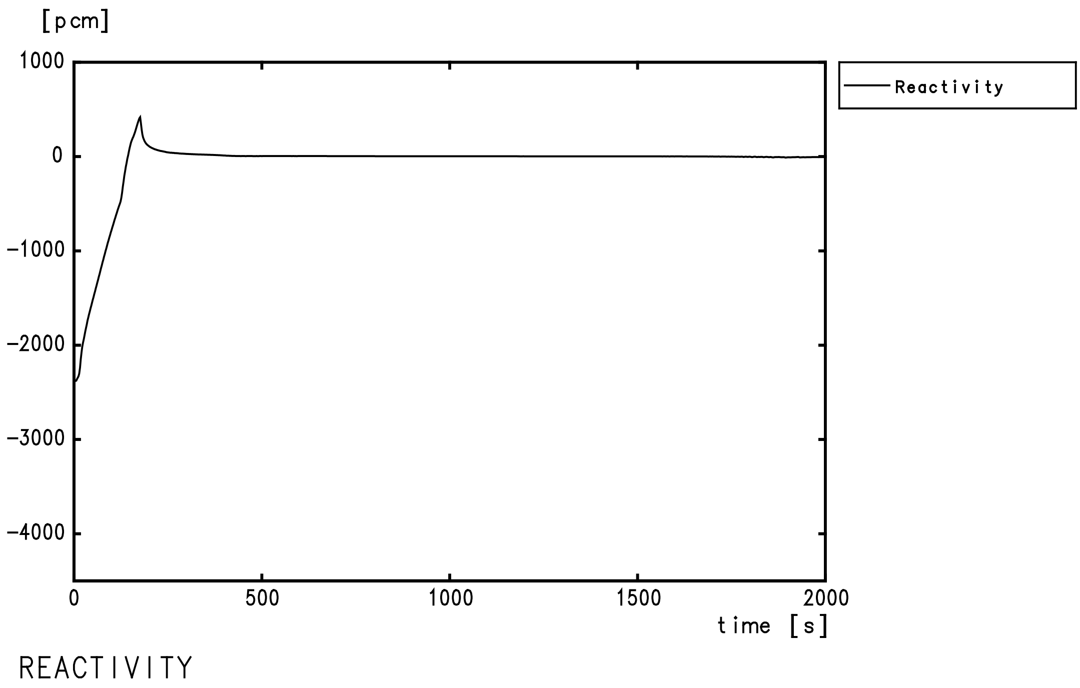
SECTION 14.5.2 - FIGURE 60

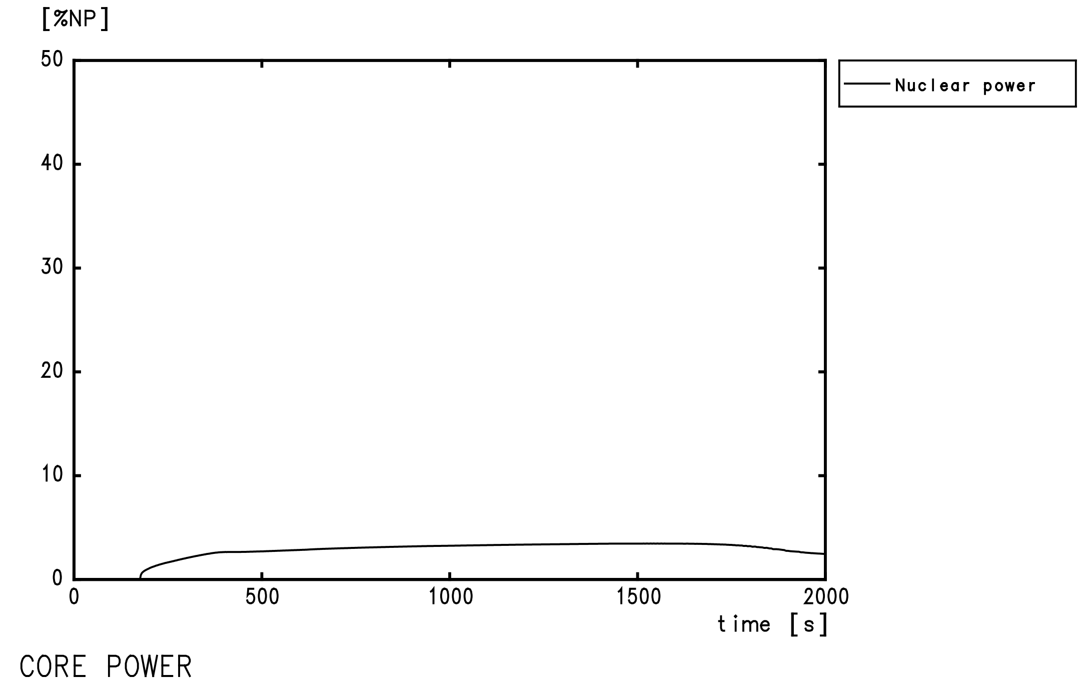
RBS [EBS] Automatic Actuation – SG Narrow Range Level



SECTION 14.5.2 - FIGURE 61

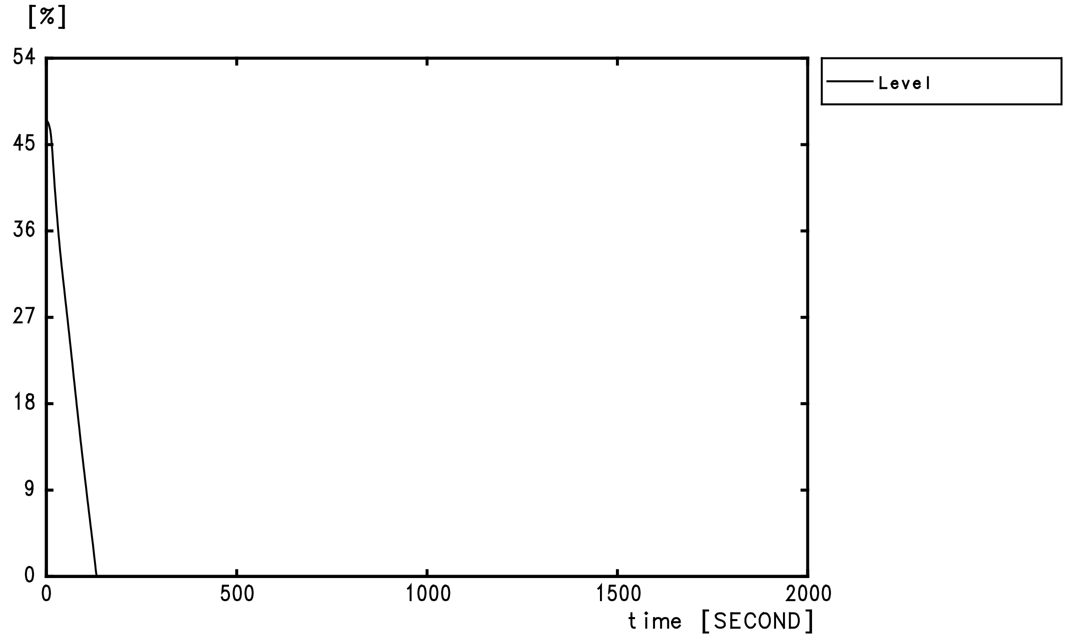
Two Stuck Rods – Reactivity and Core Power



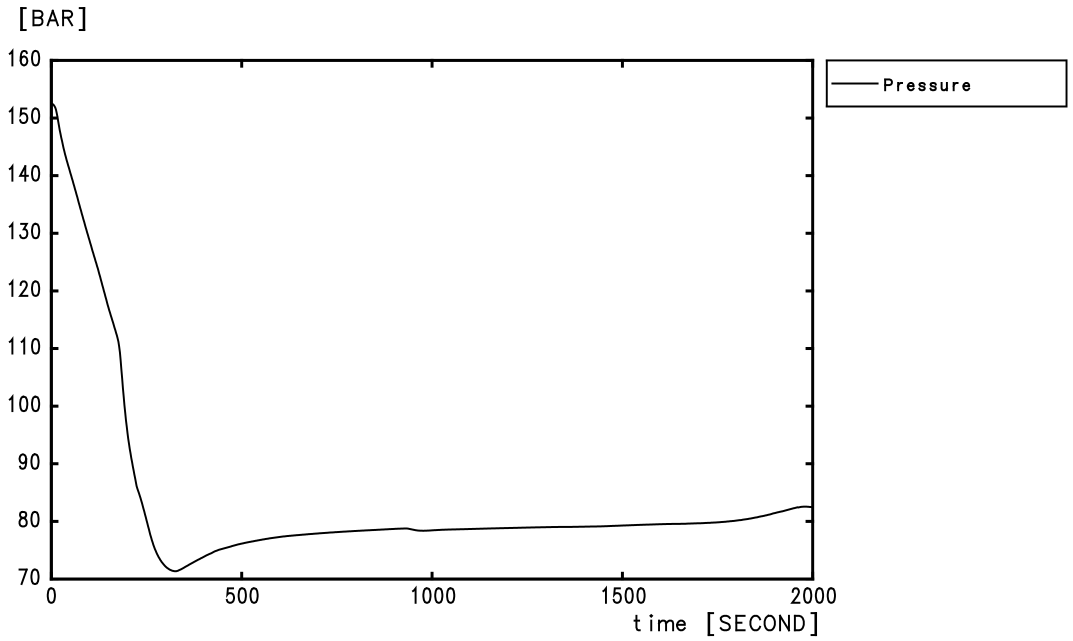


SECTION 14.5.2 - FIGURE 62

Two Stuck Rods – Pressuriser Level and Pressuriser Pressure



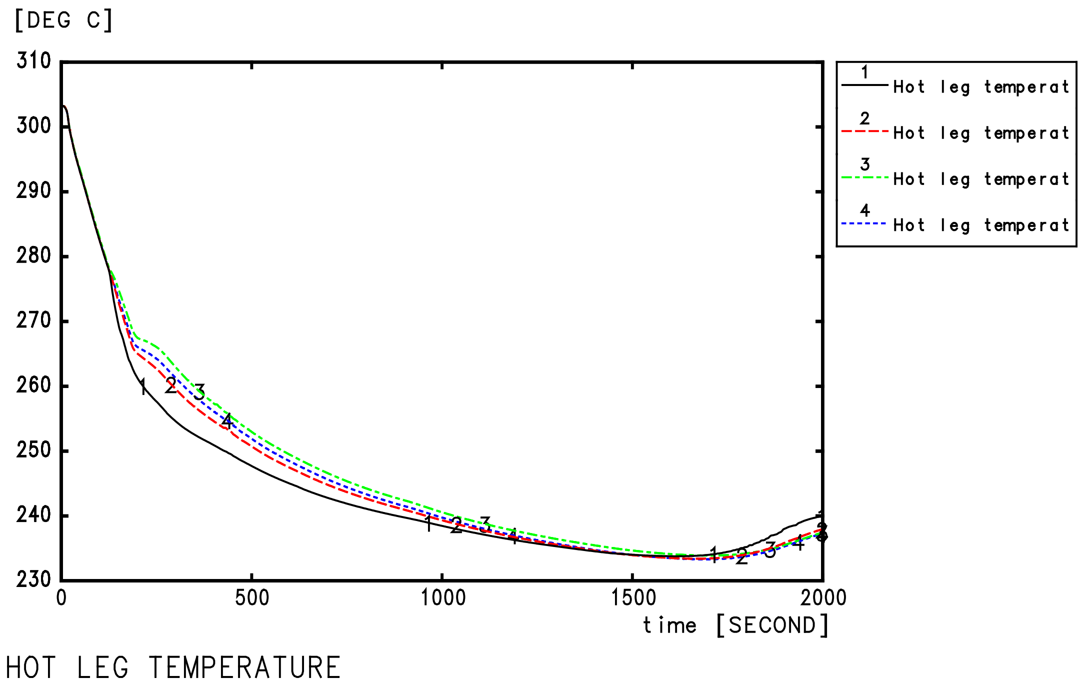
PRESSURIZER LEVEL

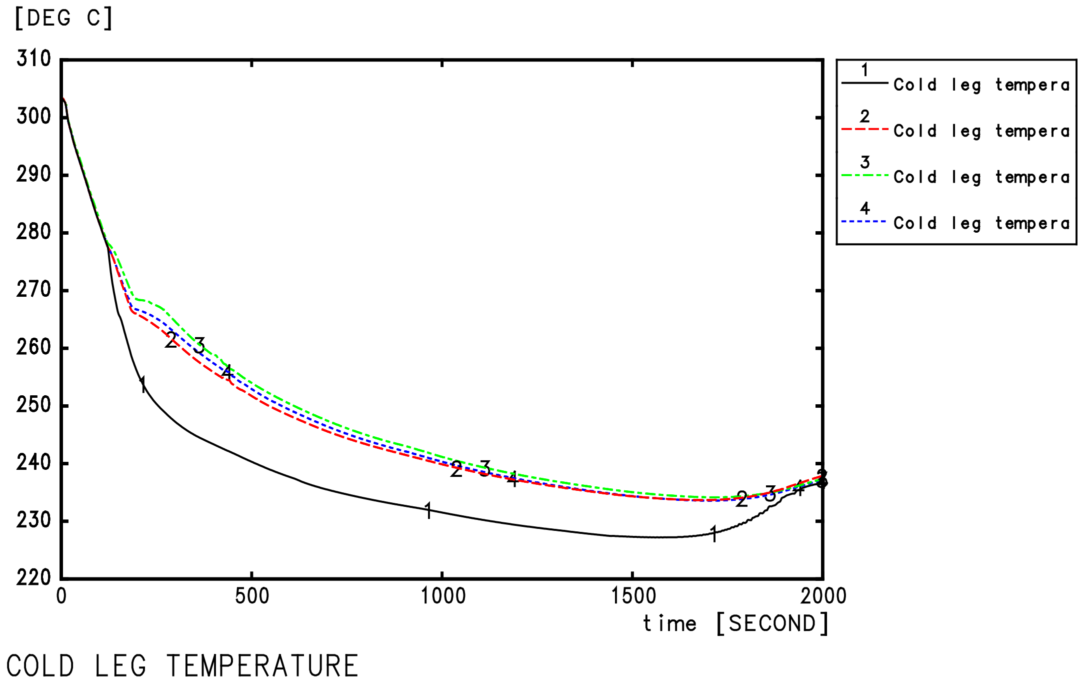


PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 63

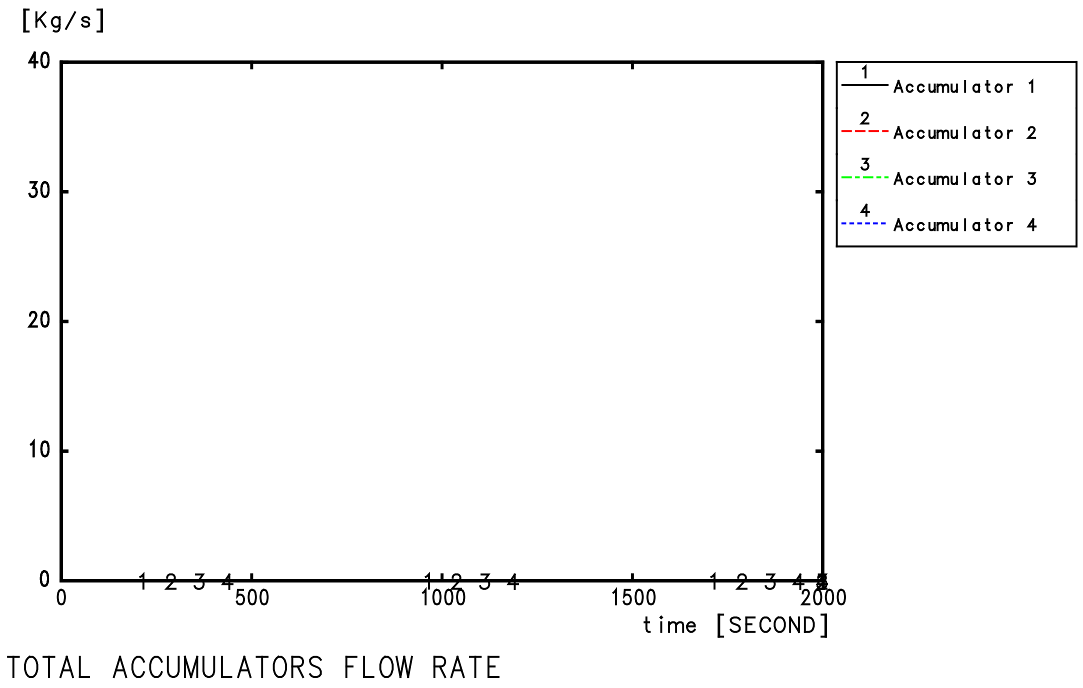
Two Stuck Rods – Hot Leg Temperature and Cold Leg Temperature

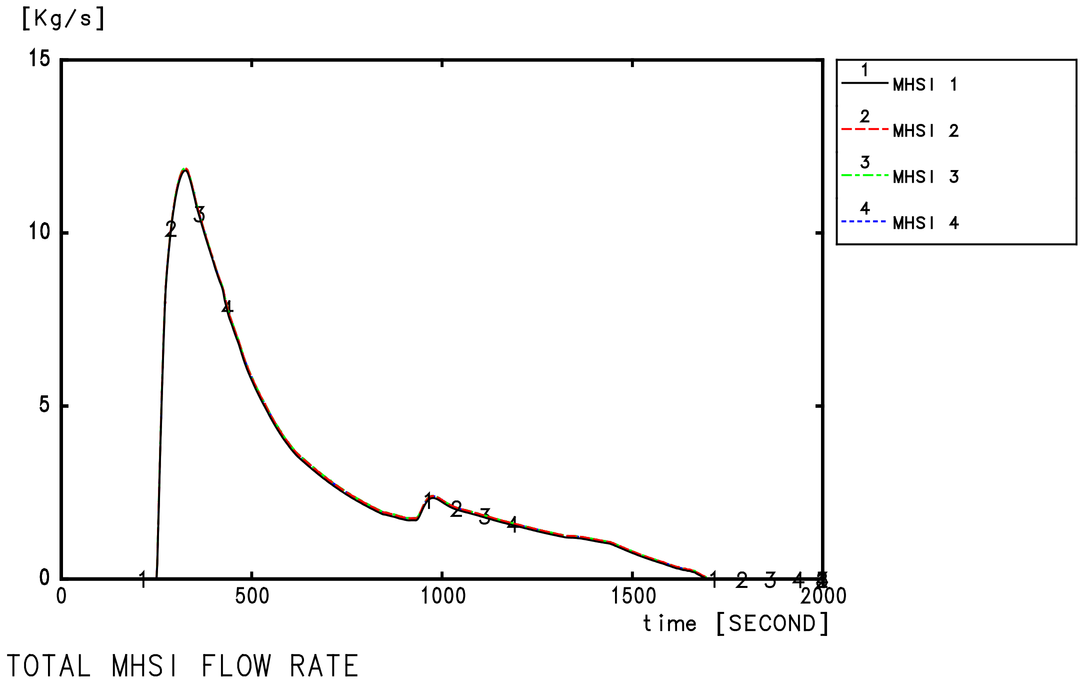




SECTION 14.5.2 - FIGURE 64

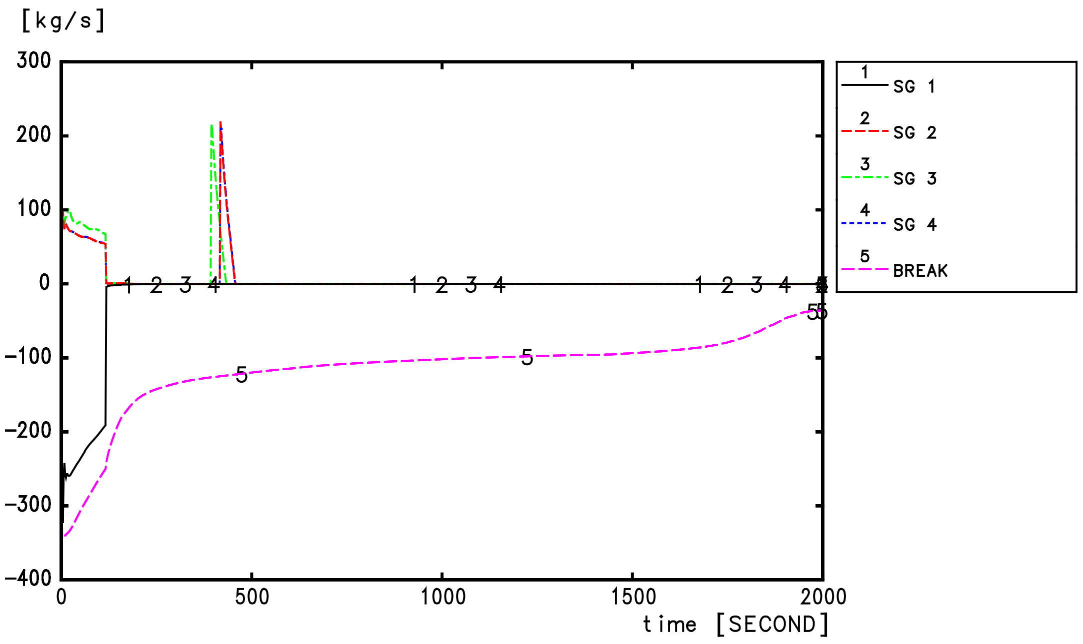
Two Stuck Rods – Total Accumulators Flow Rate and Total MHSI Flow Rate



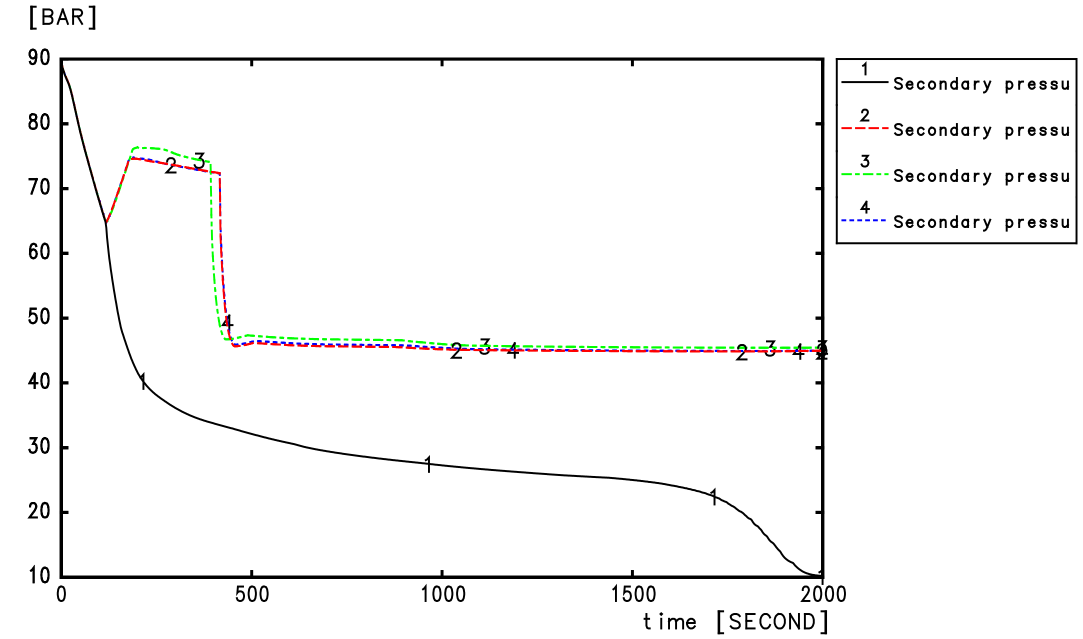


SECTION 14.5.2 - FIGURE 65

Two Stuck Rods – Vapour Mass Flow Rate and SG Pressure



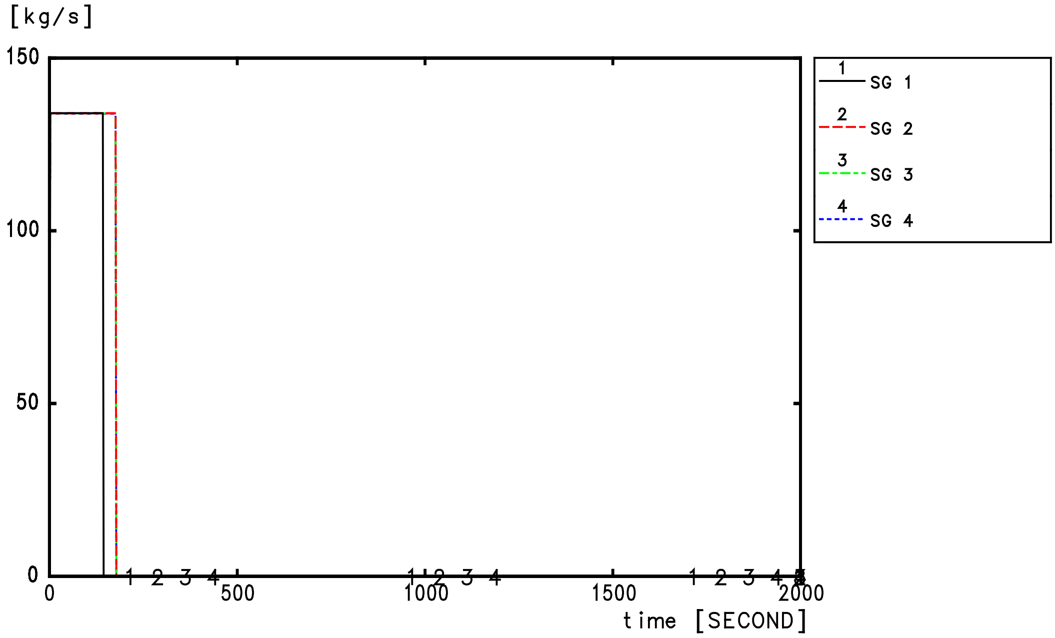
VAPOR MASS FLOWRATE



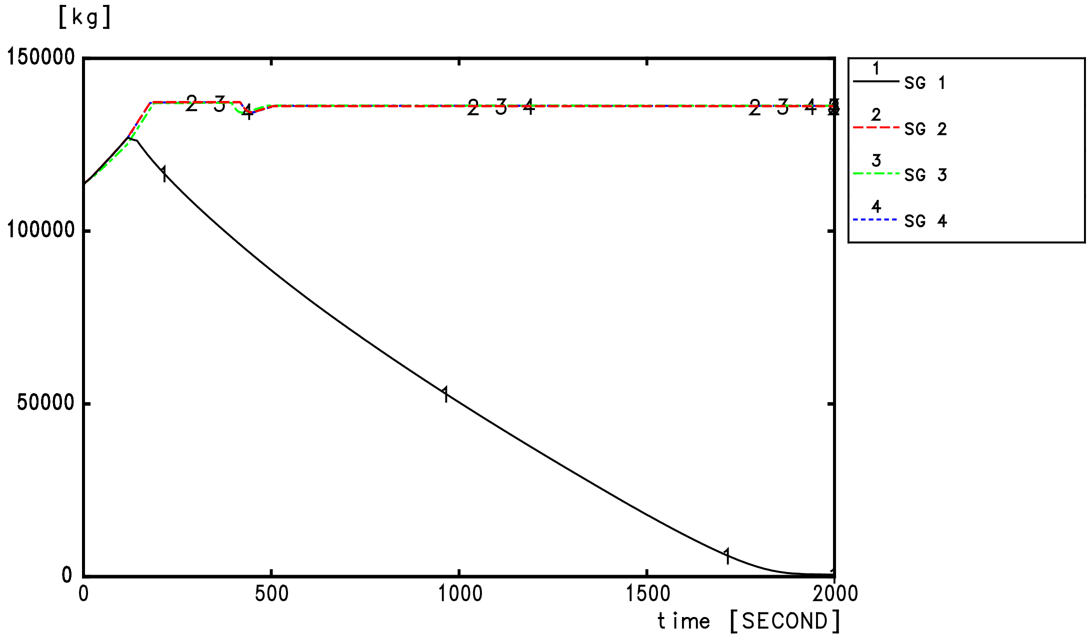
STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 66

Two Stuck Rods – Main Feedwater Flow Rate and SG Liquid



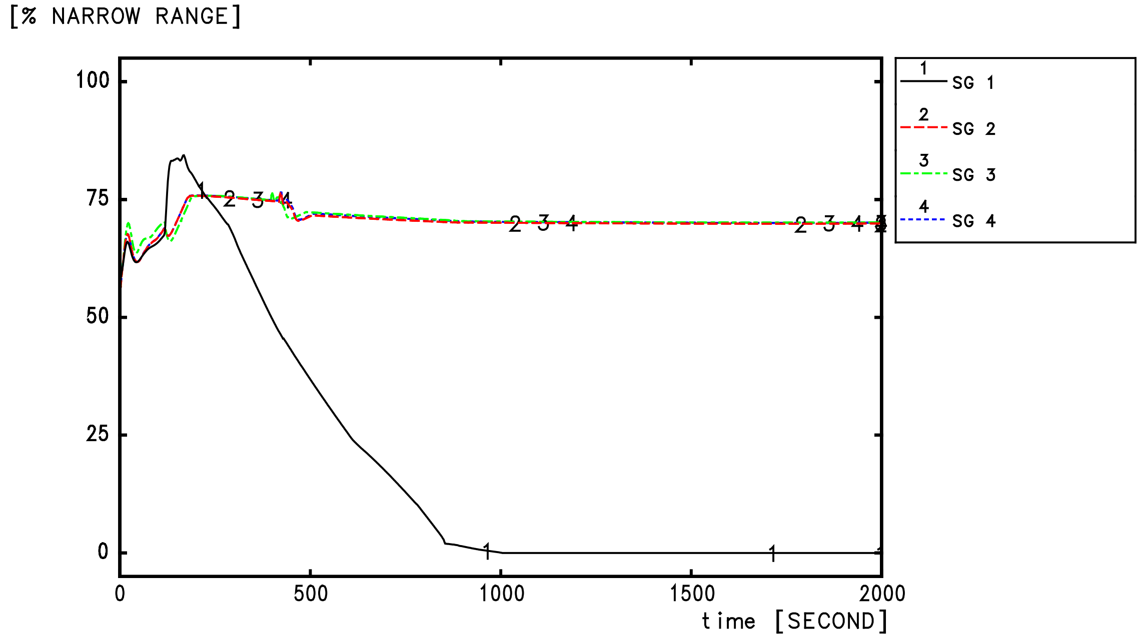
MFW FLOWRATE



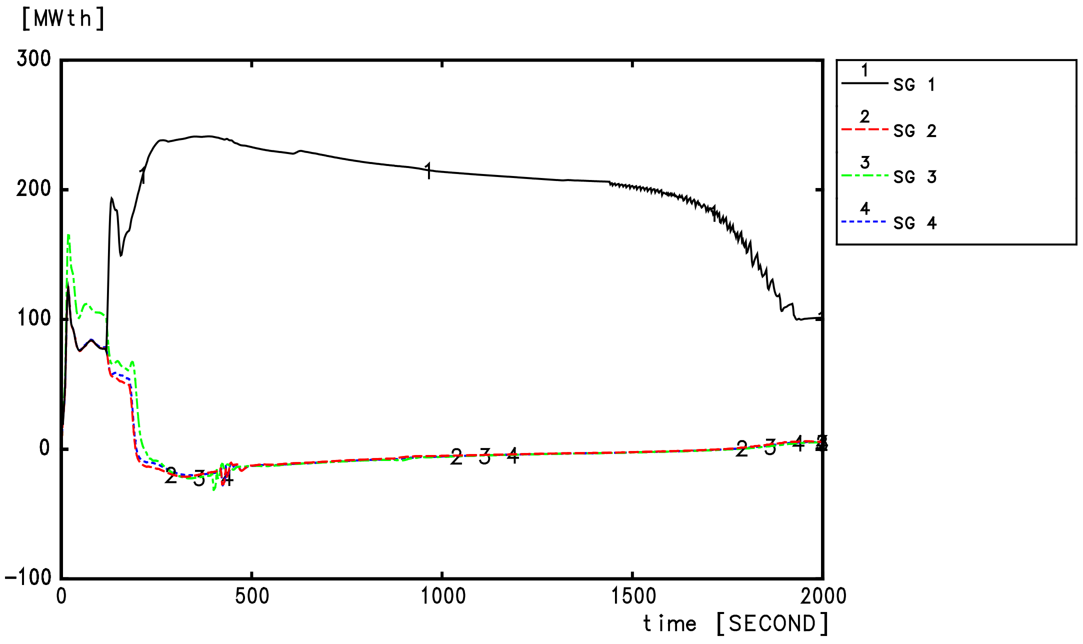
STEAM GENERATOR LIQUID MASS

SECTION 14.5.2 - FIGURE 67

Two Stuck Rods – SG NR Level and SG Power Exchanged



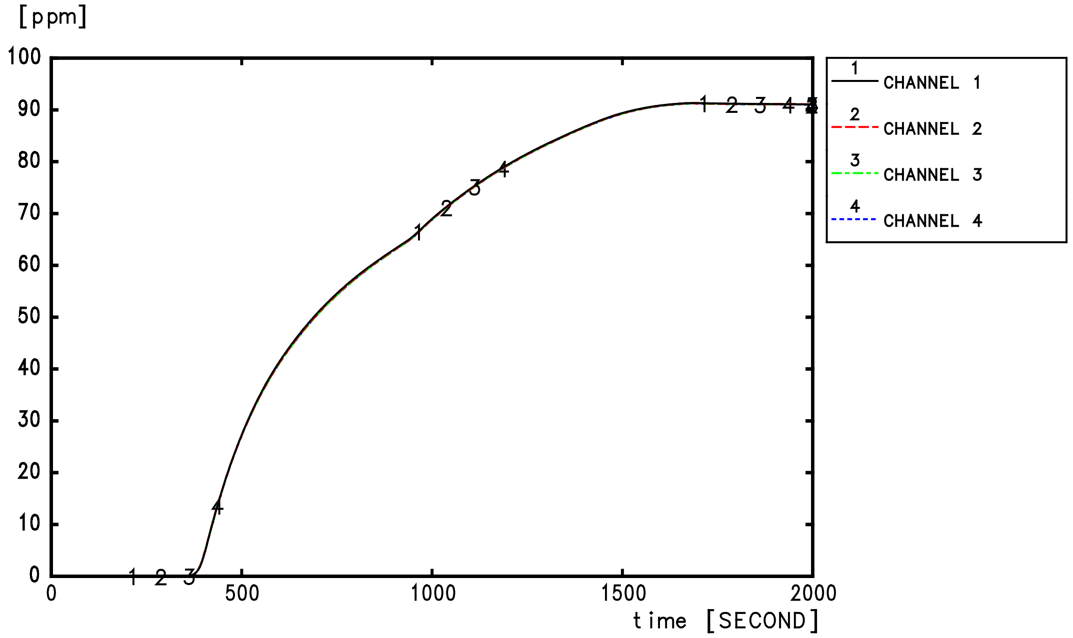
STEAM GENERATOR NARROW RANGE LEVEL



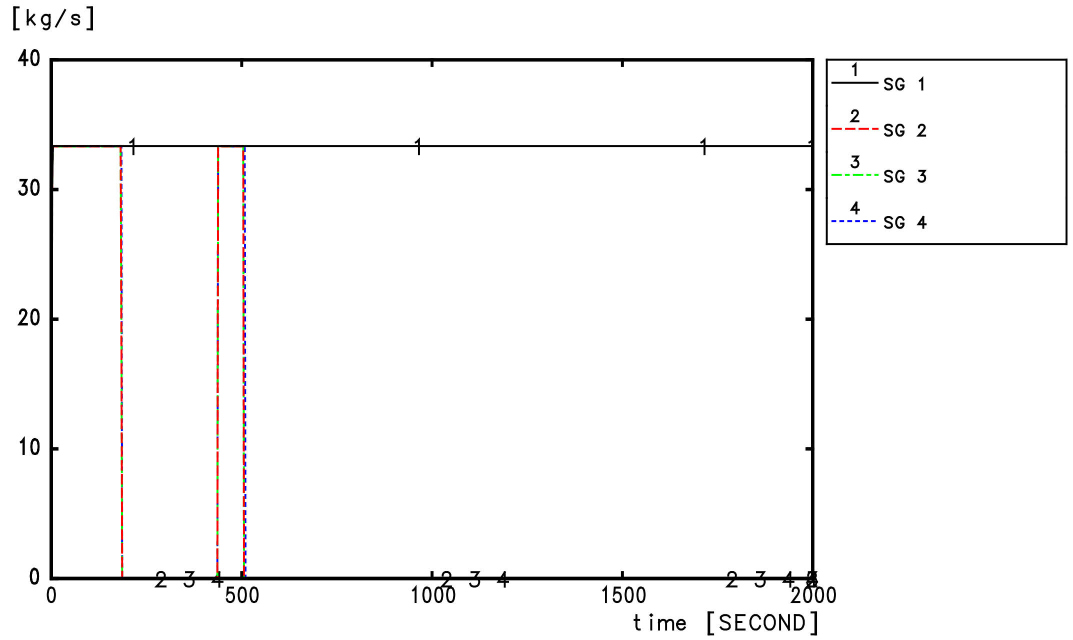
SG POWER EXCHANGED

SECTION 14.5.2 - FIGURE 68

Two Stuck Rods – Boron Concentration and ASG [EFWS] Flow Rate



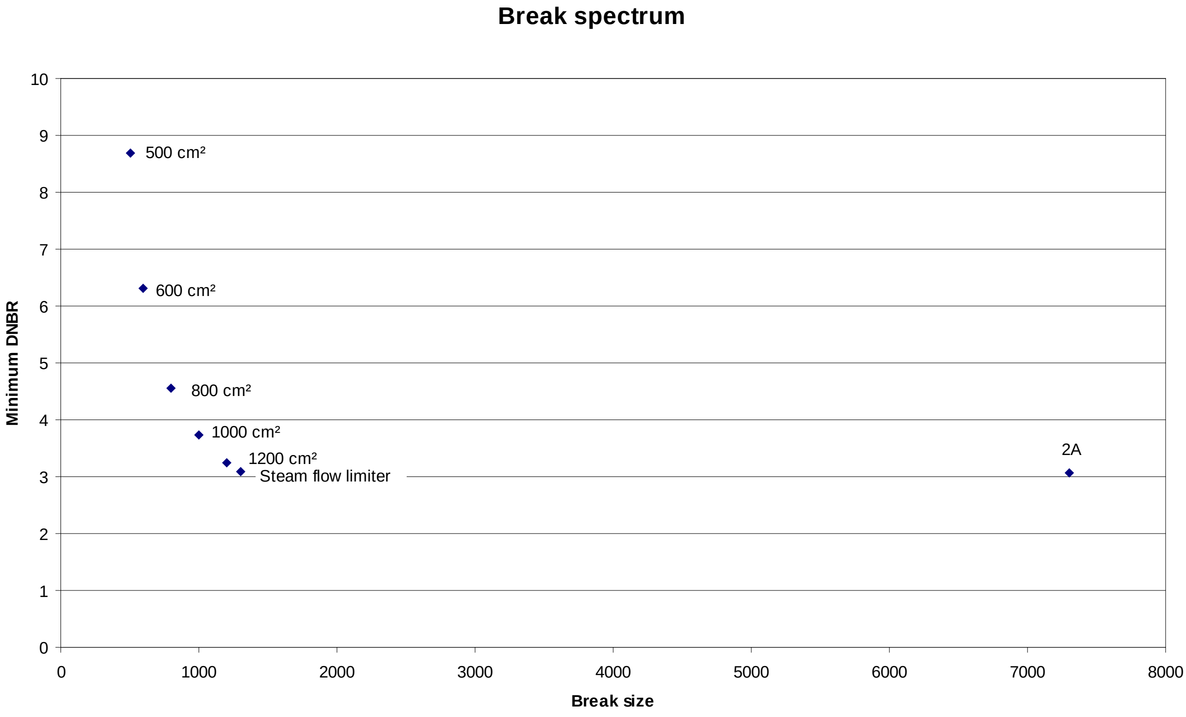
BORON CONCENTRATION AT CORE ACTIVE PART INLET



ASG FLOWRATE

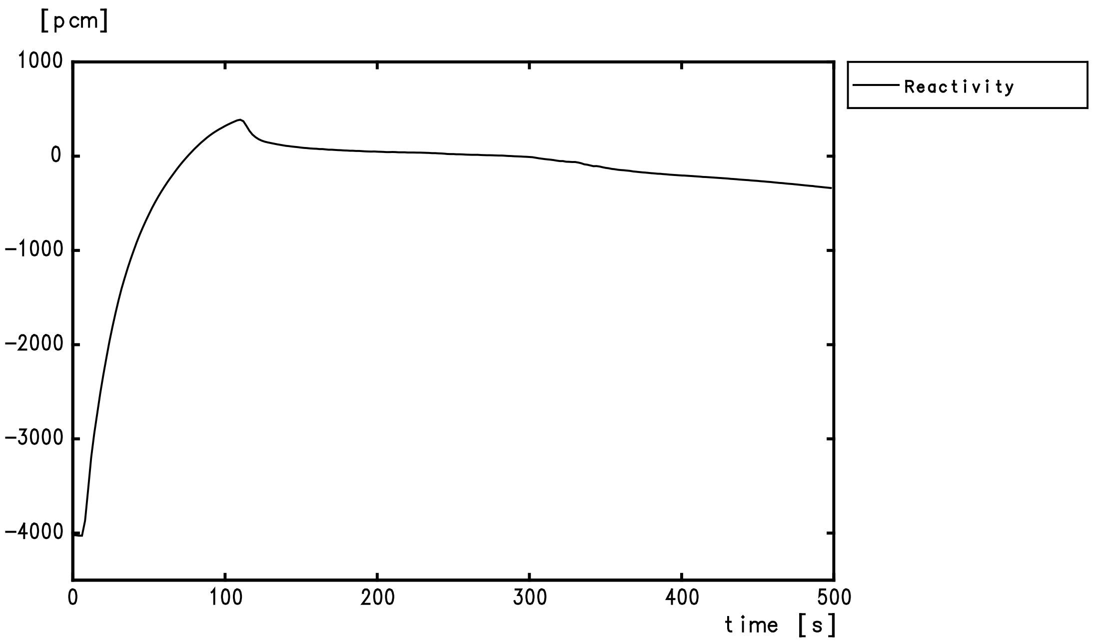
SECTION 14.5.2 - FIGURE 69

Break Spectrum at Hot Shutdown - Minimum DNBR as a Function of Steam Line Break Size

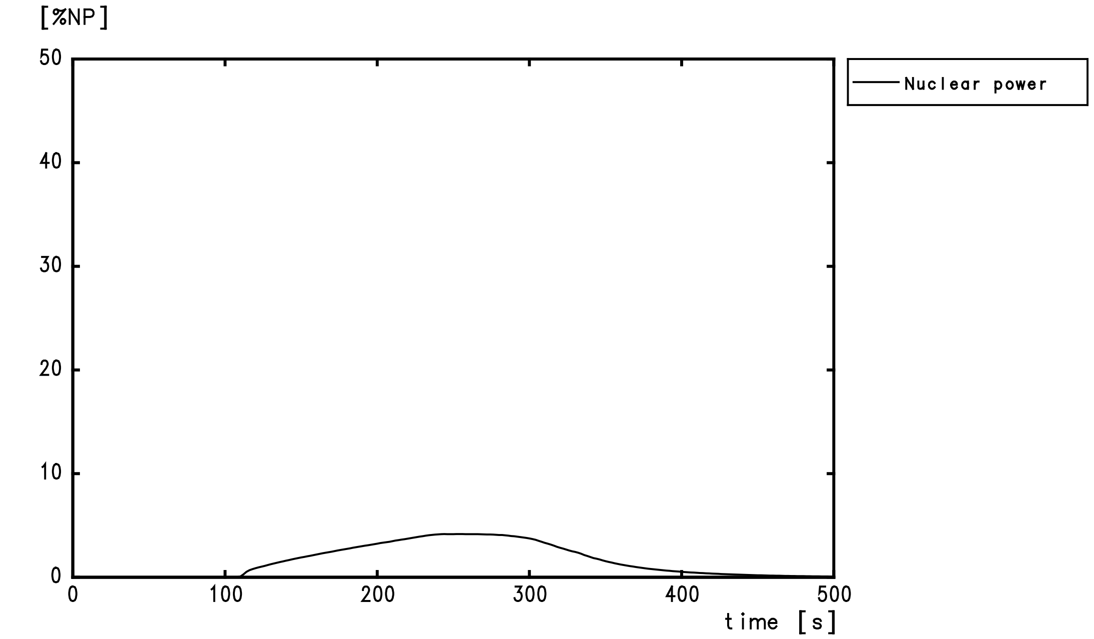


SECTION 14.5.2 - FIGURE 70

Break Spectrum – Double-Ended Guillotine Break – Reactivity and Core Power



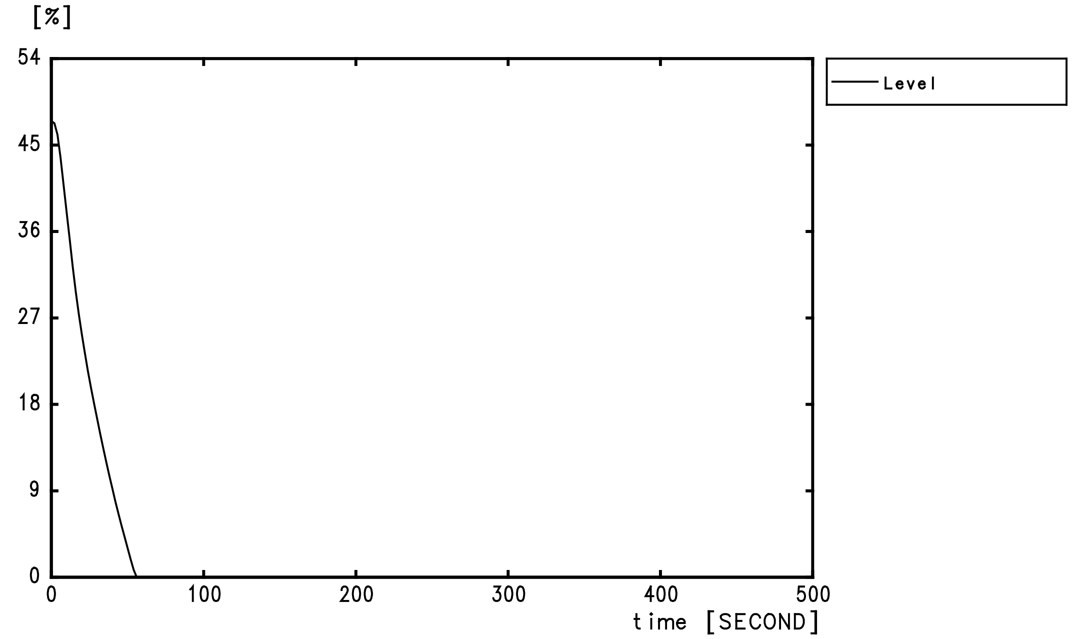
REACTIVITY



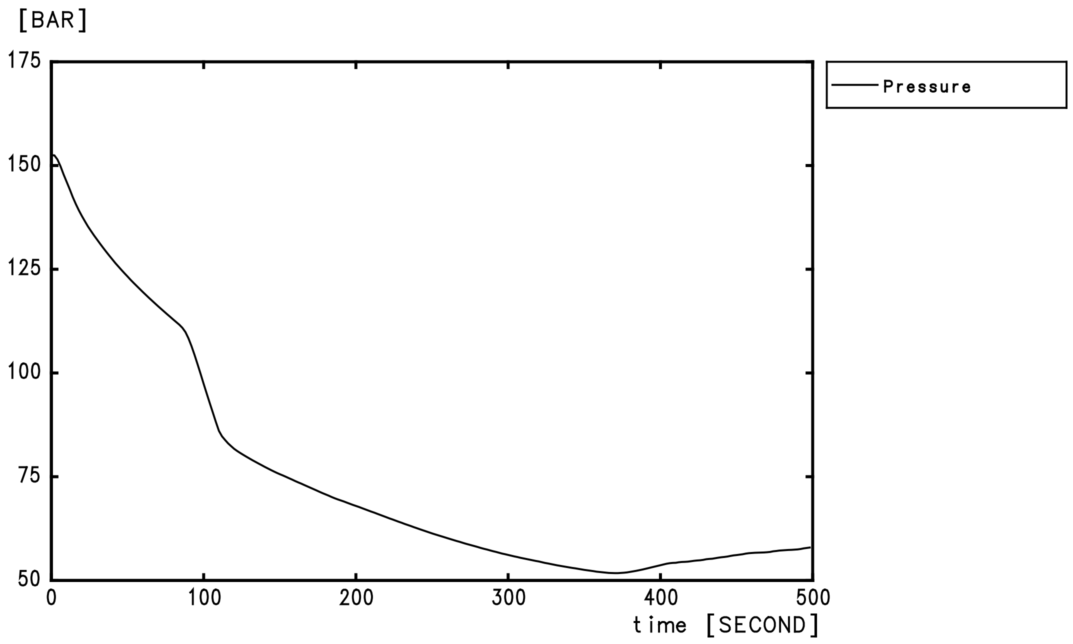
CORE POWER

SECTION 14.5.2 - FIGURE 71

Break Spectrum –Double-Ended Guillotine Break – Pressuriser Level and Pressuriser Pressure



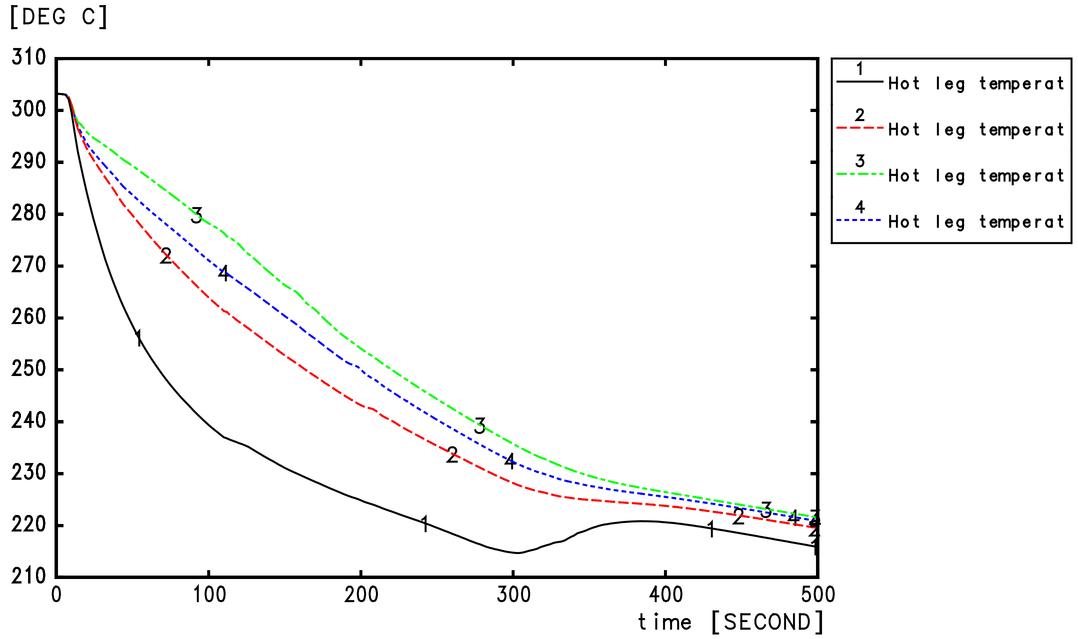
PRESSURIZER LEVEL



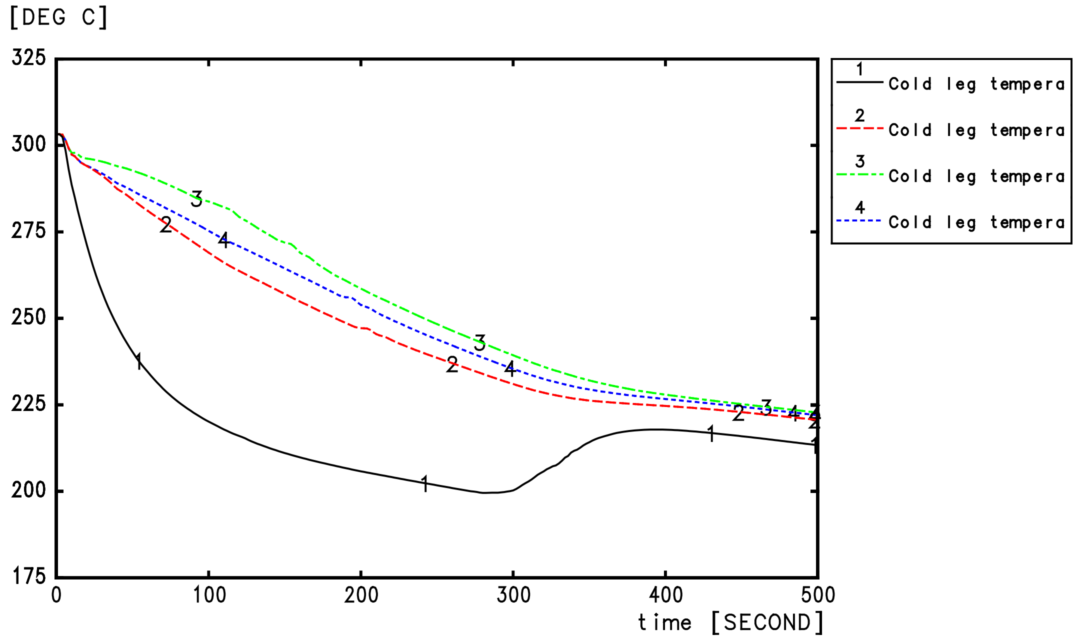
PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 72

Break Spectrum –Double-Ended Guillotine Break – Hot Leg Temperature and Cold Leg Temperature



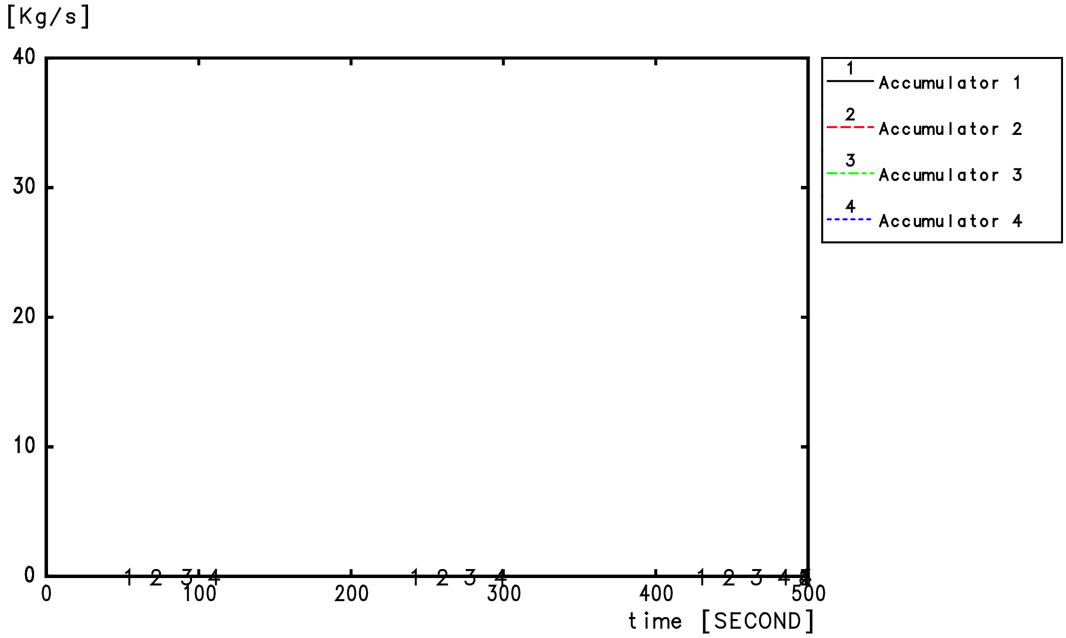
HOT LEG TEMPERATURE



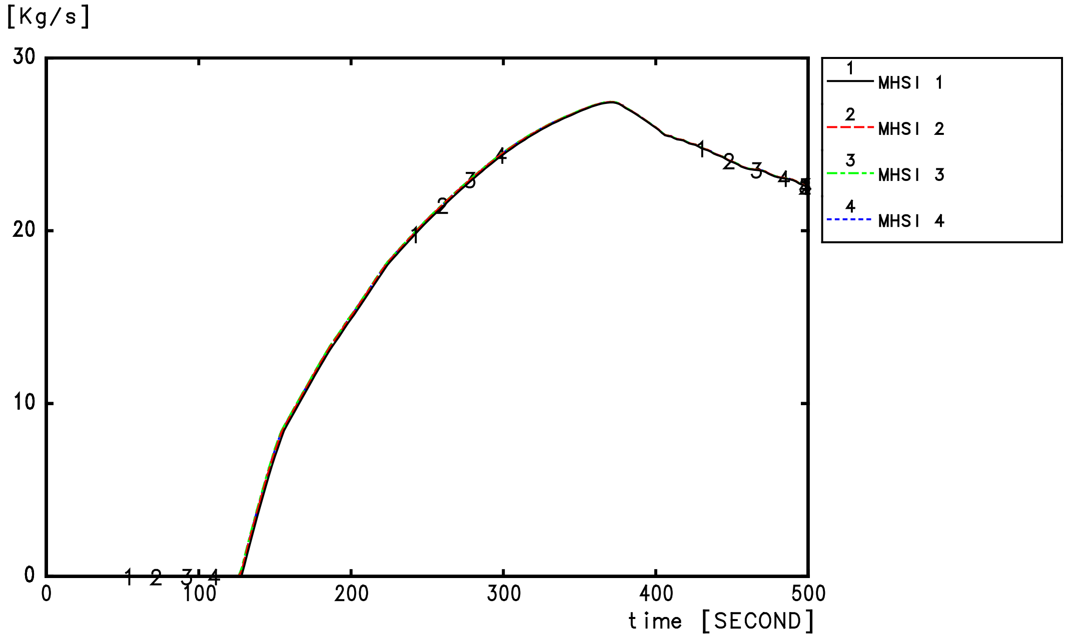
COLD LEG TEMPERATURE

SECTION 14.5.2 - FIGURE 73

Break Spectrum –Double-Ended Guillotine Break – Total Accumulators Flow Rate and Total MHSI Flow Rate



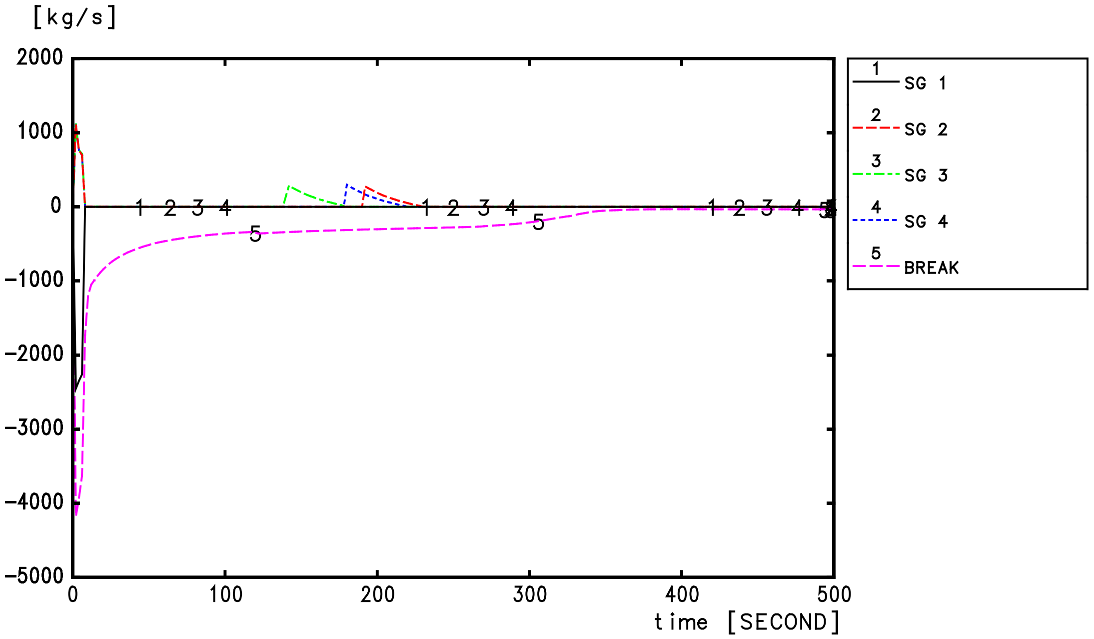
TOTAL ACCUMULATORS FLOW RATE



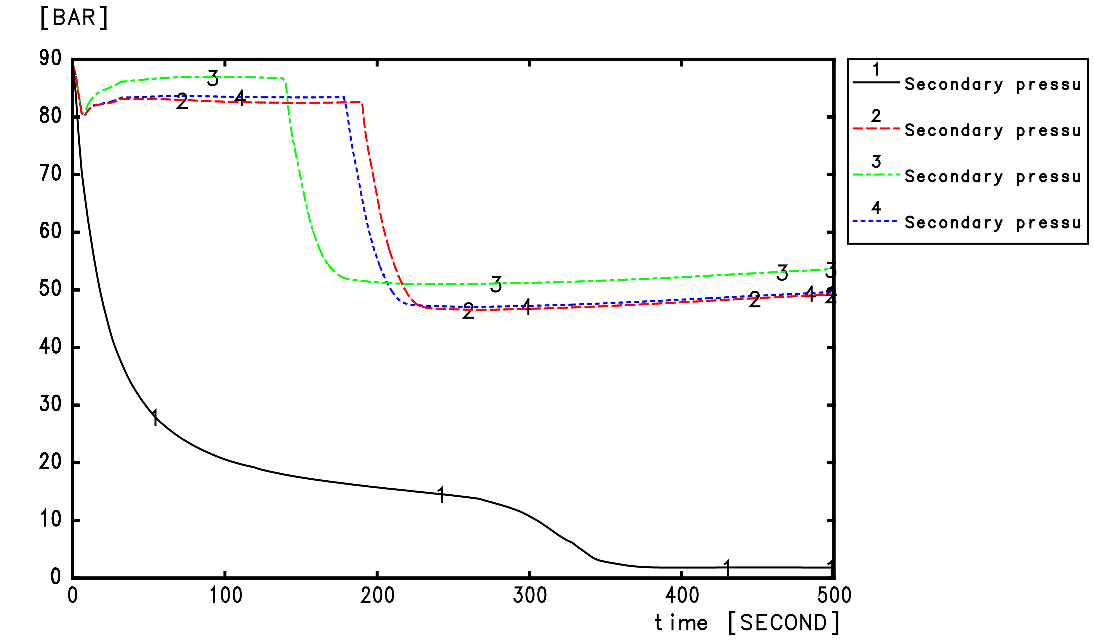
TOTAL MHSI FLOW RATE

SECTION 14.5.2 - FIGURE 74

Break Spectrum – Double-Ended Guillotine Break – Vapour Mass Flow Rate and SG Pressure



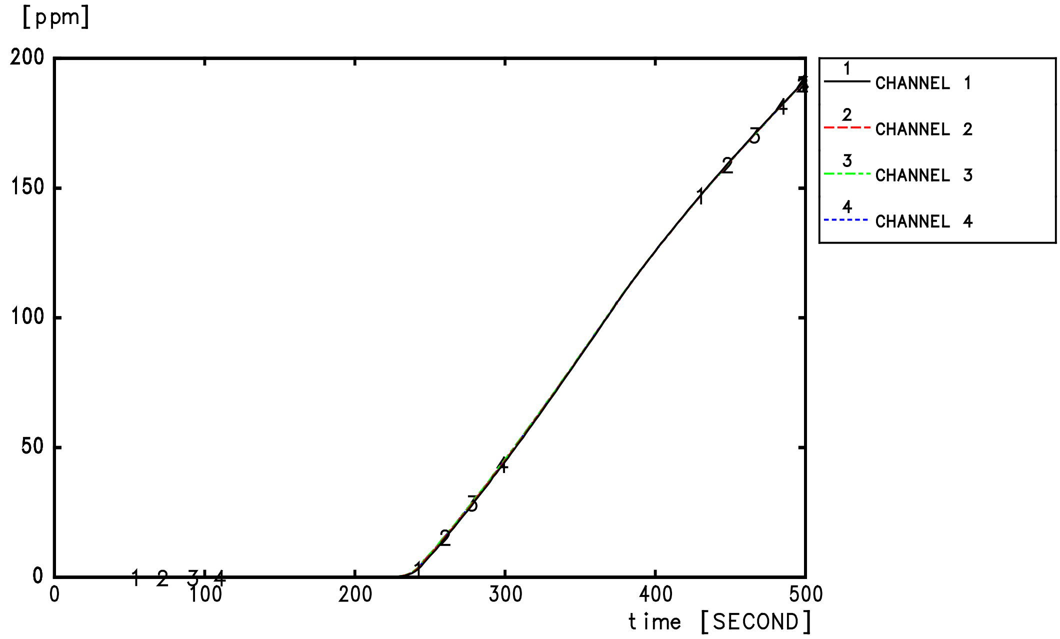
VAPOR MASS FLOWRATE



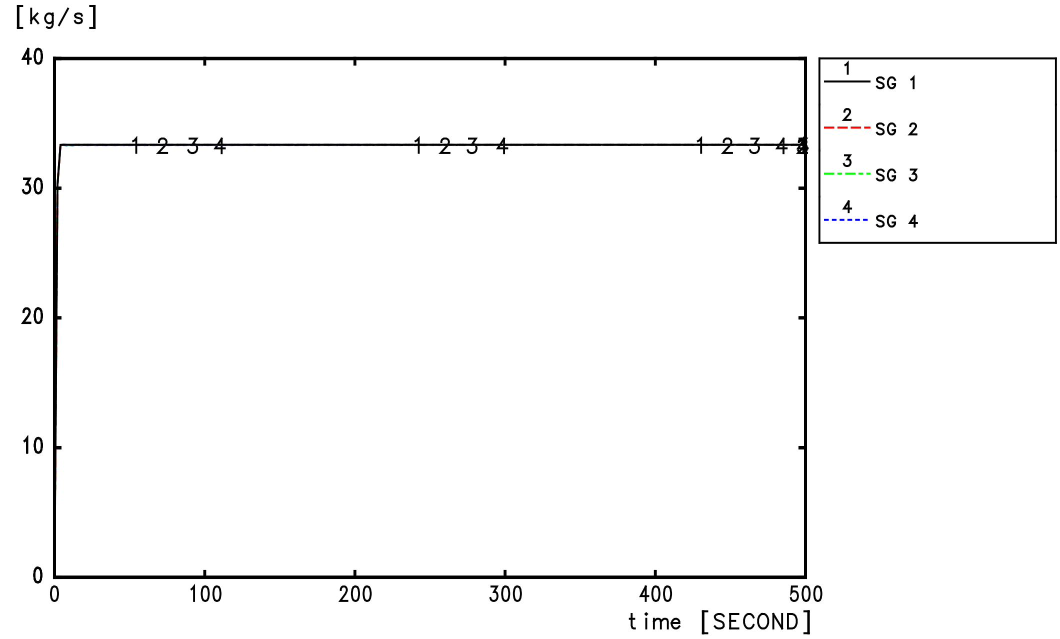
STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 75

Break Spectrum – Double-Ended Guillotine Break – Boron Concentration at Core Active Part Inlet and Emergency Feedwater Flow Rate



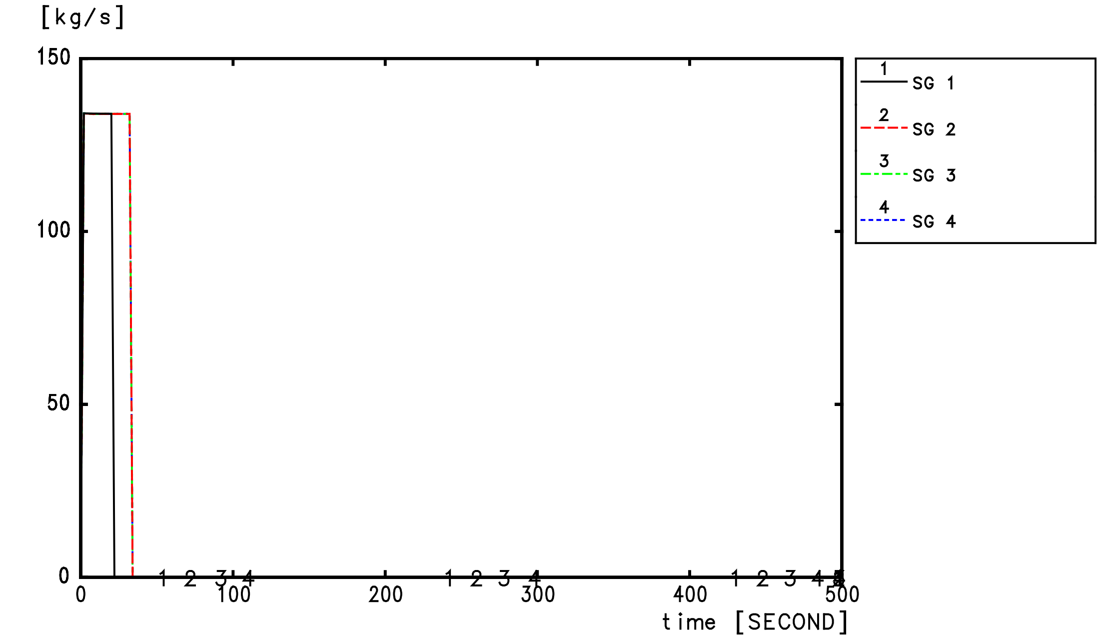
BORON CONCENTRATION AT CORE ACTIVE PART INLET



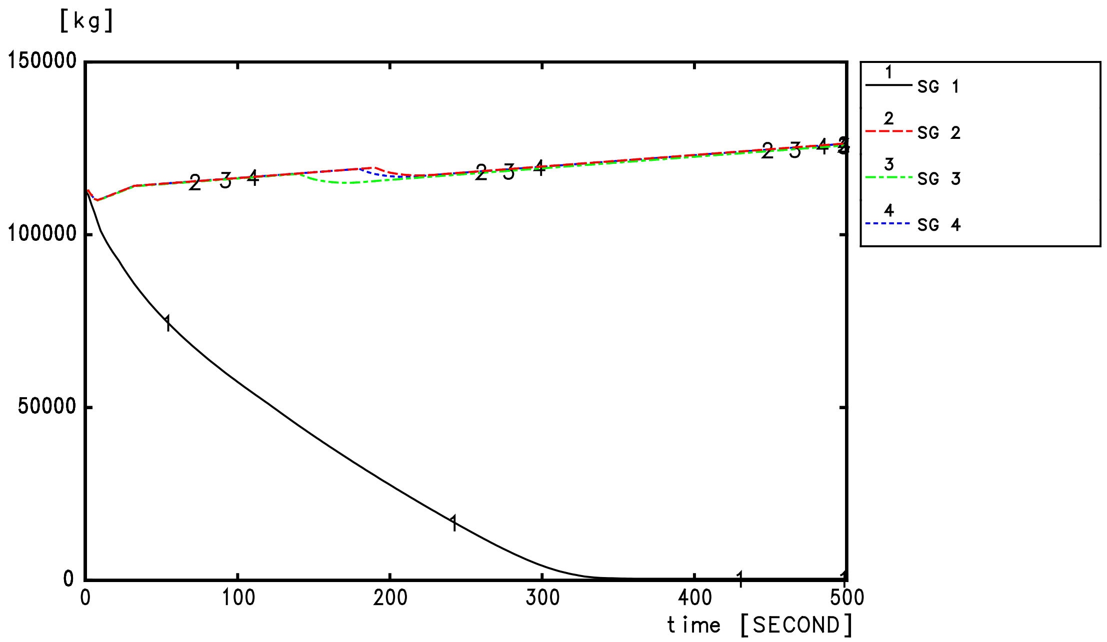
ASG FLOWRATE

SECTION 14.5.2 - FIGURE 76

Break Spectrum – Double-Ended Guillotine Break – Main Feedwater Flow Rate and SG Liquid Mass



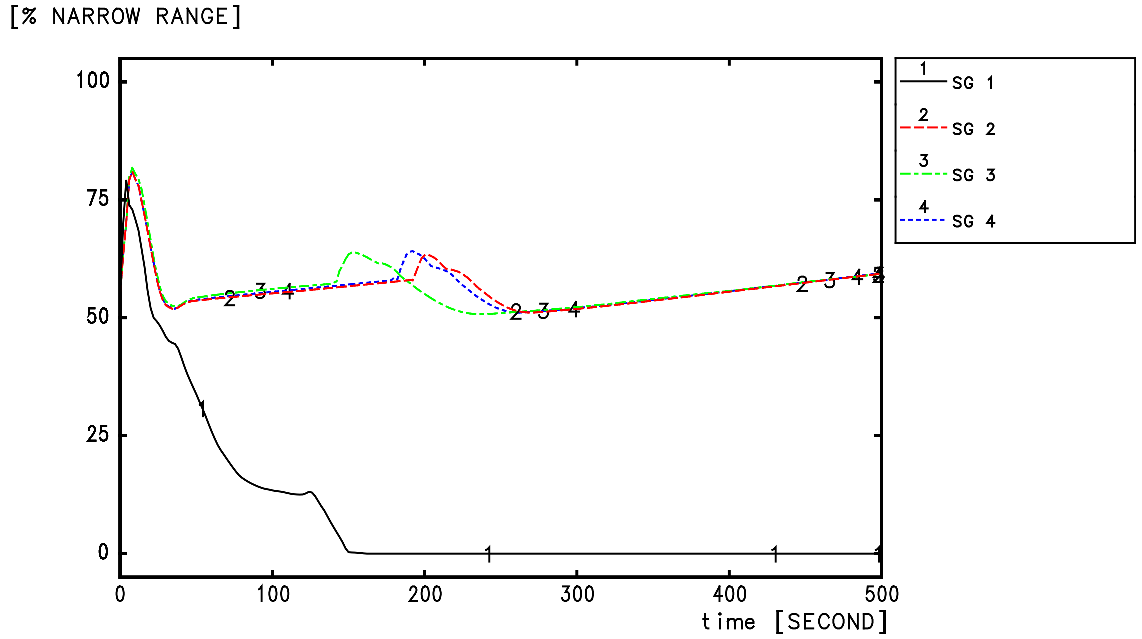
MFW FLOWRATE



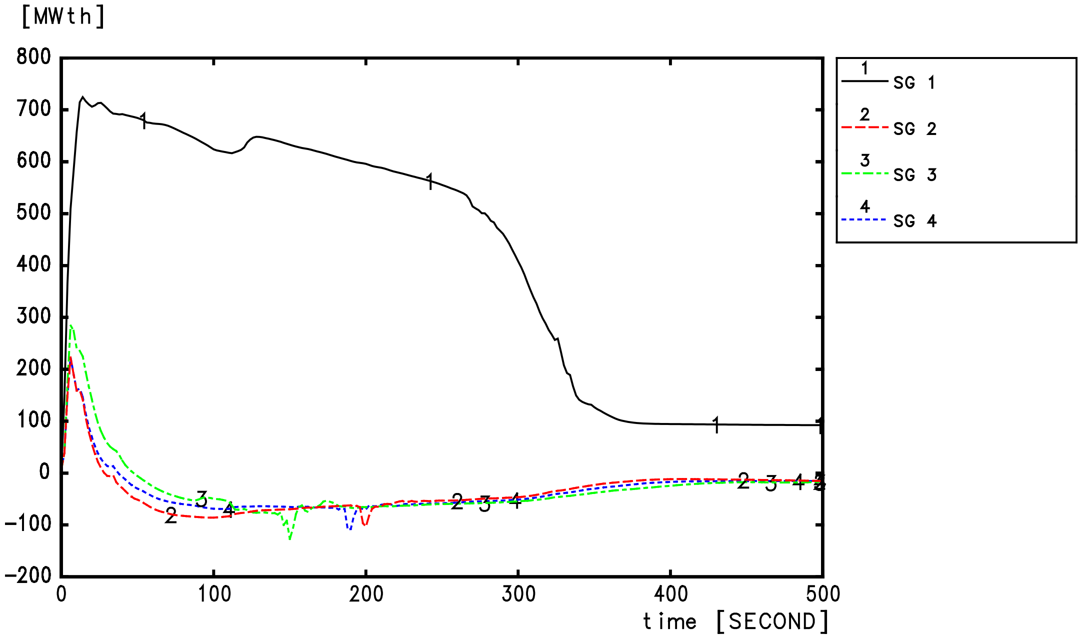
STEAM GENERATOR LIQUID MASS

SECTION 14.5.2 - FIGURE 77

Break Spectrum – Double-Ended Guillotine Break – SG Narrow Range Level and SG Power Exchanged



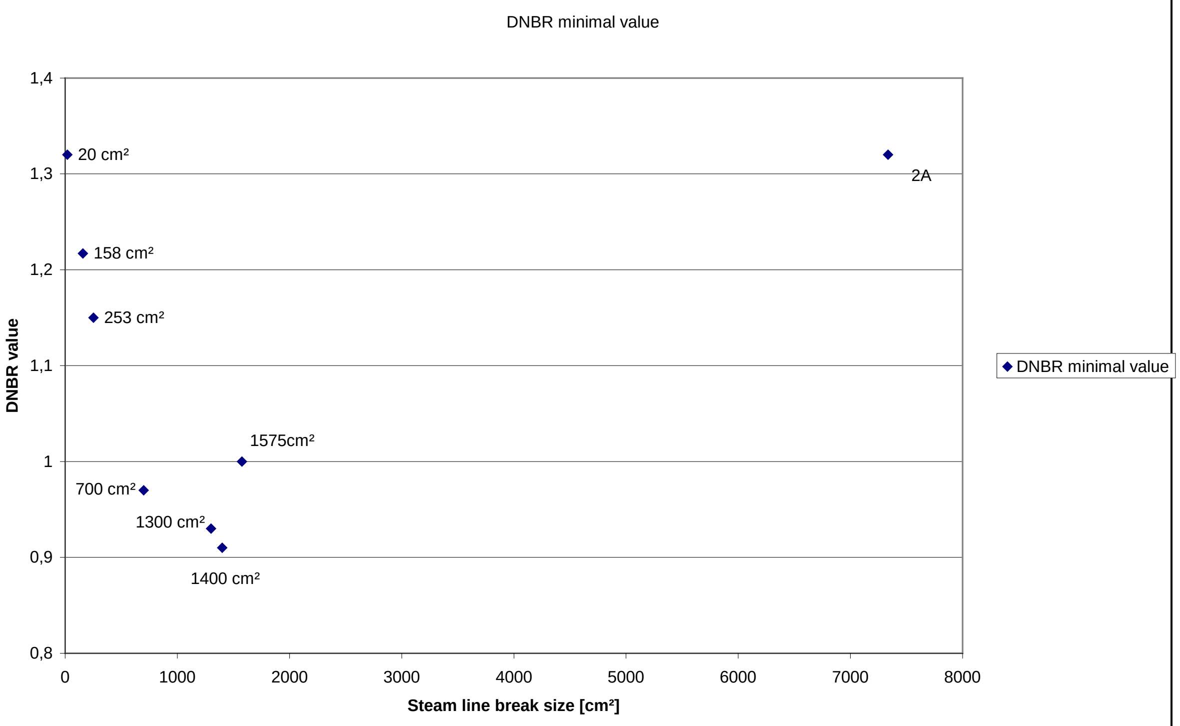
STEAM GENERATOR NARROW RANGE LEVEL



SG POWER EXCHANGED

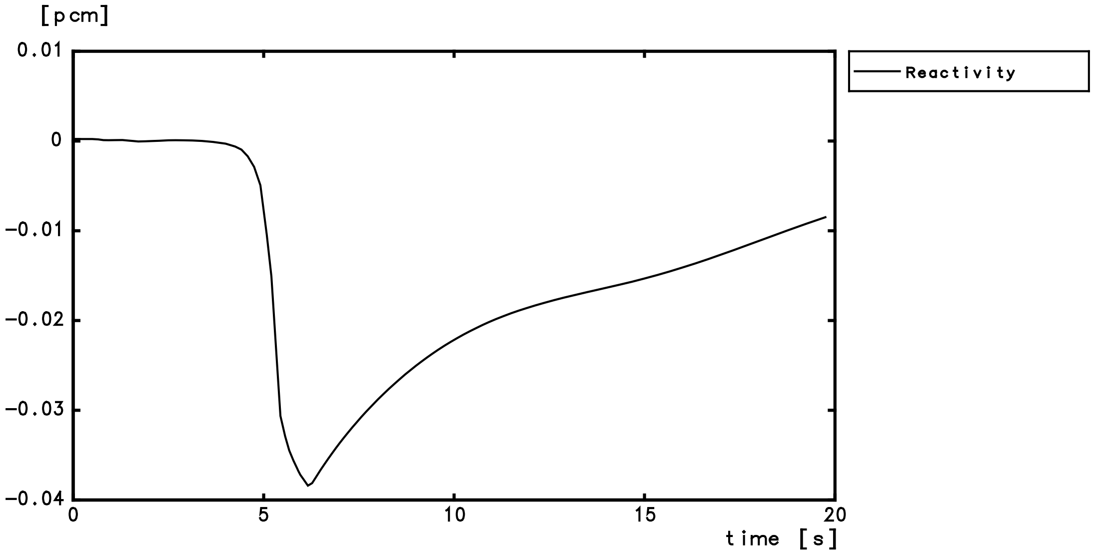
SECTION 14.5.2 - FIGURE 78

Break Spectrum at Full Power, DNBR Minimal Value as a Function of Steam Line Break Size

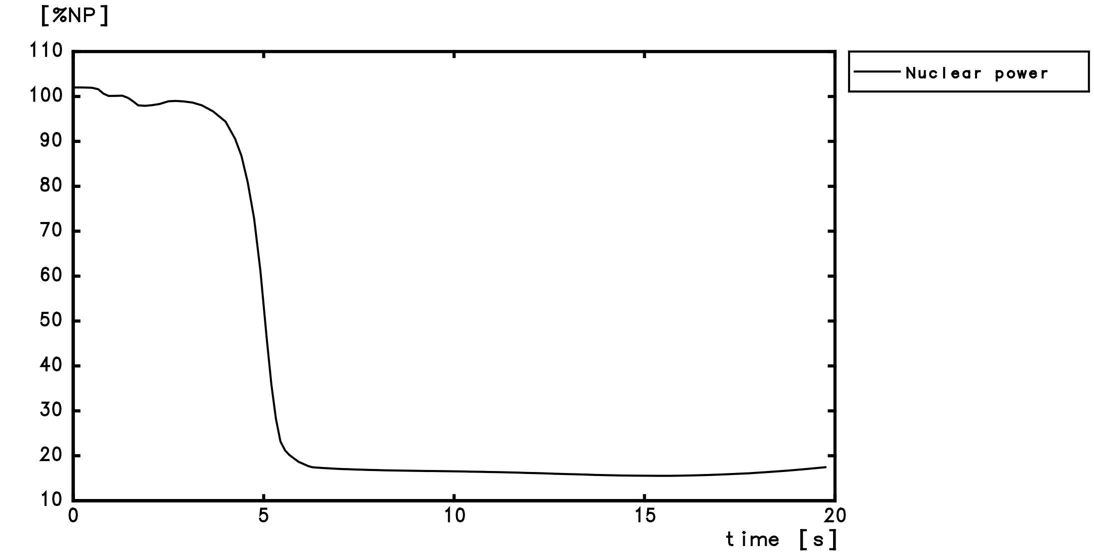


SECTION 14.5.2 - FIGURE 79

Break Spectrum at Full Power, Double Ended Guillotine Break - Reactivity and Nuclear Power



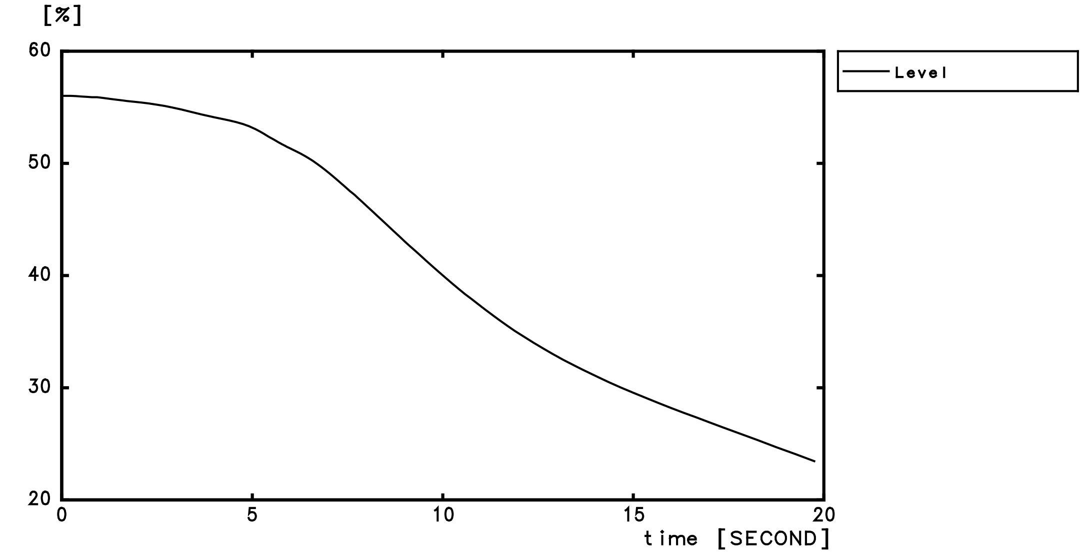
REACTIVITY



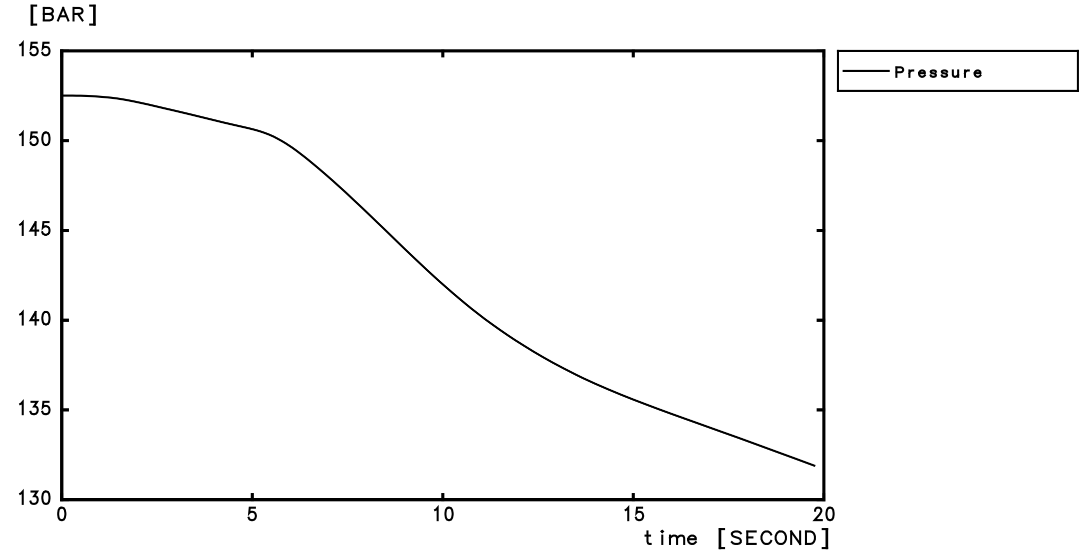
CORE POWER

SECTION 14.5.2 - FIGURE 80

Break Spectrum at Full Power, Double Ended Guillotine Break - Pressuriser Level and Pressuriser Pressure



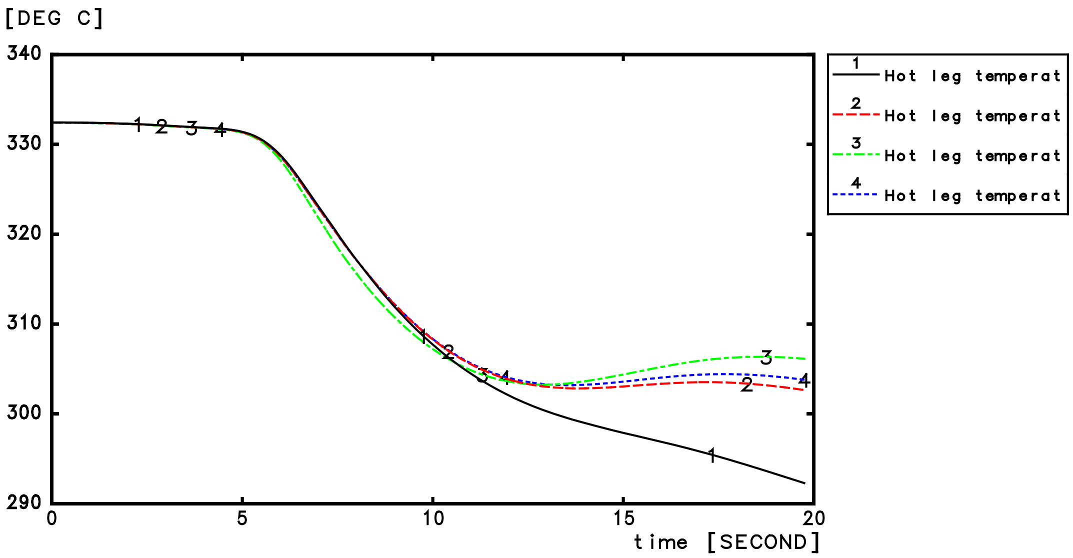
PRESSURIZER LEVEL



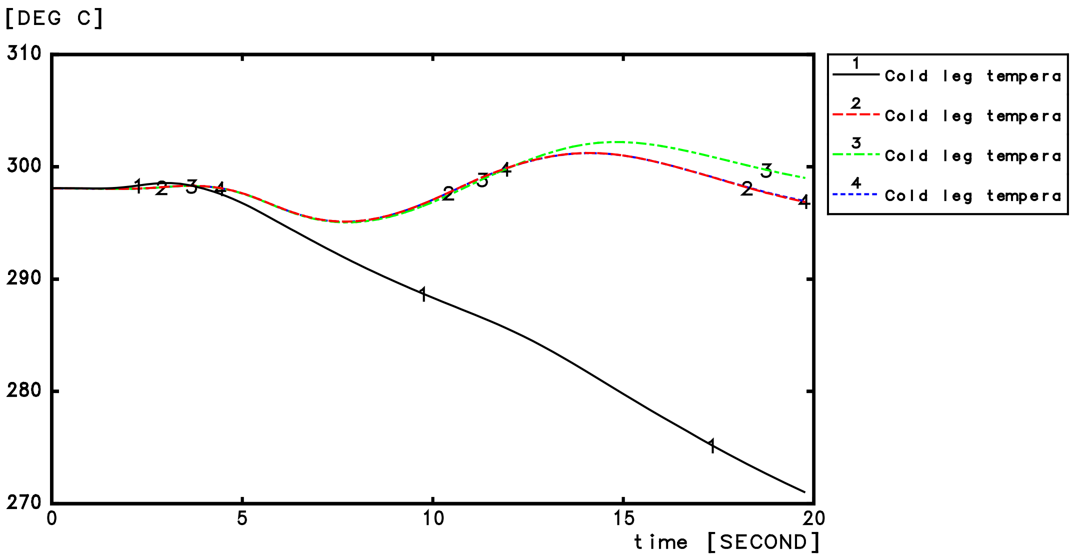
PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 81

Break Spectrum at Full Power, Double Ended Guillotine Break - Hot and Cold Leg Temperature



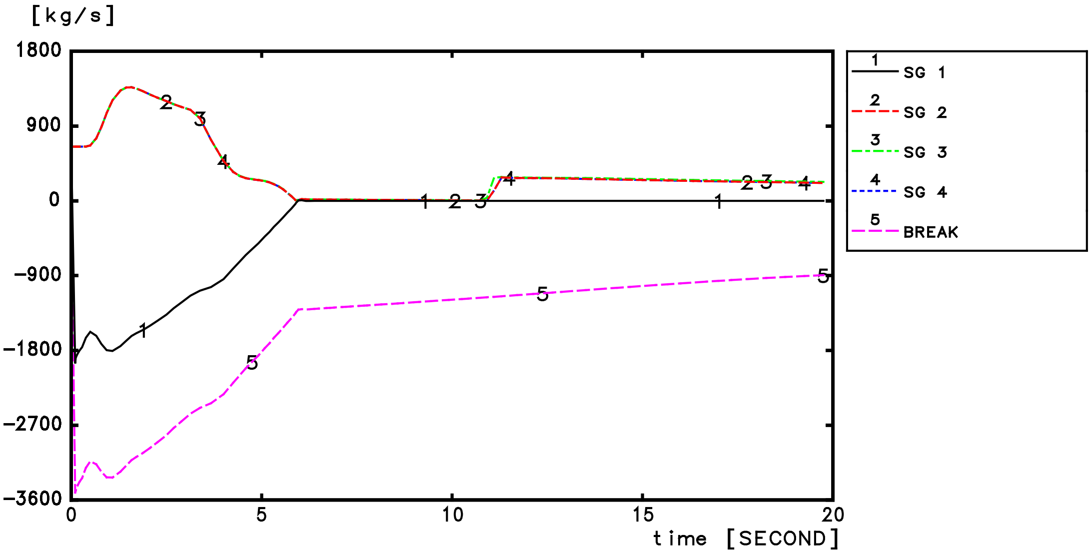
HOT LEG TEMPERATURE



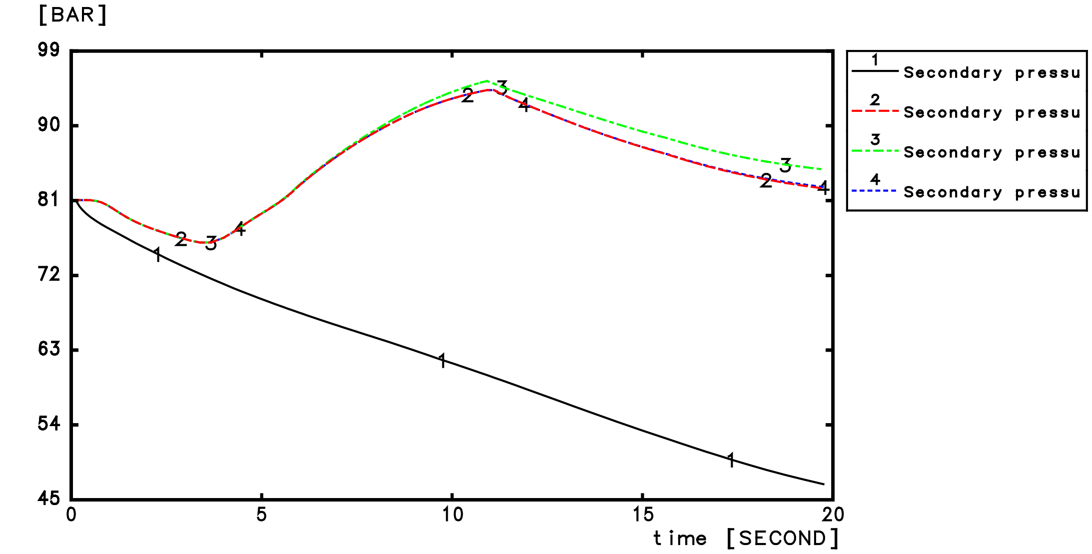
COLD LEG TEMPERATURE

SECTION 14.5.2 - FIGURE 82

Break Spectrum at Full Power, Double Ended Guillotine Break - SG Mass Flow Rate and SG Press



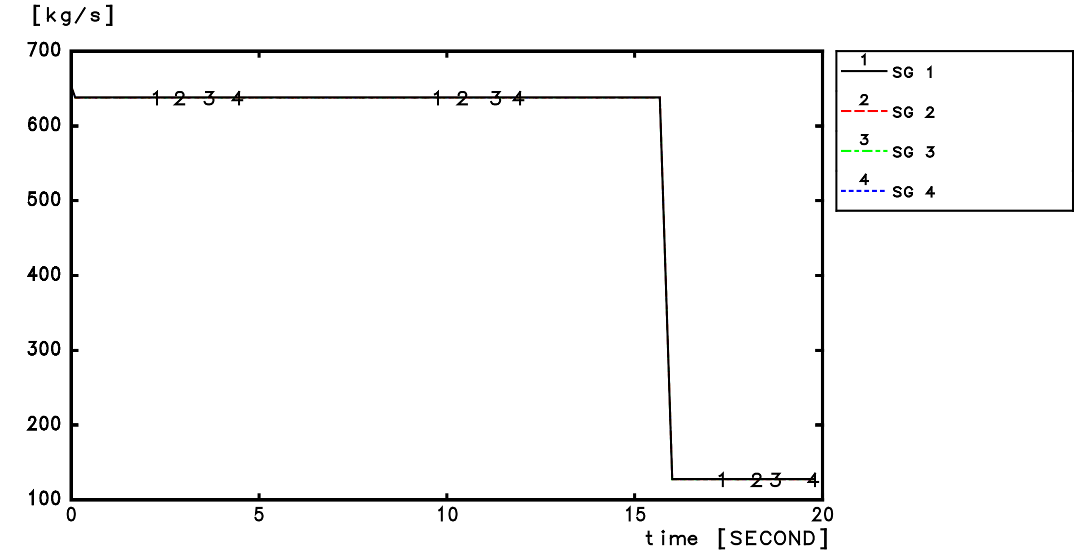
VAPOR MASS FLOWRATE



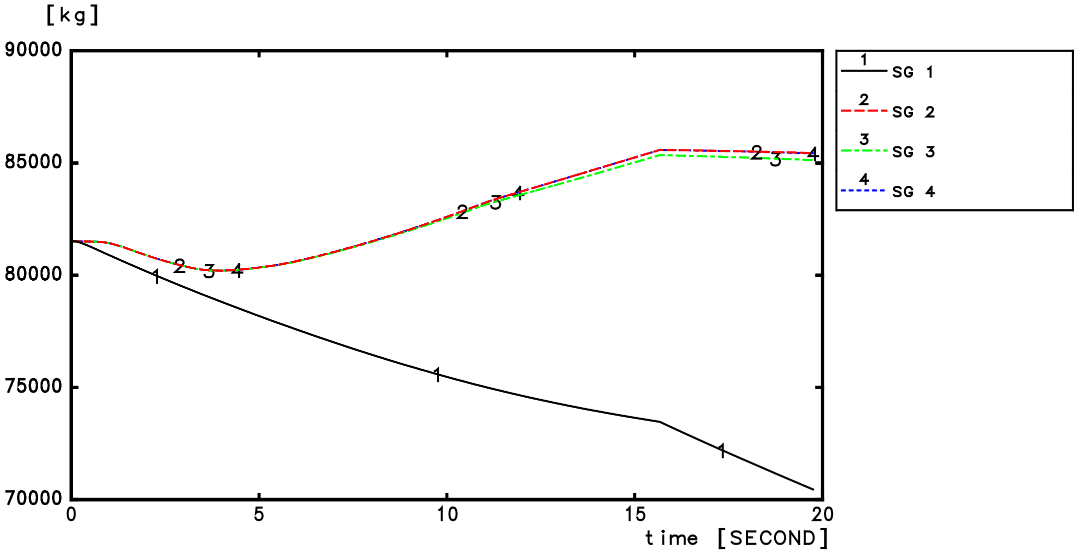
STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 83

Break Spectrum at Full Power, Double Ended Guillotine Break - Main Feed Water Flow Rate and SG Liquid Mass



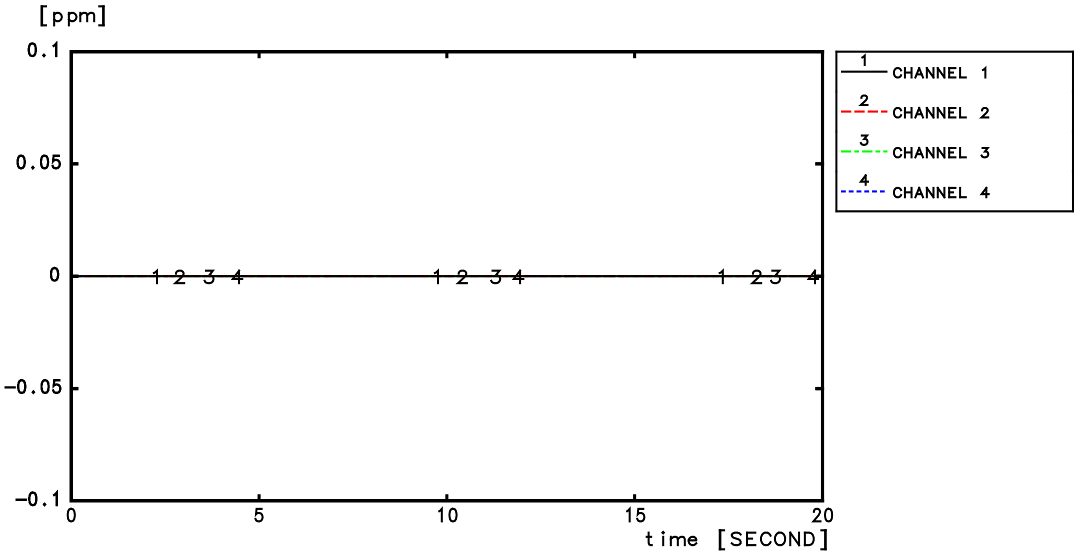
MFW FLOWRATE



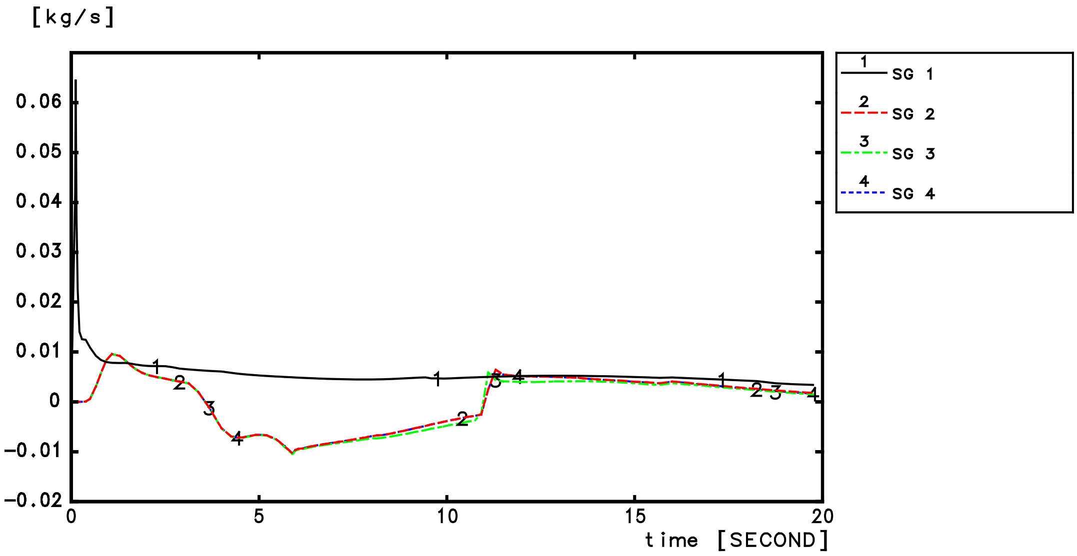
STEAM GENERATOR LIQUID MASS

SECTION 14.5.2 - FIGURE 84

Break Spectrum at Full Power, Double Ended Guillotine Break - Boron Concentration and ASG [EFWS] Flow Rate



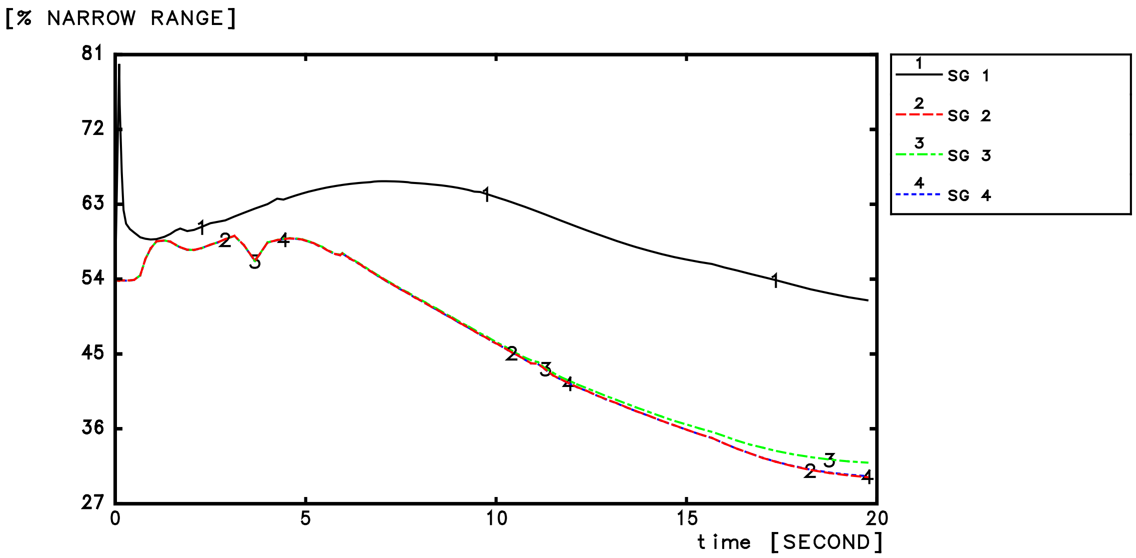
BORON CONCENTRATION AT CORE ACTIVE PART INLET



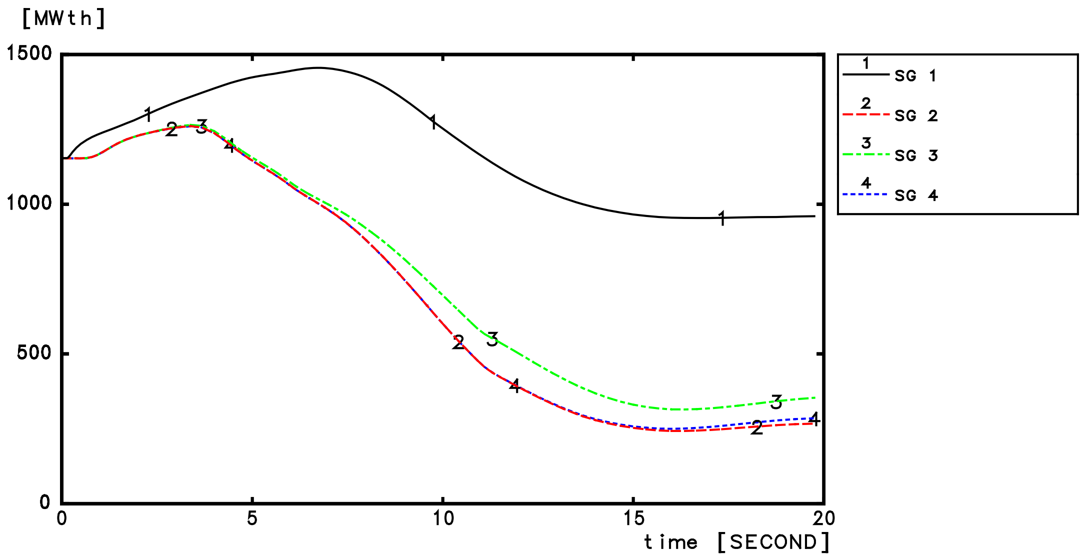
ASG FLOWRATE

SECTION 14.5.2 - FIGURE 85

Break Spectrum at Full Power, Double-Ended Guillotine Break - SG Narrow Range Level and SG Power Exchanged



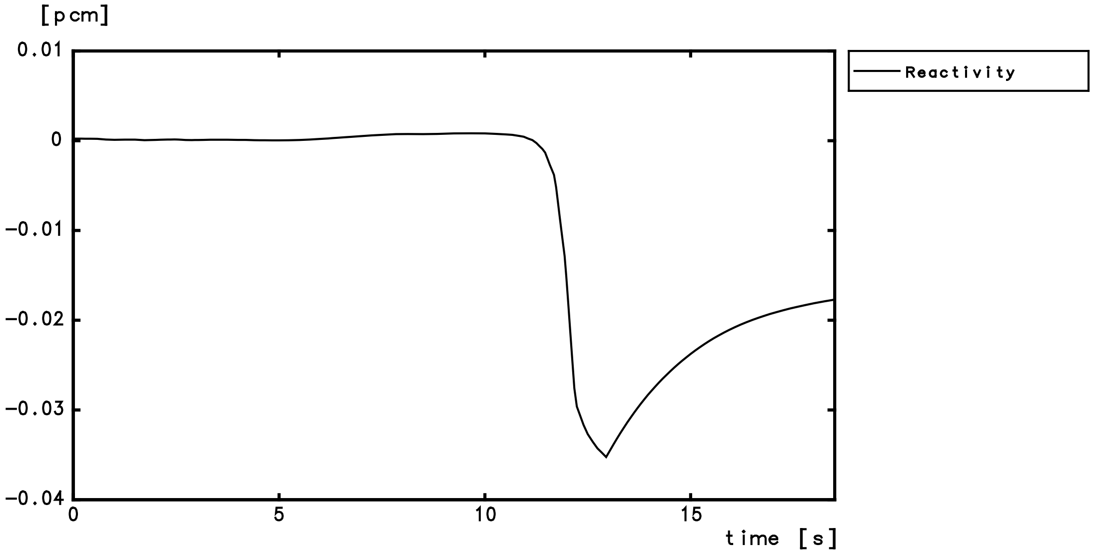
STEAM GENERATOR NARROW RANGE LEVEL



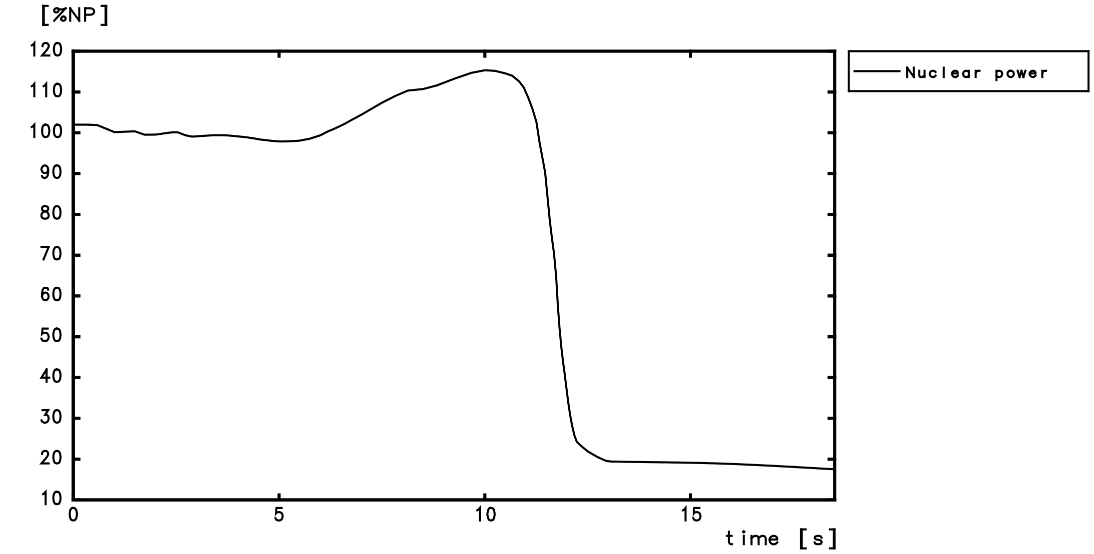
SG POWER EXCHANGED

SECTION 14.5.2 - FIGURE 86

Break Spectrum at Full Power, 1575 cm² - Reactivity and Nuclear Power



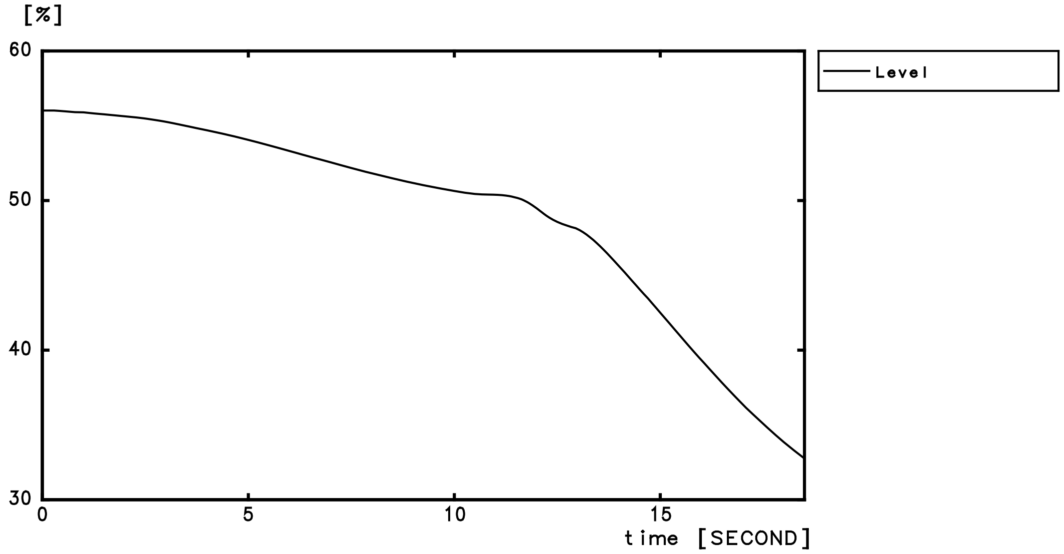
REACTIVITY



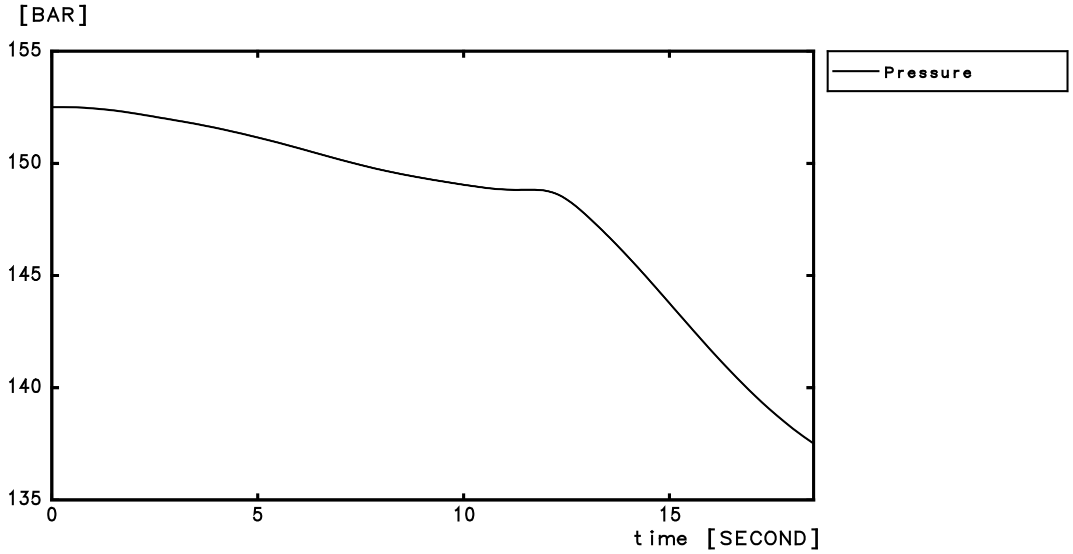
CORE POWER

SECTION 14.5.2 - FIGURE 87

Break Spectrum at Full Power, 1575 cm² - Pressuriser Level and Pressuriser Pressure



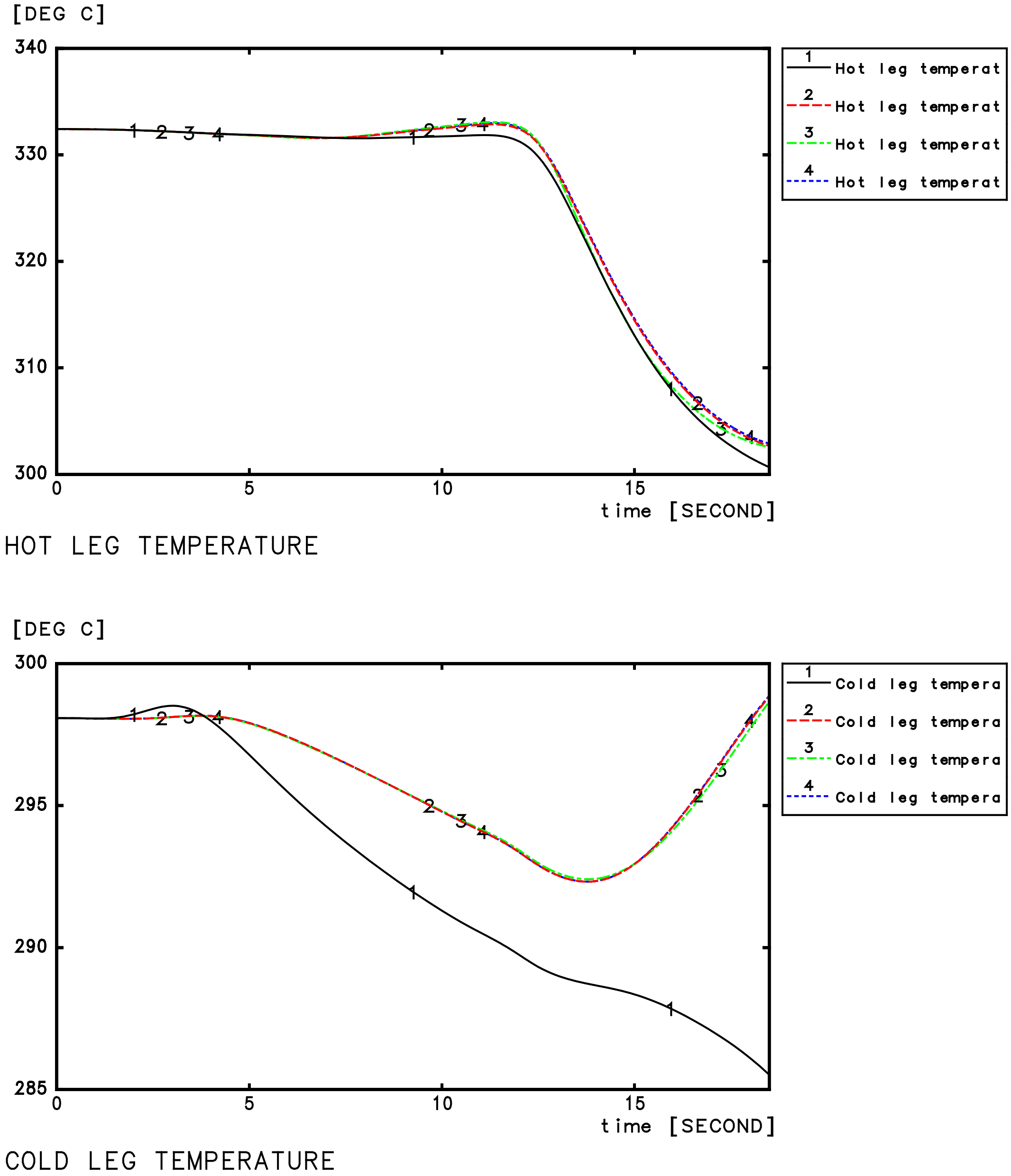
PRESSURIZER LEVEL



PRESSURIZER PRESSURE

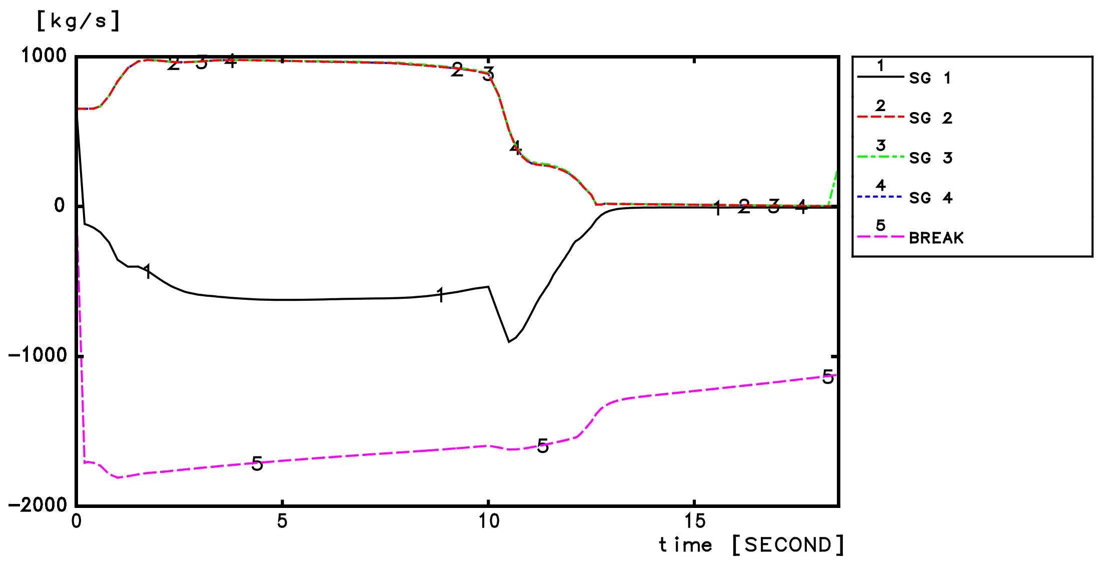
SECTION 14.5.2 - FIGURE 88

Break Spectrum at Full Power, 1575 cm² - Hot and Cold Leg Temperature

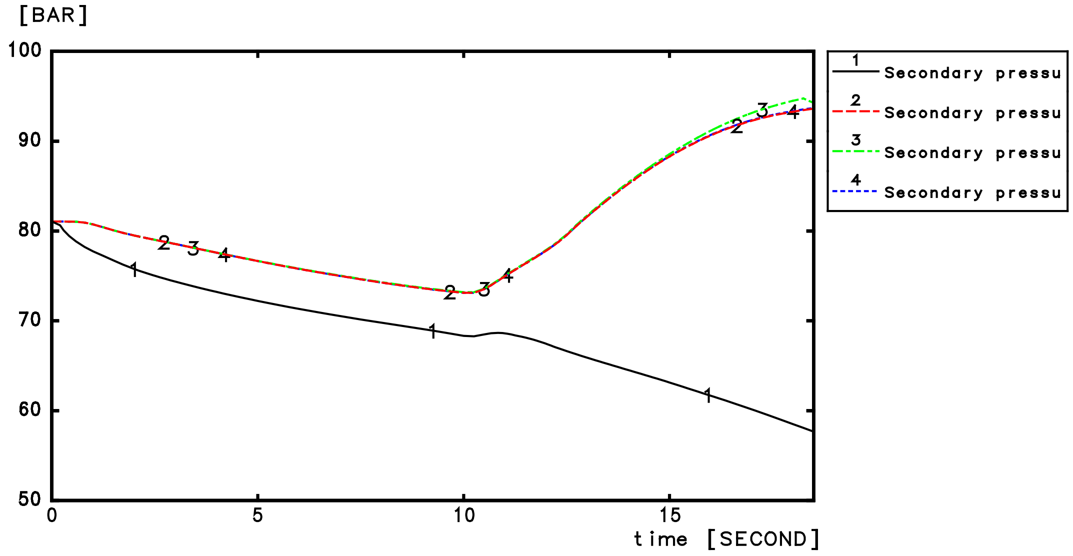


SECTION 14.5.2 - FIGURE 89

Break Spectrum at Full Power, 1575 cm² - SG Mass Flow Rate and SG Pressure



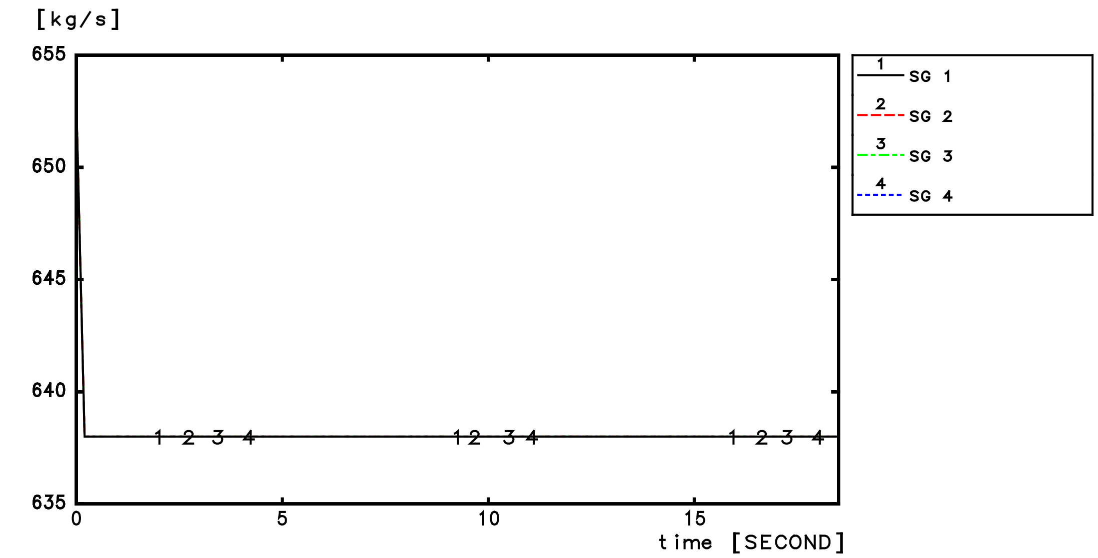
VAPOR MASS FLOWRATE



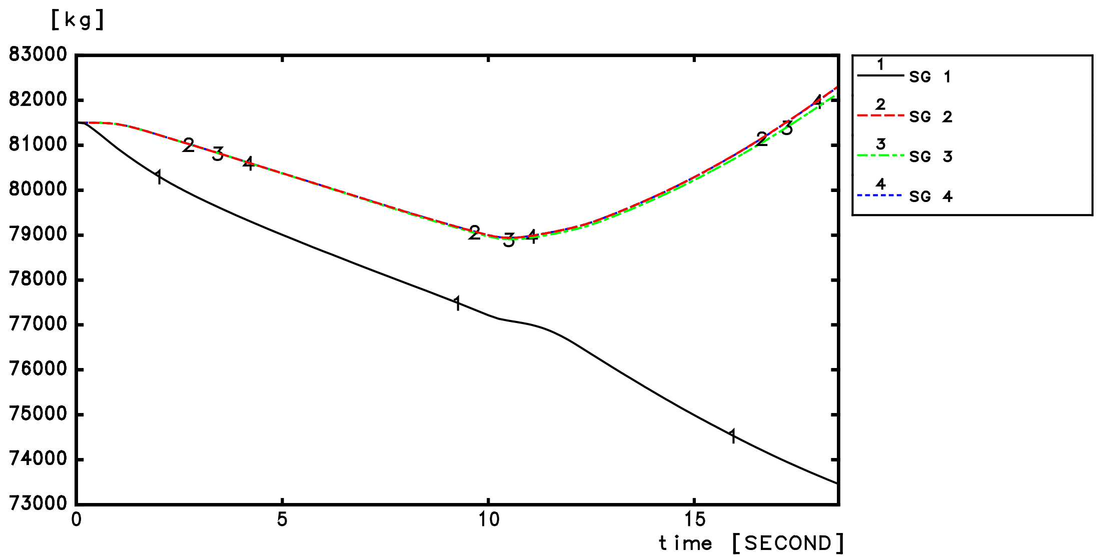
STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 90

Break Spectrum at Full Power, 1575 cm²- Main Feed Water Flow Rate and SG Liquid Mass



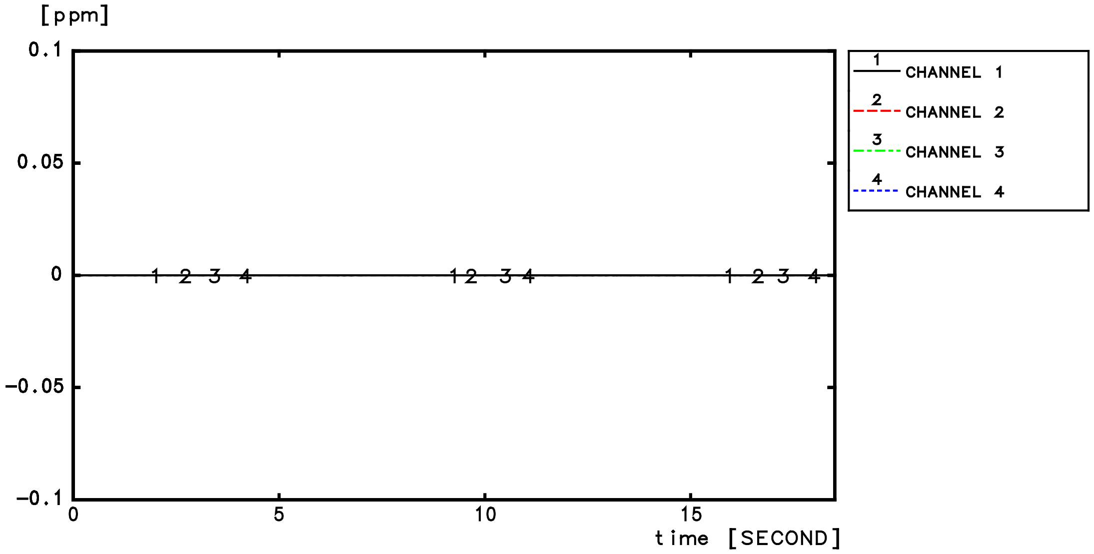
MFW FLOWRATE



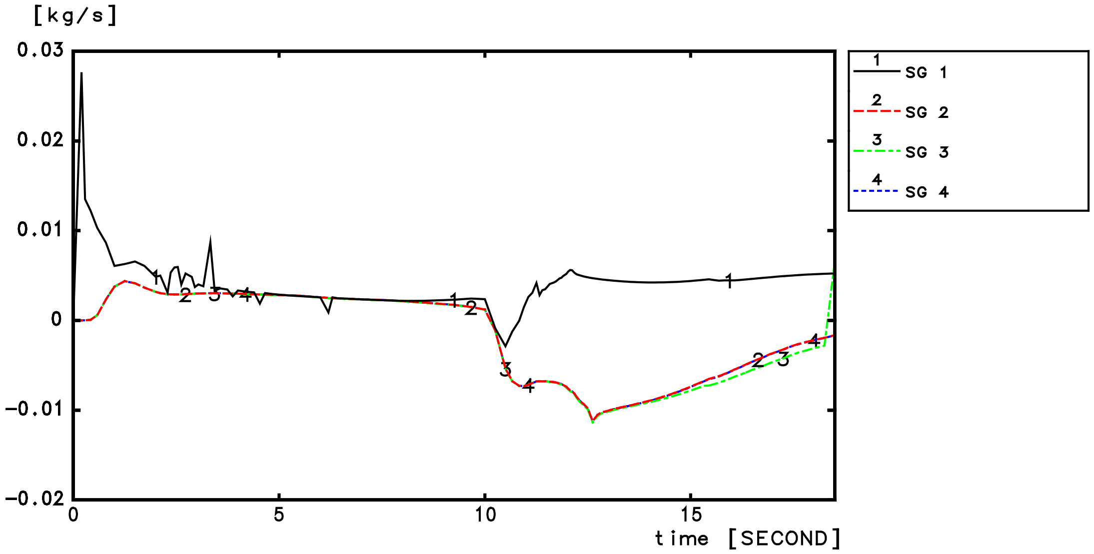
STEAM GENERATOR LIQUID MASS

SECTION 14.5.2 - FIGURE 91

Break Spectrum at Full Power, 1575 cm² - Boron Concentration and [EFWS] Flow Rate



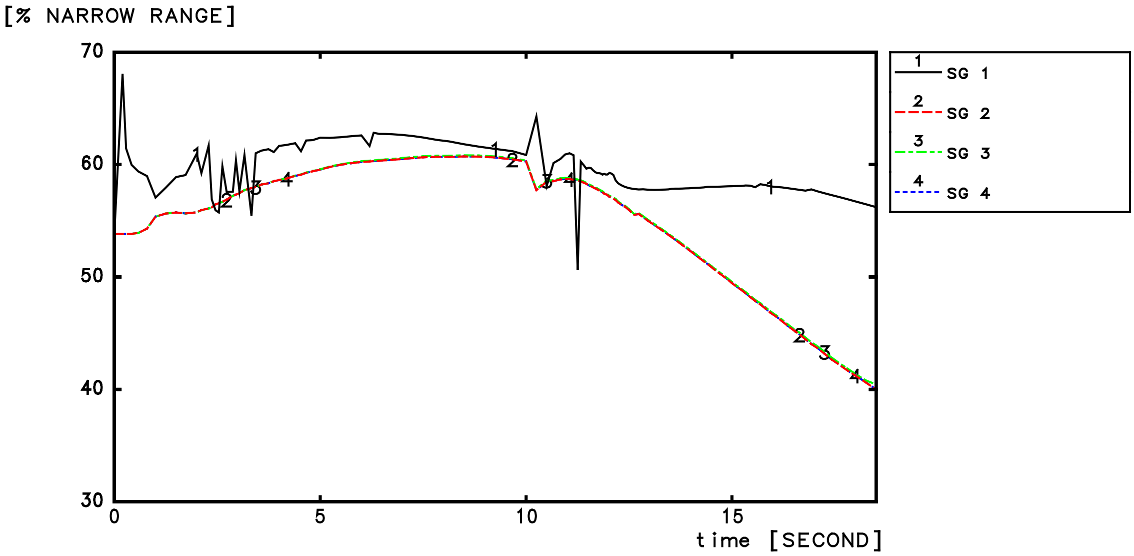
BORON CONCENTRATION AT CORE ACTIVE PART INLET



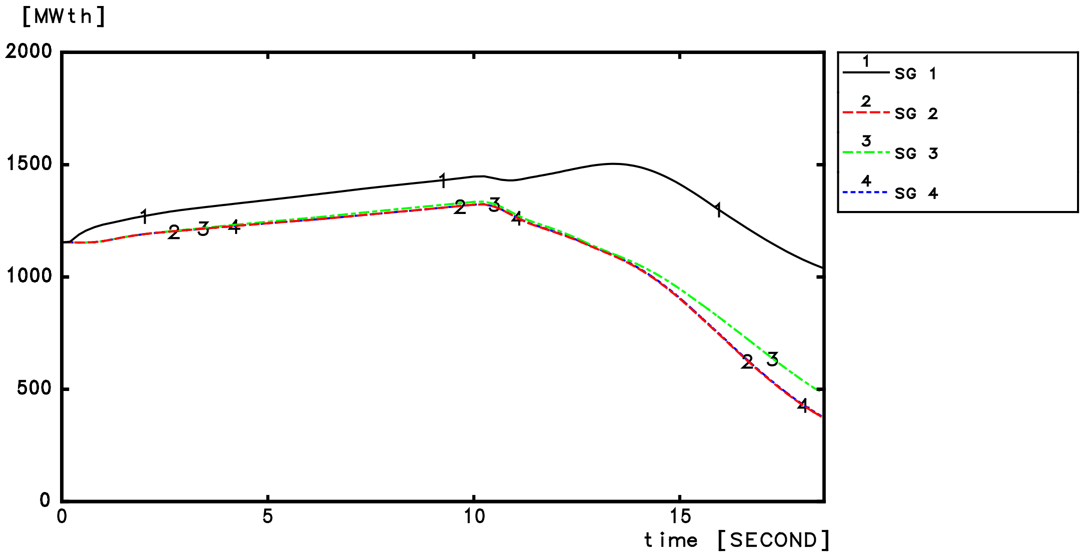
ASG FLOWRATE

SECTION 14.5.2 - FIGURE 92

Break Spectrum at Full Power, 1575 cm²- SG Narrow Range Level and SG Power Exchanged



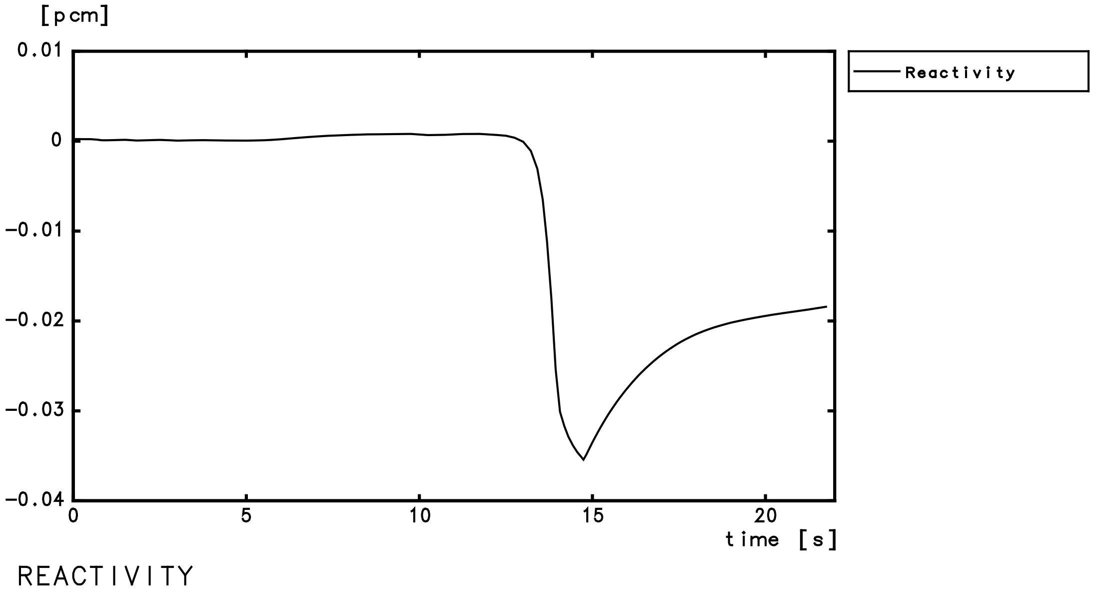
STEAM GENERATOR NARROW RANGE LEVEL

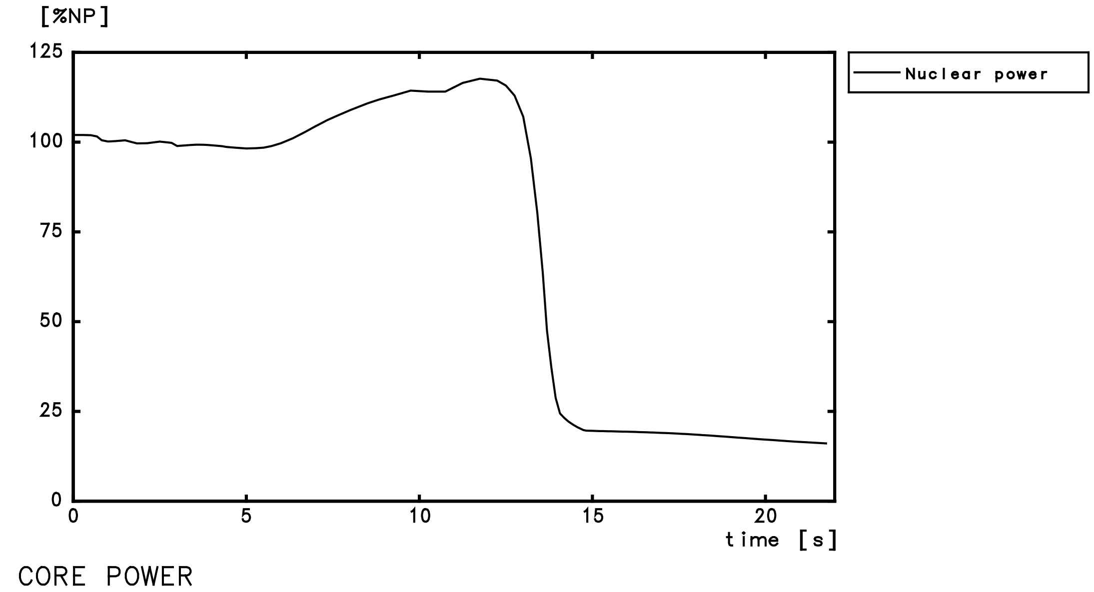


SG POWER EXCHANGED

SECTION 14.5.2 - FIGURE 93

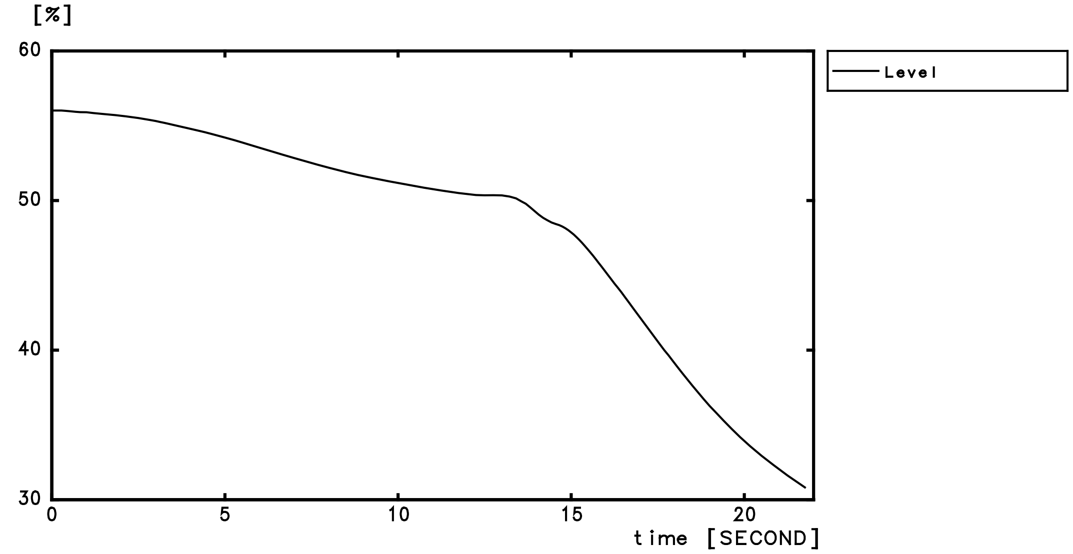
Break Spectrum at Full Power, 1400 cm² - Worst break - Reactivity and Nuclear Power



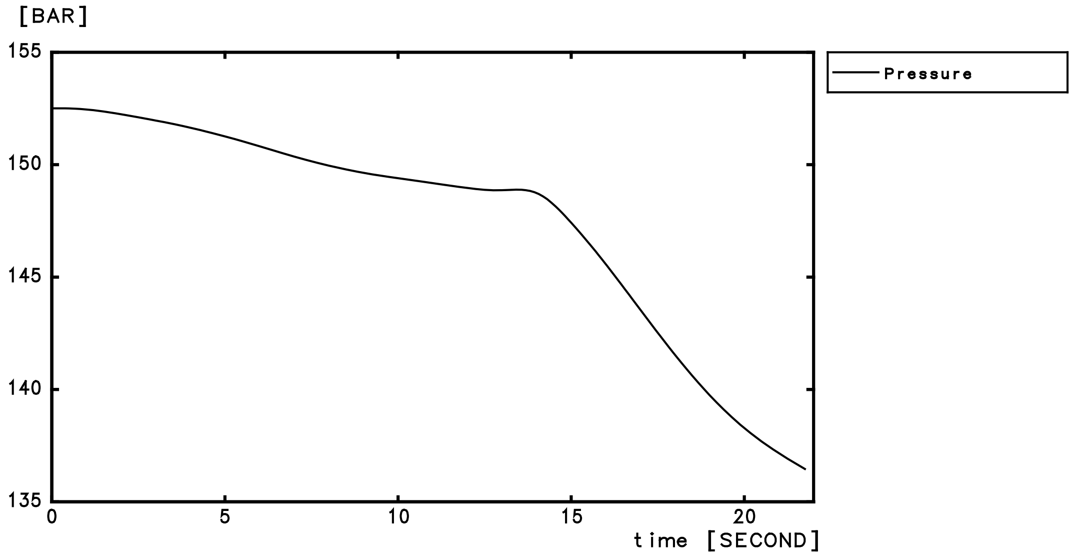


SECTION 14.5.2 - FIGURE 94

Break Spectrum at Full Power, 1400 cm² - Worst break - Pressuriser Level and Pressuriser Pressure



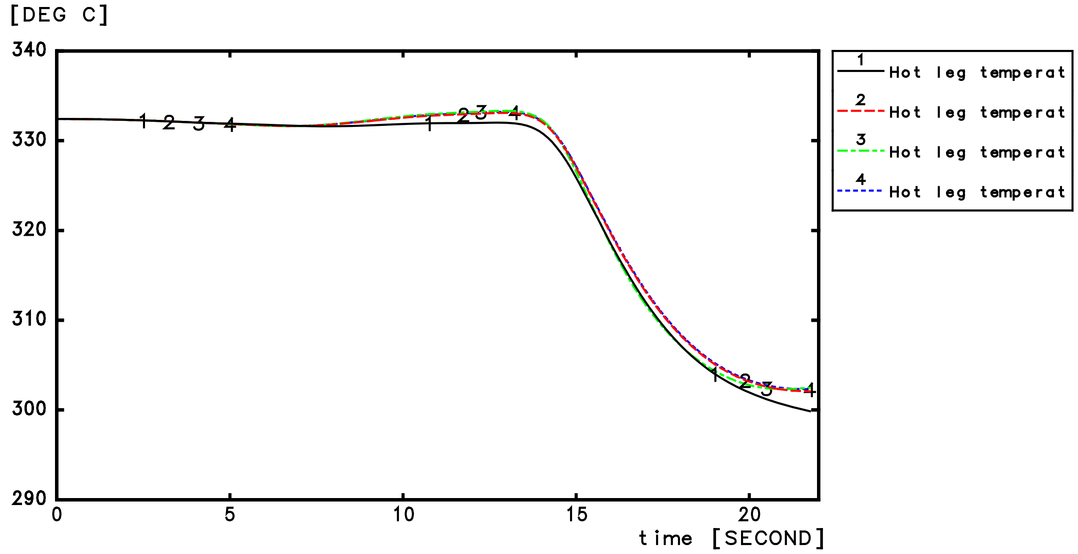
PRESSURIZER LEVEL



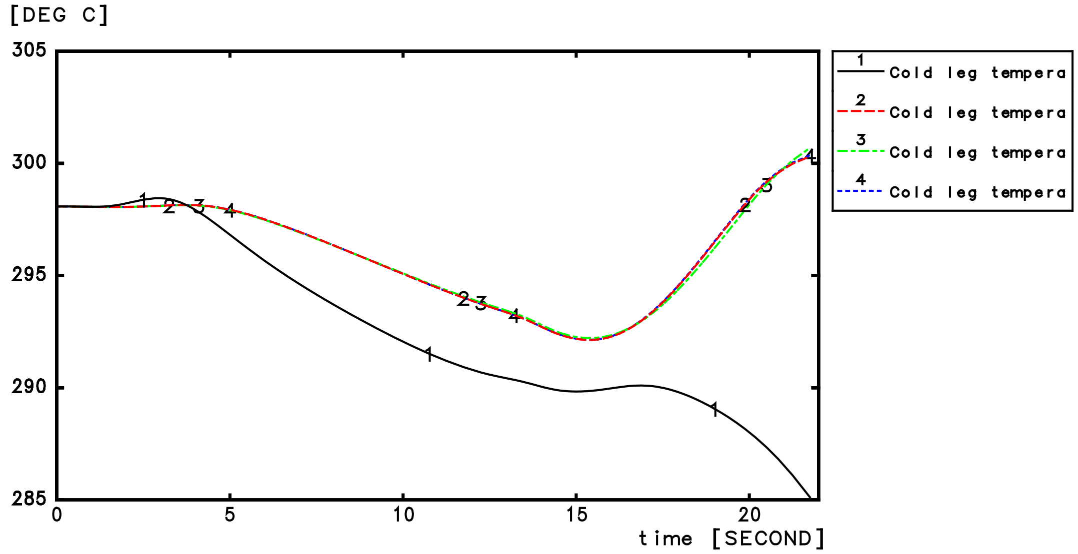
PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 95

Break Spectrum at Full Power, 1400 cm² - Worst break - Hot and Cold Leg Temperature



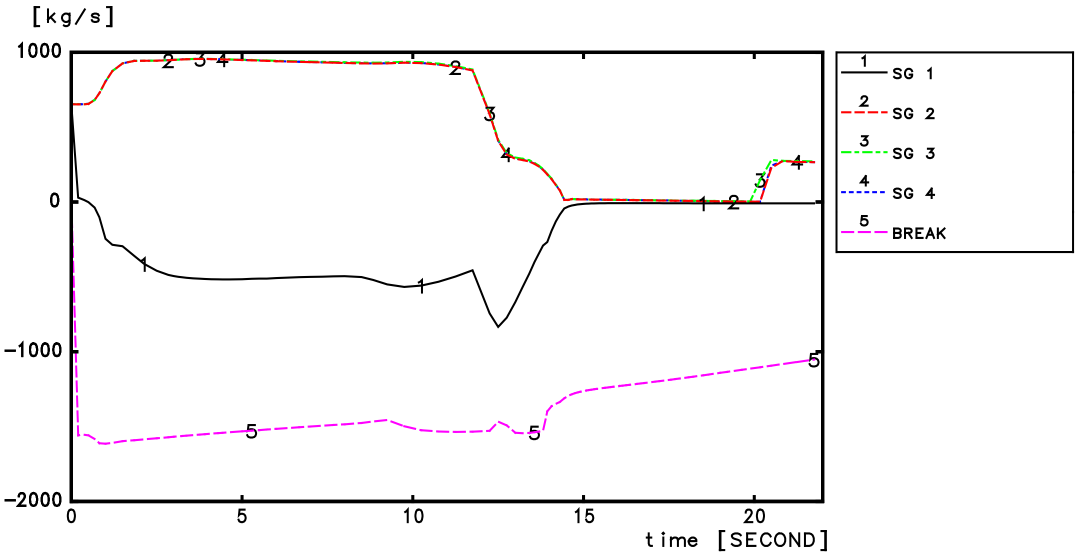
HOT LEG TEMPERATURE



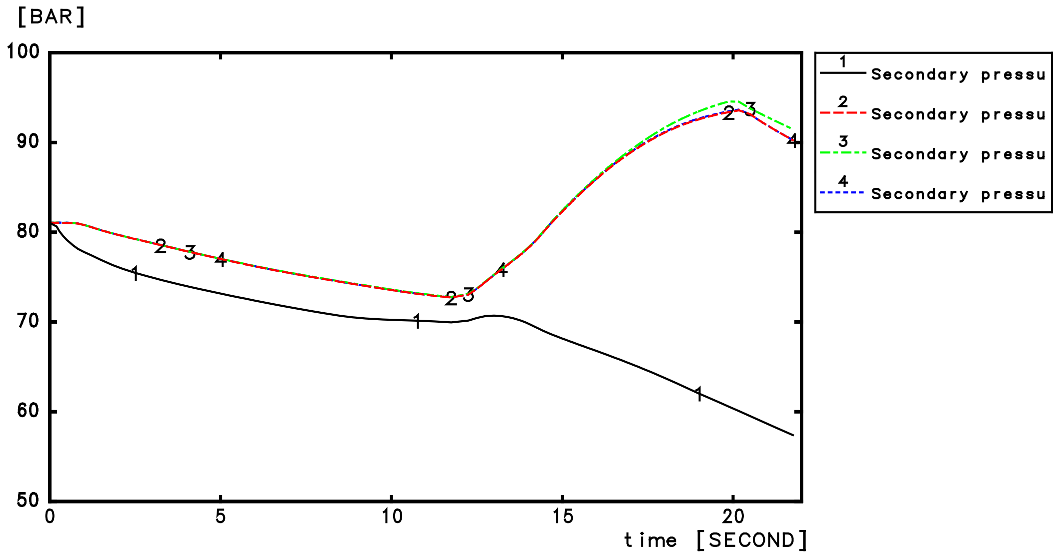
COLD LEG TEMPERATURE

SECTION 14.5.2 - FIGURE 96

Break Spectrum at Full Power, 1400 cm² - Worst break - SG Mass Flow rate and SG Pressure



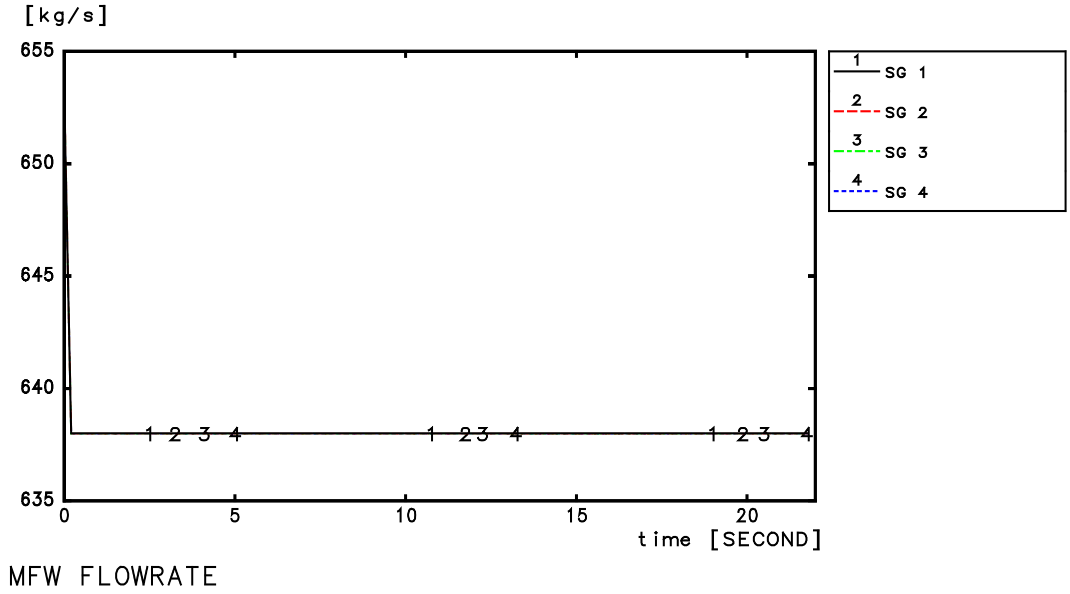
VAPOR MASS FLOWRATE

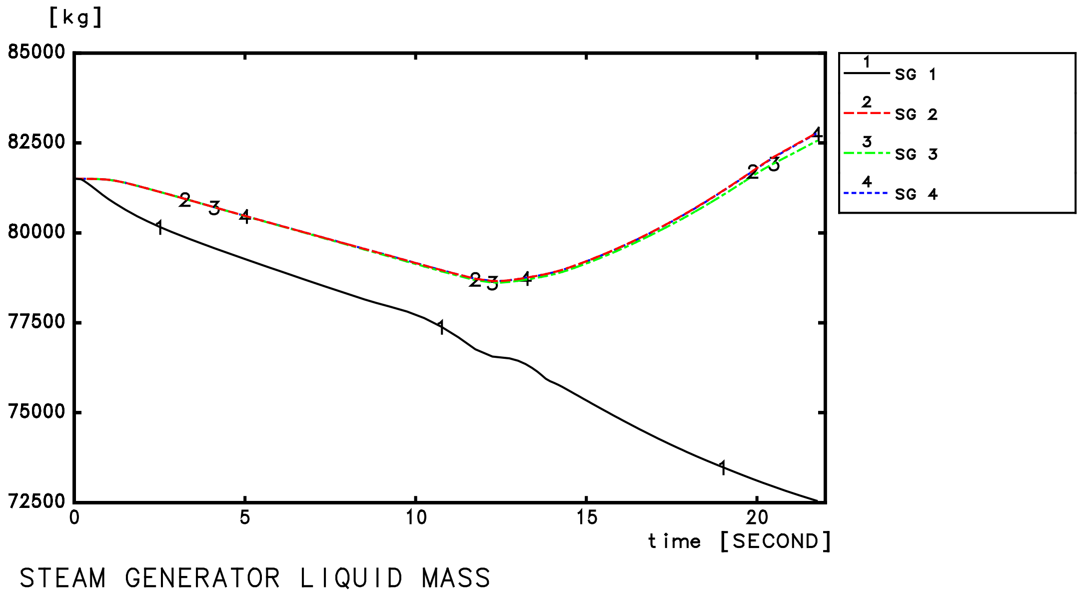


STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 97

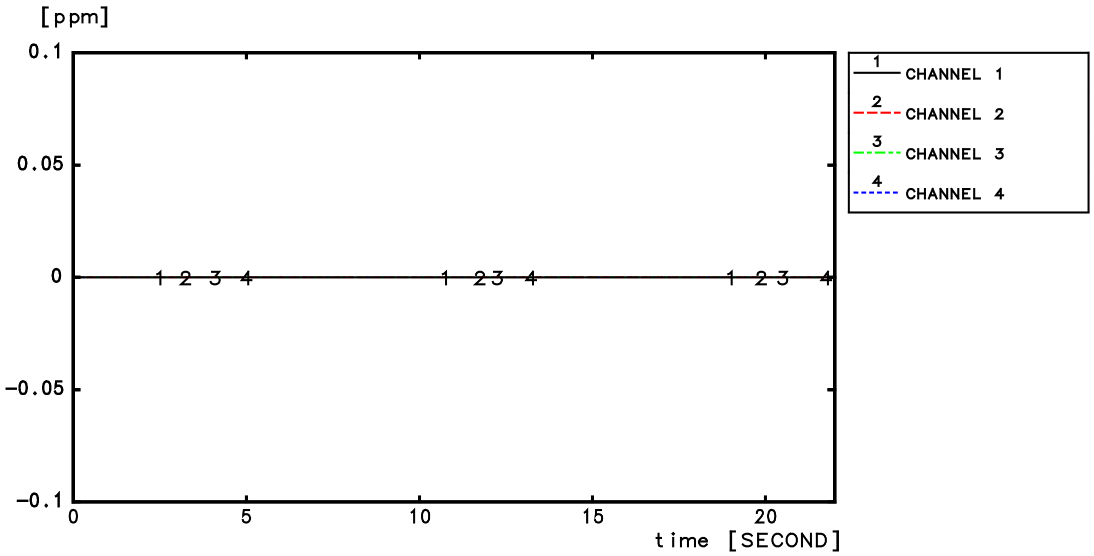
Break Spectrum at Full Power, 1400 cm² - Worst break - Main Feed Water Flow Rate and SG Liquid Mass



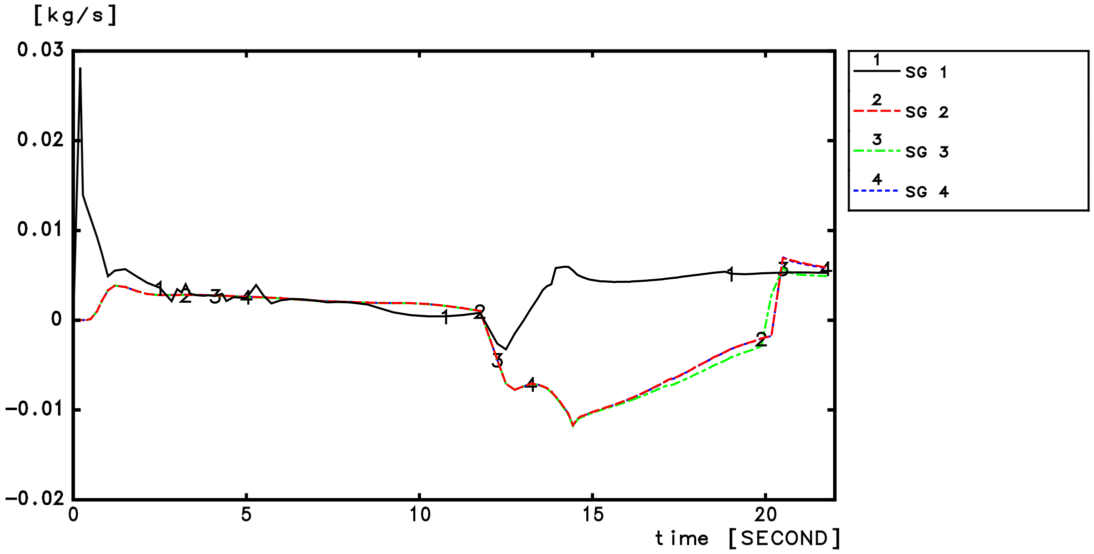


SECTION 14.5.2 - FIGURE 98

Break Spectrum at Full Power, 1400 cm² - Worst break - Boron Concentration and ASG [EFWS] Flow Rate



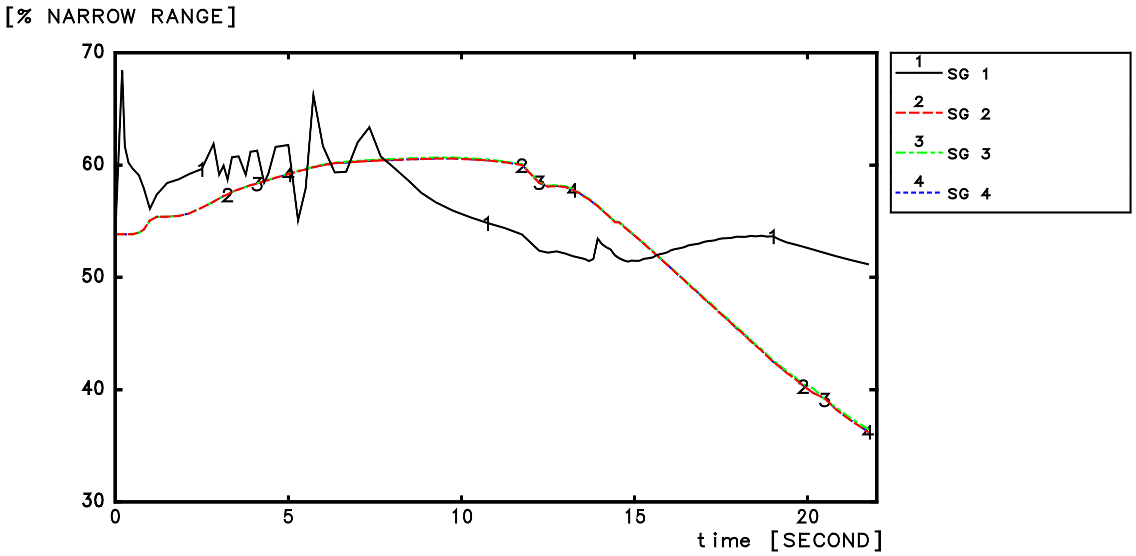
BORON CONCENTRATION AT CORE ACTIVE PART INLET



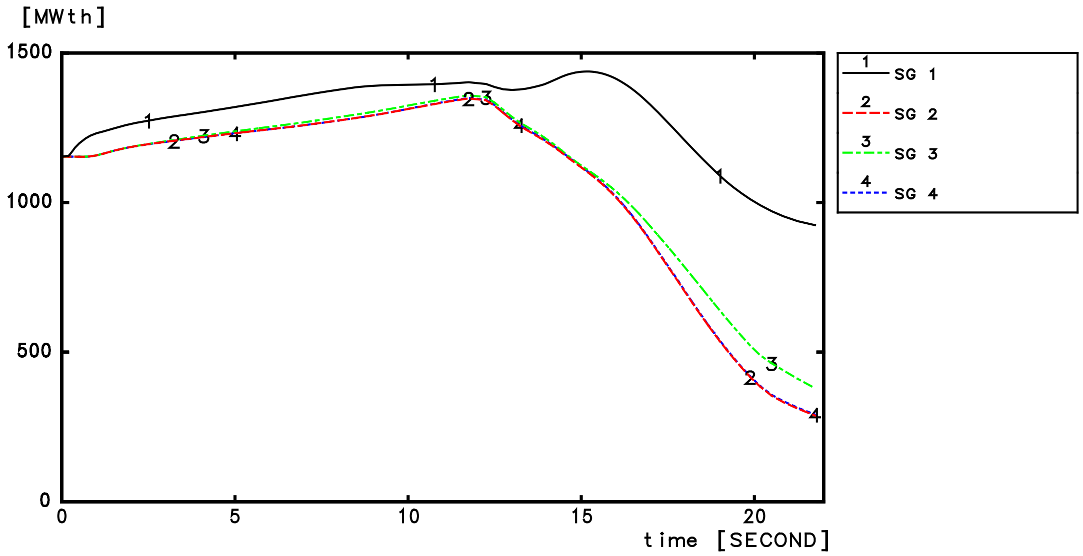
ASG FLOWRATE

SECTION 14.5.2 - FIGURE 99

Break Spectrum at Full Power, 1400 cm² - Worst break - SG Narrow Range Level and SG Power Exchanged



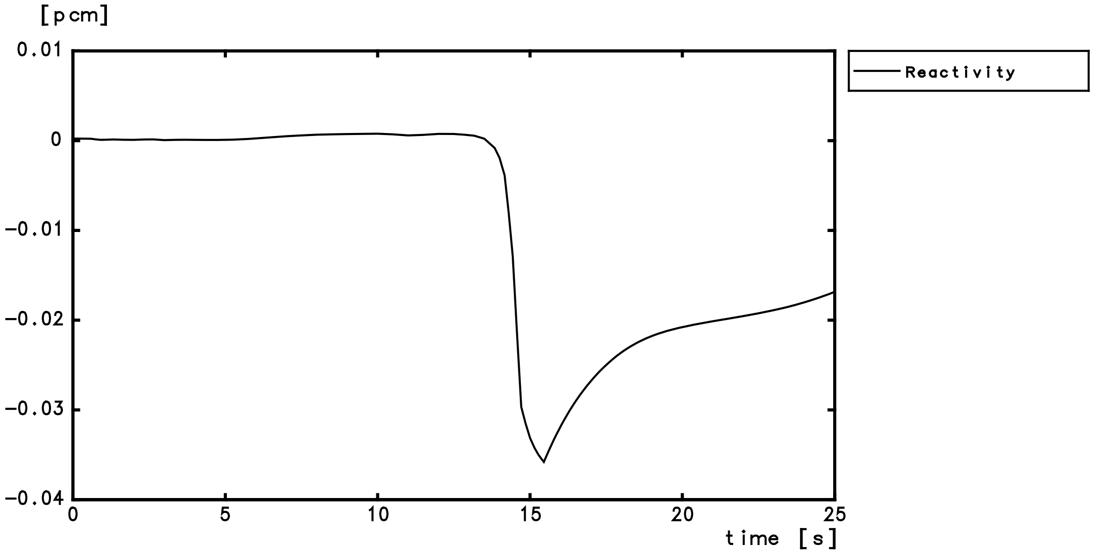
STEAM GENERATOR NARROW RANGE LEVEL



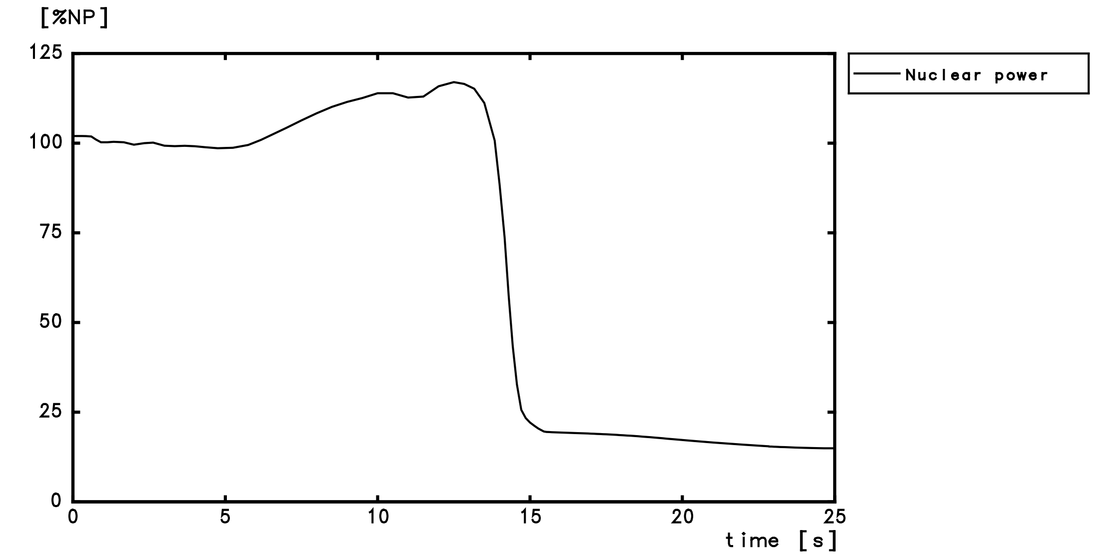
SG POWER EXCHANGED

SECTION 14.5.2 - FIGURE 100

Break Spectrum at Full Power, 1300 cm² - Corresponding to a Steam Flow Limiter - Reactivity and Nuclear Power



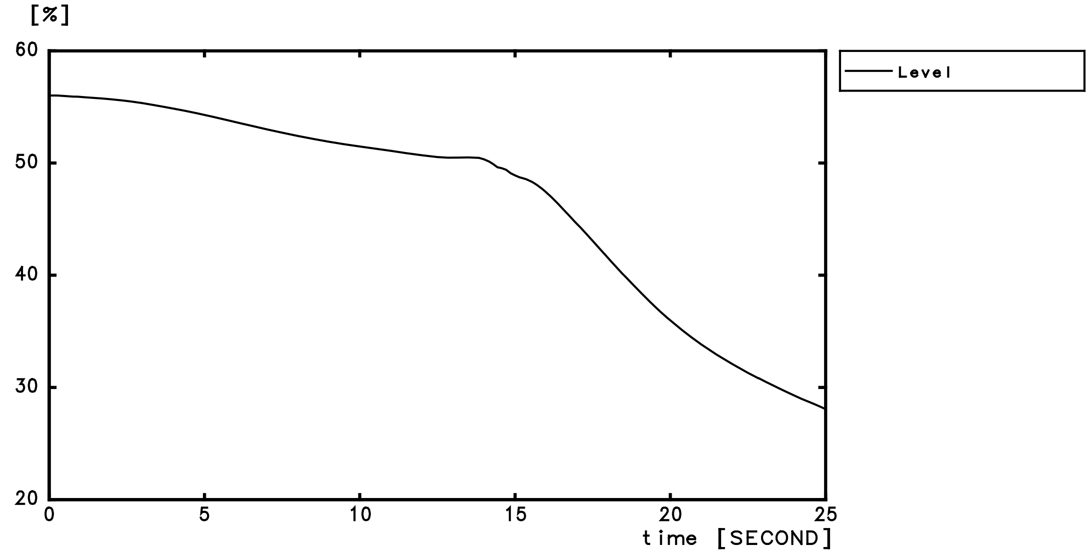
REACTIVITY



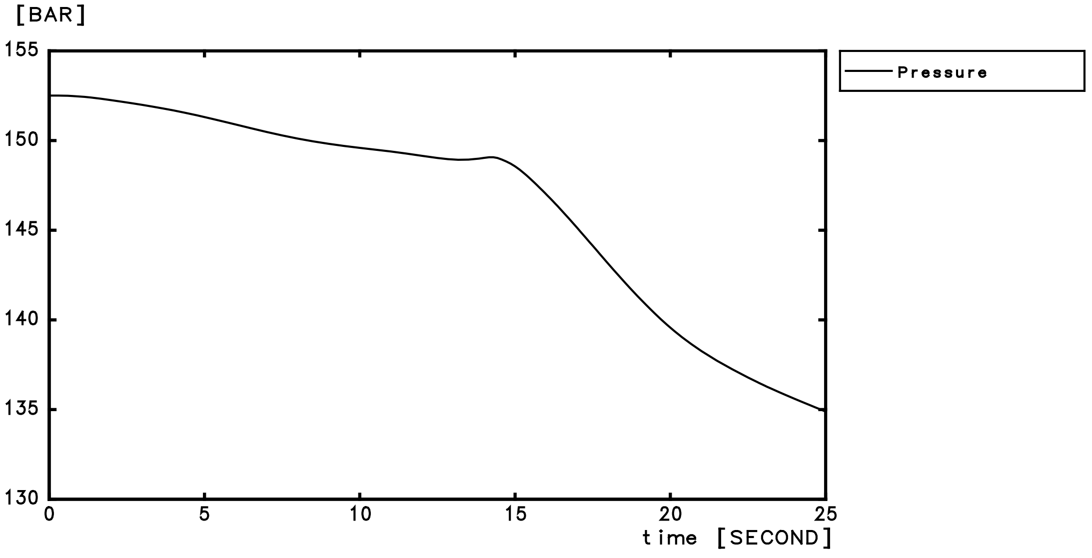
CORE POWER

SECTION 14.5.2 - FIGURE 101

Break Spectrum at Full Power, 1300 cm² - Corresponding to a Steam Flow Limiter - Pressuriser Level and Pressuriser Pressure



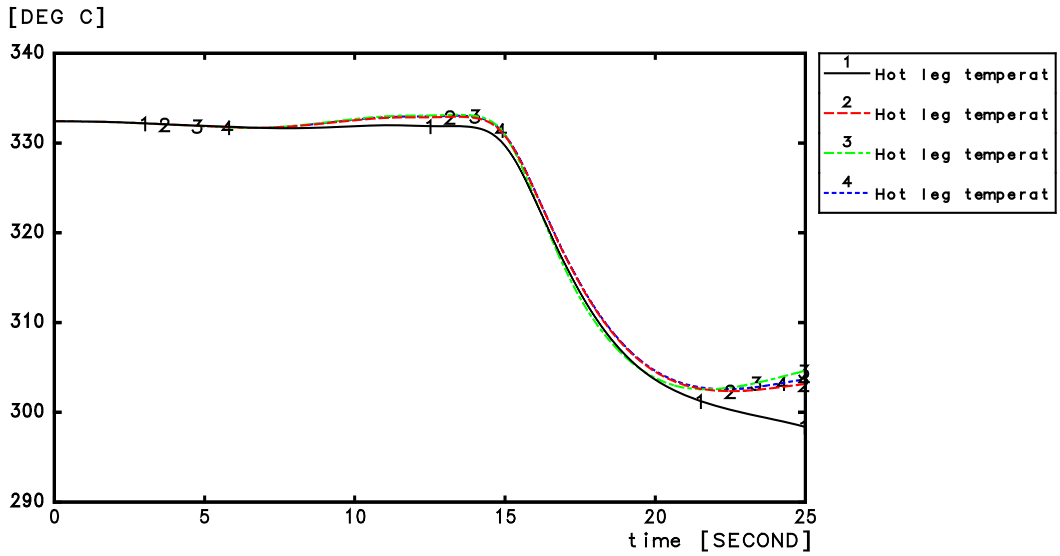
PRESSURIZER LEVEL



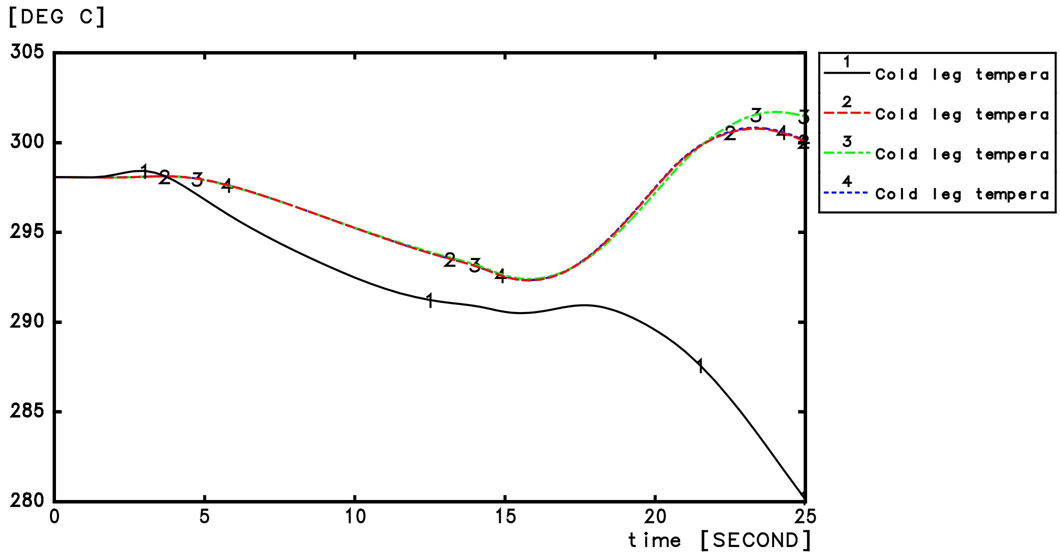
PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 102

Break Spectrum at Full Power, 1300 cm² - Corresponding to a Steam Flow Limiter - Hot and Cold Leg Temperature



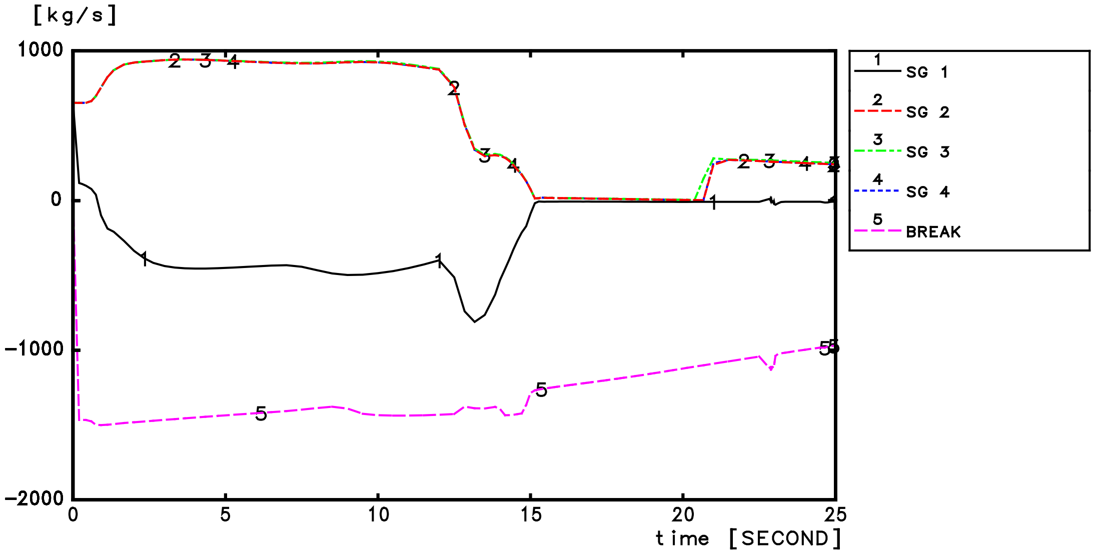
HOT LEG TEMPERATURE



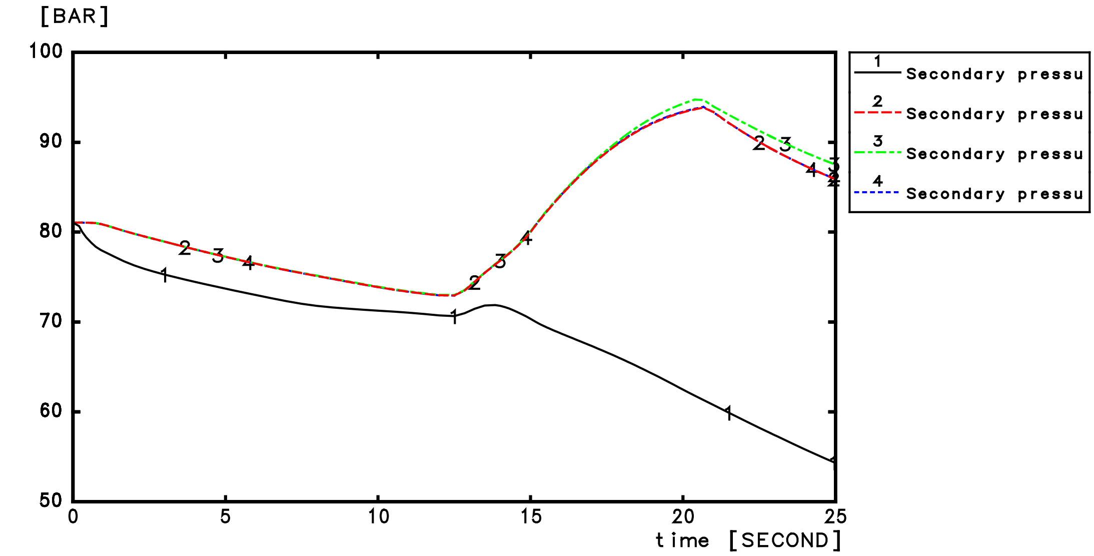
COLD LEG TEMPERATURE

SECTION 14.5.2 - FIGURE 103

Break Spectrum at Full Power, 1300 cm² - Corresponding to a Steam Flow Limiter - SG Vapour Mass Flow Rate and SG Pressure



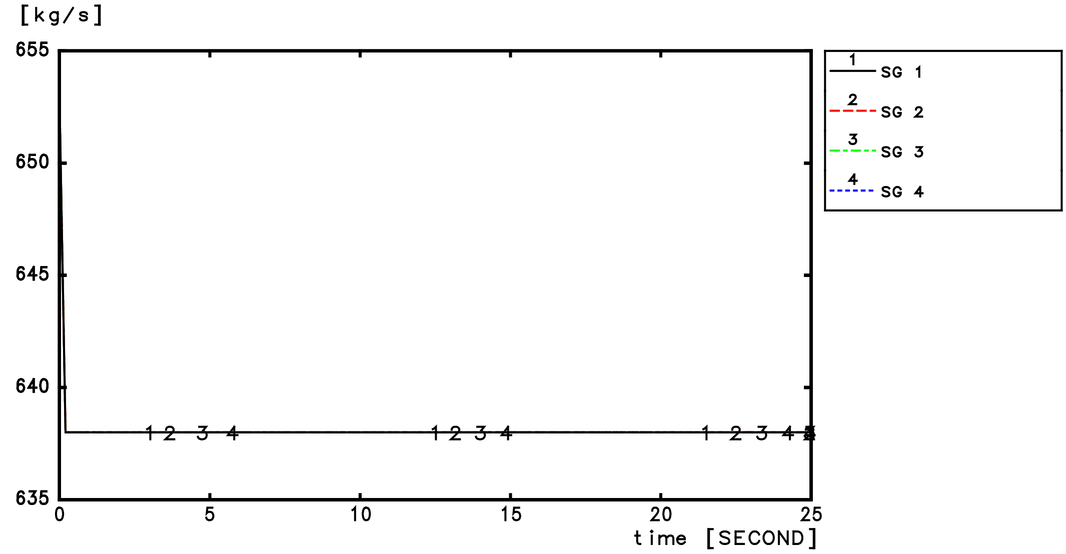
VAPOR MASS FLOWRATE



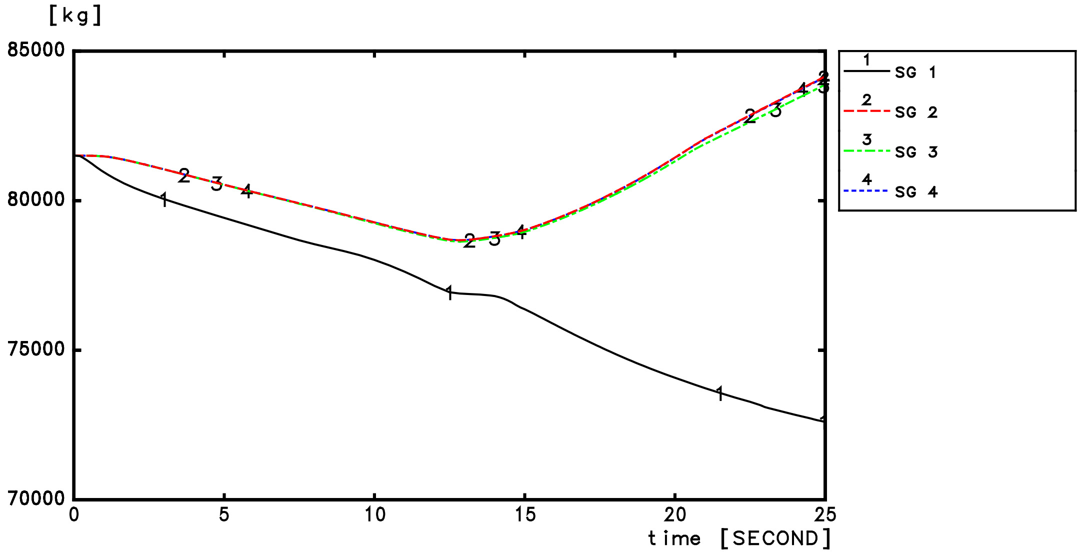
STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 104

Break Spectrum at Full Power, 1300 cm² - Corresponding to a Steam Flow Limiter - SG Main Feed Water Flow Rate and Liquid Mass



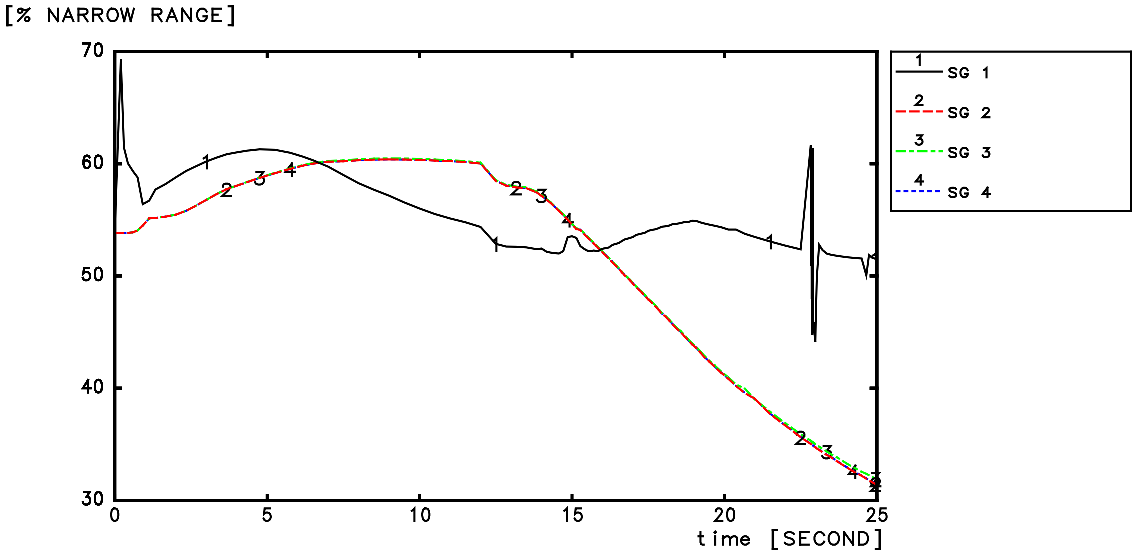
MFW FLOWRATE



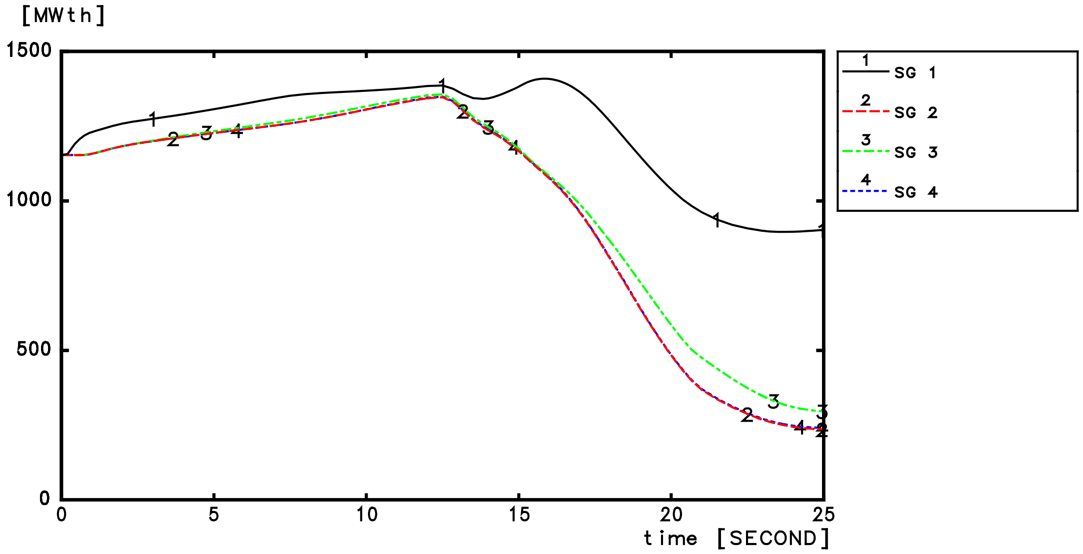
STEAM GENERATOR LIQUID MASS

SECTION 14.5.2 - FIGURE 105

Break Spectrum at Full Power, 1300 cm² - Corresponding to a Steam Flow Limiter - SG Narrow Level and SG Exchanged Power



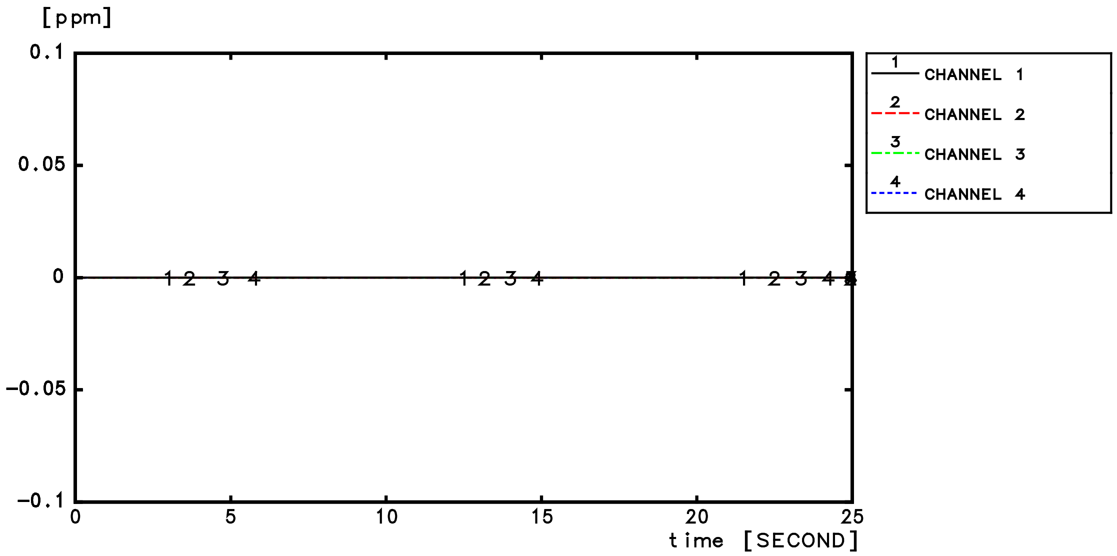
STEAM GENERATOR NARROW RANGE LEVEL



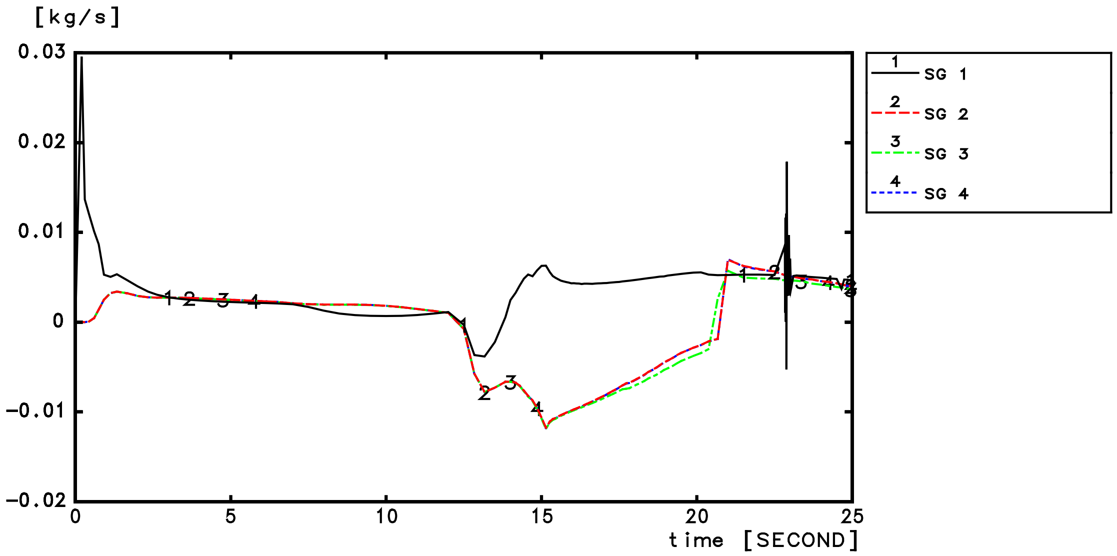
SG POWER EXCHANGED

SECTION 14.5.2 - FIGURE 106

Break Spectrum at Full Power, 1300 cm² - Corresponding to a Steam Flow Limiter - Boron Concentration and ASG [EFWS] Flow Rate



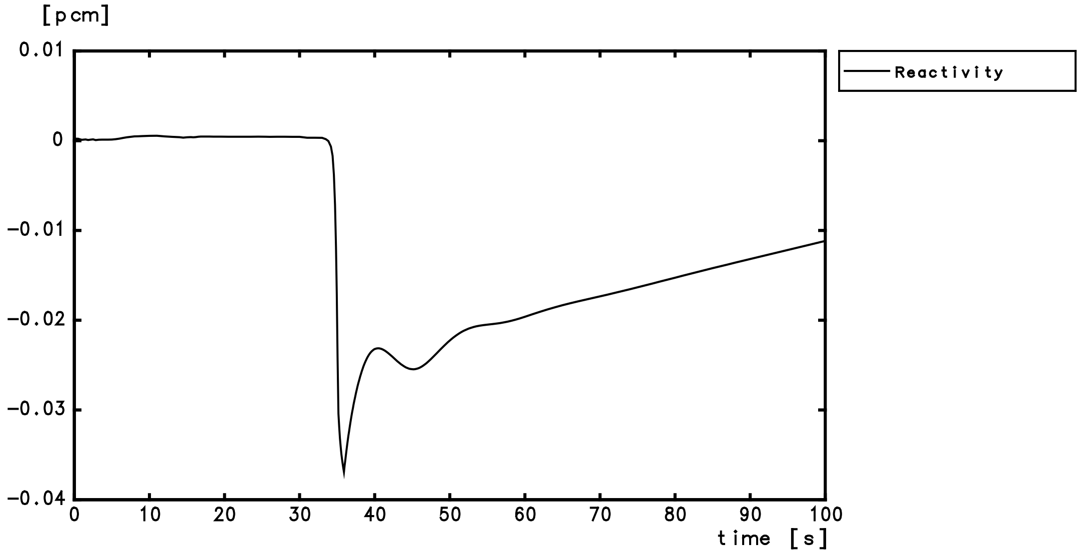
BORON CONCENTRATION AT CORE ACTIVE PART INLET



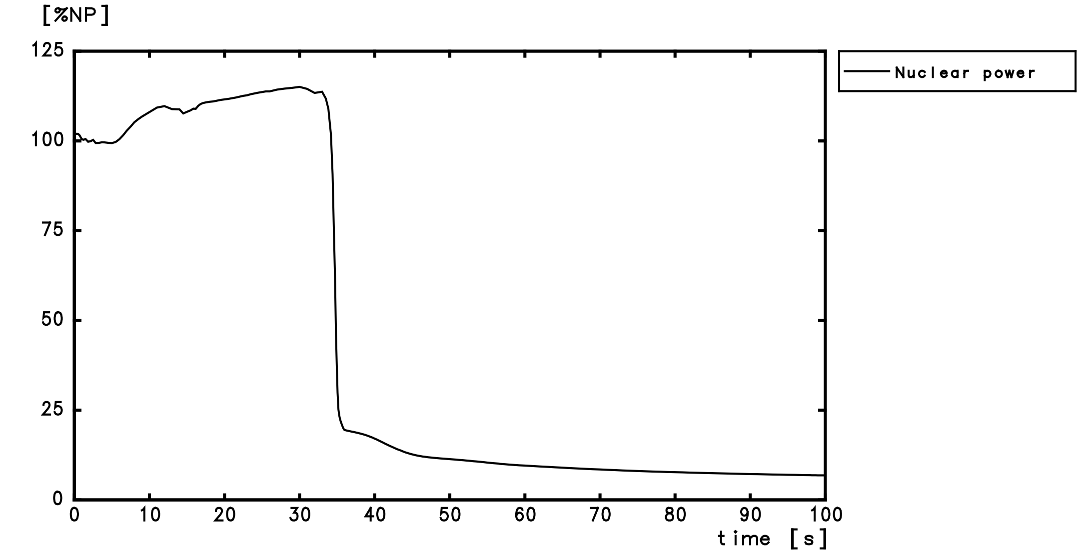
ASG FLOWRATE

SECTION 14.5.2 - FIGURE 107

Break Spectrum at Full Power, 700 cm² - Reactivity and Nuclear Power



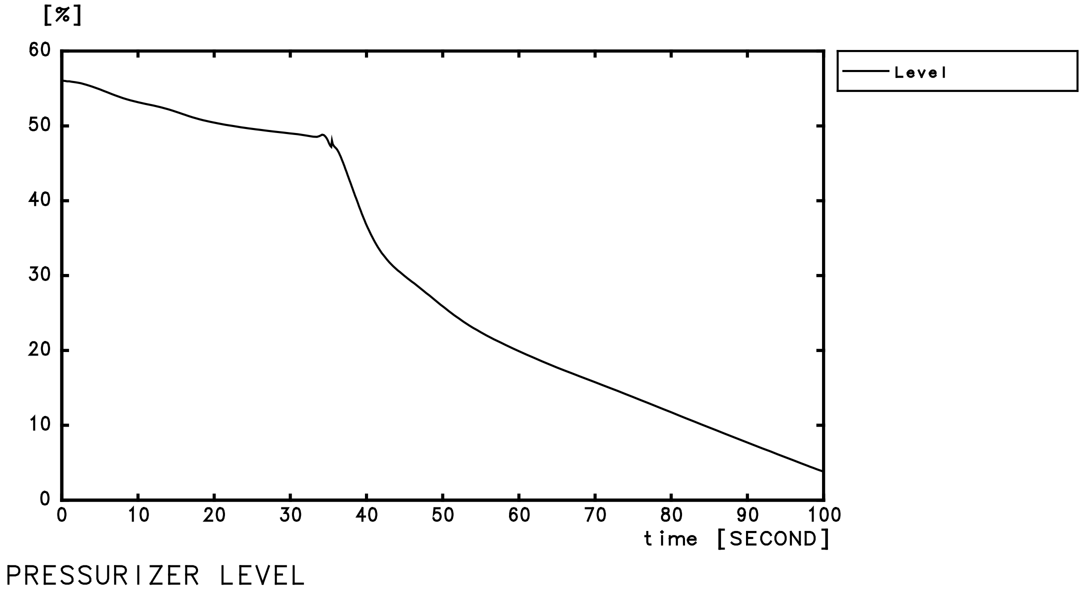
REACTIVITY

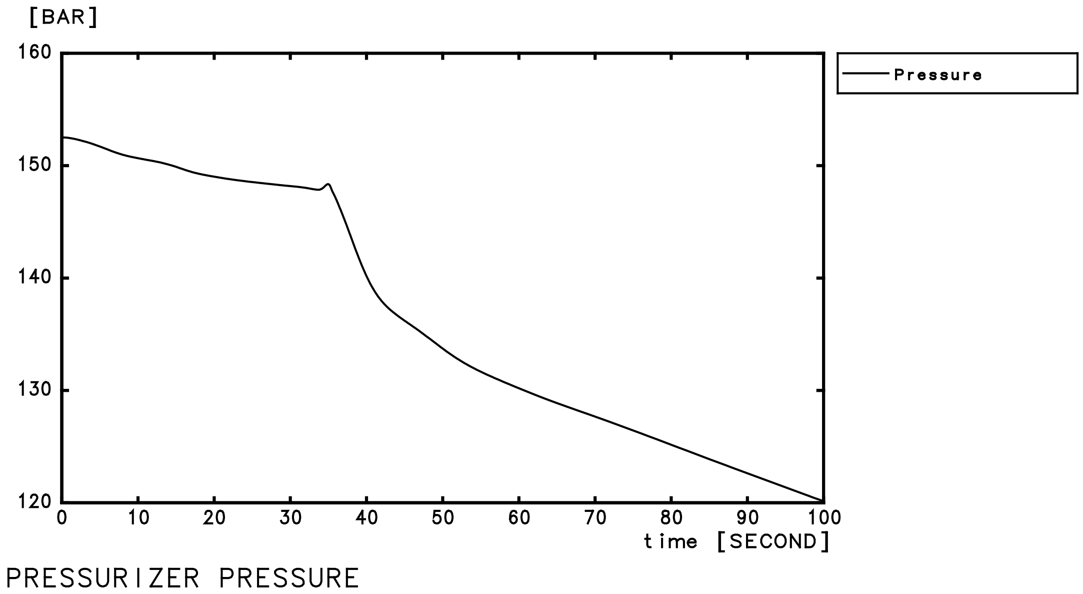


CORE POWER

SECTION 14.5.2 - FIGURE 108

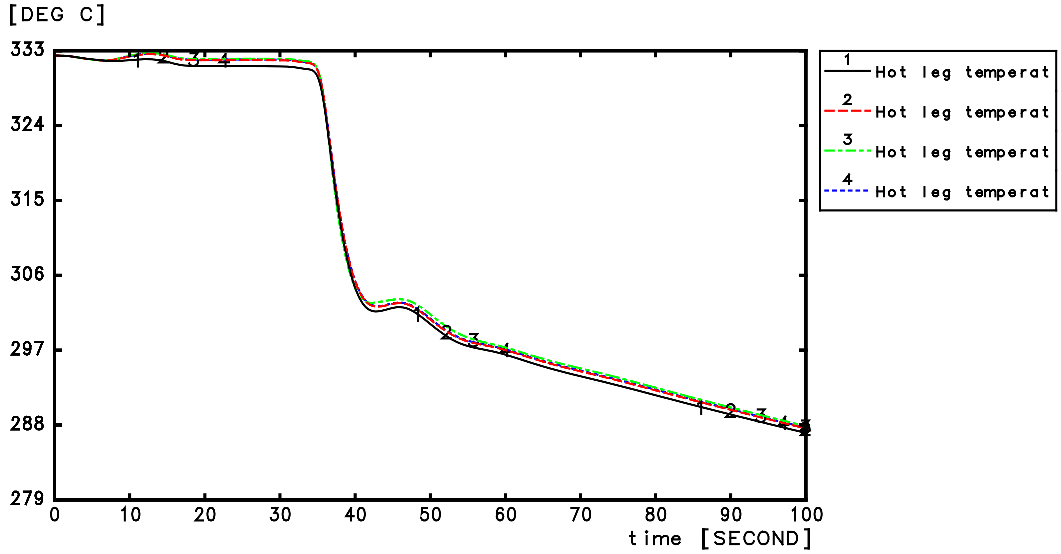
Break Spectrum at Full Power, 700 cm² - Pressuriser level and Pressuriser Pressure



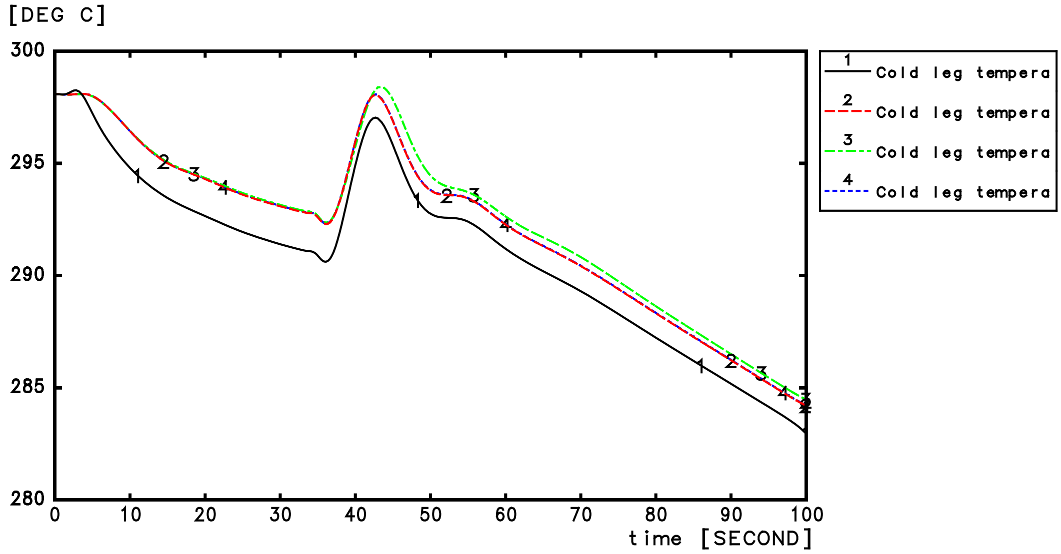


SECTION 14.5.2 - FIGURE 109

Break Spectrum at Full Power, 700 cm² - Hot and Cold Leg Temperature



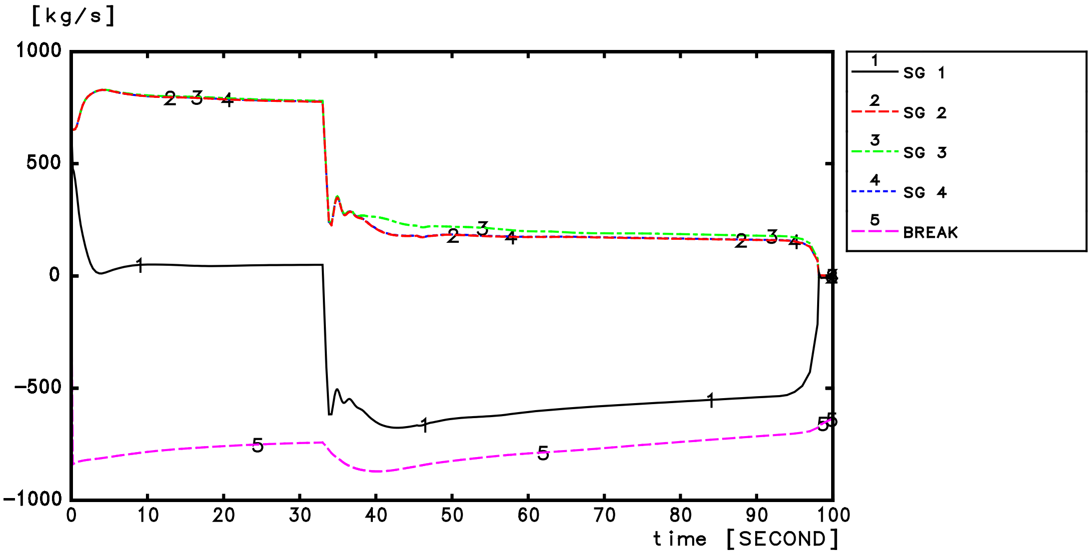
HOT LEG TEMPERATURE



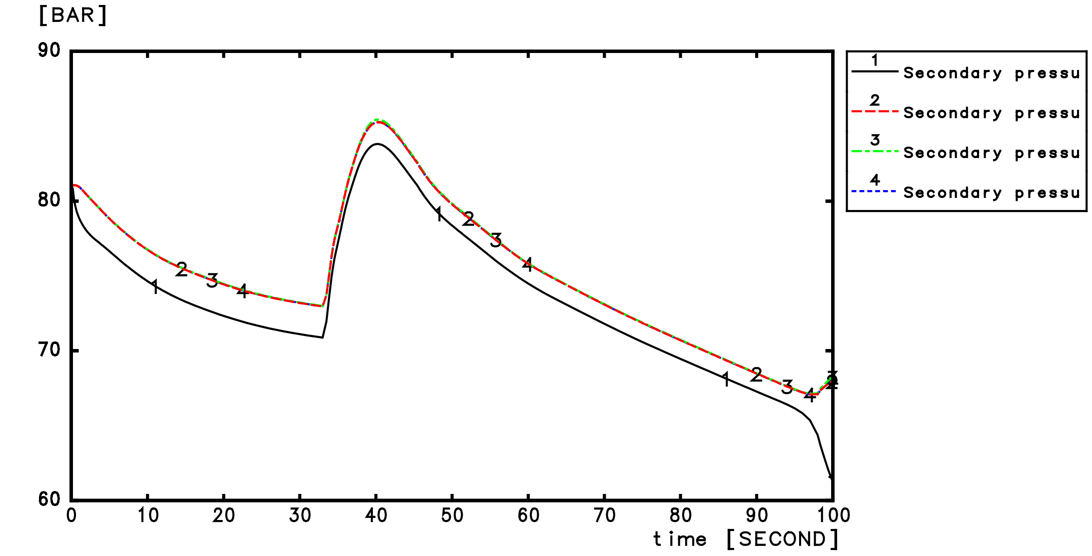
COLD LEG TEMPERATURE

SECTION 14.5.2 - FIGURE 110

Break Spectrum at Full Power, 700 cm² - SG Vapour Mass Flow Rate and SG Pressure



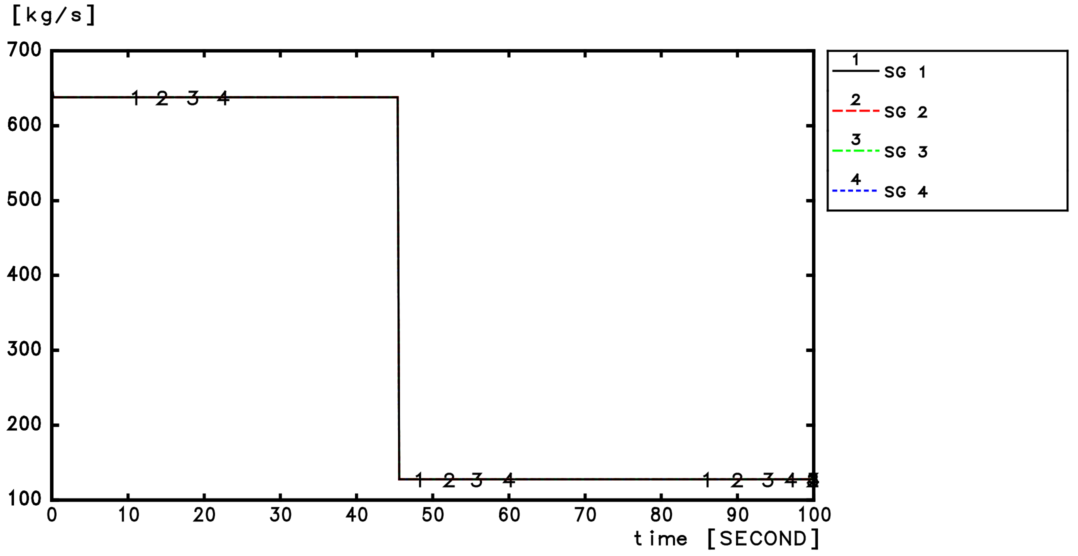
VAPOR MASS FLOWRATE



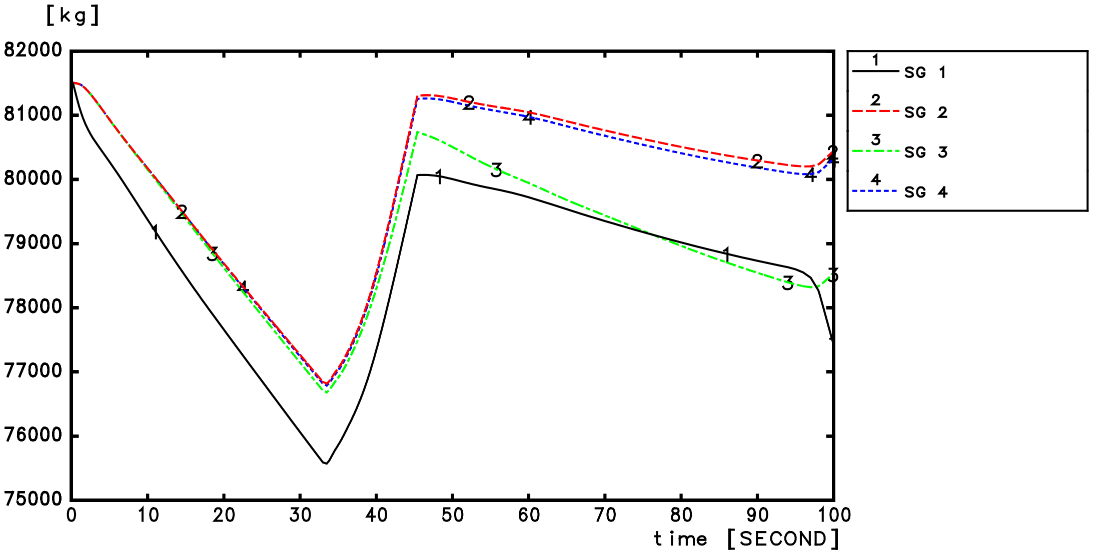
STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 111

Break Spectrum at Full Power, 700 cm² - SG Main Feed Water Flow Rate and Liquid Mass



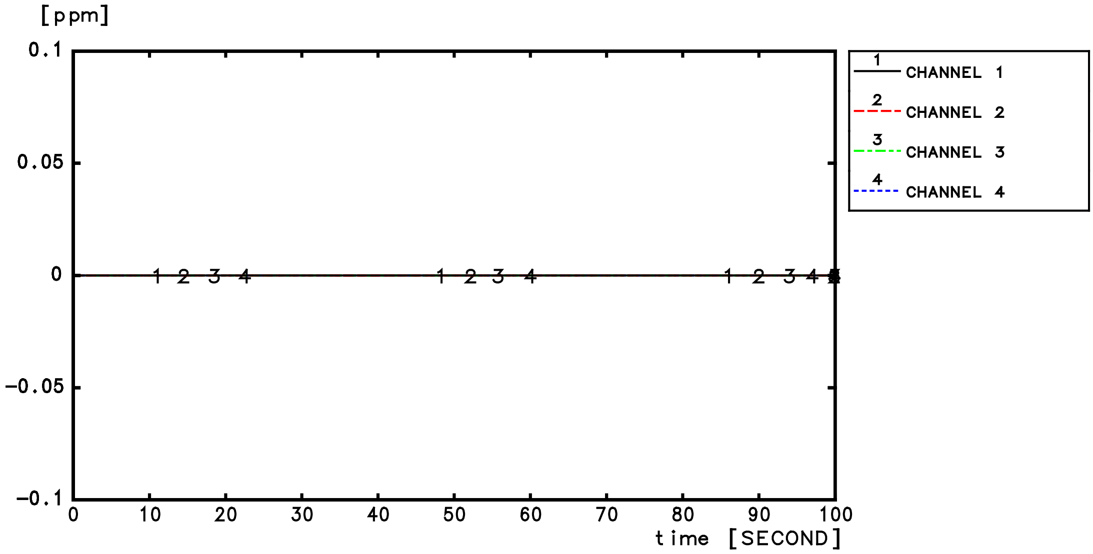
MFW FLOWRATE



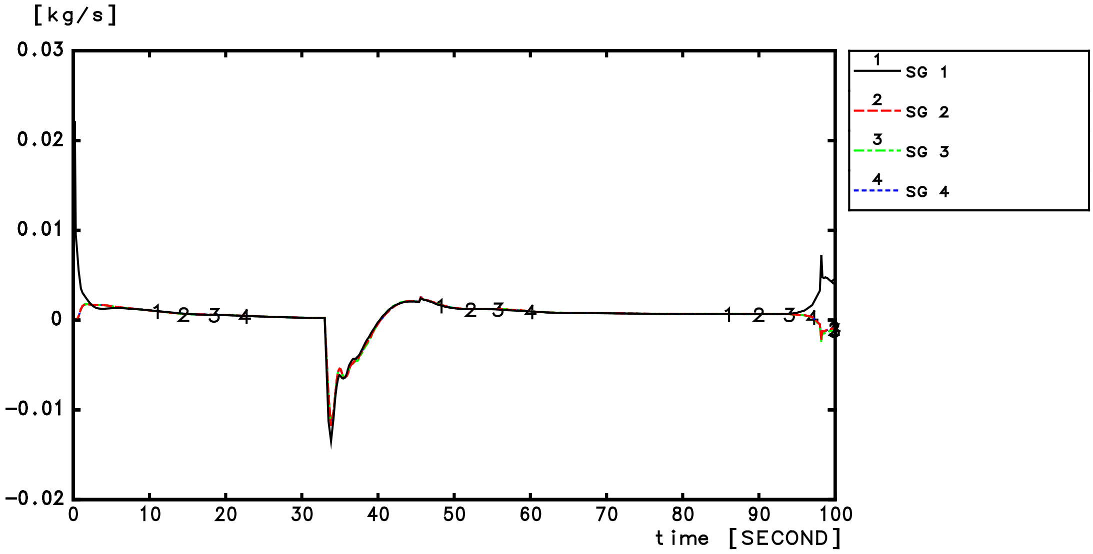
STEAM GENERATOR LIQUID MASS

SECTION 14.5.2 - FIGURE 112

Break Spectrum at Full Power, 700 cm² - Boron Concentration and ASG [EFWS] Flow Rate



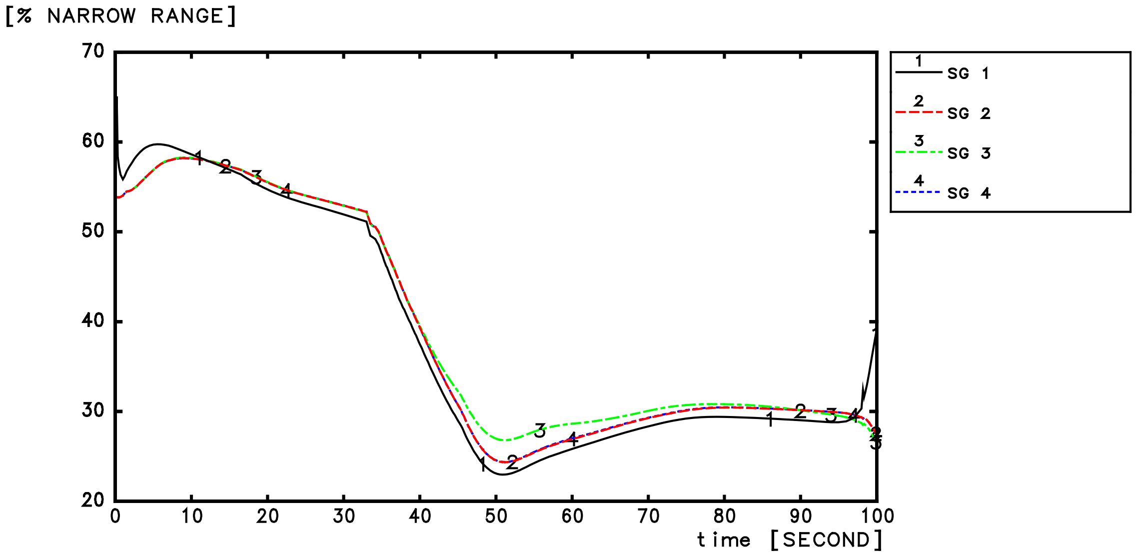
BORON CONCENTRATION AT CORE ACTIVE PART INLET



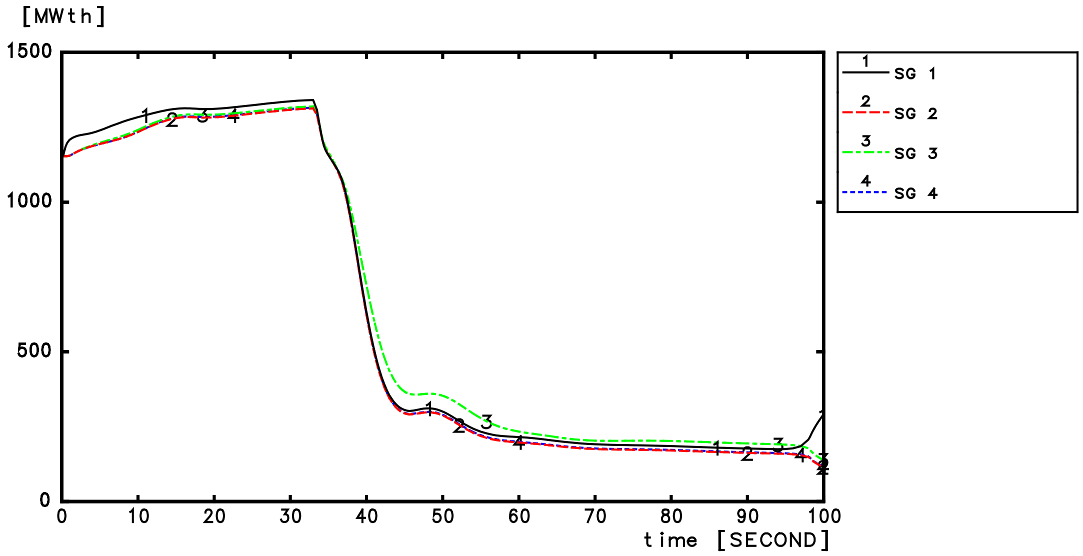
ASG FLOWRATE

SECTION 14.5.2 - FIGURE 113

Break Spectrum at Full Power, 700 cm² - SG Narrow Range Level and SG Exchanged Power



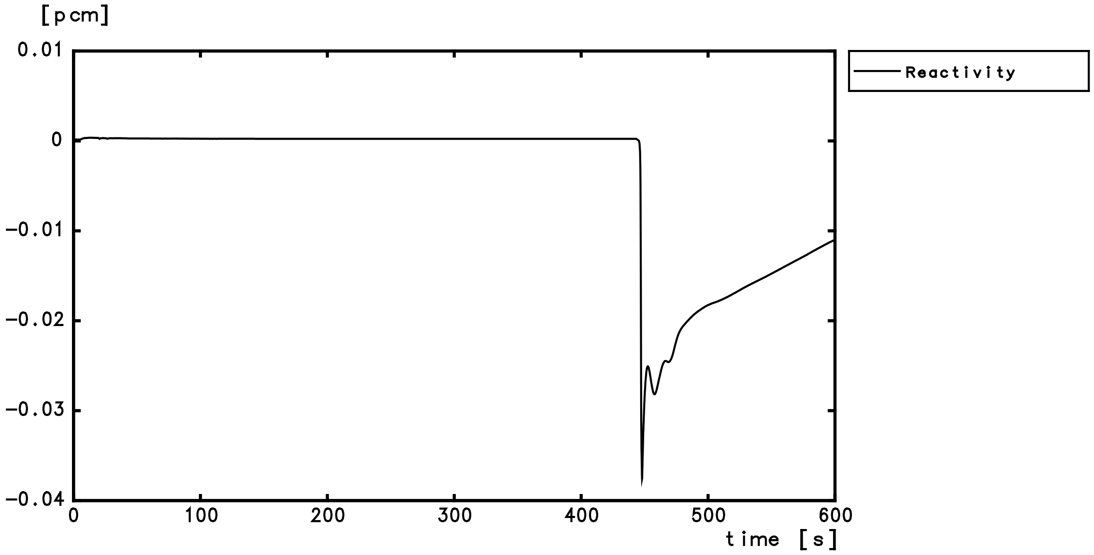
STEAM GENERATOR NARROW RANGE LEVEL



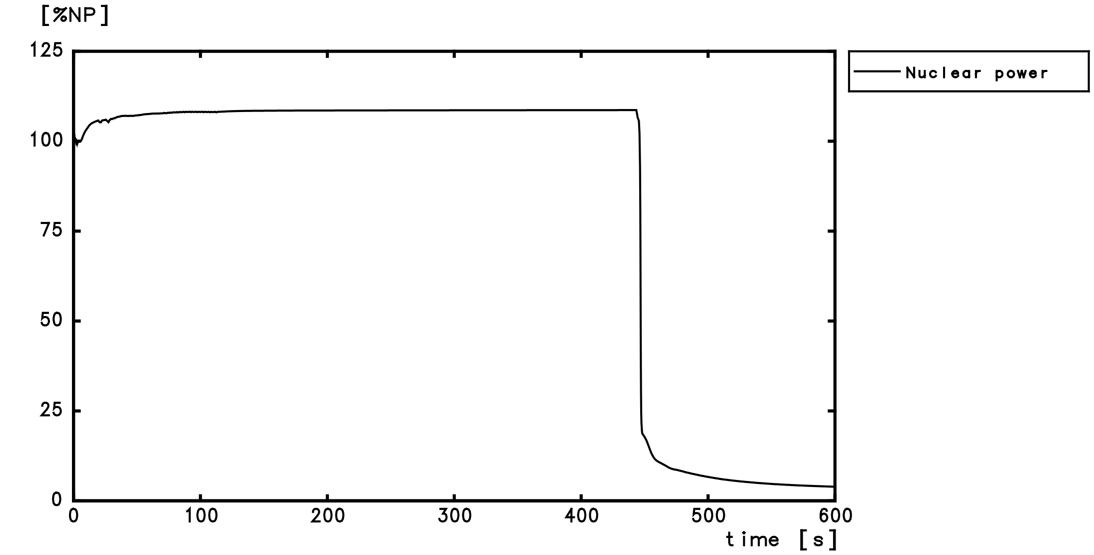
SG POWER EXCHANGED

SECTION 14.5.2 - FIGURE 114

Break Spectrum at Full Power, 253 cm² - Spurious MSRT Opening – Reactivity and Nuclear Power



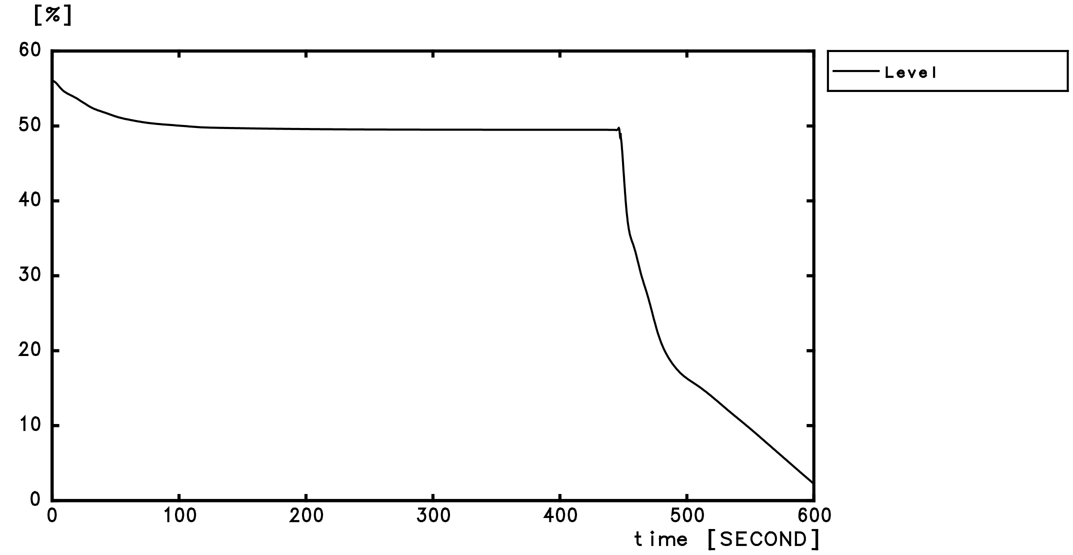
REACTIVITY



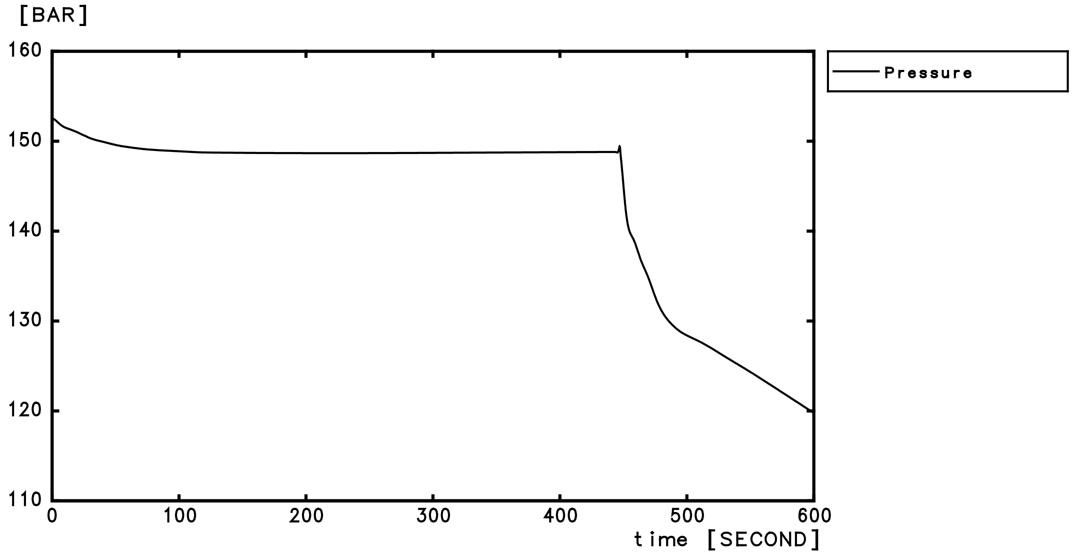
CORE POWER

SECTION 14.5.2 - FIGURE 115

Break Spectrum at Full Power, 253 cm² - Spurious MSRT Opening - Pressuriser Level and Pressuriser Pressure



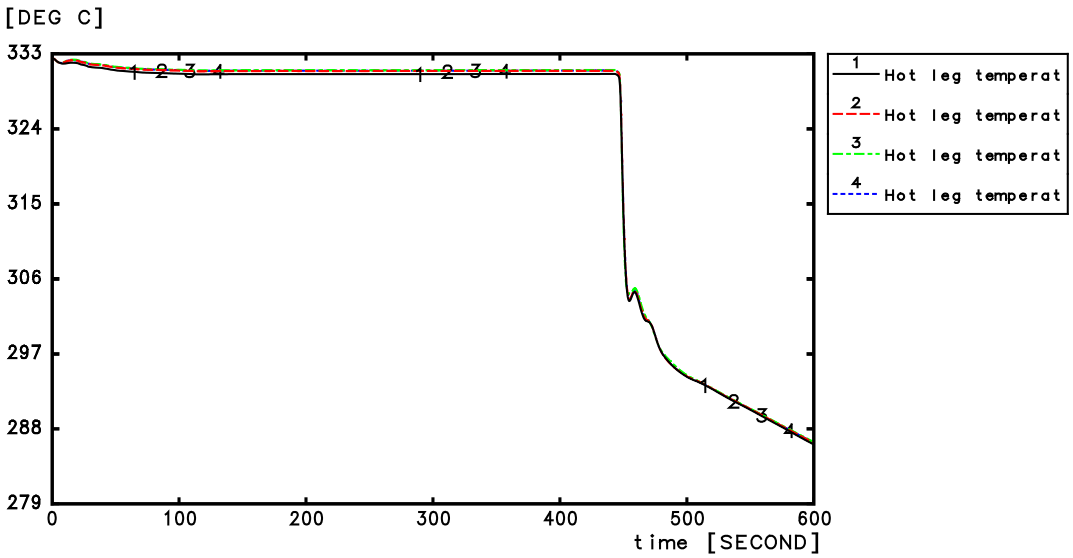
PRESSURIZER LEVEL



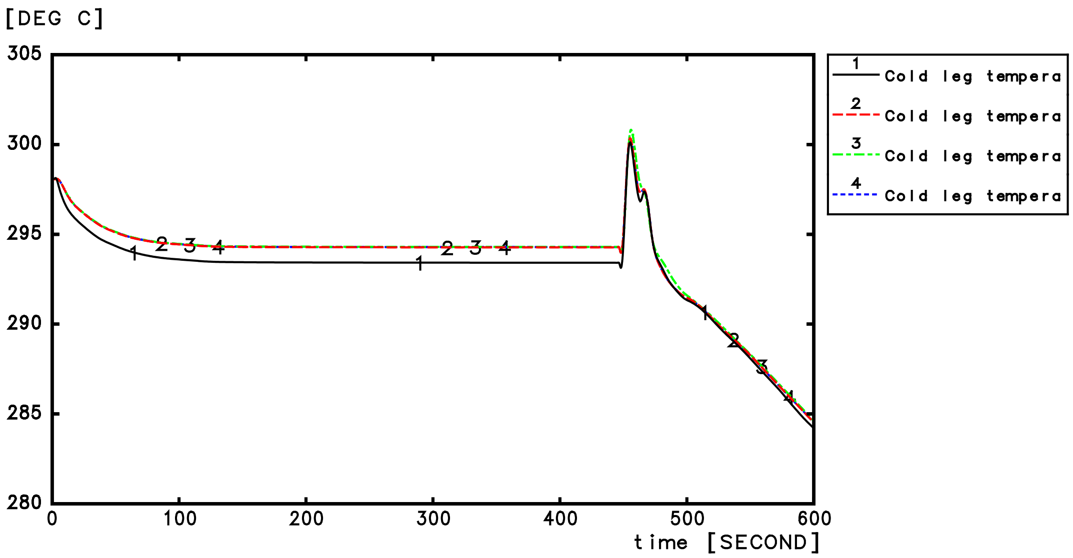
PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 116

Break Spectrum at Full Power, 253 cm² - Spurious MSRT Opening - Hot and Cold Leg Temperature



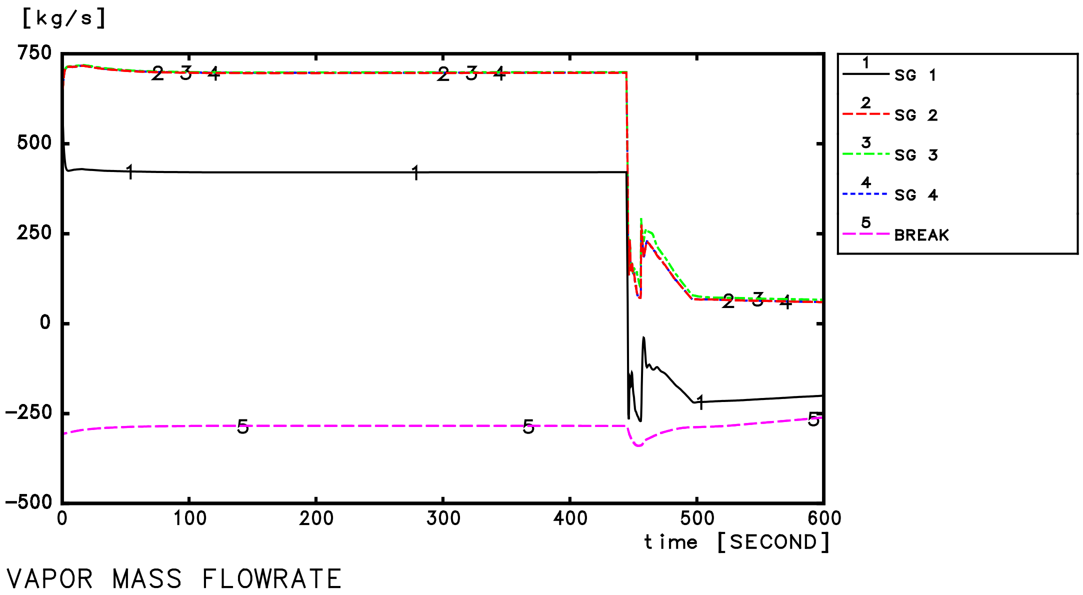
HOT LEG TEMPERATURE

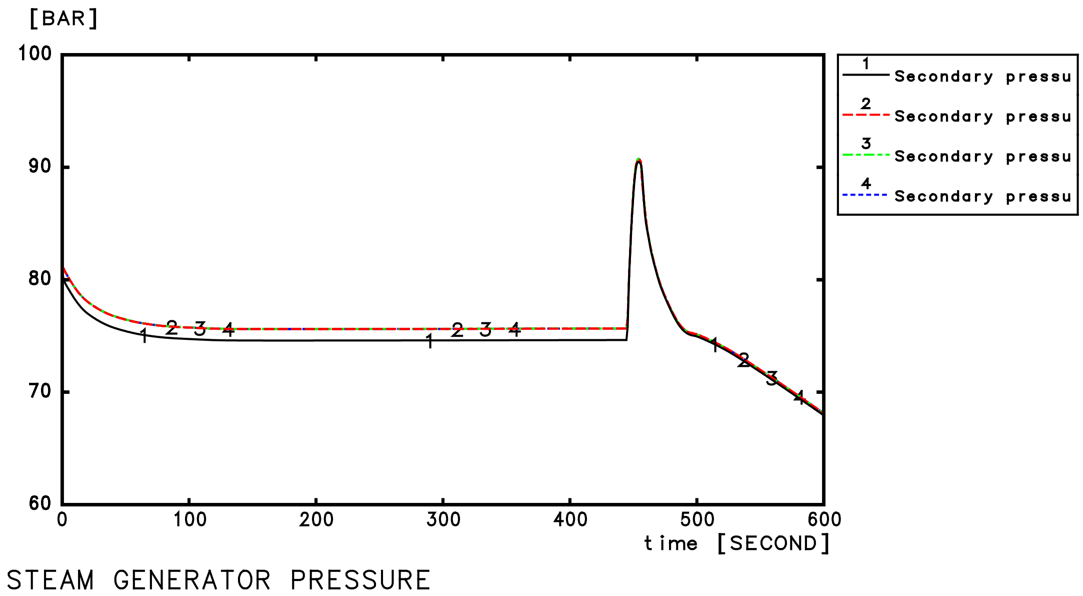


COLD LEG TEMPERATURE

SECTION 14.5.2 - FIGURE 117

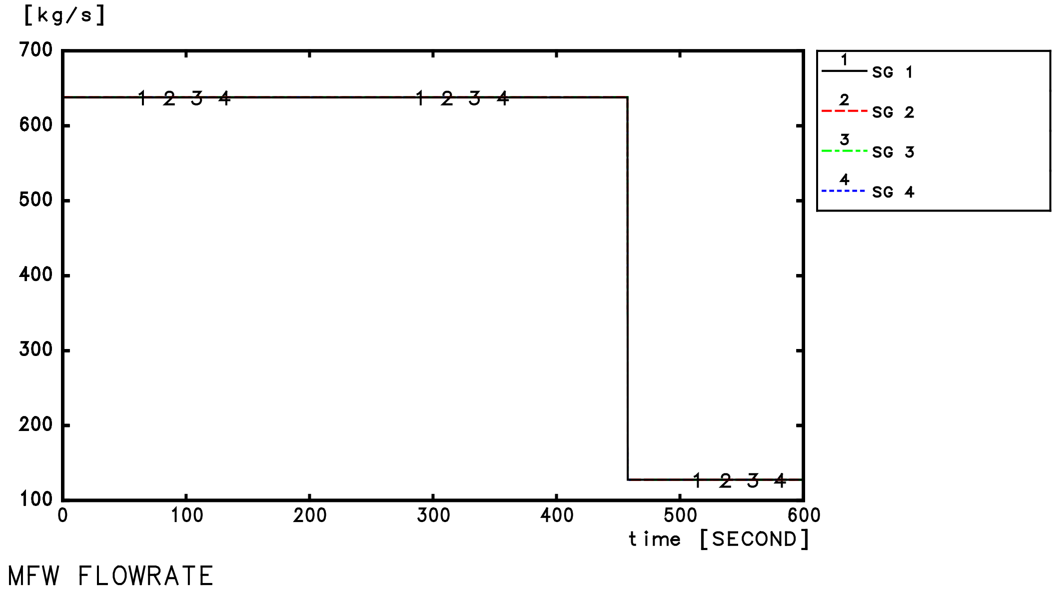
Break Spectrum at Full Power, 253 cm² - Spurious MSRT Opening – SG Vapour Mass Flow Rate and SG Pressure

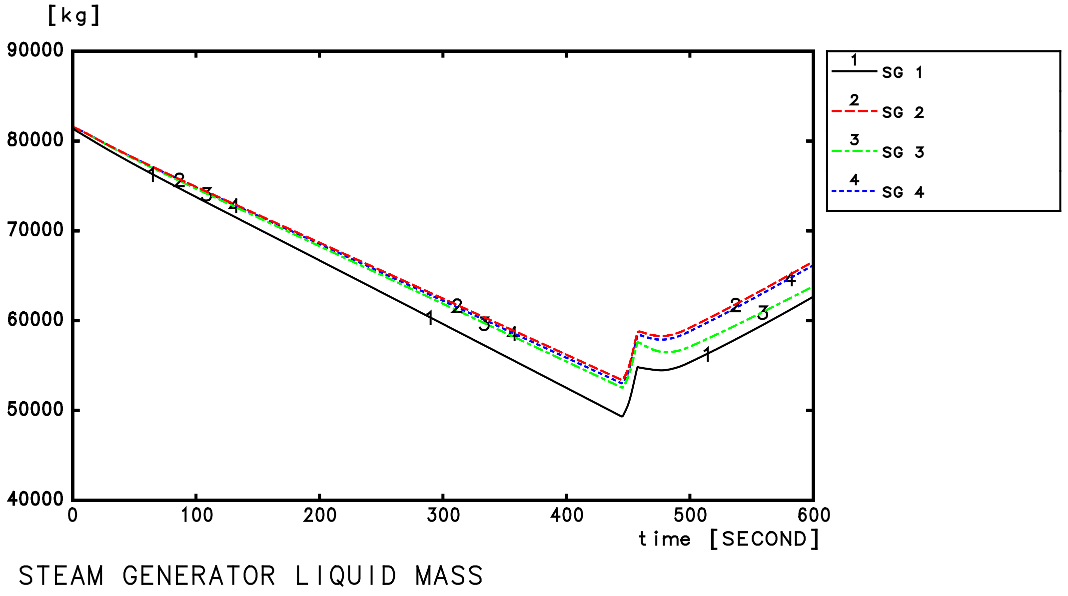




SECTION 14.5.2 - FIGURE 118

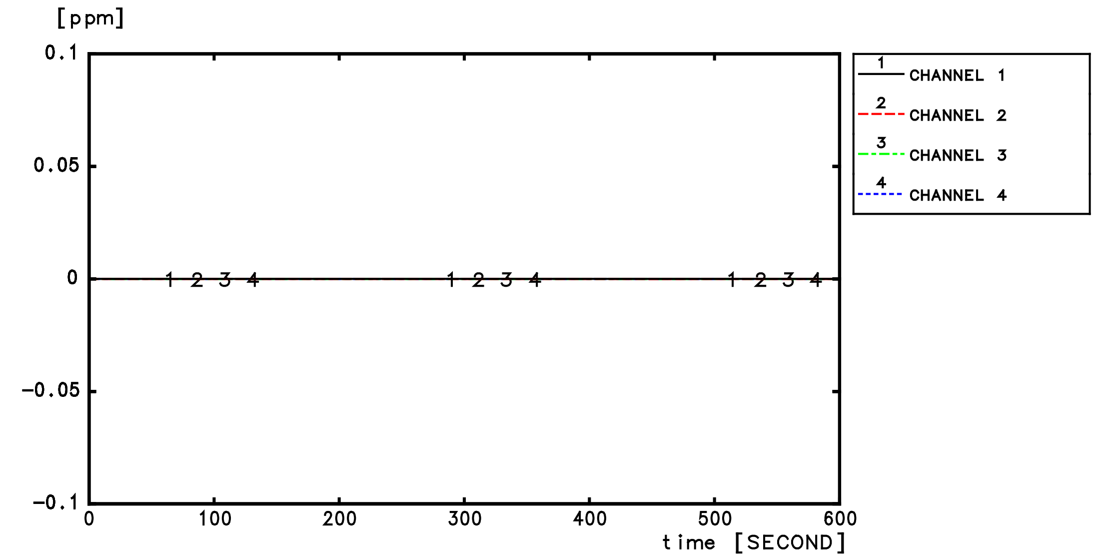
Break Spectrum at Full Power, 253 cm² - Spurious MSRT Opening - SG Main Feedwater Flow Rate and SG Liquid Mass



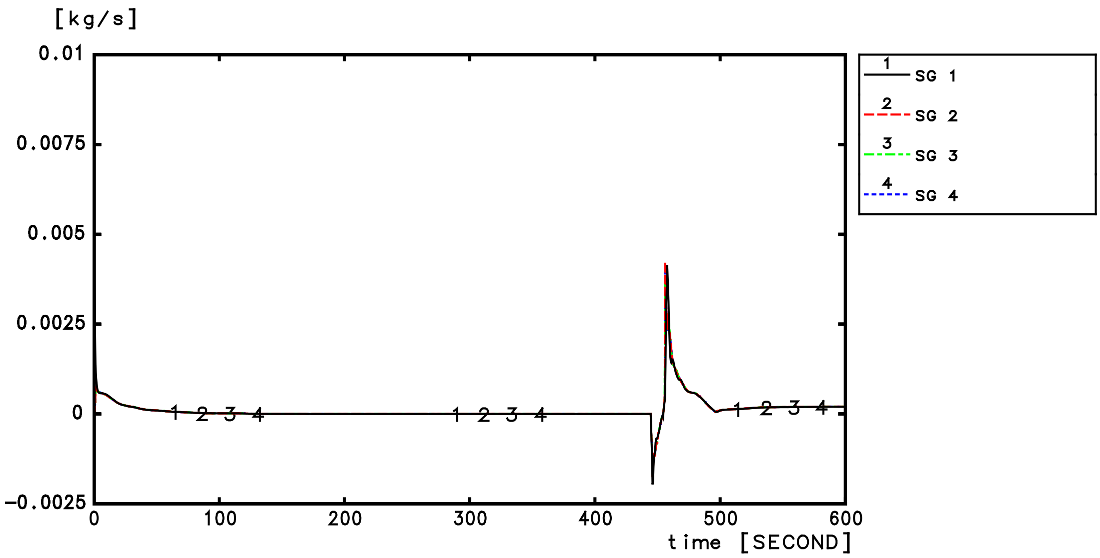


SECTION 14.5.2 - FIGURE 119

Break Spectrum at Full Power, 253 cm² - Spurious MSRT Opening – Boron Concentration and ASG [EFWS] Flow Rate



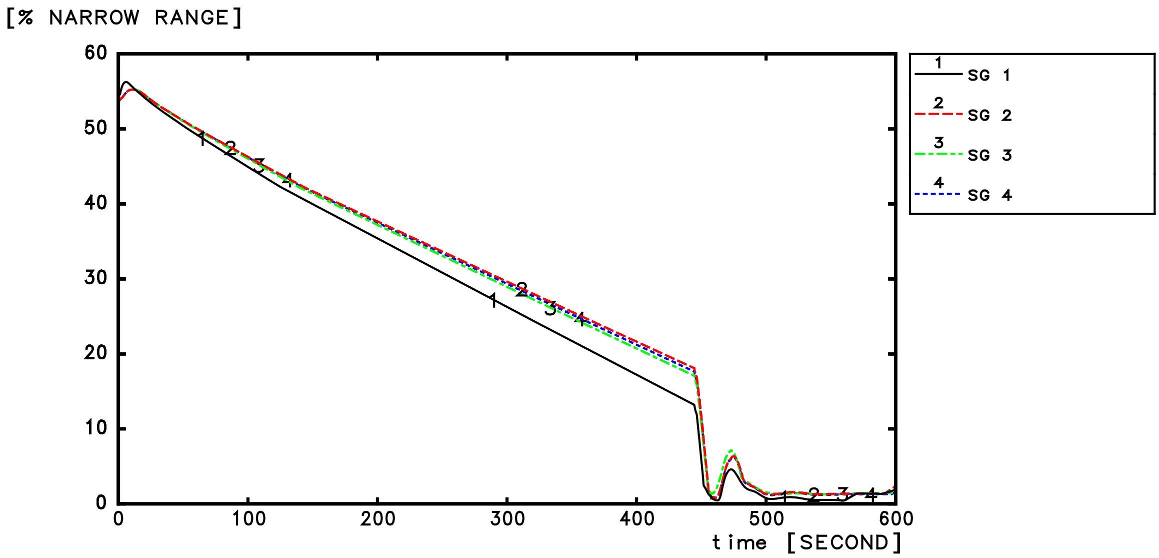
BORON CONCENTRATION AT CORE ACTIVE PART INLET



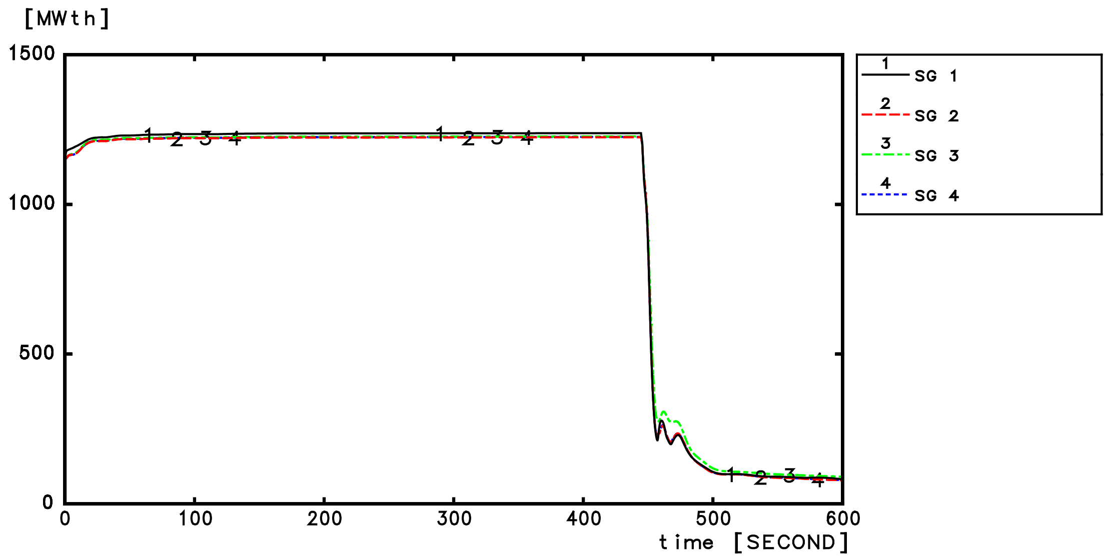
ASG FLOWRATE

SECTION 14.5.2 - FIGURE 120

Break Spectrum at Full Power, 253 cm² - Spurious MSRT Opening - SG Narrow Range Level and SG Power Exchanged



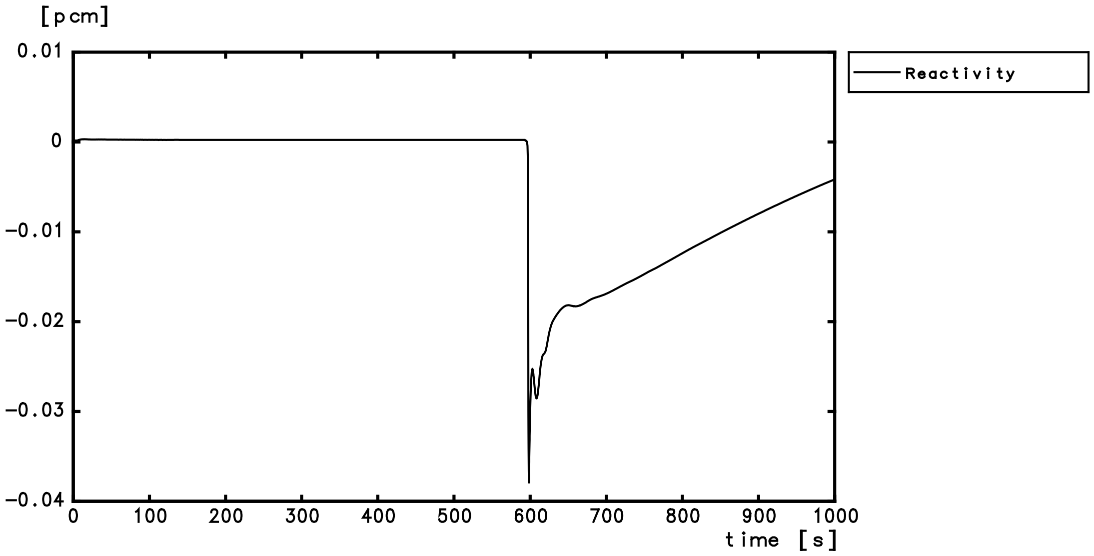
STEAM GENERATOR NARROW RANGE LEVEL



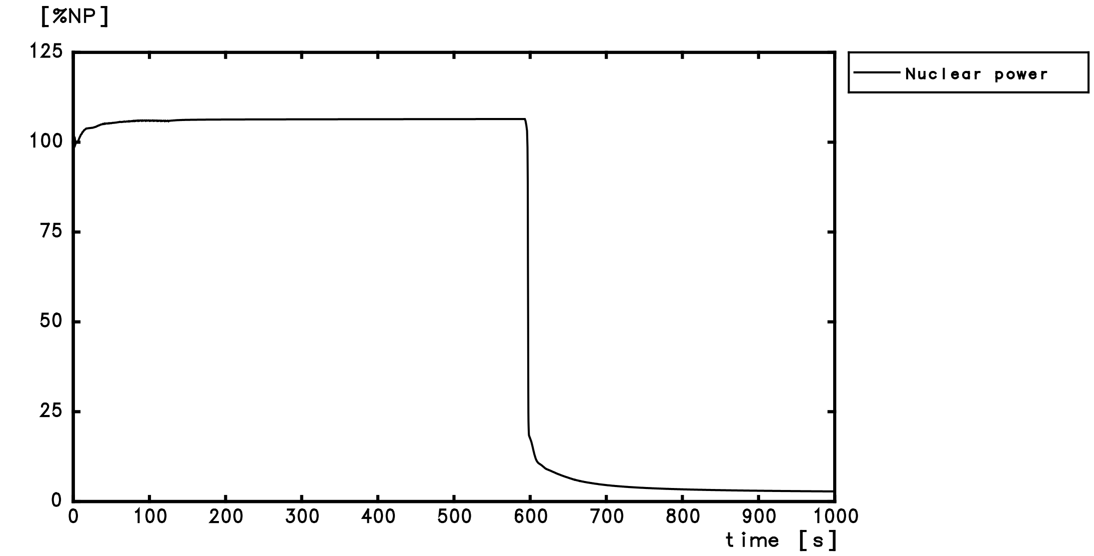
SG POWER EXCHANGED

SECTION 14.5.2 - FIGURE 121

Break Spectrum at Full Power, 158 cm² - Spurious MSIV Opening - Reactivity and Core Nuclear Power



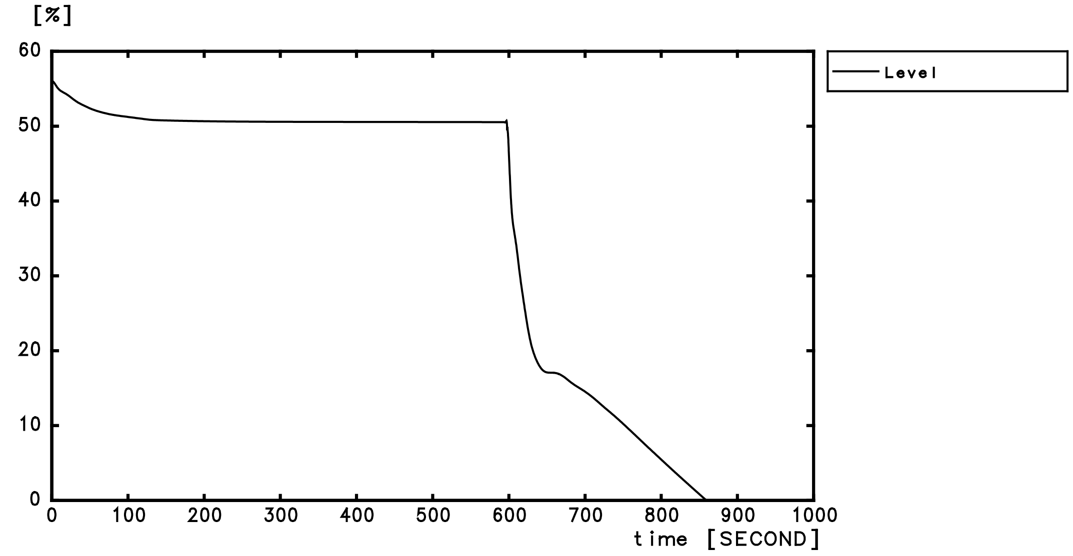
REACTIVITY



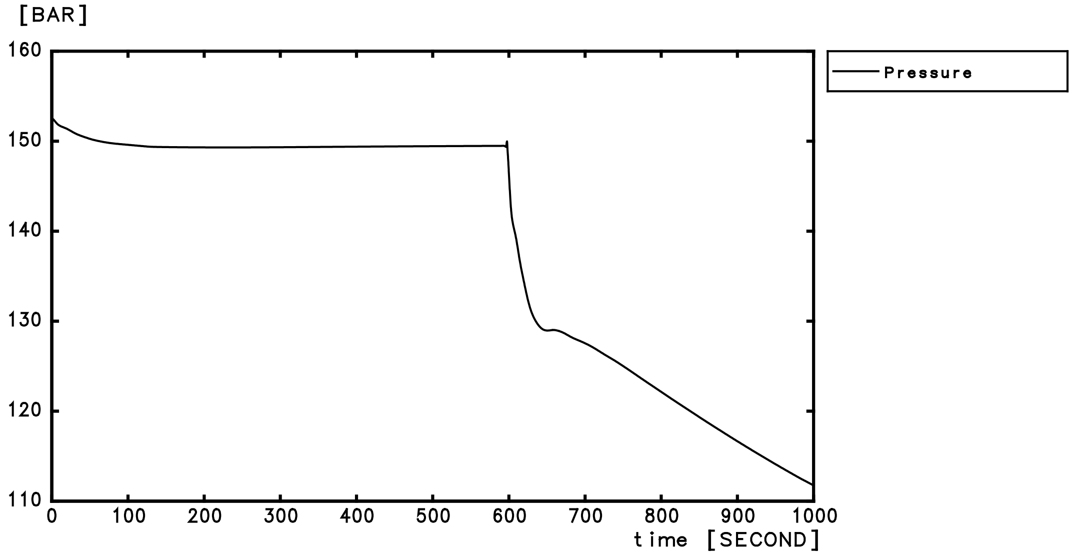
CORE POWER

SECTION 14.5.2 - FIGURE 122

Break Spectrum at Full Power, 158 cm² - Spurious MSIV Opening - Pressuriser Level and Pressuriser Pressure



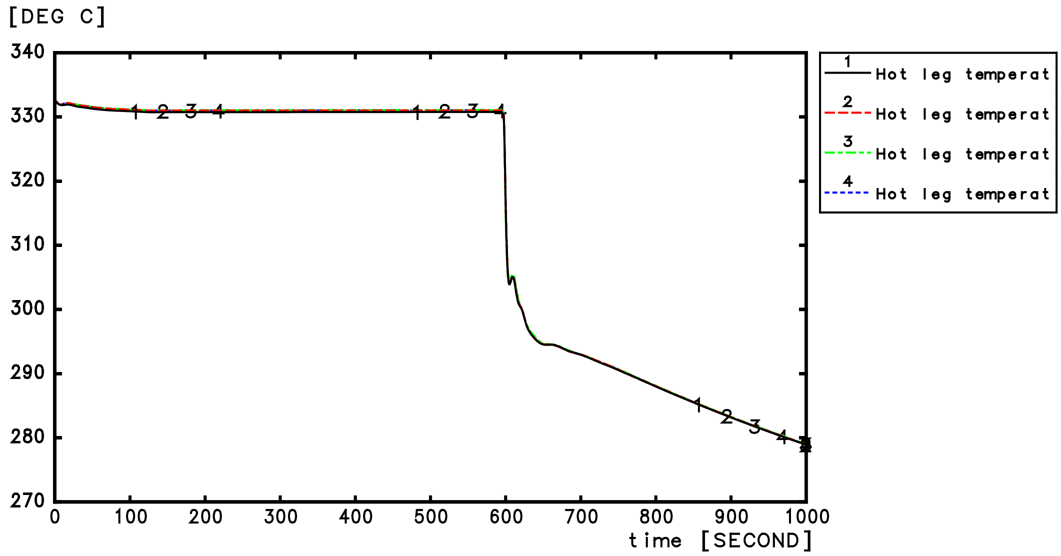
PRESSURIZER LEVEL



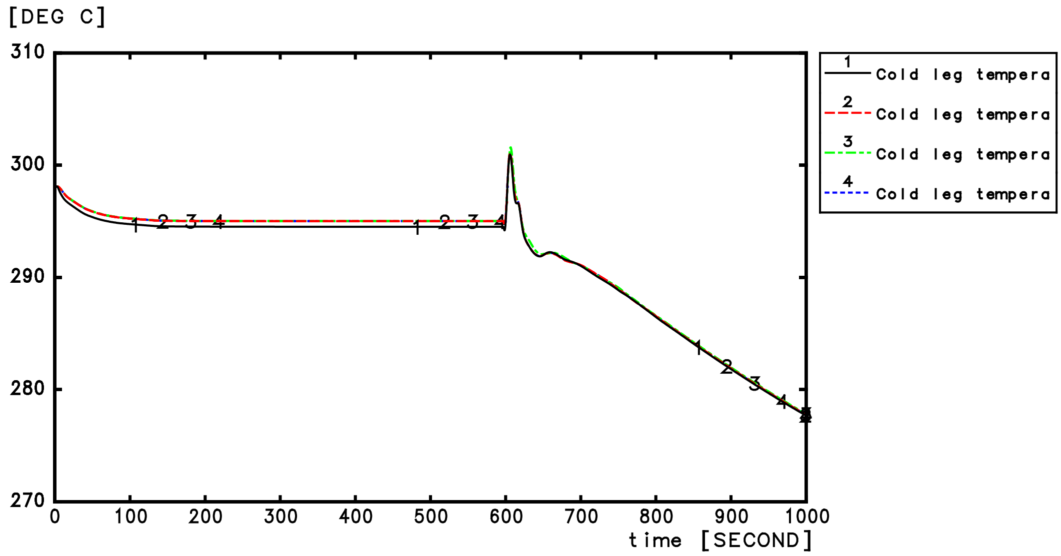
PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 123

Break Spectrum at Full Power, 158 cm² - Spurious MSIV Opening - Hot and Cold Leg Temperature



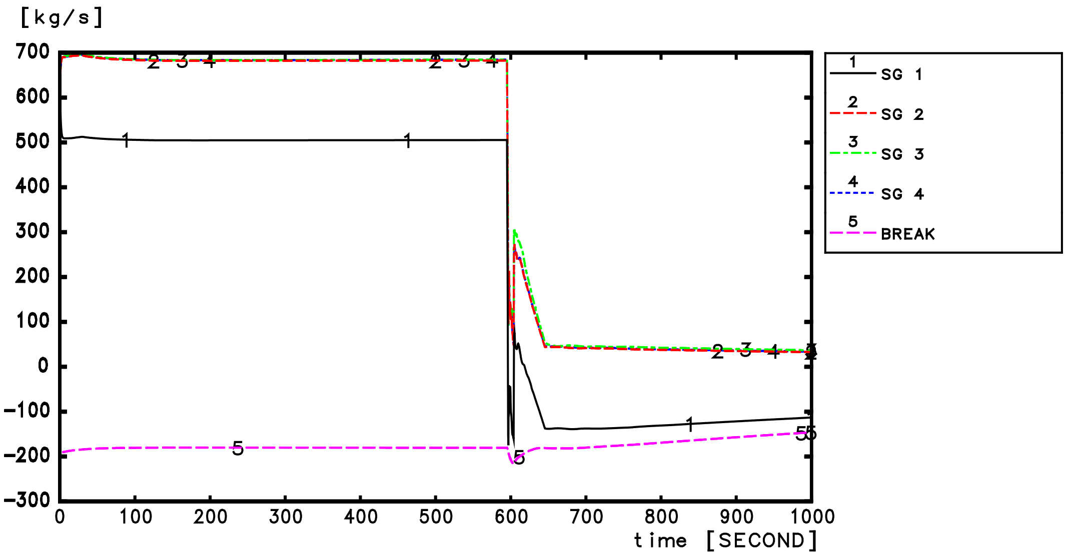
HOT LEG TEMPERATURE



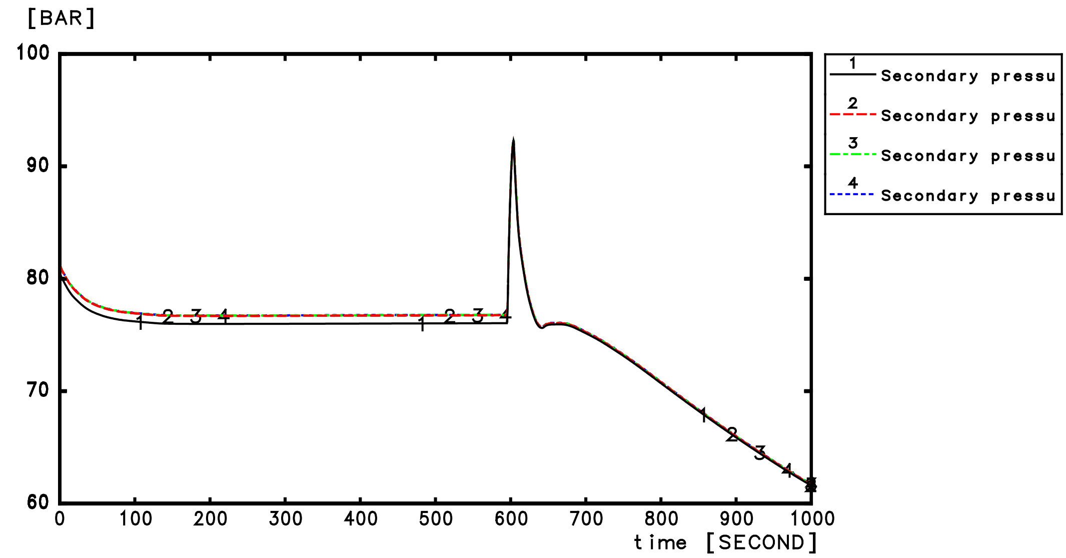
COLD LEG TEMPERATURE

SECTION 14.5.2 - FIGURE 124

Break Spectrum at Full Power, 158 cm² - Spurious MSIV Opening - SG Vapour Mass Flow Rate and SG Pressure



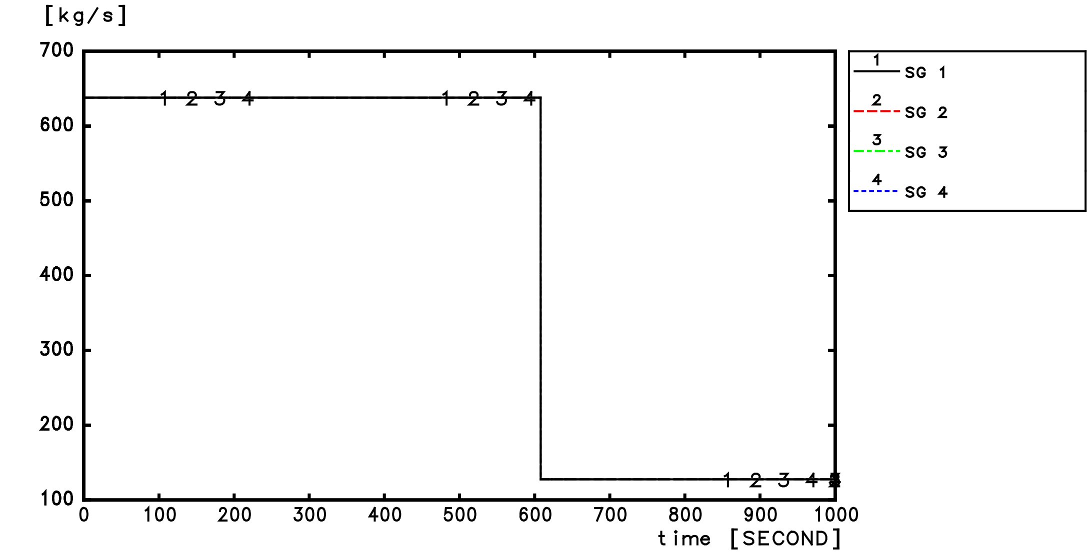
VAPOR MASS FLOWRATE



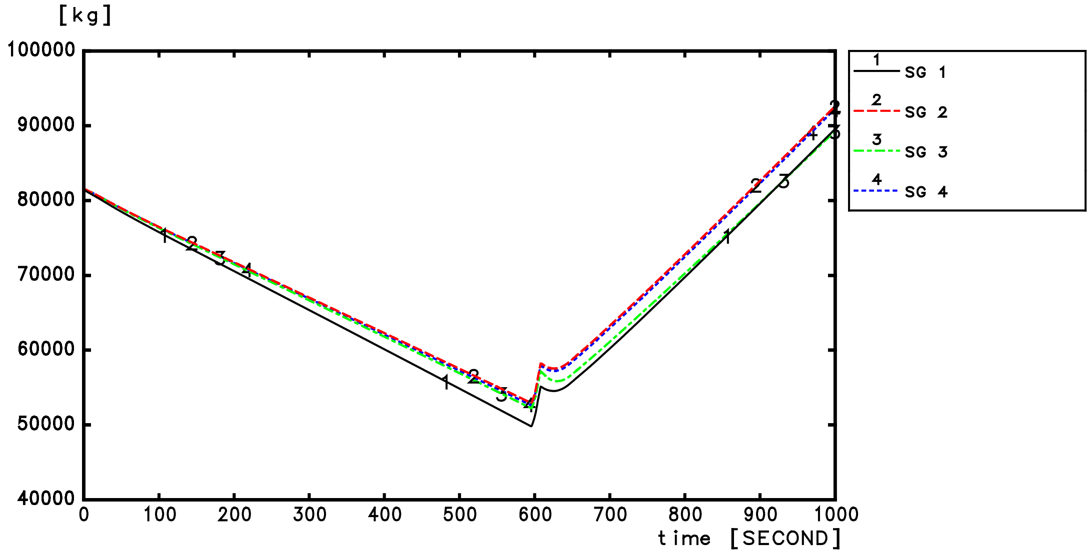
STEAM GENERATOR PRESSURE

SECTION 14.5.2 - FIGURE 125

Break Spectrum at Full Power, 158 cm² - Spurious MSIV Opening - SG Main Feedwater Flow Rate and SG Liquid Mass



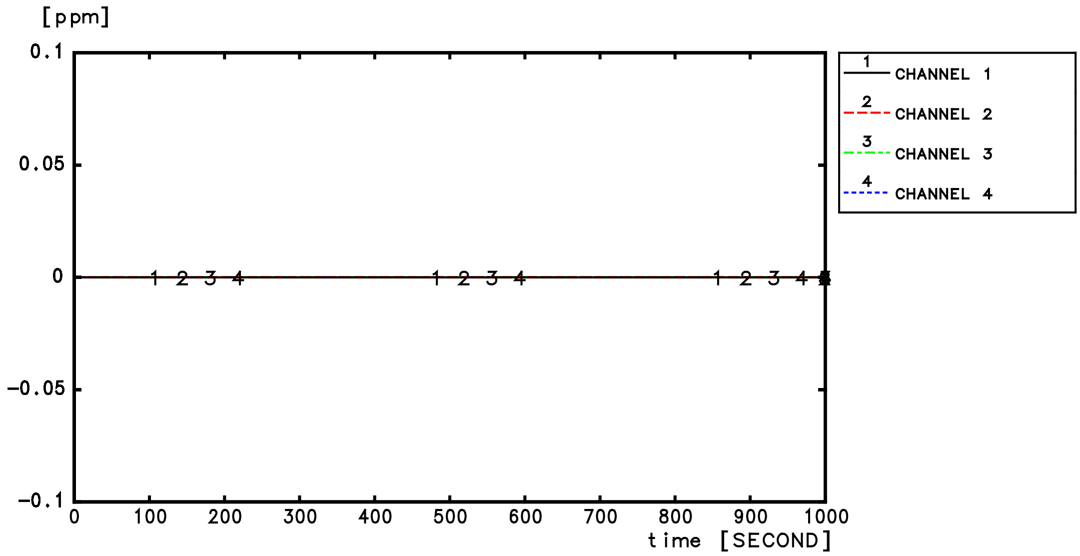
MFW FLOWRATE



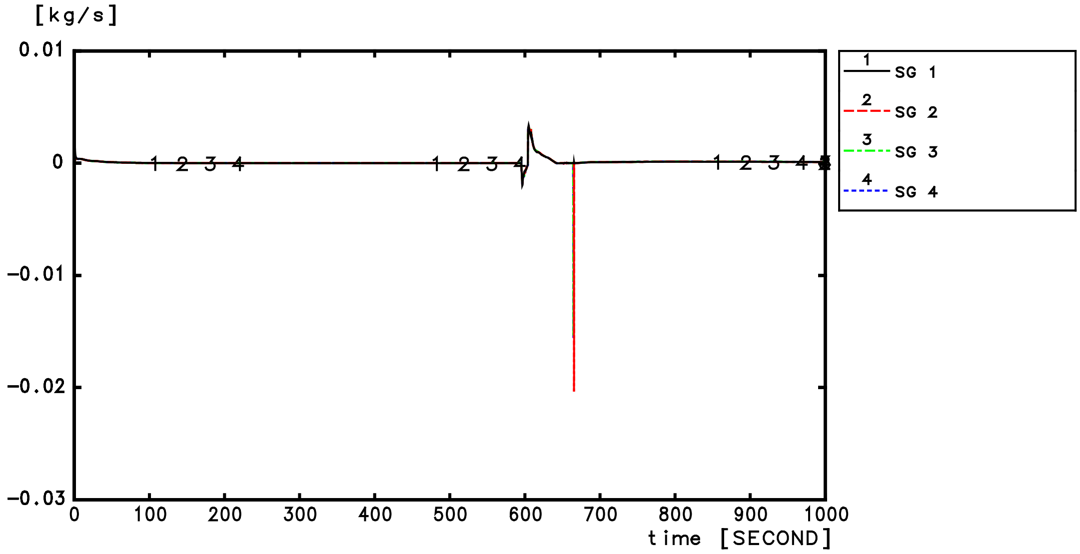
STEAM GENERATOR LIQUID MASS

SECTION 14.5.2 - FIGURE 126

Break Spectrum at Full Power, 158 cm² - Spurious MSIV Opening - Boron Concentration and ASG [EFWS] Flow Rate



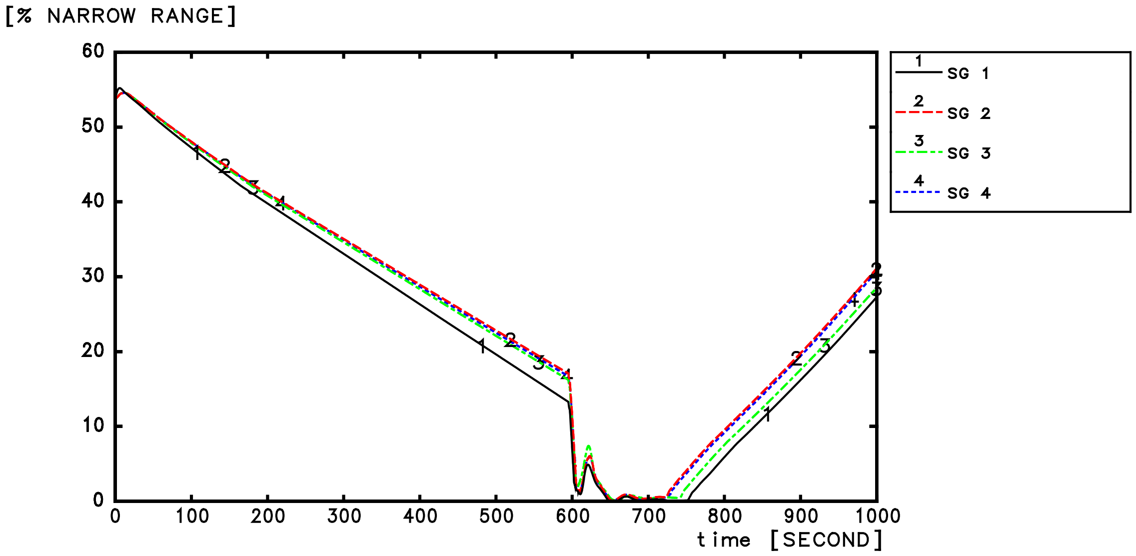
BORON CONCENTRATION AT CORE ACTIVE PART INLET



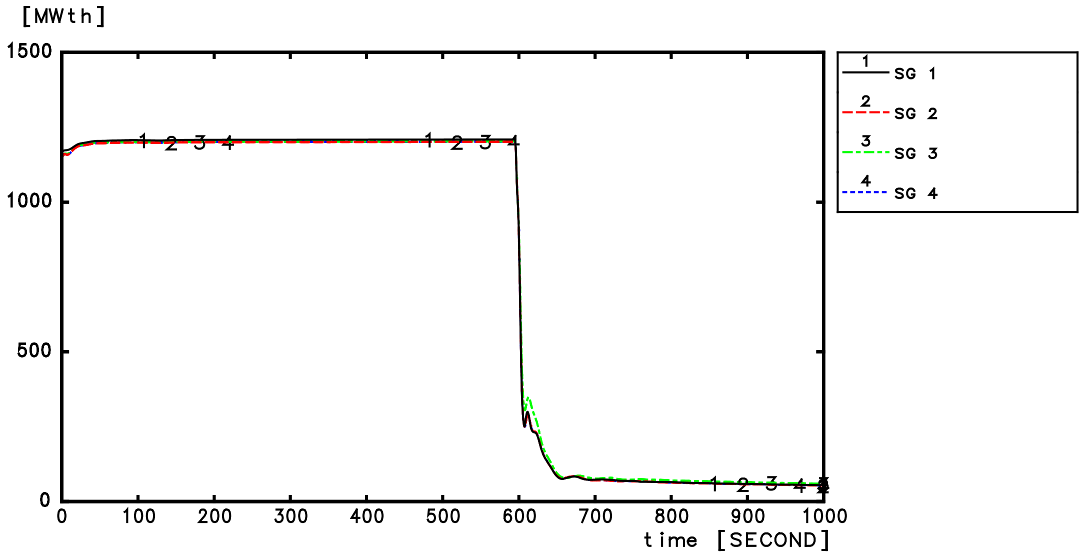
ASG FLOWRATE

SECTION 14.5.2 - FIGURE 127

Break Spectrum at Full Power, 158 cm² - Spurious MSIV Opening - SG Narrow Range Level and SG Power Exchanged



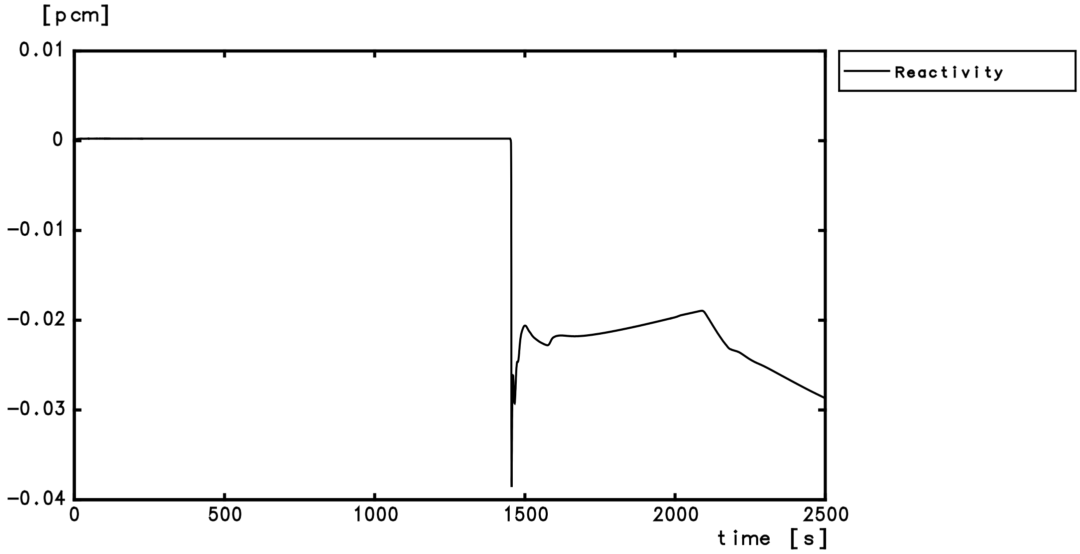
STEAM GENERATOR NARROW RANGE LEVEL



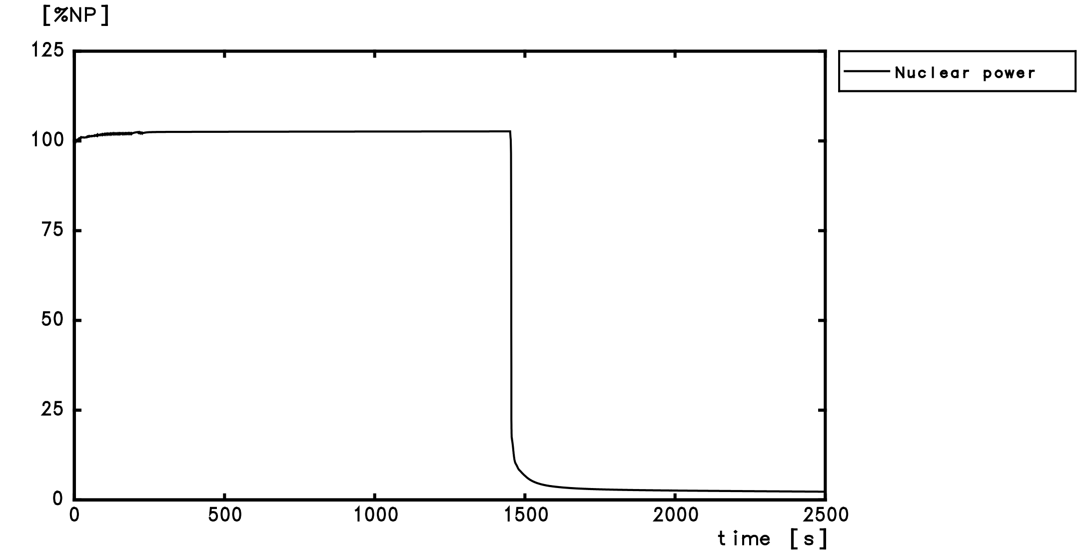
SG POWER EXCHANGED

SECTION 14.5.2 - FIGURE 128

Break Spectrum at Full Power, 20 cm² – Reactivity and Nuclear Power



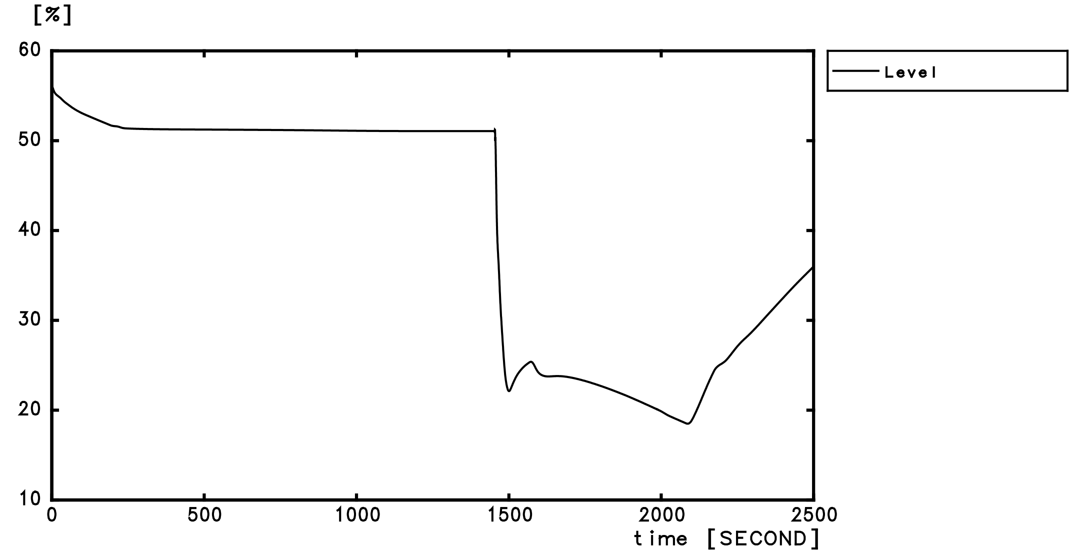
REACTIVITY



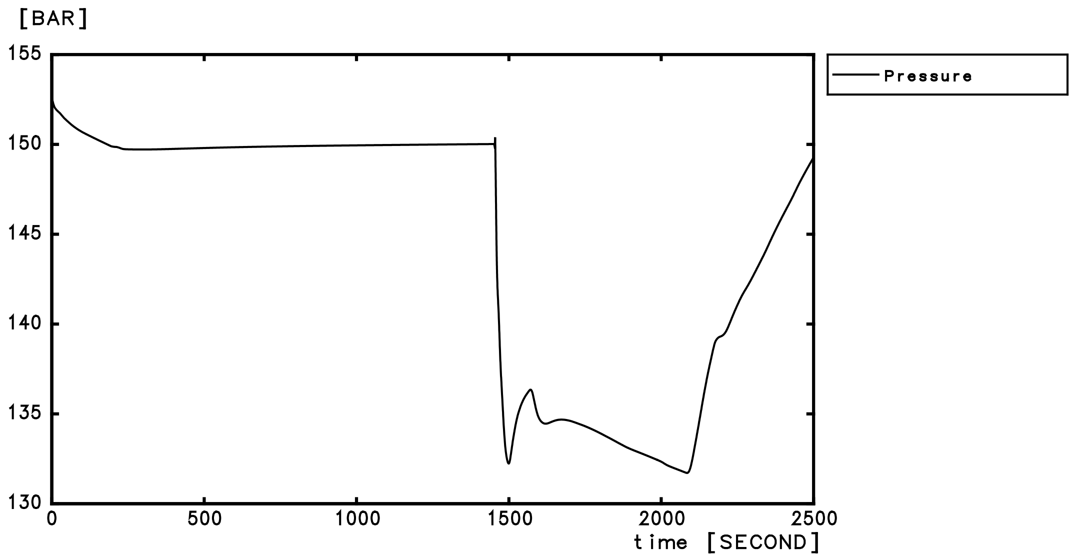
CORE POWER

SECTION 14.5.2 - FIGURE 129

Break Spectrum at Full Power, 20 cm² - Pressuriser Level and Pressuriser Pressure



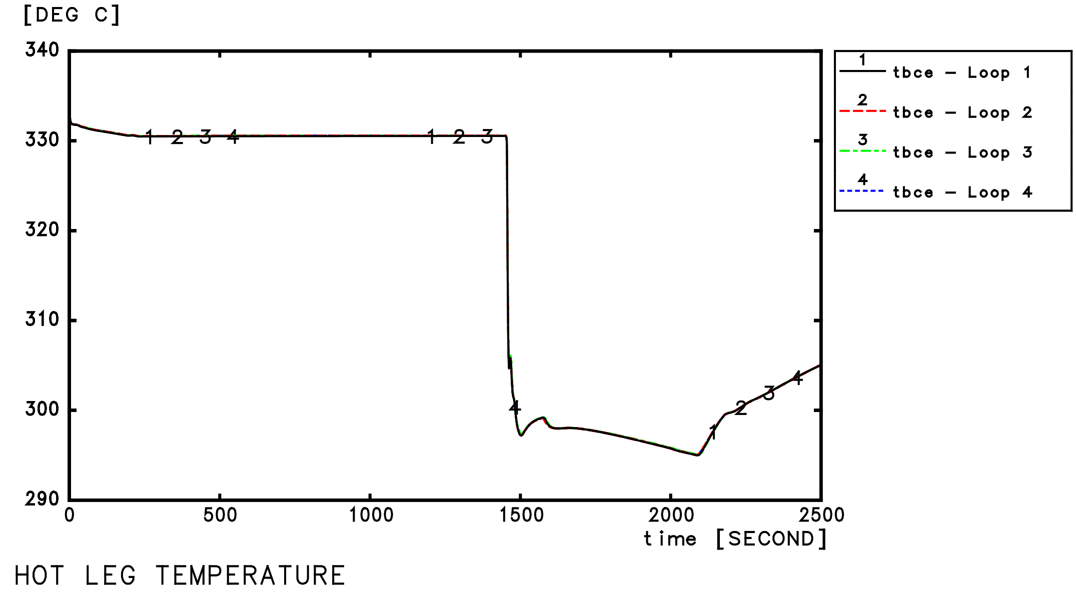
PRESSURIZER LEVEL

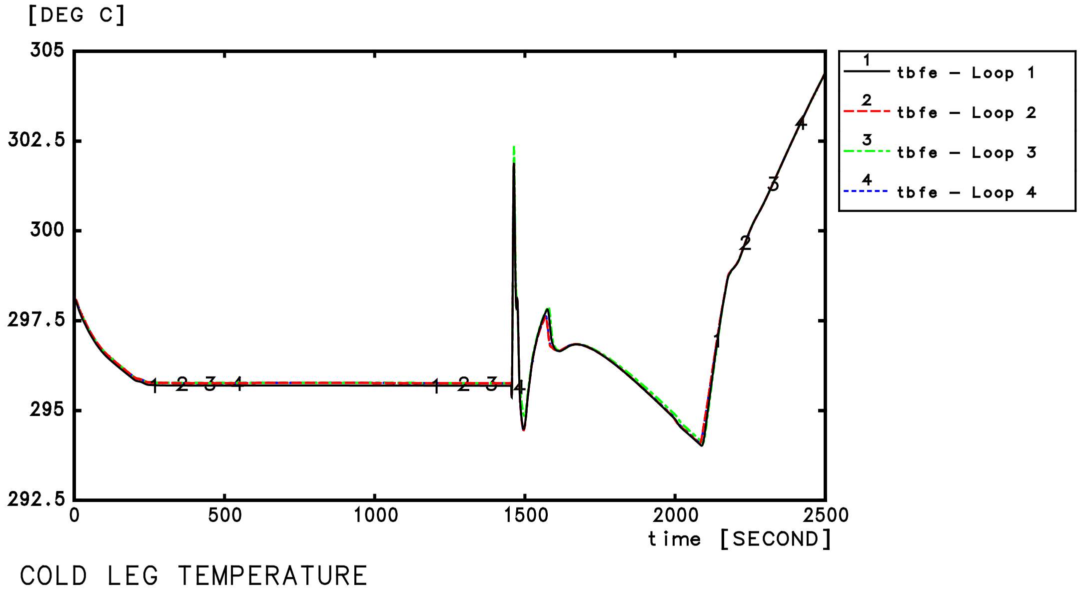


PRESSURIZER PRESSURE

SECTION 14.5.2 - FIGURE 130

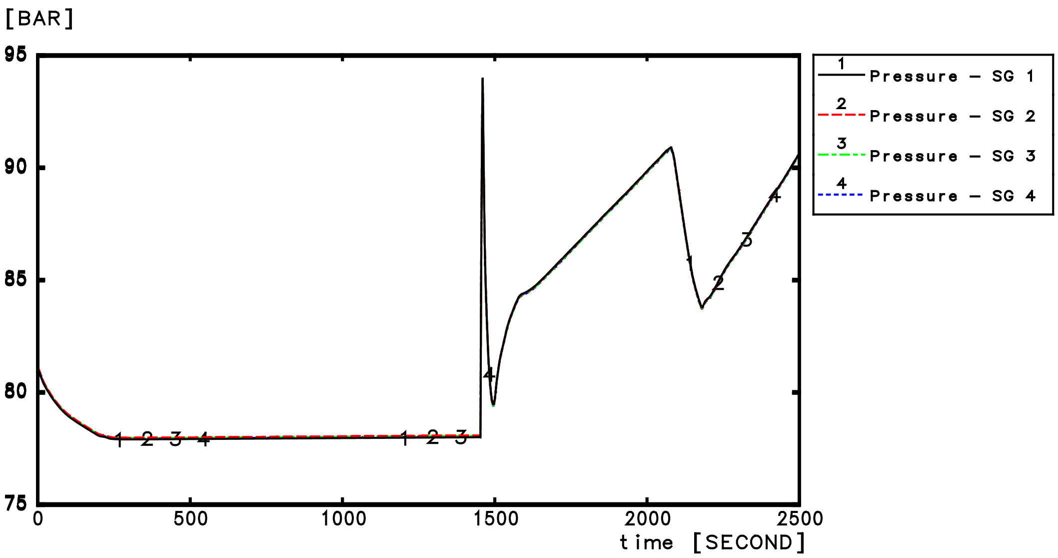
Break Spectrum at Full Power, 20 cm² - Hot and Cold Leg Temperature



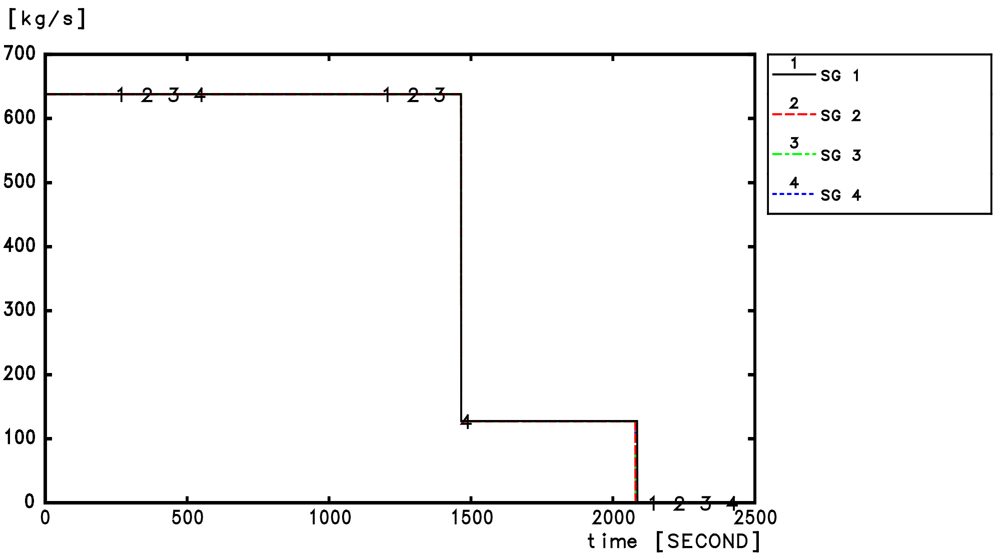


SECTION 14.5.2 - FIGURE 131

Break Spectrum at Full Power, 20 cm² - SG Pressure and Main Feedwater Flow Rate



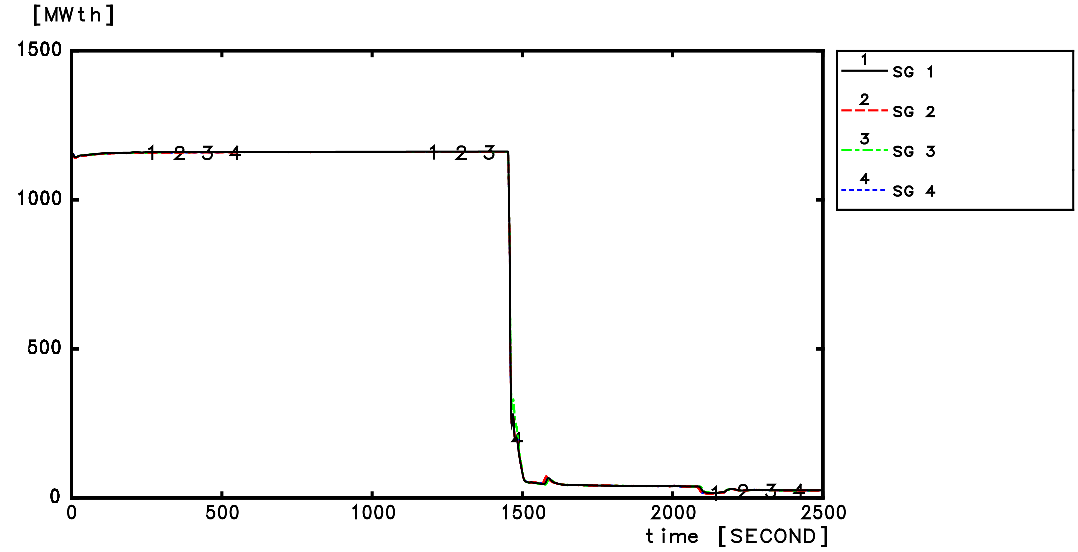
STEAM GENERATOR PRESSURE



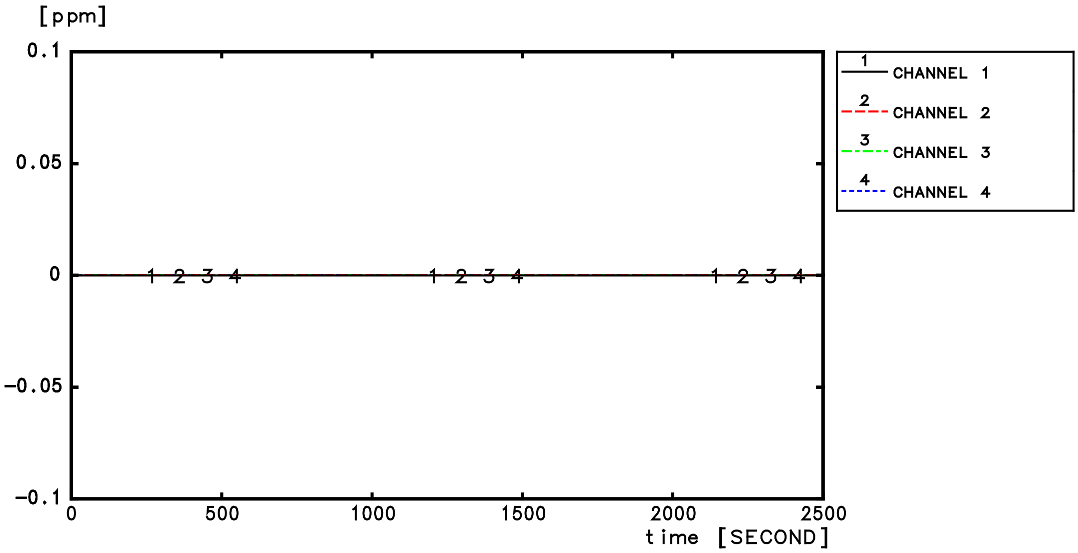
MFW FLOWRATE

SECTION 14.5.2 - FIGURE 132

Break Spectrum at Full Power, 20 cm² - SG Exchanged and Boron Concentration



SG POWER EXCHANGED



BORON CONCENTRATION AT CORE ACTIVE PART INLET

3. FEEDWATER SYSTEM PIPE BREAK

3.1. FEEDWATER SYSTEM PIPE BREAK IN STATE A

3.1.1. Identification of Causes and Description of Accident

3.1.1.1. General information

A large feedwater system (ARE [MFWS]) pipework failure in state A is classified as a PCC-4 event.

The accident examined is defined as a break located on a Main Feedwater System (ARE [MFWS]) pipe, sufficiently large to prevent the Steam Generator from being supplied with water.

The heat removal capability of the secondary system is impacted. Depending on the break size and the Nuclear Steam Supply System (VVP [MSSS]) power level at the break initiation, this accident can lead either to a RCP [RCS] overcooling or to a RCP [RCS] overheating. An RCP [RCS] cooldown would be due to an excessive steam discharge through the break. An RCP [RCS] overheating would be due to an excessive liquid discharge through the break. The effects of a RCP [RCS] overcooling are covered by the "Steam Line Break" analysis discussed in section 2 of this sub-chapter. In the present section, only the RCP [RCS] overheating aspect is considered.

The effects of the accident on the RCP [RCS] behaviour following a break located in the ARE [MFWS] line, upstream of the check valve are similar to those resulting from the ‘loss of Main Feedwater System’ accident discussed in section 7 of Sub-chapter 14.3.

Breaks located in the ARE [MFWS] line, between the SG and the check valve, result in both a loss of SG feedwater flow and a loss of SG water inventory. This leads to the maximum RCP [RCS] overheating should it occur during full power operation. The break considered in the present section is the largest located downstream the ARE [MFWS] check valve. This is a double ended guillotine break of the ARE [MFWS] line at the ARE [MFWS] nozzle inlet.

This accident bounds, in terms of RCP [RCS] overheating, a break of either an ASG [EFWS] line or an APG [SGBS] line.

3.1.1.2. Typical sequence of events

The typical sequence of events after a Feedwater Line Break (FWLB) are as follows:

a) From the initiating event to the Controlled State

Following the break initiation, the secondary system depressurises and the liquid content of the affected SG decreases as liquid is lost via the break., The liquid content of the three unaffected SG also decreases, due to the loss of main feedwater flow to the break through the ARE [MFWS] header. However, the rate of reduction is not as high as in the one affected SG.

The decrease in SG liquid mass leads to a reduction of the heat removal capability by the secondary system, and consequently to an overheating of the RCP [RCS].

During this phase, depending on the break size, a Reactor Trip (RT) occurs either on “Pressuriser pressure > MAX2” or on a reactor trip signal arising from conditions in the secondary system, namely:

- SG level < MIN1,
- SG pressure drop > MAX1,
- SG pressure < MIN1.

Following the reactor trip signal, the control/shutdown rods are automatically dropped into the core, taking the core to sub-critical conditions. The reactor trip signal automatically trips the turbine, and if available, the steam dump valves open permitting steam dump to the condenser. The GCT [MSB] may be unavailable, either due to automatic MSIV closure, because of its non F1-classification, or subsequent to a Loss Of Off-site Power (LOOP) coincident with turbine trip. In these circumstances, the steam generator pressure rapidly increases resulting in steam discharge to the atmosphere through the VDA [MSRT].

For FWLB large enough to remove the primary heat, the SG pressure increases only following automatic isolation of the main steam header. This is initiated by either a "SG pressure drop > MAX1" or "SG pressure < MIN1" signal. In that case, after VIV [MSIV] closure, the pressure in the three unaffected SG will increase, while the affected SG continues to depressurise via the break.

The ARE [MFWS] line of the affected SG is automatically isolated following a "SG pressure drop > MAX2" or "SG pressure < MIN2", signal. This signal only occurs in the affected SG. This ensures that the ARE [MFWS] or AAD [SSS] water supply continues to the unaffected SG. In the accident analysis, the ARE [MFWS] and AAD [SSS] are not credited after FWLB occurrence as the system operation would be beneficial, but is not F1 classified.

After RT, the mismatch in primary-to-secondary heat removal results in a continued decrease in liquid inventory in the affected and unaffected SG. When the “SG level MIN2” setpoint is reached in a SG, the associated ASG [EFWS] pump is started. There is a delay accounting for the diesel reloading sequence in case of LOOP. An ASG [EFWS] flow is delivered to the corresponding SG to restore its liquid inventory. For the accident analysis, it is assumed that the entire part of the ASG [EFWS] flow sent towards the affected SG is lost to the break. This conservative conservatively assumes no credit for heat removal from this flow. Only ASG [EFWS] flows sent towards the unaffected steam generator(s) are credited with supporting heat removal.

Whilst the ASG [EFWS] flow rate(s) entering the SG secondary sides remain(s) insufficient to ensure the removal from the primary system of the full heat load from decay heat and reactor coolant pumps power, temperature and pressure in the RCP [RCS] increases. When the pressuriser (PZR) pressure reaches the pressuriser safety valves (PSV) setpoints, PZR pressure MAX3, MAX4, and MAX5 respectively, the corresponding pressuriser safety valve opens. This provides an RCP [RCS] discharge to the Pressuriser Relief Tank. If the discharge is sufficient to cause the Pressuriser Relief Tank (PRT) rupture-disks to burst this fluid is released into the Reactor Building. In the accident analysis, the pressuriser spray is not credited as the system is not F1 classified.

Throughout this phase, from RT to the controlled state, both the ASG [EFWS] and VDA [MSRT] systems limit the RCP [RCS] overheating. The RCP [RCS] overpressure is limited by the PSV.

The controlled state is reached when the RCP [RCS] temperature and pressure are stabilised. At that time, the primary side heat is fully removed via the ASG [EFWS] supplied steam generator(s). The controlled state can be reached:

- Either automatically before 30 minutes, relying only on F1 systems which are started automatically. This is the case when two unaffected SG are fed, two ASG [EFWS] pumps being sufficient to remove more than the RCP [RCS] heat load of decay heat and reactor coolant pump power,

- or after operator action, 1 hour[1] after RT. This occurs when only one unaffected SG is fed due to a single failure and maintenance on the ASG [EFWS] of the two remaining unaffected SG. One ASG [EFWS] pump is not sufficient to remove the full RCP [RCS] heat load in the short-term following RT. In this case, the isolation of the ASG [EFWS] line of the affected SG, and the realignment of the ASG [EFWS] pump delivering into the affected SG towards one of those two remaining unaffected SG is undertaken by the operator. Once these actions have been performed, the entire RCP [RCS] heat load can be removed by the feed supplied.

b) From the controlled state to the safe shutdown state

The safe shutdown state is reached when the LHSI in residual heat removal mode is connected:

One out of four LHSI/RHR train is sufficient to provide the required heat removal.

Connection conditions are:

- RCP [RCS] hot leg pressure below 30 bar.

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

- ΔT_{sat}[2] and reactor pressure vessel level (RPVL) consistent with taking LHSI/RHR suction from the hot leg.

The following sequence of actions is performed by the operator to take the plant to the safe shutdown state:

Affected SG isolation

When the operator action time is reached (t_{RT} + 30 minutes in accident analysis), the affected SG is already fully depressurised (pressure about 1 bar).

The first operator action is to completely isolate this SG for steam discharge and water supply. This is performed by closing the VIV [MSIV] (if this has not occurred automatically, e.g. for small FWLB), and by isolating the associated ASG [EFWS] line. The ASG [EFWS] line is isolated to prevent the associated ASG [EFWS] tank from being complete discharged via the break, and to limit pressure increase in the containment.

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[1] In the accident analysis, operator action is assumed at RT + 30 minutes if performed from the Main Control Room, at RT + 1hour if performed outside of the Main Control Room, local to plant. ASG [EFWS] realignment is a local action, performed within the Safeguard Buildings.

[2] ΔTsat = Tsat (hot leg pressure) - T_{co}, with T_{co} = core outlet temperature.

ASG [EFWS] passive header (pump discharge) opening

After isolation of the ASG [EFWS] injection line feeding the affected SG, the corresponding ASG [EFWS] flow can be routed to another SG via the ASG [EFWS] header if necessary. This ensures the two ASG [EFWS] pumps required to remove the entire primary heat load in the short-term after RT are available in the case of single failure and preventive maintenance on two ASG [EFWS] assigned to two unaffected SG. As stated previously ARE [MFWS] and AAD [SSS] are not credited as these systems are not F1-classified.

RCP [RCS] pumps shutdown

Before starting the RCP [RCS] cooldown to LHSI/RHR connecting conditions, the operator must switch-off two out of four reactor coolant pumps, if all reactor coolant pumps are still running. This is consistent with the ASG [EFWS] tank capacity, where the water inventory matches the design assumption of at most two out of four reactor coolant pumps running during the cooling phase.

In the Emergency Operating Procedures (EOP), the reactor coolant pumps switch-off will be requested based on information of ASG [EFWS] tank water levels.

RCP [RCS] cooldown by unaffected SG

The RCP [RCS] cooldown to LHSI/RHR connecting conditions is performed using the unaffected steam generators. The EPR cooling rate is - 50°C/h if two RBS [EBS] trains are available, or - 25°C/h if only one RBS [EBS] train is available, provided the cooling rate is not limited by the VDA [MSRT] capacity at low SG pressure level. These cooldown rates are defined so that the respective RBS [EBS] boration rate matches the reactivity insertion resulting from the RCP [RCS] cooldown. As indicated here, the ASG [EFWS] tank design assumes two reactor coolant pumps remain in operation during the RCP [RCS] cooling phase.

The operator has to initiate the RCP [RCS] cooldown within 2 hours of RT, for the most onerous case where the four reactor coolant pumps are running until the end of the boration phase.

The RCP [RCS] cooldown is stopped when the LHSI/RHR connection temperature is reached of 180°C in the hot leg of the cooling loops. With two reactor coolant pumps operating, this condition is reached when the pressure in the unaffected SG reaches approximately 10 bar, the saturation pressure at 180°C. Should no reactor coolant pumps be operating, as in the case of LOOP, the unaffected SG depressurisation must be continued to around 5 bar, corresponding to a saturation temperature of 150°C. As a result, the RCP [RCS] cold leg temperature must be 150°C, with the hot leg temperature during natural circulation below 180°C.

VIV [MSIV] bypass opening

Should the VDA [MSRT] of one unaffected SG fail to open, the operator must open the associated VIV [MSIV] bypass line, the VIV [MSIV] being already closed, and the VIV [MSIV] bypass line of another unaffected SG, to depressurise this SG. This action accelerates the cooldown of the corresponding RCP [RCS] loop in the case of LOOP where, no reactor coolant pumps are running.

RCP [RCS] boration

During the cooldown the RCP [RCS] boration is performed using the RBS [EBS]. The RCV [CVCS] is not claimed as the system is not F1 classified. After completion of the required boration, the operator stops the RBS [EBS].

ASG [EFWS] passive header (pump suction) opening

The operator must open the ASG [EFWS] header at the pump suction to claim credit for all four ASG [EFWS] tanks water inventory. The ASG [EFWS] tank water supply is not claimed as this system is not F1 classified. The realignment to the ASG [EFWS] tanks previously not used, to the suction of the ASG [EFWS] pumps in operation, is required as the design of the ASG [EFWS] tanks assumes the availability of the four tanks. The inventory remains in these tanks because of the unavailability of the corresponding ASG [EFWS] pump due to single failure, preventive maintenance, or isolation towards the affected SG.

Depressurisation of RCP [RCS]

Prior to or during the RCP [RCS] cooldown, the operator isolates the accumulator injection lines. This prevents unwanted accumulator injection.

The operator must depressurise the primary side since the RCP [RCS] pressure is above the LHSI/RHR connection pressure of 30 bar after the manual cooldown phase. This depressurisation is performed by briefly opening the PSV. During this depressurisation phase, the LHSI maintains a minimum RCP [RCS] pressure level of about 20 bar so that the RCP [RCS] sub-cooling margin is kept, at least in the unaffected loops in the case of LOOP.

LHSI/RHR connection

The LHSI/RHR connecting conditions are reached when the hot leg pressure reaches 30 bar, hot leg temperature < 180°C, RPVL consistent with LHSI/RHR suction from the hot leg.

One LHSI train is connected, operating in LHSI/RHR mode with suction from a hot leg and injection into a cold leg. One LHSI/RHR train is sufficient to remove the decay heat.

c) Precautions limiting the event occurrence

- A check valve prevents the SG blowing down via the break if this break is located upstream of the valve. In this case, it results in a "loss of main feedwater" accident, less onerous than a "Feedwater system pipe break " accident.

- Main feedwater lines from and including their isolation valves outside containment up to the check valve are classified RCC-M (2007 EPR version) class 2, and from the check valve up to the SG are classified RCC-M (2007 EPR version) class 1.

3.1.2. Safety Criteria

The safety criteria are the radiological limits for PCC-4 events. The effects of a Feedwater Line Break are analysed for the following decoupling criteria:

- Fuel cladding integrity.

- Reactor coolant pressure boundary (RCPB) integrity.

- Reactor containment pressure and temperature limits.

- Amount of radioactivity released.

For the RCPB criteria, the FWLB analysis is bounded by the "primary over-pressure" analyses (see Sub-chapter 3.4).

For the criteria related to the reactor containment, the FWLB analysis is bounded by the steam line break analysis (see Appendix 6A).

For the criteria related to the amount of radioactive products released, the FWLB analysis is bounded by the ‘SGTR 2 tubes’ analysis (see section 10 of this sub-chapter).

Consequently, the FWLB analysis in this section only considers the fuel cladding integrity criteria.

It must be shown that the following two safe states are reached, by applying the safety analysis rules defined in Sub-chapter 14.0:

- The controlled state, relying only on F1A systems.
- The safe shutdown state, relying only on F1A and F1B systems.

3.1.3. Methods and Assumptions

3.1.3.1. Method of analysis

The FWLB accident analysis is performed with the CATHARE code (see Appendix 14A) using a Realistic Deterministic Methodology.

The Realistic Deterministic Methodology is characterised by the two main features:

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

The basic steps of the Realistic Deterministic Methodology consist of:

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

The dominant phenomena of the FWLB transient are:

- The faulted SG draining and depressurisation.

- The asymmetric RCP [RCS] heating, and the resultant RCP [RCS] overpressure, with the reactor coolant pumps on or off.
- The asymmetric RCP [RCS] heat removal via the unaffected SG with potentially low water level, with the reactor coolant pumps on or off.
- The asymmetric RCP [RCS] cooling, and the RCP [RCS] depressurisation down to LHSI/RHR connecting conditions, with reactor coolant pumps on or off.

All these phenomena are within the applicability range of the CATHARE code, the validation of which is based on:

- The qualification of correlations and physical laws on separate effect tests (SET) or component tests.
- The validation of the axial SG model, with an economiser of the N4 SG-type, from MEGEVE small-scale make-up tests. [Ref-1]
- The overall verification of the code by simulation of integral effect tests (IET), covering a wide range of representative PWR transients on small-scale facilities, for example:
    - Test LOBI BT-06 "partial FWLB (45%)": CATHARE gave an accurate prediction of the faulted SG, primary to secondary heat transfer, RCP [RCS] pressure and temperature trends. [Ref-1]
    - Tests BETHSY 5.1a "steady states with reduced SG mass inventories in forced and natural circulations": CATHARE trends were in good agreement with the test results as RCP [RCS] temperatures, distribution of heat transfer between ascending and descending SG U-tube sections, secondary side swell level, global P/S heat exchange, provided a fine meshing in the SG riser and downcomer volumes of CATHARE is used, as identified for such transient analyses. [Ref-1]
    - Tests BETHSY 5.2c and 5.2e “total loss of SG feedwater, Feed and Bleed”: CATHARE accurately predicted the RCP [RCS] pressure, pressuriser relief valve flows, and RCP [RCS] coolant mass inventory transients. [Ref-1]

The transient analysis relies on the application of the conservative PCC analysis rules defined in Sub-chapter 14.0. Part of these rules is the deterministic pessimism of all relevant boundary conditions, relevant to the decoupling criteria under consideration. These address at least:

- The characterisation of the initiating event (maximisation of the resulting impact).
- The plant initial conditions (control dead band limits, maximum measurement uncertainties).
- The performance of the protection and mitigation actions (maximum uncertainty on each I&C measurement and signal delay, and on each system response time and capability).

This analysis methodology provides conservative results which can be directly used to assess against the decoupling criteria.

The CATHARE code is used with separate models of each of the 4 RCP [RCS] loops.

3.1.3.2. Main assumptions

a) Accident definition

The cases studied in this section correspond to the double-ended guillotine rupture of a main feedwater pipe at the inlet of a steam generator. This leads to the fastest draining of the SG.

For accident analyses, a loss of off-site power (LOOP) is assumed, if this is pessimistic.

b) Protection and mitigation actions

In the event of a FWLB, automatic F1A classified actions trip the reactor and ensure the decay heat removal. These actions limit the overheating of the RCP [RCS].

In accordance with the rules defined for safety analyses in Sub-chapter 14.0, the Controlled State is reached relying only on F1A classified actions. The Safe Shutdown State is reached relying only on F1A and F1B classified actions.

The different automatic actions which could occur in the case of a FWLB are those linked either to the RCP [RCS] overheating or to the consequences of this event on the SG.

The potential reactor trip signals are:

- Pressuriser pressure > MAX2 (F1A),
- SG level < MIN1 (F1A),
- SG pressure drop > MAX1 (F1A),
- SG pressure < MIN1 (F1A).

To cover the entire range of break sizes, a set of conservative assumptions is used in the present analysis. The maximum break size of a main feedwater pipe is analysed with the following bounding assumptions:

- No RT signal arising on the primary side is considered.
- The "SG pressure drop > MAX1" or "SG pressure < MIN1" signals are not claimed. Consequently, the RT and VIV [MSIV] isolation occur following a "SG level < MIN1" signal in the transient calculation. This assumption ensures that the SG water content at RT is the lowest, covering the entire range of break sizes[3].

In addition, it is conservatively assumed that there are no signals generated from the affected SG.

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[3]
- For small FWLB, the RT/TT occurs on “SG level < MIN1”.
- For large FWLB, the RT/TT occurs on “SG Pressure drop > MAX1” or “SG pressure < MIN1”. At this time, the unaffected SG water content is higher than in case of small FWLB, which is beneficial.
- The PCSR analysis considers a large FWLB with RT/TT on “SG level < MIN1”. It combines the high dynamics of the large FWLB, with the low unaffected SG water content of the small FWLB. As a result, it covers the consequences of any break size within the whole FWLB spectrum.

Consequently, only the "SG level < MIN1" signals from the unaffected SG are considered to provide the RT actuation. That is pessimistic for the draining of SG.

The other automatic actions considered in the analysis are the following:

- Turbine Trip (F1A):

  On RT signal, the turbine trip is actuated.

- VIV [MSIV] isolation (F1A):

  The VIV [MSIV] closure is normally initiated on a "SG pressure < MIN1" or "SG pressure drop > MAX1" signal. However, as discussed above for RT actuation, these signals are not claimed to actuate VIV [MSIV] closure.

- ASG [EFWS] actuation (F1A):

  Following a "SG level < MIN2" signal, the ASG [EFWS] is actuated to the associated SG. This signal is specific to an individual SG. The time delay between reaching the setpoint and the effective ASG [EFWS] flow injection is defined assuming LOOP, or no LOOP, covering the effect of the emergency diesel generators (EDG) re-loading sequence.

- VDA [MSRT] actuation (F1A):

  When the SG pressure reaches the VDA [MSRT] "SG pressure > MAX1" setpoint, the VDA [MSRT] on that SG opens and provides the heat removal and pressure control.

- Main steam safety valves actuation (F1A, passive opening):

  Should the SG pressure increase to the main steam safety valves setpoint, e.g. in case of the VDA [MSRT] failing to open, the safety valves will open and provide heat removal.

- PSV actuation (F1A, passive opening):

  Should the RCP [RCS] pressure increase to the PSV setpoints, the PSV will open to limit the RCP [RCS] over-pressure. Any discharged primary coolant is delivered to the PRT and subsequently to the Reactor Building should the PRT rupture-disks burst.

c) Operator actions

No operator action is considered before 30 minutes after reactor trip. When an operator action is needed which requires local to plant action this delay is extended to 1 hour following RT.

The operator must perform the following actions identified in the Emergency Operating Procedures, as described in section 3.1.1.2 b above.

The list of F1B operator actions, with indication of the main F1B information needed, includes:

- Affected SG isolation (F1B):
    - Manual ASG [EFWS] isolation (ASG [EFWS] flow rate, SG pressure)[4].
    - Manual VIV [MSIV] isolation, if not yet done (SG pressure).
    - Manual ARE/AAD [MFWS/SSS] isolation, if not yet done (SG pressure).
- ASG [EFWS] passive header (pump discharge) opening (F1A)[5]:
    - Manual realignment of the affected SG ASG [EFWS] pump discharge to the unaffected SG, via the dedicated passive header (SG level, SG pressure, ASG [EFWS] flow rate).
- VIV [MSIV] bypass valve opening (F1B):
    - Manual opening of VIV [MSIV] bypass valve (SG pressure).
- Reactor coolant pump trip (F1B):
    - Manual reactor coolant pump trip (RCP operating status).
- RCP [RCS] cooldown using the unaffected SG (F1B):
    - Manual VDA [MSRT] opening/closing, if no F1B automatic control of SG cooldown (RCP [RCS] temperature, SG pressure).
    - Manual ASG [EFWS] opening/closing, if no F1B automatic control of SG level (SG level).
- RCP [RCS] boration (F1B):
    - Manual RBS [EBS] actuation, at the latest at RCP [RCS] cooldown initiation, with manual alignment to an unaffected loop.
    - Manual termination of RBS [EBS] (RBS [EBS] tank level).
- ASG [EFWS] passive header (pump suction) opening (F1B):
    - Manual realignment of ASG [EFWS] tank towards ASG [EFWS] pump suction via the dedicated passive header (ASG [EFWS] tank level).
- RCP [RCS] depressurisation (F1B):
    - Manual isolation of accumulator (RCP [RCS] pressure).
    - Manual MHSI shutdown, if automatically actuated during RCP [RCS] depressurisation and cooldown (MHSI flow rate).
    - Manual LHSI actuation, if not automatically actuated during RCP [RCS] depressurisation and cooldown (RCP [RCS] pressure, LHSI flow rate).

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[4] F1A classification required if Preventive Maintenance.
[5] ASG [EFWS] passive header opening only required if preventive maintenance.

    - Manual opening/closing of PSV (RCP [RCS] pressure).

- LHSI/RHR connection (F1B):

    - Manual connection of LHSI in RHR mode (LHSI/RHR) to the RCP [RCS] (RCP [RCS] pressure, RCP [RCS] temperature, RPVL, ΔTsat).

3.1.4. Definition of Cases Studied

The break flow area is 1160 cm^2, which corresponds to the inner section of the ARE [MFWS] line at the ARE [MFWS] nozzle inlet. [Ref-1]

3.1.4.1. Cases studied from the initiating event to the Controlled State

As described in section 3.1.2 of this sub-chapter, the decoupling criterion to be met is the fuel cladding integrity.

The objective of the FWLB analysis is to show that the secondary side, the four SG secondary sides taken together, remains available to provide primary side heat removal, throughout the transient, despite the loss of one SG due to the initiating event. This objective is significantly more restrictive than the "fuel cladding integrity" decoupling criteria.

To meet this objective, at least one SG must remain available for primary side heat removal during the post-accident phase. A temporary failure of the heat removal is acceptable, provided the loss of RCP [RCS] water inventory through the PSV, due to thermal expansion, remains relatively low compared to the initial RCP [RCS] water inventory. This ensures that the remaining primary coolant content is sufficient to provide the required core cooling. The Controlled State is reached when sufficient SG heat removal is re-established.

The time period extends from the initiating event to the Controlled State, relying only on F1A systems. To demonstrate meeting this objective, the FWLB transients considered are those which maximise the RCP [RCS] overheating. They utilise the following assumptions:

- The highest initial power level (i.e. 102% FP).

- The inclusion of LOOP at RT, when pessimistic.

Two cases are analysed, to assess the most onerous single failure (SF) affecting the secondary side heat removal capability. Each case also includes the most onerous Preventive Maintenance (PM) which is the unavailability of one ASG [EFWS] train as discussed in section 3.1.5.1 of this sub-chapter:

- Case 1 (102% FP, without LOOP): RCP [RCS] overheating concern, with minimum ASG [EFWS]

  The single failure is applied to one ASG [EFWS] pump feeding one unaffected SG.

  The assumption "without LOOP" is the most conservative one. Indeed, with only one ASG [EFWS] pump remaining available. This includes the effect of the preventive maintenance of one ASG [EFWS] train combined with the single failure. In these circumstances, the secondary side does not remove all the primary heat. The part of the primary heat not removed by the secondary side heats-up the primary coolant and results in PSV opening to provide the additional heat removal. The higher the primary heat load, the higher the RCP [RCS] water loss via the PSV.

Consequently, the most conservative case is thus obtained with the reactor coolant pumps operating, i.e. no LOOP.

- Case 2 (102% FP, with LOOP): RCP [RCS] overheating concern, with minimum VDA [MSRT]

  The single failure is applied on one VDA [MSRT] on one unaffected SG.

  The assumption "with LOOP" is the most conservative one. During this transient two ASG [EFWS] pumps remain available for heat removal covering the preventive maintenance on one ASG [EFWS] train combined with the single failure. This arrangement is able to remove the bulk of the primary heat. Consequently, RCP [RCS] overpressure does not result from a temporary failure in heat removal, even when the four reactor coolant pumps power input is considered.

  Consequently, the loss of reactor coolant pumps is conservative in this case because of the RCP [RCS] overpressure caused by the higher temperature difference across the core.

The other specific assumptions for those cases are described below (see section 3.1.5.3 of this sub-chapter).

3.1.4.2. Cases studied from the Controlled State to the Safe Shutdown State

As described in section 3.1.2 of this sub-chapter, it must be shown that the Safe Shutdown State is reached using only F1A and F1B measures. The objective of the FWLB analysis is to demonstrate that: "The secondary side heat removal remains available until the LHSI/RHR is connected, relying only on F1-means". It requires an adequate secondary side water supply capacity (ASG [EFWS] tanks and pumps), consistent with the performance of the other systems required for transferring the plant from the Controlled State to the Safe Shutdown State (RCP [RCS] boration, cooling and depressurisation).

During this phase, the following actions must be performed by the operator:

- Affected SG isolation and realignment of the ASG [EFWS] pump assigned to the affected SG to an un-supplied unaffected SG if necessary[6].
- Boration of RCP [RCS].
- Cooldown via the unaffected SG.
- Final depressurisation of RCP [RCS].

Two cases are analysed, concerning the RCP [RCS] cooldown, and the RCP [RCS] depressurisation respectively. In both cases, the capability to perform RCP [RCS] boration is provided:

[6] That realignment, if necessary, is needed to reach the Controlled State. The relevant actions of ASG [EFWS] line isolation towards the affected SG, and ASG [EFWS] pump alignment towards one unaffected SG not being fed, are then part of the first phase "from initiating event to the Controlled State".

- Case 3 (102% FP, without LOOP): RCP [RCS] heat removal issue

  It is assumed that all four reactor coolant pumps are running until the start of the cooldown. Subsequently, two reactor coolant pumps are switched off, and two reactor coolant pumps remain operating during the RCP [RCS] cooldown phase.

  This case is performed to demonstrate the capability of the F1 systems to reach LHSI/RHR connection conditions. It especially demonstrates the ability of the ASG [EFWS] and VDA [MSRT]) to cool the plant to the LHSI/RHR connection conditions, assuming the maximum power level to be removed, including reactor coolant pumps power.

  The cooldown has to be completed and the LHSI/RHR connected before the ASG [EFWS] tanks empty.

- Case 4 (102% FP, with LOOP): RCP [RCS] depressurisation concern

  It is assumed that the four reactor coolant pumps are not in operation.

  This case is performed to demonstrate the capability of F1 systems to depressurise the RCP [RCS] to the LHSI/RHR connection conditions, under the most difficult depressurisation conditions.

  The RCP [RCS] natural circulation which results from the LOOP assumption is pessimistic for the RCP [RCS] depressurisation. This is because of the existence of hot non-cooled loops, e.g. the loop connected to the affected SG.

3.1.5. Description of Studied Cases 1 and 2 (from the initiating event to the controlled state)

3.1.5.1. Choice of single failure and preventive maintenance

Single failure and preventive maintenance are applied on F1 systems in the most pessimistic way for the criteria to be met In accordance with the general safety rules defined in Sub-chapter 14.0.

For RCP [RCS] overheating, the most conservative assumptions reduce the secondary side heat removal capability. Consequently, preventive maintenance is assumed on the ASG [EFWS] pump connected to one unaffected SG as preventive maintenance has no effect on the VDA [MSRT] in state A, see section 3.1.7.1a of this sub-chapter.

For the single failure, as indicated previously, two different cases are considered:

- Case 1 (102% FP, without LOOP):

  The SF is applied on one ASG [EFWS] pump associated with one unaffected SG in addition to the preventive maintenance of another ASG [EFWS] pump associated with another unaffected SG.

  As discussed above for the choice of maintenance assuming the failure of one ASG [EFWS] pump associated to another unaffected SG reduces the secondary side heat removal capability and increases the RCP [RCS] overheating.

- Case 2 (102% FP, with LOOP):

  The SF is applied on one VDA [MSRT] of one unaffected SG supplied with ASG [EFWS] water in addition to the preventive maintenance on another ASG [EFWS] pump associated to another unaffected SG.

  In this case, following VIV [MSIV] closure, the SG pressure in the SG associated with the failed VDA [MSRT] increases to the main steam safety valves setpoint. As the pressure in the other unaffected SG is at the VDA [MSRT] setpoint, which is lower than main steam safety valves setpoint, the steam released by the SG affected by the VDA [MSRT] single failure decreases. Thus the decay heat removal is mainly provided by the two remaining unaffected SG, rather than by three SG. This is another way to reduce the heat removal capability via the secondary side and to increase the RCP [RCS] overheating.

  Section 14.5.3 - Figure 1 shows the SG configurations, regarding the VDA [MSRT] relief and ASG [EFWS] feeding assumptions, used in cases 1 and 2. Section 14.5.3 - Table 1 summarises the definition of cases 1 and 2.

  The intention of the analysis of cases 1 and 2 is to identify the most onerous single failure between one ASG [EFWS] pump or one VDA [MSRT], for the overheating of the RCP [RCS], and the resulting PSV coolant discharge.

3.1.5.2. Initial state

The initial state conditions, given in Section 14.5.3 - Table 5 are chosen to maximise the heat to be removed and consequently the RCP [RCS] overheating. [Ref-1] [Ref-2]

The initial power operation refers to the nominal SG heat transfer area decreased by 10%, to cover potential plugging and fouling of the SG-tubes. This assumption results in higher RCP [RCS] temperatures for the initial state, which increases the initial energy inside the RCP [RCS], and is pessimistic for the RCP [RCS] overheating transient.

3.1.5.3. Specific assumptions

a) Neutronic data and decay heat

Core power is assumed constant at 102% of full power until reactor trip. After RT, the maximum residual heat curve ‘term B+C with 1.645 σ’ as described in Sub-chapter 14.1 is assumed.

Term A (fission heat) is introduced as an input to the CATHARE code, with the CATHARE point-kinetic model not activated. This term A results from a decoupled conservative RT-simulation, (see Section 14.4.5 - Table 1).

b) Assumptions related to control systems (non F1)

Turbine: the steam flow to the turbine is simulated as decreasing linearly with secondary system pressure until the turbine is tripped. This assumes that the opening of the turbine inlet valves remains constant until Turbine Trip (TT). It minimises the turbine flow rate and consequently the heat removal before TT. This maximises the RCP [RCS] overheating.

The impact of this assumption has been compared to the situation of opening the turbine inlet valves in parallel with the secondary system depressurisation to keep the turbine flow constant. This maintains a constant heat removal via this route. This analysis shows the assumption considered results in nearly the same coolant discharges through the PSV and is a small penalty. The slightly larger RCP [RCS] overheating is compensated by a slightly larger water inventory in the unaffected SG at the time of RT. This results in a small increase in PSV coolant discharge.

GCT [MSB]: the GCT [MSB] pressure control is not considered, because it is not actuated before RT, and because the GCT [MSB] is isolated after VIV [MSIV] closure at RT.

ARE/AAD [MFWS/SSS]: all the ARE/AAD [MFWS/SSS] flows are assumed lost to the break via the ARE [MFWS] header. Neither the affected SG nor the unaffected ones are supplied by ARE/AAD [MFWS/SSS] after FWLB occurrence. Consequently, the SG level control is lost as soon as the break occurs.

Heaters/spray: the pressuriser pressure control is not considered because heaters are not actuated and spray is beneficial. In case of LOOP, normal spray would not be available.

c) Assumptions related to F1 systems

Reactor Trip (F1A): This occurs following a “SG level < MIN1” signal. The reference value for the MIN1 setpoint is 13.8 m above the tube sheet. A conservative reduction of 5% of narrow range (NR) span is assumed on the reference value, to minimise the SG liquid inventory after RT. Only SG level signals derived from unaffected SG claimed for RT actuation. The initial level in the unaffected SG is maximised, by increasing the nominal level by 5% of narrow range span, to delay the generation of the RT signal. [Ref-1] [Ref-2]

Turbine Trip (F1A): the turbine is tripped on the RT signal.

VIV [MSIV] (F1A): the VIV [MSIV] of all SG are automatically closed. It is conservatively assumed the VIV [MSIV] begins to close on the “SG level < MIN1” RT signal, not before, with a maximum closure time of 5 seconds.

VDA [MSRT] (F1A): following RT, TT and VIV [MSIV] closure, the unaffected SG pressures increase to the VDA [MSRT] setpoint. In the analysis, the minimum value is retained for the VDA [MSRT] setpoint (95.5 – 1.5 bar) to maximise the discharge and increase the likelihood of emptying the SG. [Ref-1] [Ref-2]

Main steam safety valves (MSSV) (F1A): for the case of SF applied on one VDA [MSRT], the pressure in the corresponding SG increases to the MSSV actuation pressure. The MSSV setpoint value is maximised (105.0 + 1.5 bar) in order to increase the effect of losing the VDA [MSRT]. [Ref-1] [Ref-2]

PSV (F1A): as a consequence of RCP [RCS] overheating, the RCP [RCS] pressure increases to the PSV actuation pressure. The minimum pressure setpoints are considered for PSV actuation to maximise the PSV coolant discharge (PSV1: 174-1.5 bar; PSV2: 178-1.5 bar; PSV3: 178-1.5 bar). [Ref-1] [Ref-2]

ASG [EFWS] (F1A): ASG [EFWS] is actuated following a “SG level < MIN2” signal. The reference value of MIN2 is 7.85 m above the tube sheet. A conservative reduction of the setpoint by -5% of the wide range span is assumed on the reference value. This minimises the SG liquid inventory after ASG [EFWS] actuation. In the case of LOOP, a delay of 50 seconds is assumed between the LOOP occurring and the ASG [EFWS] flow reaching the SG. This includes the effect of the emergency diesel generator (EDG) reloading sequence. [Ref-1] [Ref-2]

The ASG [EFWS] flow rate is the minimum value (see Sub-chapter 14.1).

Isolation of ASG [EFWS] line assigned to the affected SG (F1A): At operator action time of RT + 1 hour is assumed, for a local action, to isolate of the ASG [EFWS] line assigned to the affected SG. This is performed by manual closing of the respective isolation and control valves.

Opening of ASG [EFWS] header at pump discharge (F1A): At operator action time of RT + 1 hour for a local action is assumed for the realignment of the ASG [EFWS] pump of the affected SG to an unaffected and non-supplied SG. This is performed by manual opening of the ASG [EFWS] header, including the opening of at least the two isolation valves connecting the two SG together.

3.1.5.4. Results

Case 1 is calculated with EPR_{4250} characteristics. It shows that the controlled state is reached, the decoupling criteria being met with higher margin compared to case 1 of BDR-99 (see Appendix 14B) with EPR_{4900}. A lower RCP [RCS] coolant discharge via the PSV occurs.

Cases 1 and 2 are not recalculated in the PCSR for EPR_{4500}. The capability to reach the controlled state whilst meeting the associated decoupling criteria is derived from results of cases 1 and 2 analyses performed in the BDR-99 for EPR_{4900} (see section 2.16.1 of Appendix 14B). These calculations show that the decoupling criteria are met with significant margin, and from comparison of relevant characteristics between EPR_{4500} and EPR_{4900}, it is judged that the consequences for EPR_{4500} are less than those for EPR_{4900}.

3.1.5.4.1. Case 1: EPR_{4250} accident analysis

The sequence of events for case 1, FWLB from the initiating event to the controlled state – without LOOP, SF on one ASG [EFWS], maintenance on one ASG [EFWS], is given in Section 14.5.3 - Table 6. [Ref-1]

The main thermal-hydraulic trends in the transient calculation are shown in Section 14.5.3 - Figures 3 through 8:

- primary total mass / RPV upper plenum liquid level,
- hot leg liquid temperature / cold leg liquid temperature,
- SG pressure / ASG [EFWS] flow,
- narrow range SG level / wide range SG level (measurement indications),
- hot leg liquid flow / break flow,
- PSV flow rate / pressuriser pressure.

The duration of the case 1 transient is 4500 seconds, covering the manual ASG [EFWS] realignment via the ASG [EFWS]-pump header at RT+1 hour, after which the required feeding of two unaffected SG by two ASG [EFWS] pumps is operating.

One PSV briefly opens at the time of RT, approximately one minute after the FWLB occurs, for a few seconds. This is due to the RCP [RCS] overpressure peak following TT and VIV [MSIV] isolation due to reduced secondary side heat-removal. The PSV then closes, and re-opens after about 20 minutes, with a sustained discharge at a small flow rate, averaging about 1 kg/s. This PSV opening results from the continuous RCP [RCS] overheating due to the SG water inventory decrease.

The affected SG 2 empties within 1 minute of the break occurring. The two unfed SG 3 and SG 4 empty within about 15 minutes. The fed SG 1 empties within about 30 minutes. At this point its level is stabilised following ASG [EFWS] actuation until SG 3 and SG 4 empty. The ASG [EFWS] flow is sufficient to remove the primary heat actually transferred into SG 1. However, its level decreases following the emptying of SG 3 and SG 4 as the capacity of one ASG [EFWS] pump is not sufficient to remove the entire primary heat load.

At RT + 1hour, the ASG [EFWS] realignment is modelled in the CATHARE calculation. As soon as a second ASG [EFWS] pump is available to feed an unaffected SG the secondary system is able to remove the entire primary heat. In the calculation, the ASG [EFWS] pump initially feeding the affected SG 2 is aligned into the non-supplied SG 3. Consequently, the RCP [RCS] overheating is halted, the PSV closes, and the RCP [RCS] temperatures decrease to the hot shutdown temperature.

The PSV coolant discharge throughout the transient is around 3 te (≈ 0.5 te around the time of RT, 2.5 te during the RCP [RCS] overheating. This is small compared to the initial RCP [RCS] water inventory of approximately 300 te.

The RCP [RCS] loops remain full of liquid throughout the transient. Thus the core cooling is unimpaired, with a large margin, see Section 14.5.3 - Figure 3 RPV upper-plenum level. After the ASG [EFWS] realignment, the heat removal is provided by the two fed SG, the RCP [RCS] water inventory is stable, and the core is sub-critical. The controlled state is reached.

3.1.5.4.2. Case 1: EPR_{4900} accident analysis of BDR-99, transposition to EPR_{4500}

Compared to case 1 with EPR_{4900} characteristics, See section 2.16.1 of Appendix 14B, the RCP [RCS] discharge via the PSV for case 1 of the EPR_{4250} is lower: Three te instead of 35 te, representing 1% of the primary water mass instead of 12%. This decrease of primary water inventory mainly results from the lower power level for EPR_{4250} compared to EPR_{4900} (15% less power).

The power level of the EPR_{4500} is 9% lower than the EPR_{4900} power level, and 6% higher than EPR_{4250} power level. As a consequence, the RCP [RCS] discharge via the PSV for EPR_{4500} will be between 3 and 35 te. The primary loops remain full of liquid and the core cooling is maintained by a large margin.

3.1.5.4.3. Case 2: EPR_{4900} accident analysis of BDR-99, transposition to EPR_{4500}

Results of Case 2 for EPR_{4900} in BDR-99 show that the decrease of RCP [RCS] discharge via the PSV is less than 3 te. This shows that case 2 is not limiting for reaching the controlled state and the associated decoupling criteria.

Based on the result of the case 2 analysis in BDR-99, and the benefit of the lower power level of EPR_{4250} compared to EPR_{4900}, it can be concluded without need for specific calculation that the controlled state is reached in EPR_{4250} case 2. This is demonstrated by the comparison of the results of the respective case 1 calculations.

3.1.5.4.4. *Cases 1 and 2: conclusion*

The results of the analysis of case 1 on EPR_{4250}, and of case 2 on EPR_{4900} provided in BDR-99 with interpolation to EPR_{4250}, demonstrate that:

- With two ASG [EFWS] pumps available to deliver to two of the unaffected SG (single failure or preventive maintenance), the controlled state is reached without need for operator action.
- With one ASG [EFWS] pump only available to deliver to one unaffected SG, due to single failure and preventive maintenance, the controlled state is reached after manual opening of the ASG [EFWS] passive header. This claims benefit for the ASG [EFWS] pump assigned to the affected SG.
- In all cases, the controlled state is reached, despite of the most onerous single failure and of the most onerous preventive maintenance, relying only on F1A means:
    - Control/shutdown rods for core sub-criticality,
    - ASG [EFWS] and VDA [MSRT] for RCP [RCS] heat removal,
    - VIV [MSIV] and ARE [MFWS] isolation for affected-SG isolation,
    - PSV and MSSV for RCP [RCS] and SG overpressure limitation.

3.1.6. Impact of the new design of pressuriser safety valves

The BDR-99 accident analyses, presented in section 2.16.1 of Appendix 14B, have been performed assuming the SEBIM model for the Pressuriser Safety Valves (PSV). The impact of the change in the main PSV parameters for the SEMPELL-type pressuriser safety valves is as follows:

- Opening setpoint: the SEMPELL valve first opening setpoint is higher than the SEBIM valve opening setpoint, the values being 174 bar (without uncertainties) for the SEBIM valve and 175 bar for the SEMPELL valve, respectively. The PSV opening setpoint pressure will be slightly higher and the margin to core saturation will be higher, which will increase the margin to the reactor pressure upper plenum level criterion. Therefore it can be concluded that the change in valve type to the SEMPELL valve results in a less onerous transient due to the increase in the opening setpoint.
- Valve capacity: SEMPELL and SEBIM valves types have the same steam and water capacities. Thus, there is no impact on the resultant flow rates calculated due to the valve type change.
- Hysteresis: The increase in the PSV hysteresis affects the decrease in primary pressure after PSV opening. The effect of a higher PSV hysteresis (28 bars for the SEMPELL valve versus 10 bar for the SEBIM-type PSV) is to reduce the saturation margin in the RCP [RCS]. A reduction in the saturation margin results in a bigger impact on the water level of the reactor pressure upper plenum. Support studies show that the criterion margin is reduced but without any risk of core uncovery.

To conclude, the modification of the PSV model will not lead to core uncovery despite a decrease in the calculated RPV upper plenum level.

3.1.7. Description of Studied Cases 3 and 4 (from the controlled state to the safe shutdown state)

3.1.7.1. Case 3: without LOOP

a) Choice of single failure and preventive maintenance

Single failure and preventive maintenance are chosen to restrict the capacity of F1-classified systems to cool the RCP [RCS] to the LHSI/RHR mode, connection conditions.

Preventive maintenance has no impact on the VDA [MSRT] availability, as preventive maintenance is not performed on the mechanical part of the VDA [MSRT], and preventive maintenance on the electrical part of the VDA [MSRT], e.g. on emergency diesel generators in the case of LOOP, does not disable the VDA [MSRT]:

- During the first two hours, the VDA [MSRT] is operable with a power supply from batteries,
- After two hours, the VDA [MSRT] is operable in local mode.

As a consequence, the most onerous preventive maintenance, maintenance of one safety division, results in the unavailability of one ASG [EFWS] train.

The most conservative single failure for assessment of the RCP [RCS] cooldown is the failure of one VDA [MSRT]. Case 3 is performed without LOOP, since reactor coolant pumps running increases the primary heat, which similarly increases the ASG [EFWS] water consumption.

An additional failure is included, simply to limit the number of cases analysed: for RCP [RCS] boration. In this case, failure of one out of the two RBS [EBS] trains is assumed. The resulting RCP [RCS] cooling rate is restricted to 25°C/h, the minimum, which further increases the ASG [EFWS] water consumption.

It should be noted that preventive maintenance has no impact on RBS [EBS], as preventive maintenance is not performed on the mechanical part of the RBS [EBS], and the power supply is taken from the neighbouring electrical division, via a dedicated cross-connection, when preventive maintenance is taken on the normal electrical division supply.

All these assumptions remain throughout the transient.

Section 14.5.3 - Figure 2 shows the resulting SG configurations for VDA [MSRT] relief and ASG [EFWS] feeding, in both phases before and after ASG [EFWS] realignment. It also indicates the additional single failure on the RBS [EBS]. Section 14.5.3 - Table 2 summarises the definition of case 3.

Case 3 therefore considers the minimum relief capacity for RCP [RCS] cooldown, and the minimum capacity for RCP [RCS] boration. This maximises the ASG [EFWS] water consumption until LHSI/RHR connection.

b) Initial state

The assumptions considered for the calculations of the first phase, from the initiating event to the controlled state, of case 3 are identical to the assumptions described for case 2 as discussed in section 3.1.5.2 of this sub-chapter. The only differences refer to the assumption of "without LOOP" and to the aim of increasing the ASG [EFWS] water consumption:

- No LOOP at RT.
- Time delay between "SG level < MIN2" signal and ASG [EFWS] flow injection equal to 15 seconds as there is no requirement for the emergency diesel generators (EDG) reloading sequence.

c) Specific assumptions

The assumptions specific to the transfer from the Controlled State to the Safe Shutdown State are as follows. The classification indicated refers to the mechanical system. The classification of the relevant I&C is F1B when specific to the phase of post-Controlled State operation as discussed in section 3.1.3 of this sub-chapter:

RBS [EBS] (F1B): one out of two RBS [EBS] trains is manually actuated at the beginning of the RCP [RCS] cooldown phase (2 hours after RT) and shut off when the required boron concentration for RRA [RHRS] connection conditions is reached.

The minimum capacity of 2.8 kg/s is conservatively assumed, as discussed in Sub-chapter 14.1. A Minimum enriched boron concentration of 7000 ppm is assumed. This corresponds to a natural boron concentration of 11825 ppm, required for a MOX fuel management scheme.

VDA [MSRT] (F1A): consistent with the Emergency Operating Procedures, the operator uses the available VDA [MSRT] of the unaffected SG to perform the RCP [RCS] cooldown, via the secondary side. This is undertaken by decreasing the VDA [MSRT] setpoint with a gradient consistent with the required RCP [RCS] cooldown rate.

RCP [RCS] pumps trip (F1B): in accordance with the Emergency Operating Procedures, the operator trips two out of four reactor coolant pumps, before performing the RCP [RCS] cooldown using the unaffected SG.

The RCP [RCS] cooldown at -25°C/h is undertaken by decreasing the VDA [MSRT] setpoint from 94.0 bar (95.5 - 1.5; T_{sat}: 306.5°C) down to 5 bar (T_{sat}: 150°C) over 6.25 hours.

VIV [MSIV] bypass line (F1B): The VIV [MSIV] bypass line is available to decrease its pressure and temperature in an unaffected SG as a back-up to a failed-closed VDA [MSRT].

By opening this bypass, in two appropriate SG, the steam flow path is opened, via the Main Steam Header, between the SG affected by the VDA [MSRT] single failure and another unaffected SG with VDA [MSRT] available. This avoids having a hot unaffected SG without the possibility of cooling and depressurising it.

This bypass is not considered in case 3 without LOOP: with reactor coolant pumps are running, the cooling of the RCP [RCS] loop of the SG affected by the VDA [MSRT] single failure is effective due to mixing between the loops. VIV [MSIV] bypass opening is not required to support the performance of the RCP [RCS] boration and RCP [RCS] depressurisation functions.

ASG [EFWS] (F1A): the ASG [EFWS] characteristics (injection flow) are similar to that described for cases 1 and 2 in section 3.1.5 of this sub-chapter, provided the SG level has not reached its reference level value, at a nominal value of 15.7 m.

Once the reference level is reached, the SG level, as, measured using the wide range indication, is controlled by the ASG [EFWS] according to the following simple principle (CATHARE modelling):

- ASG [EFWS] flow = 0 if SG level > nominal level + 0.2 m,
- ASG [EFWS] flow = minimum ASG [EFWS] flow if SG level < nominal level – 0.2 m.

Claiming the SG level control is conservative for the ASG [EFWS] water consumption needed to reach the safe shutdown.

Isolation of ASG [EFWS] line connected to the affected SG (F1A): isolation of the ASG [EFWS] line connected to the affected SG is undertaken by manual closing of the respective isolation and control valves. This is performed at the operator action time of RT + 1 hour.

ASG [EFWS] header at pump discharge (F1A), at pump suction (F1B): The realignment of the ASG [EFWS] pump of the affected SG to an unaffected and non-supplied SG is performed at the operator action time of RT + 1 hour. This is achieved, by manual opening of the ASG [EFWS] header at the pump discharge, by the opening of at least the two isolation valves connecting the two SG together.

The opening of the ASG [EFWS] header at the pump suction to take credit for the water content of all four ARE [MFWS] tanks is not explicitly modelled in the calculation, as only the ASG [EFWS] injection flow rates are modelled. However, it is implicitly considered.

LHSI/RHR (F1B): LHSI/RHR connection conditions are reached when:

- RCP [RCS] hot leg temperature < 180°C.
- RCP [RCS] hot leg pressure < 30 bar.
- ΔT_{sat} and RPVL consistent with LHSI/RHR suction from the hot leg.

In the CATHARE calculation, at LHSI/RHR connection, the hot legs associated with the cooled loops are full of liquid, in sub-cooled conditions.

3.1.7.2. Case 4: with LOOP

a) Choice of single failure and preventive maintenance

Single failure and preventive maintenance assumptions are chosen to limit the capacity of the F1-classified systems claimed to provide RCP [RCS] boration and depressurisation during the long term phase of the FWLB with which a LOOP is coincident.

Preventive maintenance has no impact on the VDA [MSRT] and RBS [EBS] availability, as discussed for case 3 in section 3.1.7.1 above.

As a consequence, the most onerous preventive maintenance, maintenance of one safety division or one diesel, results in the unavailability of one ASG [EFWS] train.

The most onerous single failure for the RCP [RCS] depressurisation is the failure of one diesel, causing the failure of one ASG [EFWS] pump. Case 4 is performed with LOOP, since RCP [RCS] depressurisation is limited by the presence of hot non-cooled RCP [RCS] loops. The heat removal capacity of the secondary system is reduced because two SG rapidly empty, the broken SG and one unfed SG. This maximises the primary pressure at the end of the cooldown and hence the primary water mass to be lost via the PSV before reaching the LHSI/RHR connecting conditions.

As in case 3, an additional failure is considered, purely to limit the number of cases analysed. When considering RCP [RCS] boration, failure of one out of two RBS [EBS] trains is assumed. The resulting RCP [RCS] cooling rate is limited to 25°C/h.

To depressurise to the RCP [RCS] LHSI/RHR connection conditions at the end of the transfer to the Safe Shutdown State, it is necessary to open a RCP [RCS] relief path. This is due to the presence of non-cooled RCP [RCS] loops due to RCP [RCS] natural circulation restricting the mixing between loops. This depressurisation is performed using only one out of three PSV, which has sufficient capacity.

Section 14.5.3 - Figure 2 shows the resulting SG configurations for VDA [MSRT] relief and ASG [EFWS] feeding. This covers both phases, before and after ASG [EFWS] realignment. In addition, the single failure on RBS [EBS] is also indicated. Section 14.5.3 - Table 3 summarises the definition of case 4.

Case 4 therefore provides the most onerous case for RCP [RCS] depressurisation, the RCP [RCS] in natural circulation with two non-cooled loops.

> Note: *Case 4 has been defined so that only two SGs are available for RCP [RCS] cooldown to the Safe Shutdown State. It shows that in the most onerous PCC for heat removal (FWLB), only two SGs are needed to reach the Safe Shutdown State (two ASG [EFWS] plus two VDA [MSRT]).*
>
> *However, the operator would have the opportunity to use the three unaffected SGs instead of only two, via a realignment of one ASG [EFWS] pump to two SG, for feed and/or the use of the VIV [MSIV] bypass in place of a failed VDA [MSRT], for relief. This would result in only one non-cooled loop instead of two, with less restrictive RCP [RCS] conditions for the depressurisation. The effect on PSV water discharge is presented below.*

b) Initial state

The assumptions made for the calculation of the first phase, from initiating event to the controlled state, of case 4 are identical to the assumptions made for case 2, as discussed in section 3.1.5.2 of this sub-chapter.

c) Specific assumptions

The assumptions specific to the transfer from the controlled state up to the safe shutdown state are the following:

RBS [EBS] (F1B): same assumptions as in case 3 as discussed in section 3.1.7.1 of this sub-chapter. Each RBS [EBS] train injects into two cold legs, with one isolation valve in each injection line. In the case of a FWLB with reactor coolant pumps off, the operator will close the RBS [EBS] line(s) into un-cooled loop(s), in order to fully align RBS [EBS] into the cooled loop(s) which experience natural circulation flow.

VDA [MSRT] (F1A): same assumptions as in case 3 as discussed in section 3.1.7.1 of this sub-chapter.

PSV (F1A system, F1B operator action): at the end of the RCP [RCS] cooldown phase using unaffected SG, RCP [RCS] depressurisation is needed. This is performed by one PSV. In this case, the PSV setpoint is decreased to 10 bar while the LHSI is operating.

When the LHSI/RHR connection conditions are reached (P_{HL} < 30 bar, T_{HL} < 180°C), the operator closes the PSV by increasing its setpoint.

The minimum capacity of PSV is assumed, as discussed in Sub-chapter 14.1.

ASG [EFWS] isolation to the affected SG (F1A): same assumption as for case 3 as discussed in section 3.1.7.1 of this sub-chapter.

ASG [EFWS] (F1A): same assumptions as for case 3 as discussed in section 3.1.7.1 of this sub-chapter.

Opening of ASG [EFWS] header at pump discharge (F1A), at pump suction (F1B): same assumptions as in case 3 as discussed in section 3.1.7.1 of this sub-chapter.

LHSI/RHR (F1B): same assumptions as for case 3 as discussed in section 3.1.7.1 of this sub-chapter.

3.1.7.3. Results

Cases 3 and 4 are not recalculated in the PCSR for EPR_{4500}. The demonstration of the capability to reach the Safe Shutdown State whilst meeting the associated decoupling criteria is derived from results of cases 3 and 4 analyses performed in the BDR-99 for EPR_{4900}. These show that they are met with significant margin. From a comparison of relevant characteristics between EPR_{4500} and EPR $_{4900}$ EPR_{4500} is more favourable than EPR_{4900}. Thus the criteria will be met for EPR_{4500}.

3.1.7.3.1. Case 3: EPR_{4900} accident analysis of BDR-99

The results of the case 3 analysis of BDR-99 for EPR_{4900}, FWLB – transfer to Safe Shutdown State – without LOOP, are presented in section 2.16.1 of Appendix 14B. They are summarised below:

- At the end of the long term RCP [RCS] cooldown phase via the unaffected SG, pressure and temperature in the RCP [RCS] are consistent with LHSI/RHR connection conditions. No additional RCP [RCS] depressurisation e.g. via opening of RCP [RCS] a PSV is required.

- All RCP [RCS] loops, including the affected one are at a temperature lower than 180°C. This is because reactor coolant pumps are in operation. The RCP [RCS] pressure, as a result of RCP [RCS] cooldown, is lower than 30 bar.

- The boron concentration in the core necessary for LHSI/RHR connection conditions has been reached within 2 hours of the start of the cooldown.

- The water consumption by the end of the cooldown of 1430 te, including 100 te lost via the break before isolation of the affected SG by the operator at 1 hour, is less than the ASG [EFWS] tanks capacity of 1500 te. Further, at that time the two SG used for heat-removal are full of water with the SG level equal to the nominal level.

3.1.7.3.2. Case 4: EPR_{4900} accident analysis of BDR-99

The results of case 4 analysis of BDR-99 for EPR_{4900}, FWLB – transfer to safe shutdown state – with LOOP, are provided in section 2.16.1 of Appendix 14B. They are summarised below:

- At the end of the long term RCP [RCS] cooldown phase via the unaffected SG with a VDA [MSRT] pressure setpoint of 4 bar, the RCP [RCS] temperature in the two cooled loops is below 180°C. However, the RCP [RCS] pressure is higher than the LHSI/RHR connecting conditions (P ≈110 bar > 30 bar).

- This high pressure level is caused by the pressurising effect of the hot legs associated with the affected SG and unfed intact SG. In these hot legs, saturation conditions have been reached. After the momentarily opening of one PSV by the operator[7], to depressurise the RCP [RCS] and to cool down the RCP [RCS] loops, the LHSI/RHR connection conditions are reached in the heat removing loops (T < 180°C, P < 30 bar).

- The boron concentration in the core required for LHSI/RHR connection conditions is reached within 2 hours of the start of the cooldown.

3.1.7.3.3. Cases 3 and 4: transposition of BDR-99 for EPR_{4900} results to EPR_{4500}

The results of analysis of both cases 3 and 4 of BDR-99 for EPR_{4900}, demonstrate that the F1-classified systems are sufficient to reach the safe shutdown RCP [RCS] boron concentration and to cool and depressurise the RCP [RCS] down to LHSI/RHR connection conditions within a time period consistent with the ASG [EFWS] tanks capacity. This is achieved despite the most onerous single failure and the most onerous preventive maintenance.

The F1-classified systems involved in the achievement of the safe shutdown state are:

- ASG [EFWS] and VDA [MSRT] for RCP [RCS] cooling.

- RBS [EBS] for RCP [RCS] boration.

- PSV for RCP [RCS] depressurisation.

- LHSI for long term RCP [RCS] heat removal.

When compared to the EPR_{4900} characteristics, the EPR_{4500} is more favourable for RCP [RCS] cooling capability, and similar for the RCP [RCS] boration capability:

- Power level 9% less in EPR_{4500} compared to EPR_{4900}.

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[7] PSV discharge is about 50 te if RCP [RCS] cooling has been performed by only two SG, and less than 5 te if RCS cooling has been performed by three SG (via MSIV bypass opening), to be compared with an actual RCP [RCS] water inventory of approximately 320 te.

- ASG [EFWS] tanks water inventory slightly higher in EPR_{4500} compared to EPR_{4900}, thus providing higher heat removal capacity with respect to the power level.

- Similar ASG [EFWS] and VDA [MSRT] flow rates between EPR_{4500} and EPR_{4900}, in terms of percent of nominal power level.

- Same RBS [EBS] boration capacity between EPR_{4500} and EPR_{4900}.

- Same PSV relief capacity between EPR_{4500} and EPR_{4900}.

Based on the presented results of cases 3 and 4 analyses in BDR-99 for EPR_{4900}, and on the beneficial aspect of EPR_{4500} compared to EPR_{4900} for the RCP [RCS] cooldown, and similar aspect for the RCP [RCS] boration, it can be concluded, without need for separate calculation, that the Safe Shutdown State is achieved in EPR_{4500} cases 3 and 4.

3.1.7.4. Impact of the change to the safety classification of the normal spray operations

The F1B safety classification for the normal spray allows this system to be claimed in the transition to reach the LHSI/RHR connecting conditions. As a consequence, the pressuriser safety valves do not need to be used during the transfer between the controlled and the safe shutdown states. This modification requires the energy formerly released via the PSV to be removed by the steam generators. Thus, this modification can affect the ASG [EFWS] water consumption requirements.

The feedwater line break from the initiating event to the safe shutdown state without LOOP is used as the ASG [EFWS] tanks sizing transient, as the reactor coolant pumps are kept running.

The BDR-99 tank sizing calculations are performed at 4900 MW. The PSV flow behaviour during the feedwater line break presented in Appendix 14B.2.16.1 - Figure 16 shows that the PSV are not required to open to reach the safe shutdown state. As a result, there is no impact from the change in the normal spray classification to F1B on the ASG [EFWS] tank sizing.

3.1.8. Conclusion

The present analysis of the FWLB accident shows that despite of the most onerous single failure and preventive maintenance:

- The Controlled State is reached, using only F1A means:
    - Control/shutdown rods for core sub-criticality.
    - ASG [EFWS] and VDA [MSRT] for RCP [RCS] heat removal.
    - VIV [MSIV] and ARE [MFWS] isolation for affected-SG isolation.
    - PSV and MSSV for RCP [RCS] and SG overpressure limitation.
- The Safe Shutdown State is reached, using only F1A and F1B means:
    - RBS [EBS] for RCP [RCS] boration.
    - ASG [EFWS] and VDA [MSRT] for RCP [RCS] cooling.

  - o PSV, and VIV [MSIV] bypass useful but not necessary, for RCP [RCS] depressurisation in the case of LOOP.
  - o LHSI for long term heat removal.

- The RCP [RCS] water inventory is always sufficient for the core cooling requirements. The decoupling criterion of "fuel cladding integrity" is met, with a large margin.

3.2. FEEDWATER SYSTEM PIPE BREAK IN STATE B

3.2.1. Accident definition

A feedwater line break in state B is classified as a PCC-4 event.

The accident examined is defined as a break located on a feedwater system pipe, sufficiently large to prevent the steam generator from being supplied with water. Breaks located between the SG and the check valve result in both a loss of SG feedwater flow and a loss of SG water inventory. The SG water content of the unaffected SG decreases as well, but not as rapidly as in the affected SG. This is due to the feedwater flow decrease, for a break located in the ARE [MFWS] header, and to the secondary system depressurisation whilst the main steam header (MSH) is unisolated.

The heat removal capability of the secondary system is impacted. Depending on the break size and location, the accident can lead either to a RCP [RCS] overcooling, due to an excessive steam discharge through the break, or to a RCP [RCS] overheating, due to an excessive water discharge through the break. The effect of a RCP [RCS] overcooling is bounded by the "steam line break" analysis discussed in section 2 above. Therefore only the RCP [RCS] overheating aspect is considered here.

The main difference, in terms of protection and mitigation actions, between a feedwater line break in state A, analysed in section 3.1 above and a feedwater line break in state B, is the following one:

- The "SG pressure < MIN1” or "SG pressure < MIN2" signals, leading to all VIV [MSIV] closure and closure of the ARE [MFWS] line of the affected steam generator respectively, have been deactivated for RCP [RCS] normal cooldown purpose, when leaving hot shutdown conditions.
- In state B, such isolations occur only following a “SG pressure drop” signal.
- Consequently, there is no difference for the larger feedwater line breaks in state B compared to the mitigation actions in state A, which actuate the pressure drop signal at - 2 bar/min .However, the VIV [MSIV] closure and ARE [MFWS] isolation signals from the secondary side are no longer available for the smaller breaks which do not actuate the pressure drop signal.

The main differences, in terms of plant operating conditions, between a feedwater line break in state A, analysed in section 3.1 of this sub-chapter, and a feedwater line break in state B, are the following:

- The initial power level is lower in state B.
- The initial SG water inventory is larger in state B.

Such differences are of significant benefit for the feedwater line break transient in state B.

3.2.2. Safety criteria

The safety criteria and decoupling criteria are the same as those described in section 3.1.2 of this sub-chapter, for a feedwater line break in state A.

The effect of feedwater line break are analysed for the following decoupling criteria:

- Fuel cladding integrity.
- Reactor coolant pressure boundary (RCPB) integrity.
- Reactor containment pressure and temperature boundaries.
- Amount of radioactive products released to the environment.

,The FWLB accident is not limiting for the RCPB overpressure criteria, being bounded by the ‘primary overpressure’ analyses addressed in the sub-chapter related to ‘Topics specific to mechanical components’, see Sub-chapter 3.4.

The FWLB accident is bounded by the ‘steam line break’ analysis for the reactor containment pressure and temperature boundaries, as discussed in Appendix 6A.

The FWLB accident is covered by the ‘SGTR 2 tubes’ analysis for the criteria assessing the amount of radioactive products released, as discussed in section 10 of this sub-chapter.

Consequence, the FWLB is analysed in this section to address only the fuel cladding integrity criteria.

3.2.3. Method and assumptions

a) Method of analysis

A quantitative argument is used to demonstrate that the core cooling is not impaired.

The conservative PCC analysis rules defined in Sub-chapter 14.0, ‘PCC accident analysis rules’ are used.

b) Assumptions

The F1A systems defined in section 3.1 of this sub-chapter for the FWLB in state A are available for the FWLB in state B. The F1A automatic actions are the same as in state A with the following exceptions. The RT/TT signals are no longer required as the plant is shutdown. The VIV [MSIV] and ARE [MFWS] isolation signals on "SG pressure < MIN1" and "SG pressure < MIN2" are unavailable as these are deactivated in state B, as discussed in section 3.2.1 of this sub-chapter.

The F1A automatic I&C functions relevant for a FWLB in state B are:

- First operator information:
    - SG level < MIN1.

- Closure of all main steam isolation valves (F1A) on:
  - SG pressure drop > MAX1.
- Closure of main feedwater line of the affected steam generator (F1A, SG specific) on:
  - SG pressure drop > MAX2.
- Emergency feedwater actuation in each steam generator (F1A, SG specific) on:
  - SG water level < MIN2.
- VDA [MSRT] actuation (F1A) on:
  - SG pressure > actual set point.

Signal coming from the affected SG are not claimed in the analysis. This is a conservative approach, as followed for the FWLB in state A.

- The F1B systems defined in section 3.1 of this sub-chapter for the FWLB in state A are available for the FWLB in state B. The operator actions needed to transfer the plant from the controlled state to the safe shutdown state are the same as in state A, except that the ASG [EFWS] passive header (pump discharge) opening is not required. As the heat removal requirements are lower in State B, the ASG [EFWS] flow injection is already sufficient.

As for FWLB in state A, no operator action is considered before 30 minutes following the first F1A information provided to the operator.

In state B, LOOP is not included with the initiating event.

3.2.4. Definition of the cases studied

The analysis covers all break sizes, located either on the ARE [MFWS] or ASG [EFWS] or APG [SGBS] line, between the SG and the feed line isolation valve. Should the break be located upstream of the ARE [MFWS] or ASG [EFWS] check valve or downstream of the APG [SGBS] isolation valves, the consequences would be less severe.

The spectrum of FWLB sizes and locations can be split into two groups:

- Large breaks for which the SG pressure drop signal is reached

  Following the break initiation, the secondary system rapidly depressurises. The SG pressure drop signal setpoint is reached, which results in all VIV [MSIV] closure and ARE [MFWS] isolation. Following this isolation the unaffected SGs are decoupled from the affected SG and water is no longer drained from the unaffected SG because of the break.

  The automatic protection and mitigation actions are the same in state B as in state A, with less severe plant conditions in state B:

  - Lower initial power level.
  - Larger initial SG water inventory.

Therefore the consequences of an FWLB in state B that initiates a SG pressure drop signal are bounded by the consequences of the event in state A. Separate analyses are not required.

- Small breaks for which the SG pressure drop signal is not reached

  Following the break initiation, the secondary system slowly depressurises. The VIV [MSIV] are not closed and ARE [MFWS] is not isolated as the SG pressure drop signal (- 2 bar/min) is not reached and the "SG pressure < MIN1" (50 bar) or "SG pressure < MIN2" (40 bar) signals are deactivated.

  Therefore, the unaffected SG remain coupled to the affected SG via the open main steam header during a longer time period than for FWLB in state A. VIV [MSIV] isolation is only possible by operator action, assumed to occur 30 minutes after the first relevant information.

The following "FWLB in state B" study demonstrates that this late VIV [MSIV] isolation does not impair meeting the secondary side heat removal requirements.

3.2.5. Description of cases studied (from the initiating event to the controlled state)

The controlled state is defined as a state where the RCP [RCS] temperature and pressure are stabilised, so that the primary side heat is fully removed via the ASG [EFWS] supplied SG.

a) Choice of single failure and preventive maintenance

The most onerous single failure is the failure of one ASG [EFWS] pump to start on demand, leading to the minimum capacity of emergency feeding for SG water inventory recovery.

The most restrictive preventive maintenance is maintenance on one safety division which leads to the unavailability of another ASG [EFWS] pump.

Consequently, one ASG [EFWS] pump is available for injection into one unaffected SG, the SG configuration being described in Section 14.5.3 - Table 4.

b) Initial state

State B is an intermediate shutdown above 120°C, as defined in Sub-chapter 14.0. The LHSI/RHR is not connected to the RCP [RCS], in normal operation, above this temperature.

At the time of the break, the complete loss of ARE/AAD [MFWS/SSS] flow rate is conservatively assumed in all four SG. In addition, no credit is taken for any further feeding of the affected SG by the associated ASG [EFWS] pump.

The initial power level is the maximum decay heat value, discussed in Sub-chapter 14.1 ‘Term B+C with 1.645 σ’ at the beginning of state B, assumed to be 5 hours after reactor shutdown. The maximum core power level of 102% nominal power is assumed prior to the reactor shutdown. The resulting maximum decay heat power level in state B is:

- Maximum decay heat at 5 hours = 0.91%.
- Maximum core power before shutdown = 102% x 4500 MW_{th}.

- Maximum core power in state B = 42 MW_{th}.

The four reactor coolant pumps are running, leading to the maximum power input into the primary system. The associated power level is:

- Four reactor coolant pumps power, transferred to RCP [RCS] coolant = 30 MW_{th}. [Ref-1]

c) Result

The FWLB considered is of the maximum size which does not activate the "pressure drop" signal. Therefore, there is no automatic VIV [MSIV] isolation.

Following the break initiation, the secondary system depressurises and the affected SG liquid level decreases as liquid mass is lost via the break. Due to the assumed loss of main feedwater flow via the break, the unaffected SG liquid levels also decrease, but not as rapidly as the affected SG liquid level.

The decrease of liquid mass in the SG leads to a reduction of the heat removal capability by the secondary system and consequently may lead to overheating of a RCP [RCS].

If no signal is assumed to come from the affected SG, a F1A signal is sent to the operator (1st information) once the SG level in one unaffected SG reaches the MIN1 setpoint.

For the break size modelled, the SG depressurisation is not sufficient to generate a "pressure drop" signal (- 2 bar/min) signal. Therefore, there is no VIV [MSIV] isolation, and depressurisation of all four SG continues. The loss of SG water inventory continues until the automatic ASG [EFWS] actuation signal setpoint "SG level < MIN2" (7.85 m) is reached in the unaffected SG. At this time, the primary side heat removal has not been impaired as there is still a high water inventory in the unaffected SG.

Only one ASG [EFWS] pump is actuated on SG level MIN2 (SG specific). This accounts for the single failure of one ASG [EFWS] pump assigned to one unaffected SG, and the preventive maintenance of one ASG [EFWS] pump assigned to another one unaffected SG. The one ASG [EFWS] pump delivers the minimum flow of 90 t/h of cold water, at a maximum temperature of 50°C to one unaffected SG. [Ref-2] [Ref-3]

<u>The resulting heat removal capability is:</u>

- One ASG [EFWS] pump flow rate (min) = 25 kg/s or 90 t/h.
- Heat of vaporisation, from 50°C = 2525 kJ/kg. [Ref-2] [Ref-3]
- One ASG [EFWS] pump heat removal (min) = 63 MW_{th}.

<u>The actual primary side heat removal requirements are:</u>

- Maximum core power in state B = 42 MW_{th}.
- Four RCP power, transferred to RCP [RCS] = 30 MW_{th}.
- Total heat removal needs (maximum) = 72 MW_{th}.

The resulting heat balance is:

- Fraction of primary heat removed by the ASG [EFWS] pump = 88%, i.e. 63 MW_{th}.
- Fraction of primary heat stored in the RCP [RCS] coolant = 12%, i.e. 9 MW_{th}.

Therefore, once the "SG level MIN2" is reached, the ASG [EFWS] flow sent towards the unaffected SG is able to remove more than 88% of the primary side heat including the power from four reactor coolant pumps. The remaining power is stored in the RCP [RCS] coolant, causing a coolant temperature increase.

The RCP [RCS] coolant temperature increase over 30 minutes is:

- Power stored within the RCP [RCS] coolant = 9 MW_{th}
- RCP [RCS] coolant mass = 300 te
- RCP [RCS] coolant enthalpy raise over 30 minutes = 54 kJ/kg [8]
- Heat capacity of RCP [RCS] coolant = 5.4 kJ/°C-kg
- RCP [RCS] temperature increase over 30 min = 10°C [9]

The RCP [RCS] coolant temperature increase before operator action after 30 minutes is thus limited to 10°C. At the time of operator action, shutdown of two reactor coolant pumps by the operator is sufficient to halt the RCP [RCS] coolant temperature increase. The ASG [EFWS] pump flow is then able to remove the whole primary heat.

The heat balance at ASG [EFWS] pump actuation + 30 minutes is:

- Core power (maximum) = 42 MW_{th}
- Two reactor coolant pumps power, transferred to RCP [RCS] = 15 MW_{th}
- Total heat removal needs (maximum) = 57 MW_{th}
- One ASG [EFWS] pump heat removal (minimum) = 63 MW_{th}

Assuming a maximum coolant temperature in state B of 300°C, the temperature rise of the RCP [RCS] is stopped at 310°C. This is well below the saturation temperature of 350°C. The RCP [RCS] water inventory has not been affected. The primary side heat removal has always been sufficient to ensure core cooling.

The controlled state is reached 30 minutes after ASG [EFWS] pumps actuation.

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[8] 9000 kJ/s x 1800 s / 300,000 kg.
[9] 54 kJ/kg / 5.4 kJ/°C-kg.

3.2.6. Description of studied cases (from the controlled state to the safe shutdown state)

The safe shutdown state is defined as a state where the core is sub-critical, the LHSI/RHR operating conditions have been reached and the affected steam generator is isolated. In this state, the heat removal function is performed by the LHSI/RHR.

The sequence of actions to be performed by the operator to reach the LHSI/RHR operating conditions are identical to those presented in section 3.1 of this sub-chapter for a feedwater line break in state A. The only exception is that ASG [EFWS] passive header opening at the pump discharge may not be needed. However, this may still be useful if ASG [EFWS] pump failure has occurred, but is not required).

The analysis of the plant transfer from the controlled state to the safe shutdown state following a FWLB in state B is bounded by the equivalent analysis following a FWLB in state A, for the following reasons:

- Decay heat to remove is lower.
- Secondary side water inventory is larger.
- ASG [EFWS] tanks water inventory is higher compared to the actual core power level.
- RCP [RCS] boron concentration is higher.

3.2.7. Conclusion

The consequences of a feedwater line break in state B are less severe than in state A.

At least one SG remains available for heat removal and RCP [RCS] cooling throughout the transient.

The RCP [RCS] water inventory and then the core cooling capability are never challenged. The decoupling criterion of "fuel cladding integrity" is met by a large margin.

3.3. SYSTEM SIZING

The ASG [EFWS] minimum flow rate is designed to reach the controlled state following a feedwater line break.

The ASG [EFWS] tanks are designed to allow the RIS/RRA [SIS/RHRS] connection following a feedwater line break.

SECTION 14.5.3 - TABLE 1

Definition of cases 1 and 2

From Initiating Event up to the Controlled State:	
Case 1 (102% FP, no LOOP)	**Case 2 (102% FP, LOOP)**
min ASG [EFWS] - 1 broken SG - 1 SG with relief and feeding - 2 SG with relief	**min VDA [MSRT]** - 1 broken SG - 1 SG with relief and feeding - 1 SG with relief - 1 SG with feeding
Results: max RCP [RCS] overheating	

SECTION 14.5.3 - TABLE 2

Definition of case 3

Case 3 (102 % FP, w/o LOOP)
<u>**min VDA [MSRT], min RBS [EBS]**</u> **From Initiating Event up to the Controlled State:** 1 broken SG 1 SG with relief and feeding 1 SG with relief 1 SG with feeding **From the Controlled State up to the Safe Shutdown State:** 1 broken SG 1 SG with relief and feeding 1 SG with relief and feeding (after ASG [EFWS] realignment) 1 SG with feeding
Result: max ASG [EFWS] needs (RCP [RCS] cooling concern)

SECTION 14.5.3 - TABLE 3

Definition of case 4

Case 4 (102 % FP, with LOOP)
<u>min VDA [MSRT], min RBS [EBS]</u> **From Initiating Event up to the Controlled State:** 1 broken 1 SG with relief and feeding 2 SG with relief **From the Controlled State up to the Safe Shutdown State:** 1 broken SG 1 SG with relief and feeding 1 SG with relief and feeding (after ASG [EFWS] realignment) 1 SG with relief
Result: max PSV water discharge (RCP [RCS] depressurisation concern)

SECTION 14.5.3 - TABLE 4

SG configuration

From Initiating Event up to the Controlled State
FWLB in state B
<u>Min ASG [EFWS]</u> - 1 broken SG (no feeding considered) - 1 SG with relief - 1 SG with relief - 1 SG with relief and feeding
Result: max RCP [RCS] overheating

SECTION 14.5.3 - TABLE 5 [Ref-1] [Ref-2]

FWLB (state A): Initial Conditions for CASE 1 (EPR 4250MWth)

Parameters	**Limiting values used**
Reactor coolant system	
- Initial reactor power (% of nominal power)	100 + 2 = 102% (4335 MW_{th})
- Initial average RCP [RCS] temperature	313.6 + 2.5 = 316.1°C
- Initial reactor coolant pressure	155 + 2.5 = 157.5 bar
- Reactor coolant flow	22135 kg/s (T/H design flow rate)
- Pressuriser water volume / level	43.4 m^3 (nominal + 5% MR)
Steam generators	
- Initial steam pressure (bar)	Consistent with RCP [RCS] temperature
- Initial SG level in unaffected SG	16.0 m (nominal + 5 % NR)
- Initial SG level in affected SG	15.4 m (nominal - 5 % NR)
Feedwater	
- Main feedwater flow (% of nominal flow)	100 + 2 = 102%
- Initial ARE [MFWS] temperature	232°C

SECTION 14.5.3 - TABLE 6 [Ref-1]

Sequence of Events - Case 1 (FWLB - from INITIATING EVENT up to the Controlled State - without LOOP - SF on 1 ASG [EFWS])

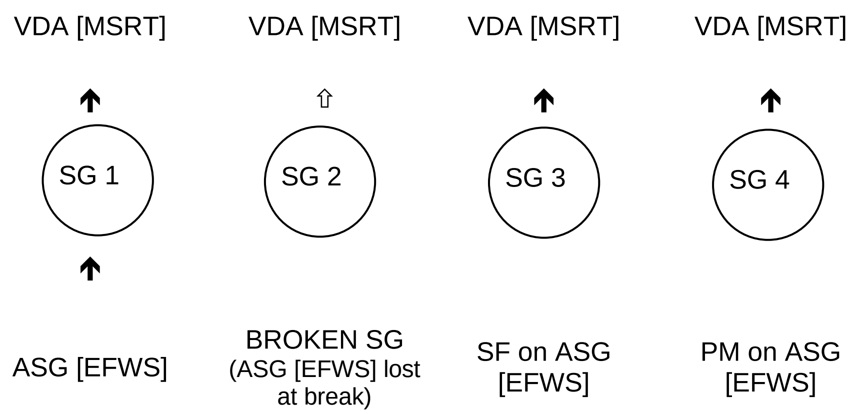
0 s	FWLB occurrs in SG 2 All four ARE [MFWS] flows are lost via the break
14 s	Signal "dP/dt > MAX1" generated in SG 2 *signal not claimed in the accident analysis*
41 s	Signal "dP/dt > MAX2" generated in SG 2 *signal not claimed in the accident analysis*
55.8 s	Setpoint "SG level < MIN1" reached in unaffected SG
57.3 s	Signal RT "SG level < MIN1" generated in unaffected SG Signal TT Signal VIV [MSIV] isolation
57.6 s	Beginning of rod drop Turbine inlet valve closed
62.3 s	VIV [MSIV] closed
100 s	VDA [MSRT] setpoints are reached in all unaffected SG, VDA [MSRT] opening demanded.
109 s	Signal ASG [EFWS] actuation "SG level < Min2" generated in SG 1
123 s	Beginning of ASG [EFWS] flow injection to SG 1 (15 s delay between actuation signal and beginning of injection for case without LOOP). ASG [EFWS] pump connected to SG 4 unavailable due to PM. ASG [EFWS] pump connected to SG 3 unavailable due to SF. Note: ASG [EFWS] flow towards affected SG 2 fully lost via the break.
≈1000s	SG 3 and SG 4 empty (unfed SG)
≈1200 s	PSV sustained opening due to RCP [RCS] overheating (small discharge flow rate: average value app. 1 kg/s)
≈2000s	SG 1 empty (fed SG)
3660 s	The operator performs the realignment of the ASG [EFWS] pump connected to the affected SG (SG 2) to one unaffected and non-supplied SG (SG 3). PSV closes (two ASG [EFWS] pumps able to remove the entire primary heat) The Controlled State is reached.

SECTION 14.5.3 - FIGURE 1

FWLB (State A): Illustration of the Different FWLB Cases Analysed

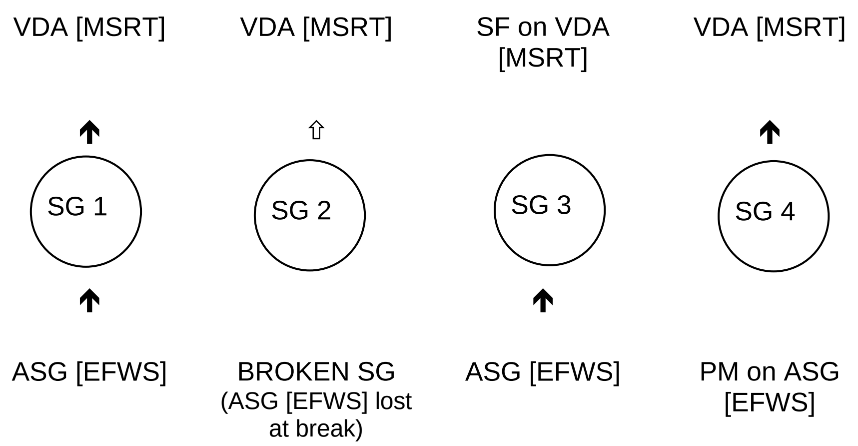
Cases 1 and 2 (Objective = Max RCP [RCS] Overheating)

CASE 1 :	**From Initiating Event up to Controlled State (without LOOP)**



Note: at t_{RT} + 1 hour, realignment of the ASG [EFWS] pump associated to SG 2 towards the SG 3

CASE 2 :	**From Initiating Event up to Controlled State (with LOOP)**

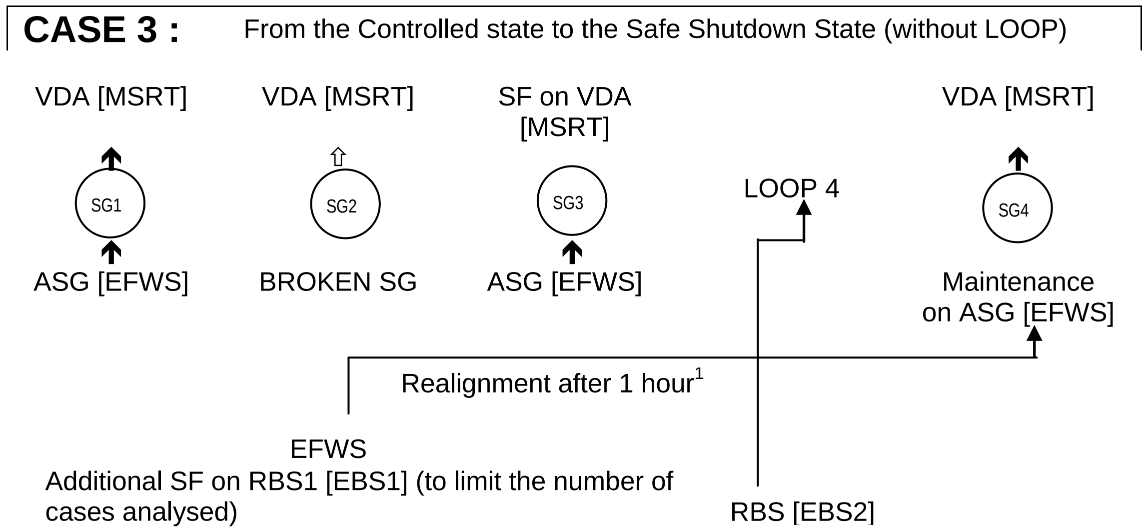


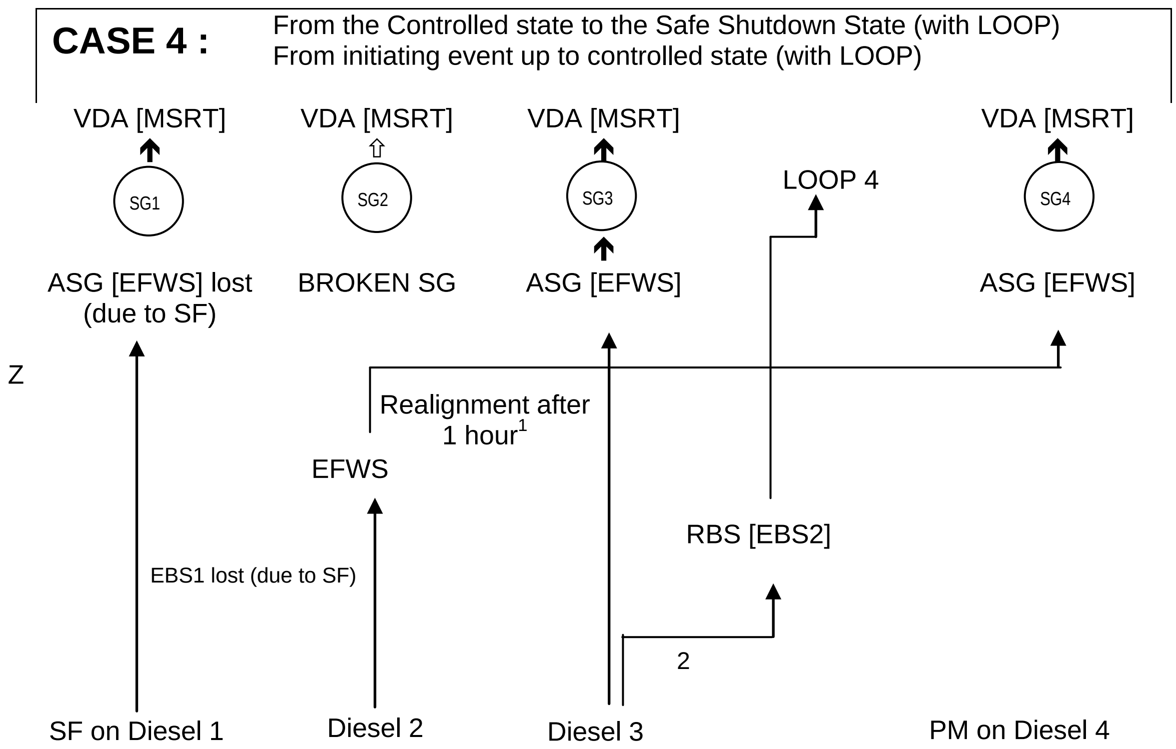
SECTION 14.5.3 - FIGURE 2

FWLB (State A) - Illustration of the Different Cases Analysed

Case 3 (Criterion = ASG [EFWS] Tanks Capacity)

Case 4 (Criterion = RCP [RCS] Boration and Depressurisation)

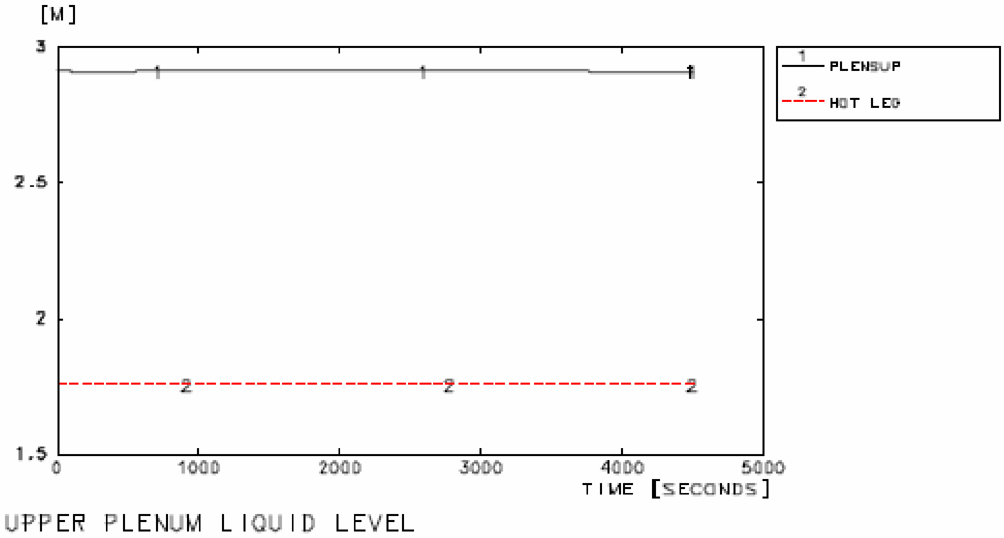


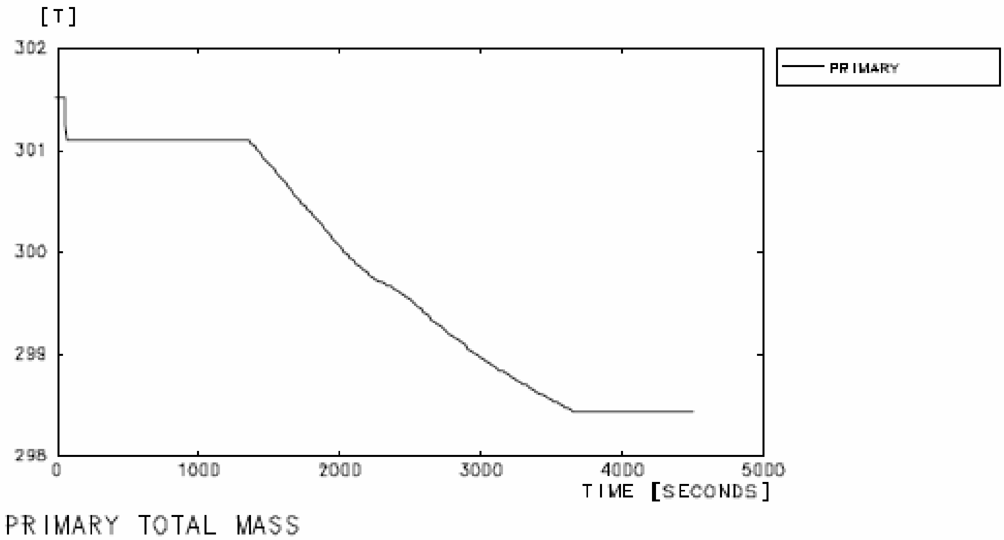


[1] via ASG [EFWS] passive header
[2] via RBS [EBS] power supply cross connection

SECTION 14.5.3 - FIGURE 3 [Ref-1]

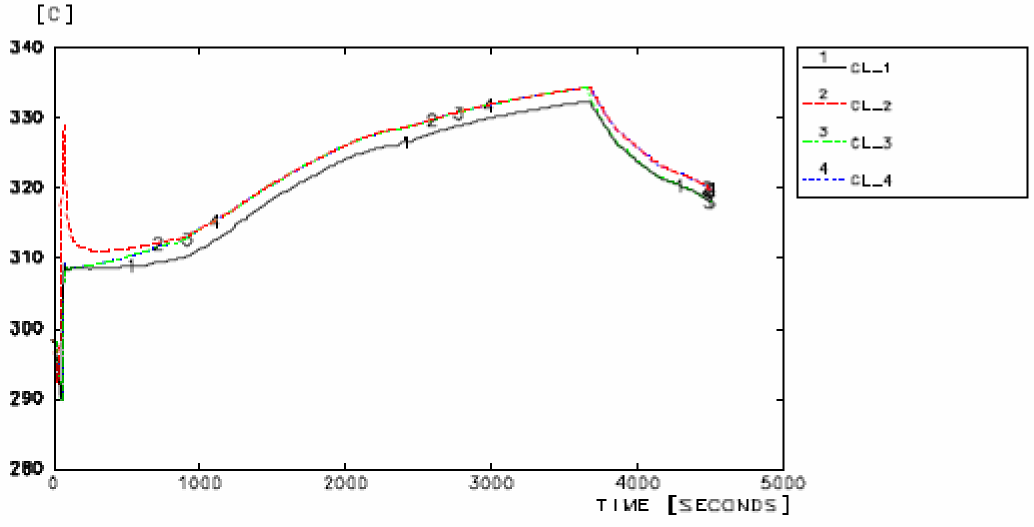
Case 1 - 1 EPR 4250MWth - from Initiating Event up to Controlled State - without LOOP SF on One ASG [EFWS]



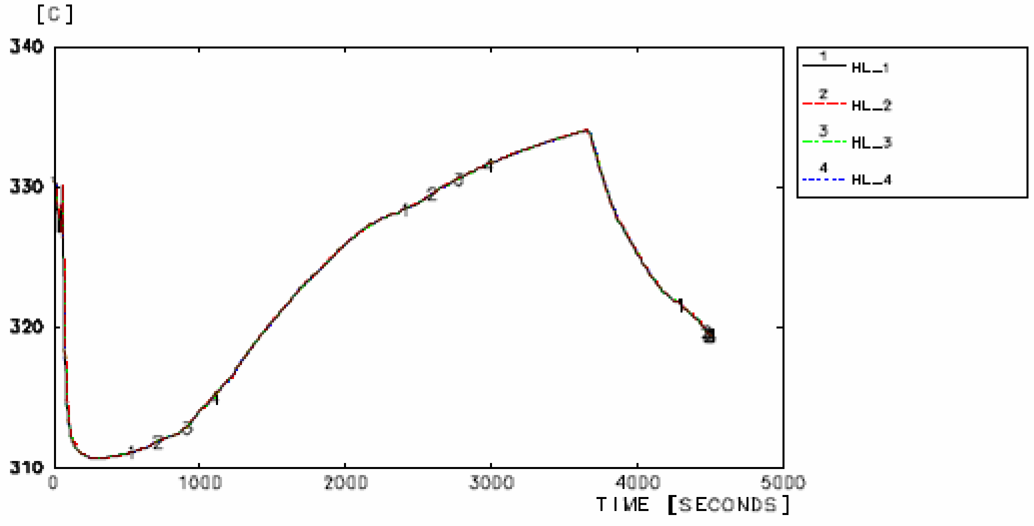


SECTION 14.5.3 - FIGURE 4 [Ref-1]

Case 1 - 1 EPR 4250MWth - from Initiating Event up to Controlled State - without LOOP SF on One ASG [EFWS]



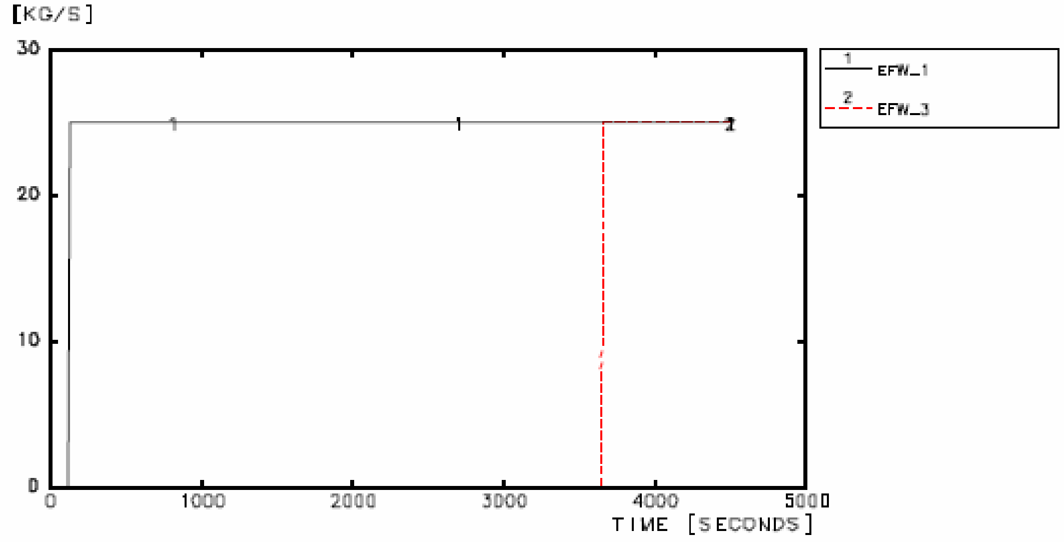
COLD LEG LIQUID TEMPERATURE



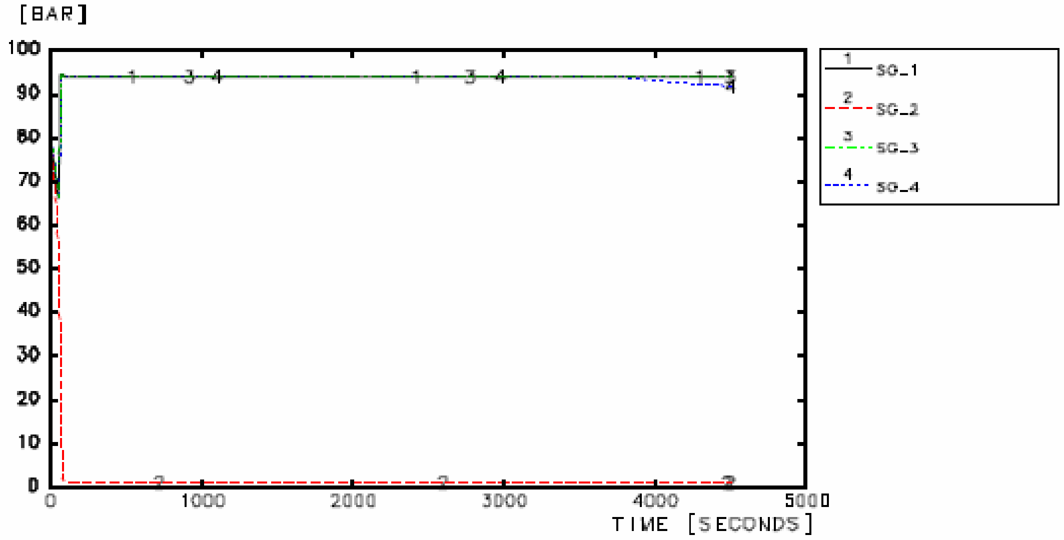
HOT LEG LIQUID TEMPERATURE

SECTION 14.5.3 - FIGURE 5 [Ref-1]

Case 1 - 1 EPR 4250MWth - from Initiating Event up to Controlled State - without LOOP SF on One ASG [EFWS]



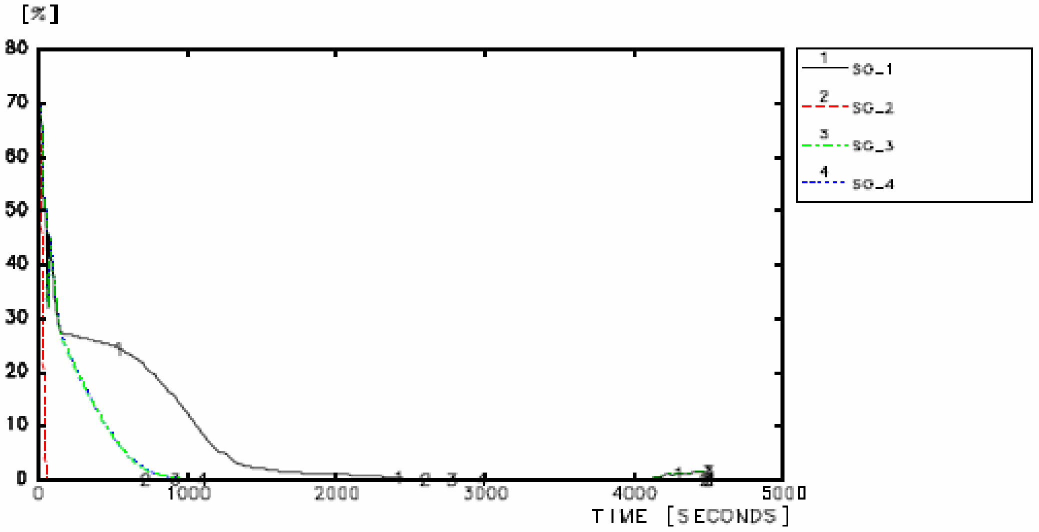
EFWS FLOW



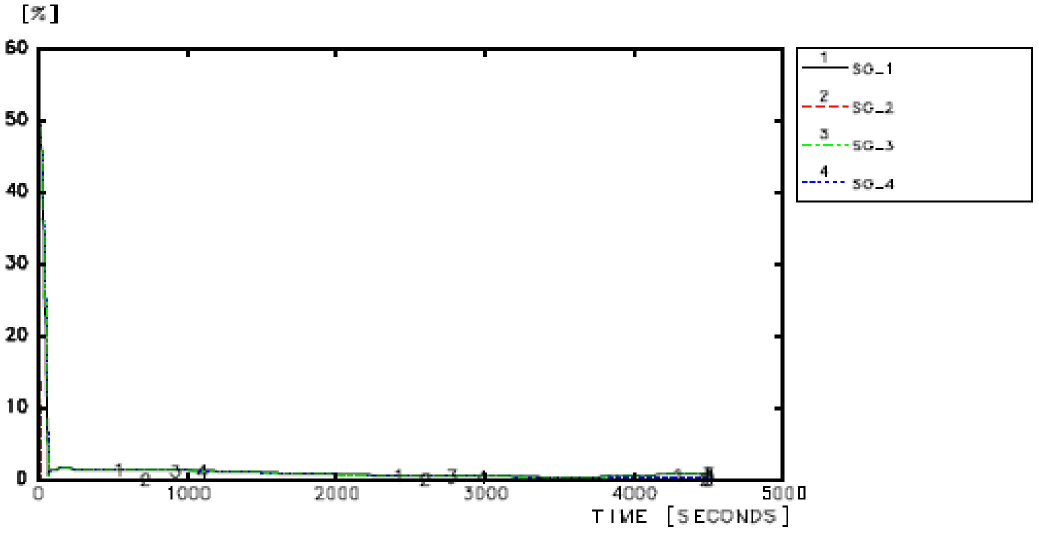
SG PRESSURE

SECTION 14.5.3 - FIGURE 6 [Ref-1]

Case 1 - 1 EPR 4250MWth - From Initiating Event up to Controlled State - without LOOP SF on One ASG [EFWS]



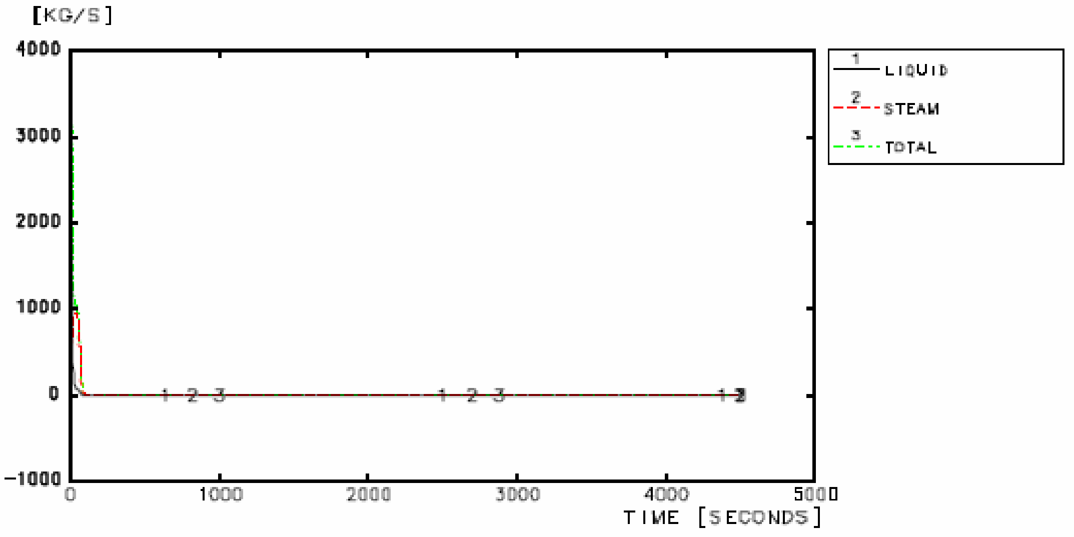
WIDE RANGE SG MEASURED LEVEL



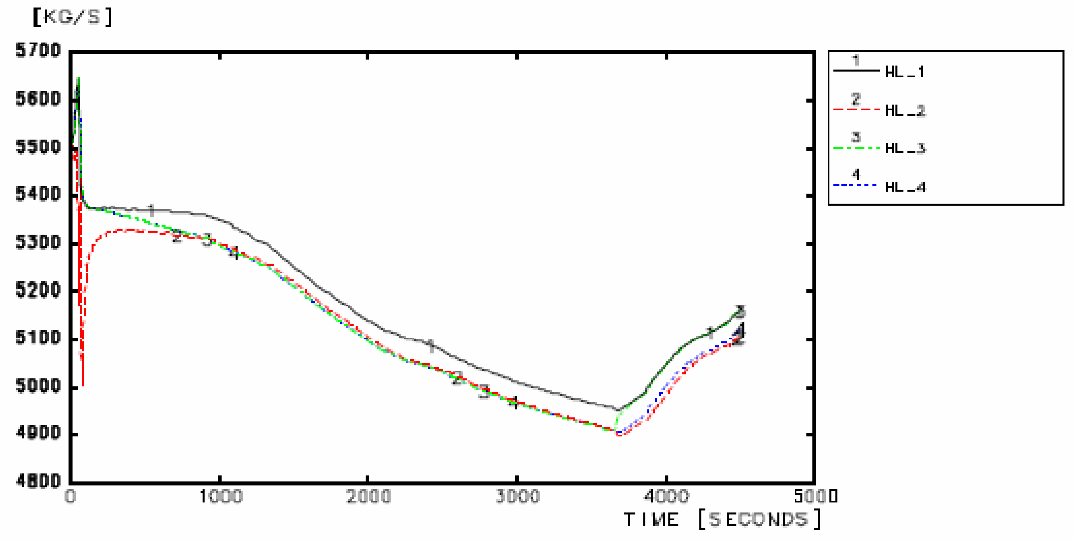
NARROW RANGE SG MEASURED LEVEL

SECTION 14.5.3 - FIGURE 7 [Ref-1]

Case 1 - 1 EPR 4250MWth - From Initiating Event up to Controlled State - without LOOP SF on One ASG [EFWS]



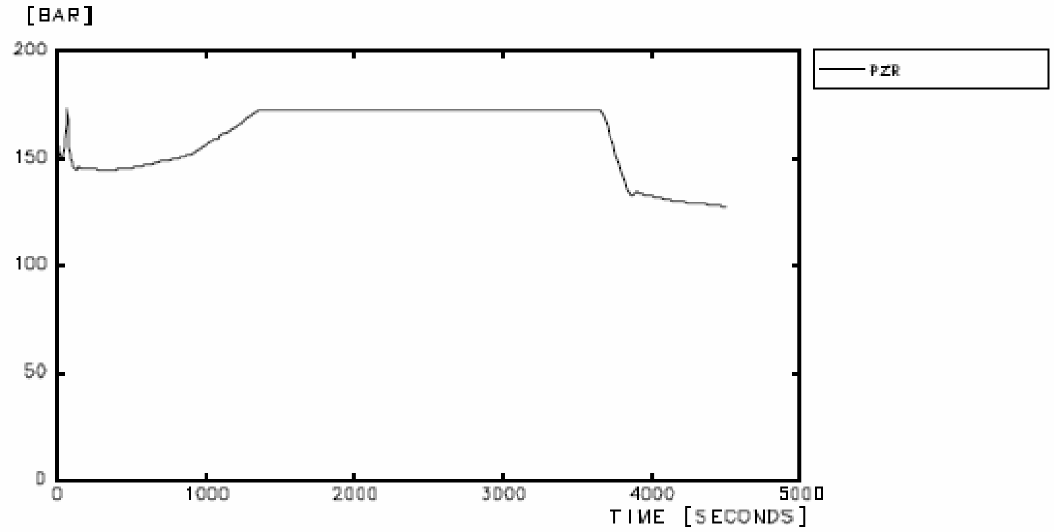
BREAK FLOW



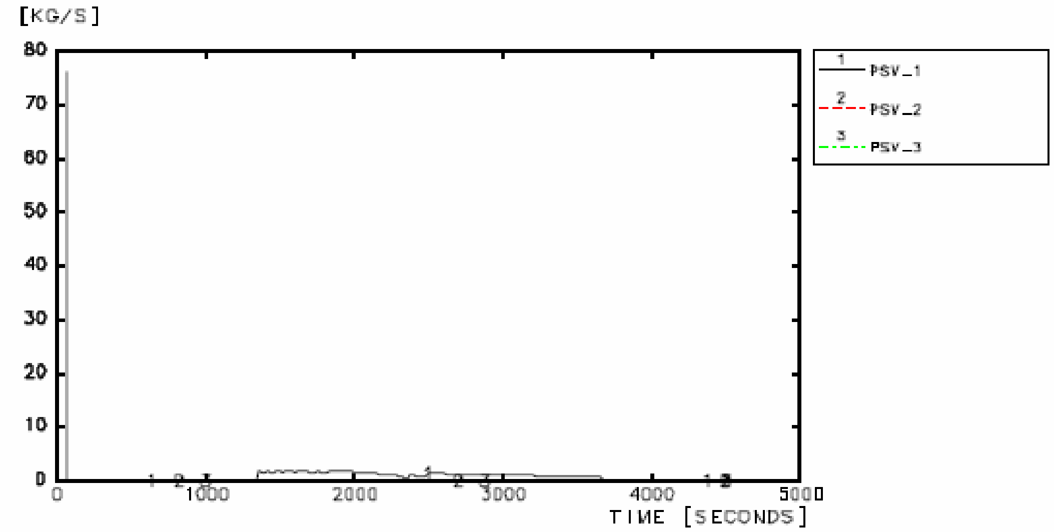
HOT LEG LIQUID FLOW

SECTION 14.5.3 - FIGURE 8 [Ref-1]

Case 1 - 1 EPR 4250MWth - From Initiating Event up to Controlled State - without LOOP SF on One ASG [EFWS]



PRESSURIZER PRESSURE



PSV FLOW

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

4.1. INTRODUCTION

The inadvertent opening of a steam generator relief or safety valve involves the spurious opening of either a VDA [MSRT] or main steam safety valve. These spurious openings will result in the depressurisation of the secondary system and cooldown and depressurisation of the RCP [RCS]. The cooldown resulting from these events is less severe than that occurring in the case of main steam line break (2A SLB), as the equivalent break area is smaller.

Therefore the consequences of such inadvertent core cooling in state B are bounded by those occurring following a 2A SLB in state B discussed in section 2 of this sub-chapter.

The analysis of the corresponding uncontrolled core cooling transient following a 2A SLB shows that the fuel damage is prevented as none of the core experiences departure from nucleate boiling (DNB).

The inadvertent or spurious opening of a VDA [MSRT] or a main steam safety valve in state B is classified as a PCC-4 event.

4.2. SYSTEM SIZING

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

5. SPECTRUM OF RCCA EJECTION ACCIDENTS

The rod ejection accident is assessed in reactor states A and B.

For the reactor operating from Hot Zero Power (HZP) with all rods in the core (state A) to state B, the required boron concentrations are defined to maintain core sub-criticality following a rod ejection. Consequently, a transient analysis calculation is performed only for initial power conditions at hot standby.

5.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

The Rod Ejection Accident (REA) is defined as the mechanical failure of a Rod Cluster Control Assembly (RCCA) drive mechanism casing (RGL [CRDM]), located on top of the reactor pressure vessel. The RCCA is ejected vertically from the reactor core due to the high coolant pressure.

This accident leads to a loss of reactor coolant through the ruptured RGL [CRDM] housing. This situation is addressed in the Loss Of Coolant Accident (LOCA) analysis in section 5 of Sub-chapter 14.4, and sections 6 and 7 of this sub-chapter.

The REA leads to a very fast transient with an ejection time of typically 0.1 second. It causes a fast positive reactivity insertion followed by a power excursion and a localised perturbation of the power distribution. In addition to a power overshoot, an increase in the hot channel factor due to the skewed power distribution after the rod ejection may occur. The magnitude of the reactivity insertion depends on the initial insertion of the bank to which this RCCA belongs as well as the number of RCCAs that make up the bank concerned.

The uncontrolled addition of reactivity to the reactor core by REA leads to a power excursion. The neutron flux response to the fast reactivity insertion is characterised by a very fast rise limited by the reactivity feedback effect of the negative Doppler coefficient. This self limitation of the power excursion is of importance because it limits the power during the delay time for any protection actions. The neutron flux is measured during the transient. If the detected flux exceeds a certain threshold value, reactor trip will be triggered causing the drop of all remaining RCCAs.

The transient is terminated by the following possible reactor trip signals:

- High positive neutron flux rate of change (Power Range Detectors, PRD).
- Intermediate range high neutron flux (Intermediate Range Detectors, IRD).

5.1.1. Controlled and Safe Shutdown State

It must be demonstrated that the controlled state can be reached with F1A functions and the safe shutdown state based on F1A or F1B functions by assessing the following safety and decoupling criteria:

Safety Criteria

Radiological limits for PCC-3 and PCC-4.

Decoupling Criteria (see section 2.1 of Sub-chapter 14.0):

- The number of fuel rods over 25 GWd/te average burnup experiencing Departure from Nucleate Boiling (DNB) must remain below 10%.
- The cladding failure limit is expressed in terms of average fuel enthalpy rise (cal/g) and depends on the initial linear power density. For every average rod burnup between 0 and 69 GWd/te, the maximum average fuel enthalpy rise (cal/g) at the peak power node is expressed with the following formulae:

$$\Delta H_{\max}(BU) = Min\left(162.26\,;141.4 - 29 \times \tanh\left(\frac{BU-49}{8.5}\right)\right) \text{ if } LPD_{init} = 0 \text{ W/cm}$$

$$\Delta H_{\max}(BU) = Min\left(151.56\,;151.56 - 48.5 \times \tanh\left(\frac{BU-45}{12}\right)\right) \text{ if } LPD_{init} < 100 \text{ W/cm}$$

$$\Delta H_{\max}(BU) = Min\left(134.49\,;149.19 - 60 \times \tanh\left(\frac{BU-44.7}{15.1}\right)\right) \text{ if } LPD_{init} < 200 \text{ W/cm}$$

$$\Delta H_{\max}(BU) = Min\left(123.68\,;108.66 - 35.6 \times \tanh\left(\frac{BU-54.6}{11.1}\right)\right) \text{ if } LPD_{init} < 300 \text{ W/cm}$$

$$\Delta H_{\max}(BU) = Min\left(108.43\,;110.75 - 51 \times \tanh\left(\frac{BU-49.9}{9.9}\right)\right) \text{ if } LPD_{init} < 400 \text{ W/cm}$$

$$\Delta H_{\max}(BU) = Min\left(102.12\,;99.15 - 49 \times \tanh\left(\frac{BU-49.9}{9.9}\right)\right) \text{ if } LPD_{init} < 450 \text{ W/cm}$$

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

Reactor trip

The core nuclear power excursion is initially limited by the Doppler feedback effect; the reactor trip occurs on high neutron flux rate of change (power range detectors) or on high neutron flux (intermediate range detectors) signals. As the REA is a PCC-4 transient no I&C diversity is required.

Systems

Because of the speed of the REA transient, no control system will have any effect within the timescale of the transient; consequently, control systems are not taken into account.

5.2. METHODS AND ASSUMPTIONS

5.2.1. Methods of Analysis

The REA analysis is based on a 3D simulation of the transient with the SCIENCE nuclear code package. It includes the following steps:

- 3D core model conservatism, to pessimise the initial conditions using bounding nuclear and thermal-hydraulic parameters;
- nuclear data calculations to identify bounding cases;

- transient calculations for thermal and thermal-hydraulic analysis of bounding cases.

5.2.1.1. First stage – 3D core model pessimism

The 3D core model is adjusted to reach bounding nuclear and thermal-hydraulic parameters. This pessimisation is performed at Beginning of Cycle with Xenon (BCX), Middle Of Cycle (MOC) and End Of Cycle (EOC) for 18 and 22 months equilibrium cycles to determine the bounding initial conditions for the REA analysis.

5.2.1.2. Second stage – Nuclear data calculations

The calculations aim at selecting bounding cases to be considered in the transient analysis.

For a given cycle, at a given burnup and for a given initial state, the following nuclear data are determined:

- the ejected rod worth;
- the initial hot spot factor, FQ;
- the hot spot factor, FQ, in post-ejection state;
- the hot channel rising factor, FΔH, in post-ejection state.

The static rod ejection calculation consists of the following two calculations:

- a static core calculation performed with equilibrium feedback for a given core configuration;
- a static core calculation performed with the same Doppler and moderator feedback (frozen feedback) but with the ejected rod out of the core.

3D static calculations with the SCIENCE nuclear code package are performed during this stage. The bounding cases correspond to those with significant values of FQ, FΔH or $\Delta\rho$.

5.2.1.3. Third stage – Transient simulations for thermal and thermal-hydraulic analysis

These calculations verify that the thermal and thermal-hydraulic criteria are met, by comparison against the safety decoupling criteria.

3D kinetic calculations are performed using the SCIENCE nuclear code package (including thermal and thermal-hydraulic modules) during this stage.

5.2.2. Boundary conditions and assumptions

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

The REA is investigated for both 18 and 22 month fuel cycles at EOC with each of the four RCCA sequences shown in Section 14.5.5 – Figure 1. Sensitivity studies are performed at BCX for the limiting cases. Power levels of 0% NP and 100% NP are considered, as well as intermediate power levels for each 20% level step.

The initial conditions and other fault-specific input data are listed in Section 14.5.5 – Table 1 to Table 5.

5.2.2.1. 3D core model pessimism

The pessimisation of the 3D core model, for the initial nuclear and thermal parameters, is:

- Core power: The initial power is increased by its uncertainty, except for HZP where the power is minimised to increase the prompt critical power excursion.
- Moderator density: The minimum moderator density is considered. This means that the maximum moderator inlet temperature on the part load diagram, corresponding to the studied initial conditions including uncertainties, associated with a thermal-hydraulic flow rate (minimum flow rate), is considered.
- Primary flow rate: A Loss Of Offsite Power (LOOP) is conservatively considered to occur at the start of the transient.
- Pressuriser pressure: The pressuriser pressure is decreased by its uncertainty.
- Xenon distribution: The Xenon distribution is chosen to maximise the worth of the ejected RCCA. This can be an equilibrium distribution or the distribution to obtain the axial power shapes skewed at the top of the core without exceeding the limiting value of Axial Offset.
- Doppler Effect: The Doppler temperature coefficient is adjusted to cover all burnup (BU) and all cycles (minimum absolute value), including uncertainties, penalties and allowances.
- Moderator effect: the moderator temperature coefficient is adjusted to the limiting value of the studied BU and covers all cycles (minimum absolute value), including uncertainties, penalties and allowances.
- Effective delayed neutrons fraction (β_{eff}): The effective delayed neutron fraction is adjusted to the limiting value to cover all BU and all cycles (minimum absolute value), including uncertainties, penalties and allowances.
- Prompt neutrons generation time (l*): The prompt neutron generation time is reduced by its uncertainty.
- Thermal parameters: The clad-to-pellet gap heat conductance and the pellet thermal conductivity are taken as best estimate values. The heat capacity is reduced by its uncertainty.
- Ejected rod worth: The ejected rod worth is increased by its uncertainty.

5.2.2.2. Nuclear data calculation

Nuclear data are calculated at EOC for conditions before and after the RCCA ejection without reactivity feedback effects. The specific nuclear data evaluated are the following:

- the ejected rod worth($\Delta\rho$);
- the initial hot spot factor (FQ_i);
- the hot spot factor in post-ejection state (FQ_{pe});
- the hot channel enthalpy rising factor in post-ejection state ($F\Delta H_{pe}$).

Among the approximately 11,000 rod ejection cases studied, at several power levels and several RCCA configurations for the different fuel management scenarios considered, approximately 60 cases are potentially bounding and are retained for transient calculation.

5.2.2.3. Transient simulations for thermal and thermal-hydraulic analysis

5.2.2.3.1. Main parameters and assumptions for the nuclear transient

The main parameters considered for the nuclear transient are:

- Transient duration: The transient duration is set to 1 second for the thermal-hydraulic analysis and set to 3 seconds for the thermal analysis.
- Ejection time: The rod is typically ejected in 0.1 seconds.
- External reactivity insertion: An external reactivity is inserted during the transient accounting for the ejected rod worth uncertainty. The external reactivity is inserted linearly from 0 to 0.1 seconds.
- Reactor trip (RT): The reactor trip is triggered on high neutron flux rate of change. A 3 out of 4 actuation logic is considered accounting for the loss of one Power Range Detector (PRD). A delay time between RT actuation and the beginning of RCCA drop is considered. Earthquake is accounted for in the time assumed for the RCCA to fall from top to bottom. No stuck rod is considered.
- Nuclear power: The nuclear power used for the thermal-hydraulic calculations is the maximum value of the nuclear power including uncertainties and allowances, at each time step of the transient.
- Linear Power Density (LPD) operating range limit: If the local linear power density at the initial time (including uncertainties and allowances) is lower than the Limiting Condition of Operation (LCO) linear power density, then no adjustment is made to the LCO linear power,

  If the local linear power density at the initial time (including uncertainties and allowances) is higher than the LCO linear power density, then an adjustment is made to the LCO linear power.
- Thermal-hydraulic correlation: The FC correlation is used.

5.2.2.3.2. Main parameters and assumptions for the thermal transient

The main parameters considered for the thermal transient are:

- Thermal parameters:

  The uranium dioxide heat capacity is minimised.

  The sensitivity to thermal transfer heat flux is studied for the maximum clad temperature and the fuel enthalpy rise calculations. Thus, both the maximum and minimum uranium dioxide conductivity and the clad-to-pellet gap heat conductance are assessed. The radial power distribution is calculated and pessimised by coefficients in order to maximise or minimise the depression in the pellet.

- Clad-to-pellet gap heat conductance: The value corresponds to a closed clad-to-pellet gap heat conductance.

- Fuel enthalpy: The fuel enthalpy is maximised.

5.2.2.3.3. Main parameters and assumptions for the thermal-hydraulic analysis

The main parameters to be considered for the thermal-hydraulic analysis are the following:

- Thermal parameters:

  The uranium dioxide heat capacity is minimised.

  The uranium dioxide conductivity and the clad-to-pellet gap heat conductance are maximised. The radial power distribution is calculated and pessimised by coefficients in order to maximise the depression in the pellet.

- Clad-to-pellet gap heat conductance: The value corresponds to a closed clad-to-pellet gap heat conductance.

- BU limit: a BU limit is considered for the calculations of the number of rods experiencing DNB. The calculation only considers the clad failure for rods with an average BU higher than the BU limit.

- Fuel enthalpy: The nominal fuel enthalpy is assumed.

5.2.3. CHOICE OF SINGLE FAILURE AND PREVENTIVE MAINTENANCE

- Single failure: A failure of an ex-core Power Range Detector results in a 3 out of 4 actuation logic for reactor trip.

- Preventive maintenance: preventive maintenance is considered for the analysis from the initiating event to the controlled state, but as it is already taken into account in the reactor protection system logic design (as degradation of the voting logic is assumed in the case of preventive maintenance), reactor trip is completed without modifying the event time sequence for the present analysis despite unavailability resulting from preventive maintenance. For the analysis of the transition from the controlled to safe shutdown state, the assumptions for the reference cases discussed in section 5.3.2 of this sub-chapter apply.

5.3. RESULTS AND CONCLUSIONS

5.3.1. From Initiation to Controlled State

The rod ejection transient is modelled by withdrawing the most reactive rod in 0.1 second. This results in a rapid reactivity insertion with large local power peaks. The associated power excursion is limited by Doppler reactivity feedback caused by the increased fuel temperatures. The transient is ultimately terminated by the reactor protection system.

The reactor trip is triggered on the high neutron flux rate of change signal (PRD). After a conservative time delay of 0.6 seconds all control rods (except the ejected one) are dropped. They start dropping into the core according to their time dependent rod insertion characteristic. After 5 seconds control rod insertion is complete.

The maximum (or minimum) values for the safety relevant parameters obtained during the transient calculations are given in Section 14.5.5 - Table 6. The transient behaviour of some key parameters is shown for the most onerous transients to demonstrate that a controlled state is achieved; Section 14.5.5 - Figure 2 for 82% NP at EOC and Section 14.5.5 - Figure 3 for 42% NP at EOC.

The maximum enthalpy rise for each case with regard to the fuel enthalpy rise criterion is shown in Section 14.5.5 – Figure 4. The decoupling safety criterion on enthalpy rise is met with large margins.

The safety criterion concerning clad failure due to DNB is met for all power levels with large margins.

5.3.2. From the Controlled State to the Safe Shutdown State

This transition is not analysed explicitly as it is covered by equivalent analyses of other events (reference cases). In the table below these reference cases for the demonstration of safe shutdown and compliance with the three criteria "sub-criticality", "decay heat removal provided by RIS/RRA [SIS/RHRS]" and "activity release/barrier integrity" within the PCC limits are given:

Criteria	Reference case	Remark/Reason
Sub-criticality	Section 5 of Sub-chapter 14.3 Loss of condenser vacuum	Due to the RBS [EBS], the necessary boration is not impacted by the additional RCCA not inserted
Max. Activity release	Section 5 of Sub-chapter 14.3 Loss of condenser vacuum	In the reference case one SG is completely emptied in addition
Heat removal	Section 3 of this sub-chapter Feedwater system piping break	In the reference case only one train is available for cooldown

5.4. SYSTEM SIZING

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

SECTION 14.5.5 - TABLE 1

Rod Ejection Accident
RCCA Limits of Insertion of BCX and EOC

RCCA limits of insertion at BCX (inserted steps)					
Power level (%NP)	**P1**	**P2**	**P3**	**P4**	**P5**
100	200	100	100	100	100
87	411	204	100	100	100
80	411	244	128	100	100
61	411	411	242	100	100
40	411	411	346	200	100
30	411	411	411	263	100
20	411	411	411	306	100
2	411	411	411	411	100
0	411	411	411	411	120

RCCA limits of insertion at EOC (inserted steps)					
Power level (%NP)	**P1**	**P2**	**P3**	**P4**	**P5**
100	100	50	50	50	50
87	411	206	50	50	50
80	411	267	93	50	50
70	411	411	196	50	50
60	411	411	265	114	50
45	411	411	411	251	50
40	411	411	411	275	50
25	411	411	411	411	50
20	411	411	411	411	107
6	411	411	411	411	411
0	411	411	411	411	411

SECTION 14.5.5 - TABLE 2

Rod Ejection Accident
Initial Conditions

Parameter	Nominal Value	Uncertainty	Retained Value
Primary side			
Core power (%NP)	0	-	0
	from 0 to 100 (20% NP step)	2	from 2 to 102% (20% NP step)
Boron concentration	-	-	BCX : Critical EOC : Critical or null
Core inlet temperature (°C)	0%PN 303.3 20%PN : 304.4	4.0	0%PN : 307.3 22%PN : 308.4
	40%PN : 302.8 60%PN : 302.7 80%PN : 299.1 100%PN : 295.6	2.5	42%PN : 305.3 62%PN : 305.2 82%PN : 301.6 102%PN : 298.1
Pressuriser pressure (bar abs.)	155	2.5	152.5
Core by-pass (%)	-	-	5.5
Core flow rate (m^3/h)	-	-	102,740
Axial Power Distribution			
Axial Power Shape ΔI	Rightmost (12%) or ΔI corresponding to the equilibrium xenon state		

SECTION 14.5.5 - TABLE 3

Rod Ejection Accident
Neutronic Data Retained for Calculations

<table>
<tr><th>Parameter</th><th colspan="3">Calculated value
w/o uncertainties</th><th>Uncertainties</th><th colspan="3">Retained value</th></tr>
<tr><td rowspan="2">Doppler effect</td><td>BCX</td><td>MOC</td><td>EOC</td><td rowspan="2">20%</td><td colspan="3" rowspan="2">-1.9</td></tr>
<tr><td>-2.13</td><td>-2.32</td><td>-2.44</td></tr>
<tr><td rowspan="2">Moderator effect</td><td>BCX</td><td>MOC</td><td>EOC</td><td rowspan="2">3.6 pcm/°C</td><td>BCX</td><td>MOC</td><td>EOC</td></tr>
<tr><td>-1.21</td><td>-17.89</td><td>-41.70</td><td>0</td><td>-17.89</td><td>-41.7</td></tr>
<tr><td rowspan="2">Effective delayed neutron fraction (β_{eff})</td><td>BCX</td><td>MOC</td><td>EOC</td><td rowspan="2"></td><td colspan="3" rowspan="2">440</td></tr>
<tr><td>580.7</td><td>519.3</td><td>500.2</td></tr>
<tr><td>Prompt neutrons generation time (l*)</td><td colspan="3">BE value</td><td>-10%</td><td colspan="3">BE value -10%</td></tr>
<tr><td>Clad-to-pellet gap heat conductance</td><td colspan="3">BE value</td><td>-</td><td colspan="3">BE value</td></tr>
<tr><td>Clad-to-pellet gap conductivity</td><td colspan="3">BE value</td><td>-</td><td colspan="3">BE value</td></tr>
<tr><td>Clad-to-pellet gap heat capacity</td><td colspan="3">BE value</td><td>- 2%</td><td colspan="3">BE value -2%</td></tr>
<tr><td>Rod worth</td><td colspan="3">BE Value</td><td>10%</td><td colspan="3">BE value +10%</td></tr>
</table>

These values are calculated for the 18 month and the 22 month equilibrium fuel cycles presented in the PCSR.

SECTION 14.5.5 - TABLE 4

Rod Ejection Accident
Main Parameters and Assumptions

Parameter	Retained assumption or value
Nuclear Transient	
Transient duration	1.0s for thermal-hydraulic analysis 3.0 s for thermal analysis
Ejection time	0.1 s
Time delay between RT actuation and beginning of RCCA drop	0.6 s
Time for RCCA to fall (from top to bottom)	5 s
Ejected rod worth uncertainty	10%
Uncertainty applied on FQ	$(1+F_{Xe})\times(1+\sqrt{(F_Q^E-1)^2+(F_Q^B-1)^2+F_U^{N\,2}})$ $F_{Xe} = 4\%$ $F_Q^E = 3\%$ $F_Q^B = 5.6\%$ $F_U^N \geq 5\%$
Uncertainty applied on FΔH	$(1+F_{Xe})\times(1+\sqrt{(F_{DH}^E-1)^2+F_U^{N\,2}})$ $F_{Xe} = 4\%$ $F_{DH}^E = 2.1\%$ $F_U^N \geq 4\%$
BU limit for clad failure with DNB	25 GWD/te
Linear Power Density LCO	470 W/cm
Thermal-hydraulic correlation	FC correlation
Thermal analysis	
Closed clad-to-pellet gap heat conductance	150,000 W/(m².K)
Fuel enthalpy	Maximum
Thermal-hydraulic analysis	
Closed clad-to-pellet gap heat conductance	150,000 W/(m².K)
Fuel enthalpy	Best-Estimate

SECTION 14.5.5 - TABLE 5

Rod Ejection Accident
Initial Reactor State before Rod Ejection at EOC
(Limiting case at each power level)

Fuel Management	UO2 in/out 22mths	UO2 in/out 18mths	UO2 in/out 18mths	UO2 in/out 18mths	UO2 in/out 18mths	UO2 in/out 18mths	UO2 in/out 18mths
Initial reactor power (%NP)	102	82	62	42	22	2	0
Ejected rod	J03	N03	N03	N03	N03	N03	N03
Ejected rod worth pcm (incl. uncertainty)	42	323	400	703	656	619	623
Initial FQ (incl. uncertainty)	1.93	3.25	3.58	2.92	5.16	8.17	8.32
RCCA configuration	P1P5	P1P5	P1P5	P1P5	P1P5	P1P4	P1P4
RCCA sequence	# 3	# 1	# 1	# 1	# 1	# 1	# 1
Initial Bank insertion (inserted steps)	P1 : 100	P1 : 411	P1 : 411	P1 : 411	P1 : 411	P1 : 411	P1 : 411
	P2 : 50	P2 : 267	P2 : 411	P2 : 411	P2 : 411	P2 : 411	P2 : 411
	P3 : 50	P3 : 93	P3 : 265	P3 : 411	P3 : 411	P3 : 411	P3 : 411
	P4 : 50	P4 : 50	P4 : 114	P4 : 275	P4 : 411	P4 : 206	P4 : 206
	P5 : 50	P5 : 50	P5 : 50	P5 : 50	P5 : 107	P5 : 0	P5 : 0
Delayed neutron fraction (pcm, incl. uncertainty)	440	440	440	440	440	440	440
Doppler coefficient (pcm/°C, incl. uncertainty)	-1.9	-1.9	-1.9	-1.9	-1.9	-1.9	-1.9
Moderator coefficient (pcm/°C, incl. uncertainty)	-41.7	-41.7	-41.7	-41.7	-41.7	-41.7	-41.7
Primary pressure (bar)	152.5	152.5	152.5	152.5	152.5	152.5	152.5
Core inlet temperature (°C)	298.1	301.7	305.2	305.3	308.4	307.3	307.3
Core flow rate (m^3/h)	102,740	102,740	102,740	102,740	102,740	102,740	102,740

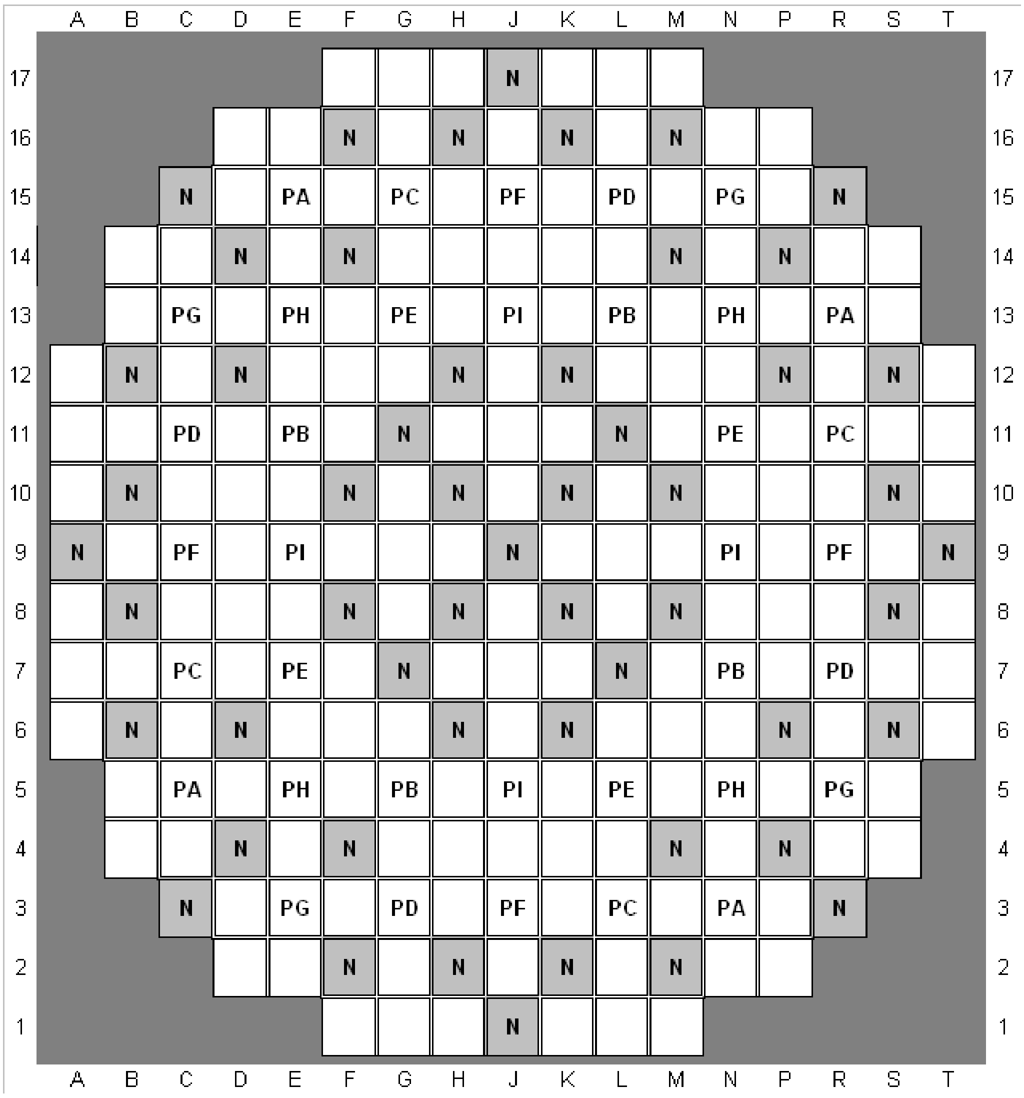
SECTION 14.5.5 - TABLE 6

Rod Ejection Accident
Results of Transient Calculations
REA Decoupled Acceptance Criterion

Fuel management	UO2 in/out 22mths	UO2 in/out 18mths	UO2 in/out 18mths	UO2 in/out 18mths	UO2 in/out 18mths	UO2 in/out 18mths	UO2 in/out 18mths
Initial reactor power (%NP)	102	**82**	62	**42**	22	2	0
Burn up	EOC	EOC	EOC	EOC	EOC	EOC	EOC
Maximum enthalpy rise (J/g)	28	90	82	129	139	104	98
Minimum margin (J/g) (ΔH criterion)	311	**258**	303	335	364	393	430
Percentage of rods in DNB - with an average BU higher than 25 GWd/te	0	0	0	**2.43**	1.61	0	0

SECTION 14.5.5 - FIGURE 1

Rod Ejection Accident
RCCA Pattern

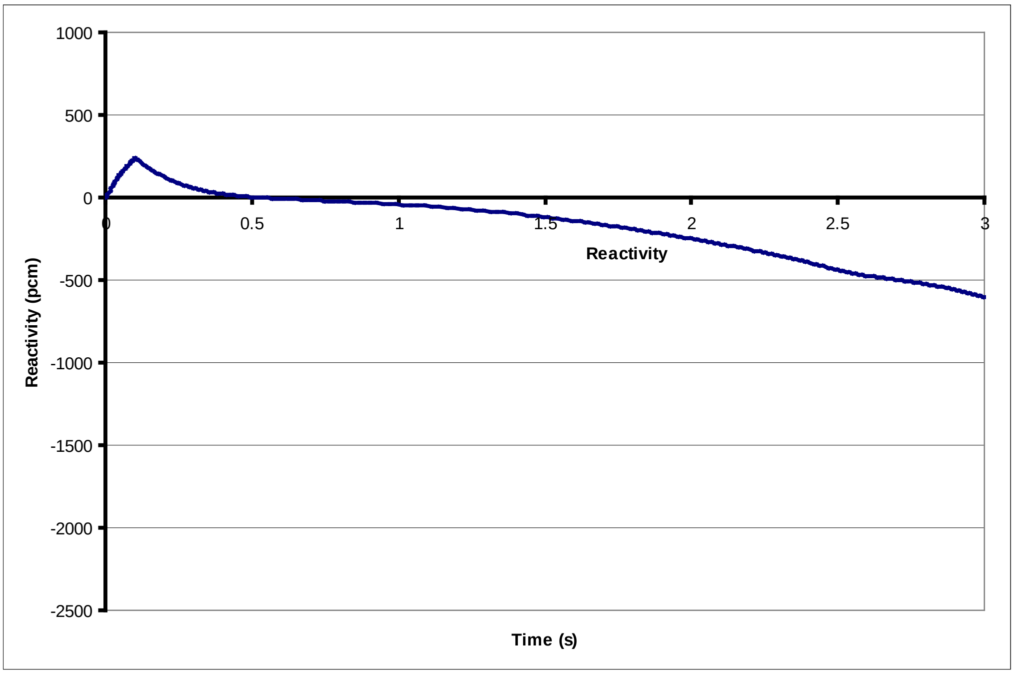


	P1	P2	P3	P4	P5
S1	PA	PB	PC PD	PF PE	PG PH PI
S2	PD	PH	PC PB	PA PF	PE PG PI
S3	PF	PB	PA PE	PC PD	PG PH PI
S4	PC	PH	PD PB	PA PE	PF PG PI

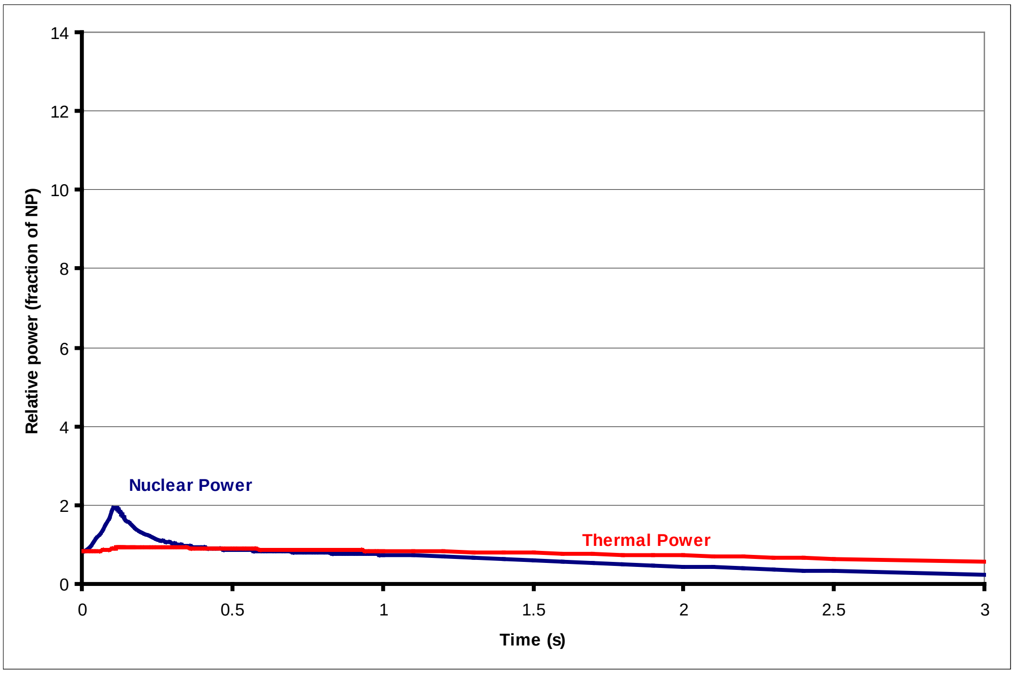
SECTION 14.5.5 - FIGURE 2 (1/3)

Rod Ejection at 82%NP

Reactivity

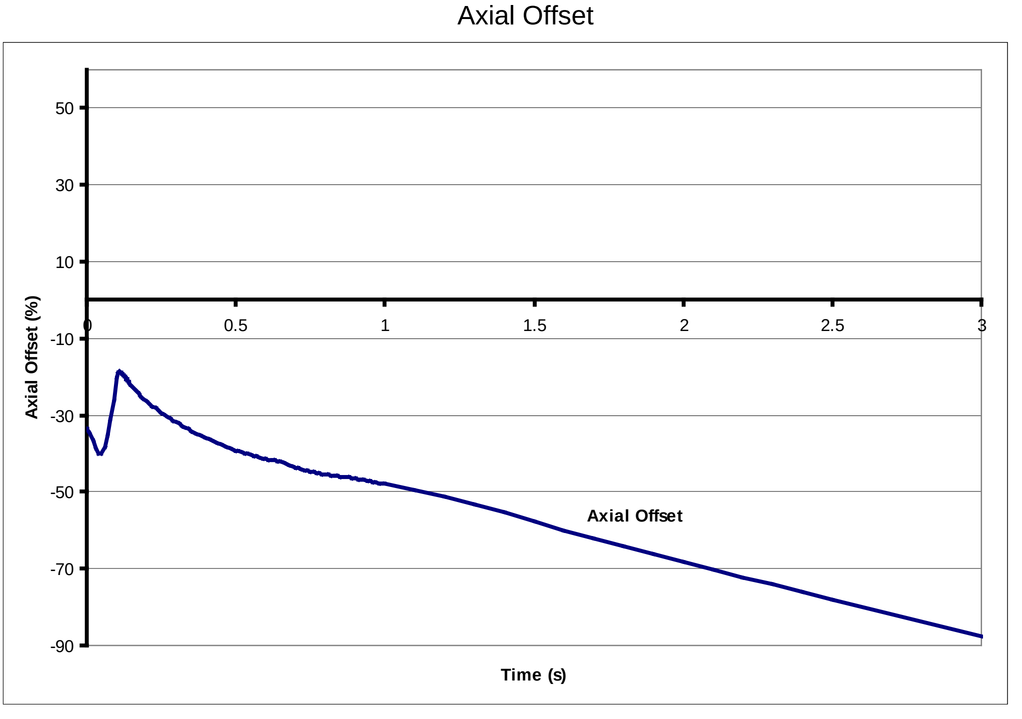


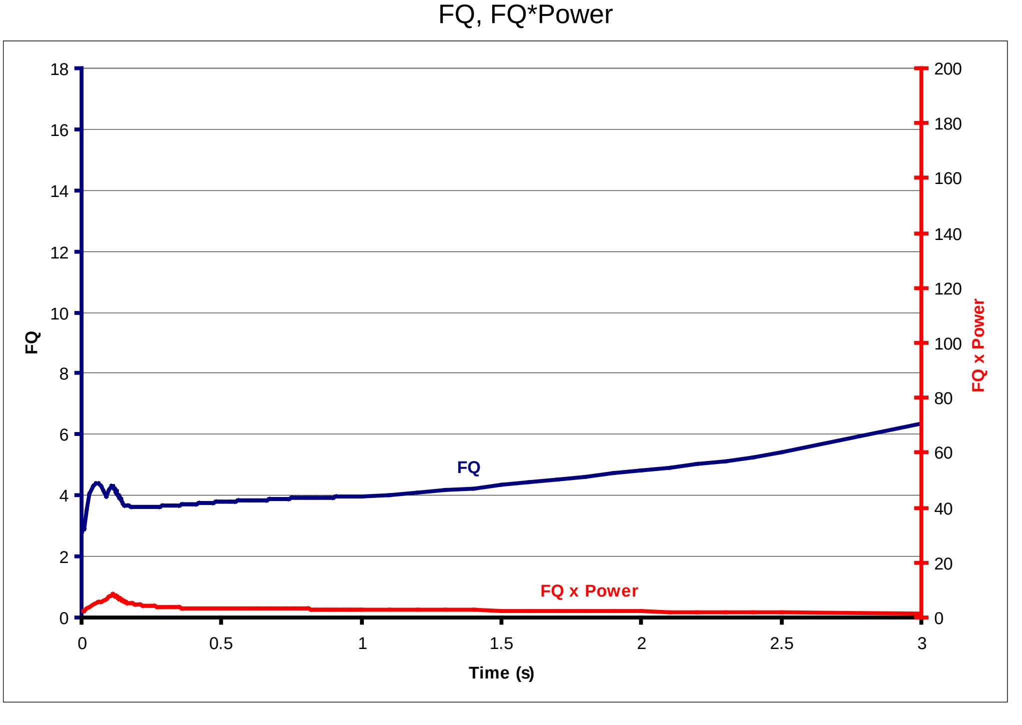
Relative Power



SECTION 14.5.5 - FIGURE 2 (2/3)

Rod Ejection at 82%NP

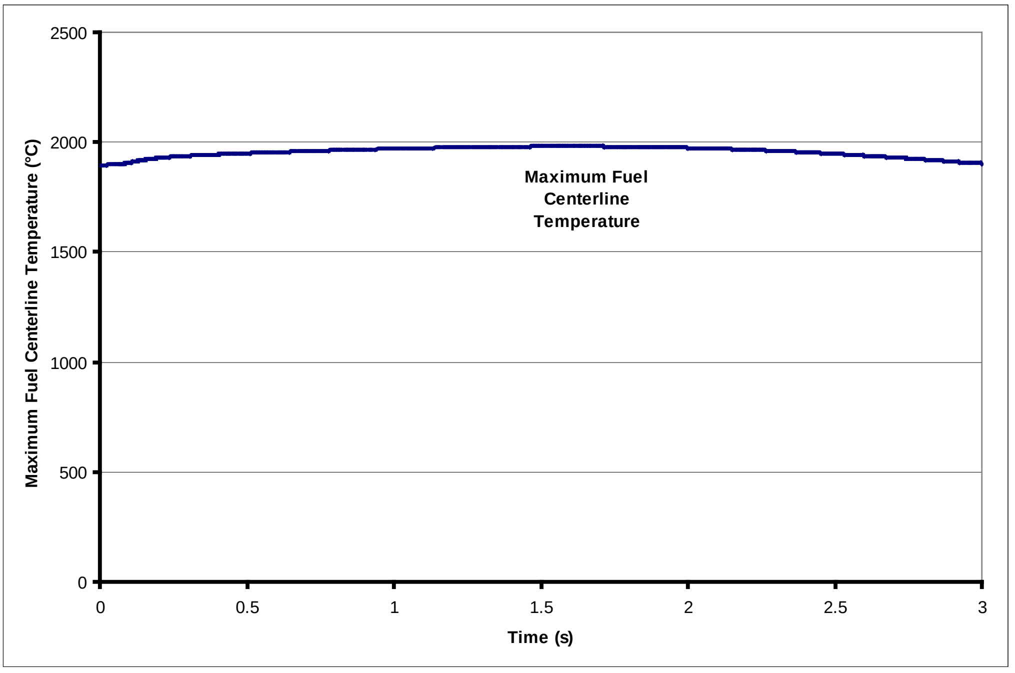




SECTION 14.5.5 - FIGURE 2 (3/3)

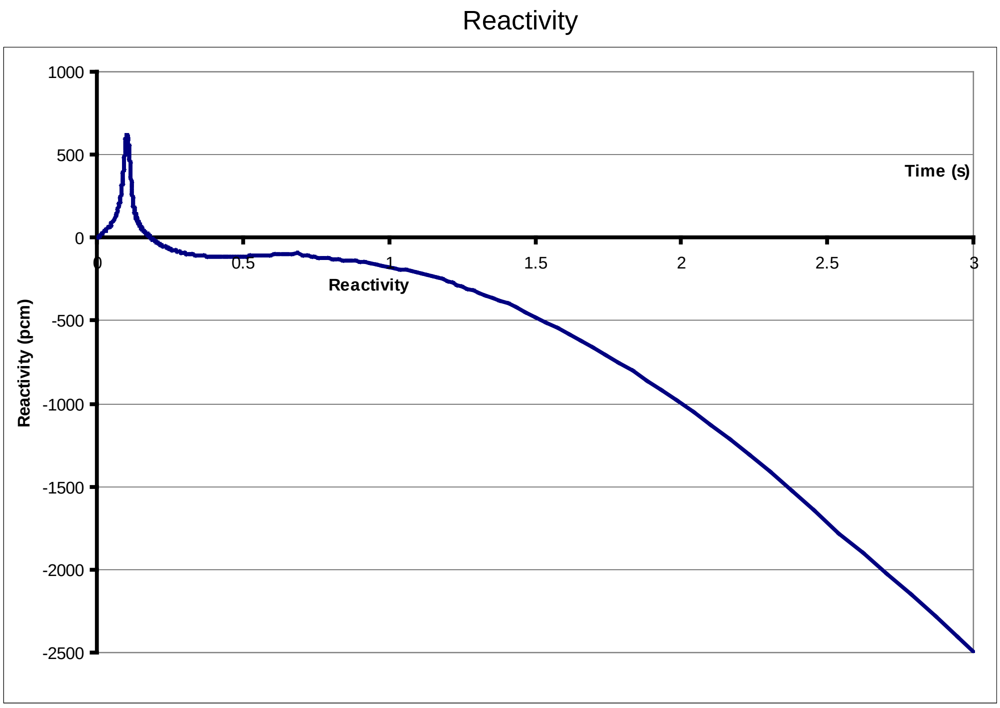
Rod Ejection at 82%NP

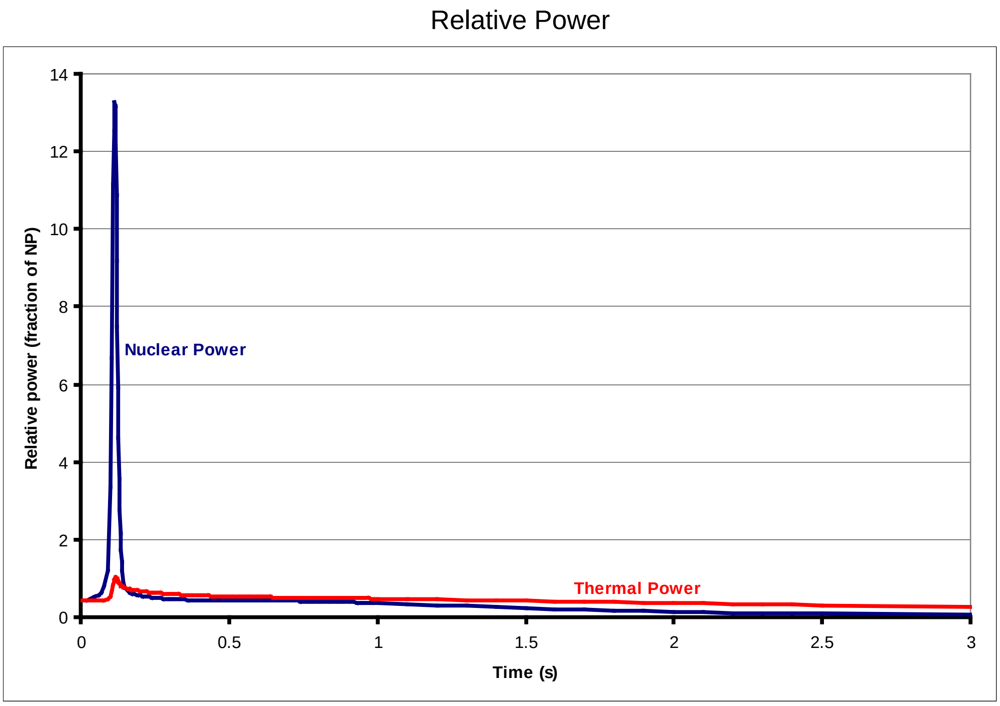
Maximum Fuel Centreline Temperature



SECTION 14.5.5 - FIGURE 3 (1/3)

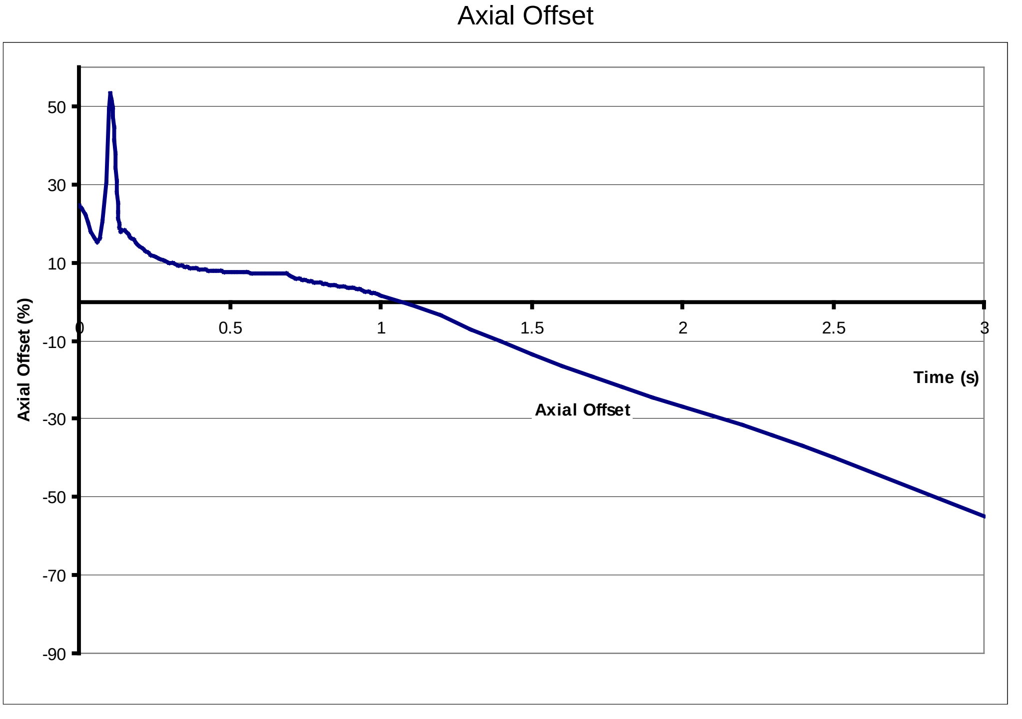
Rod Ejection at 42%NP

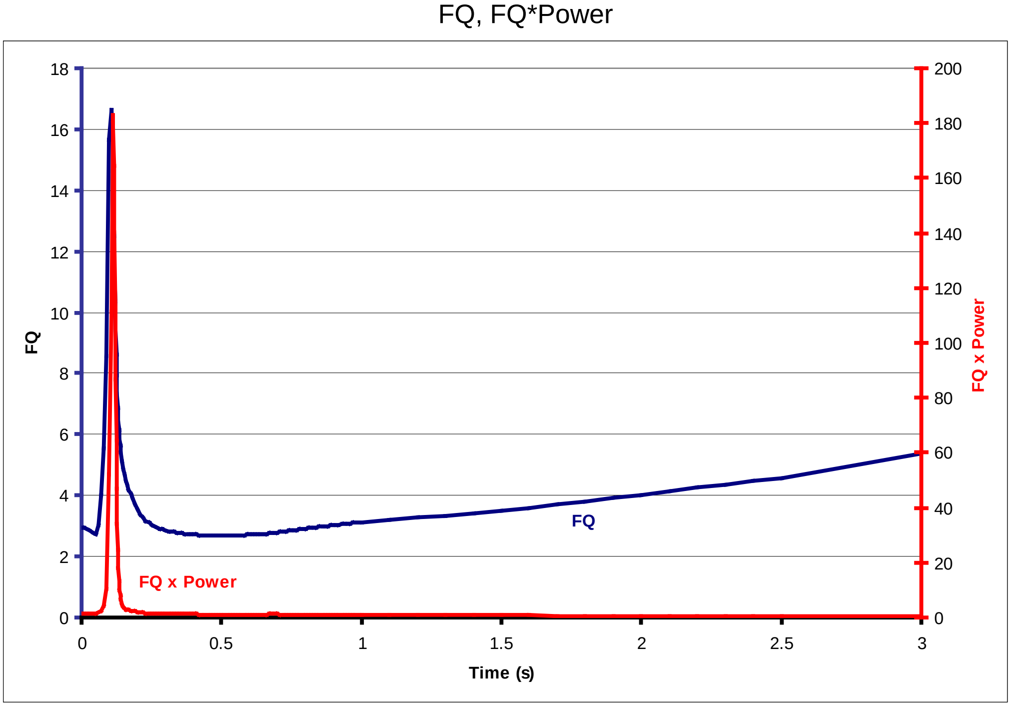




SECTION 14.5.5 - FIGURE 3 (2/3)

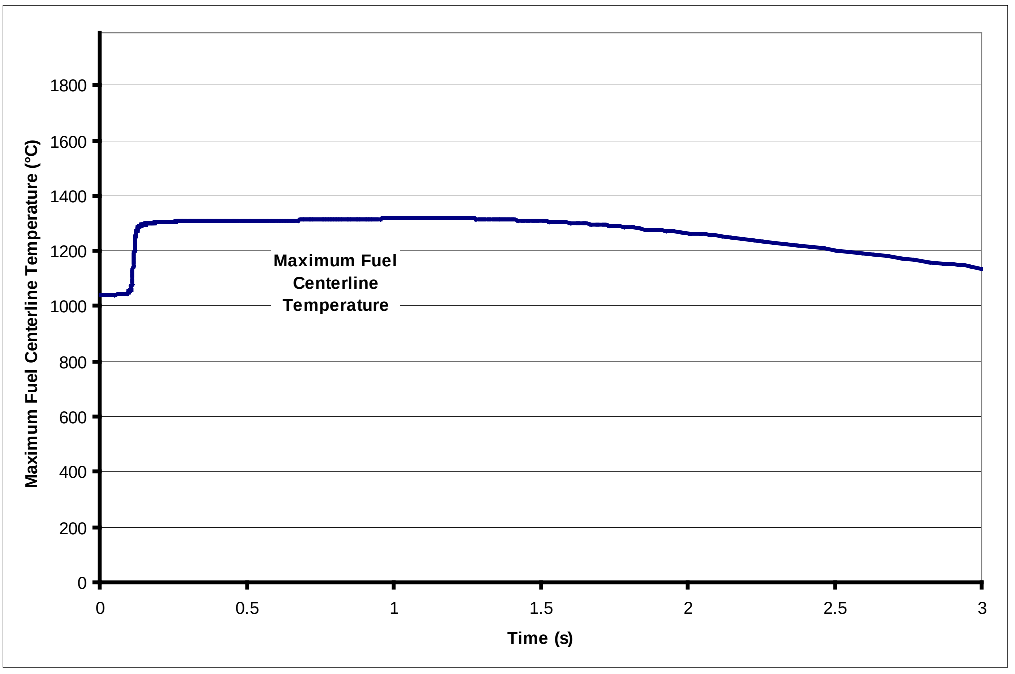
Rod Ejection at 42%NP





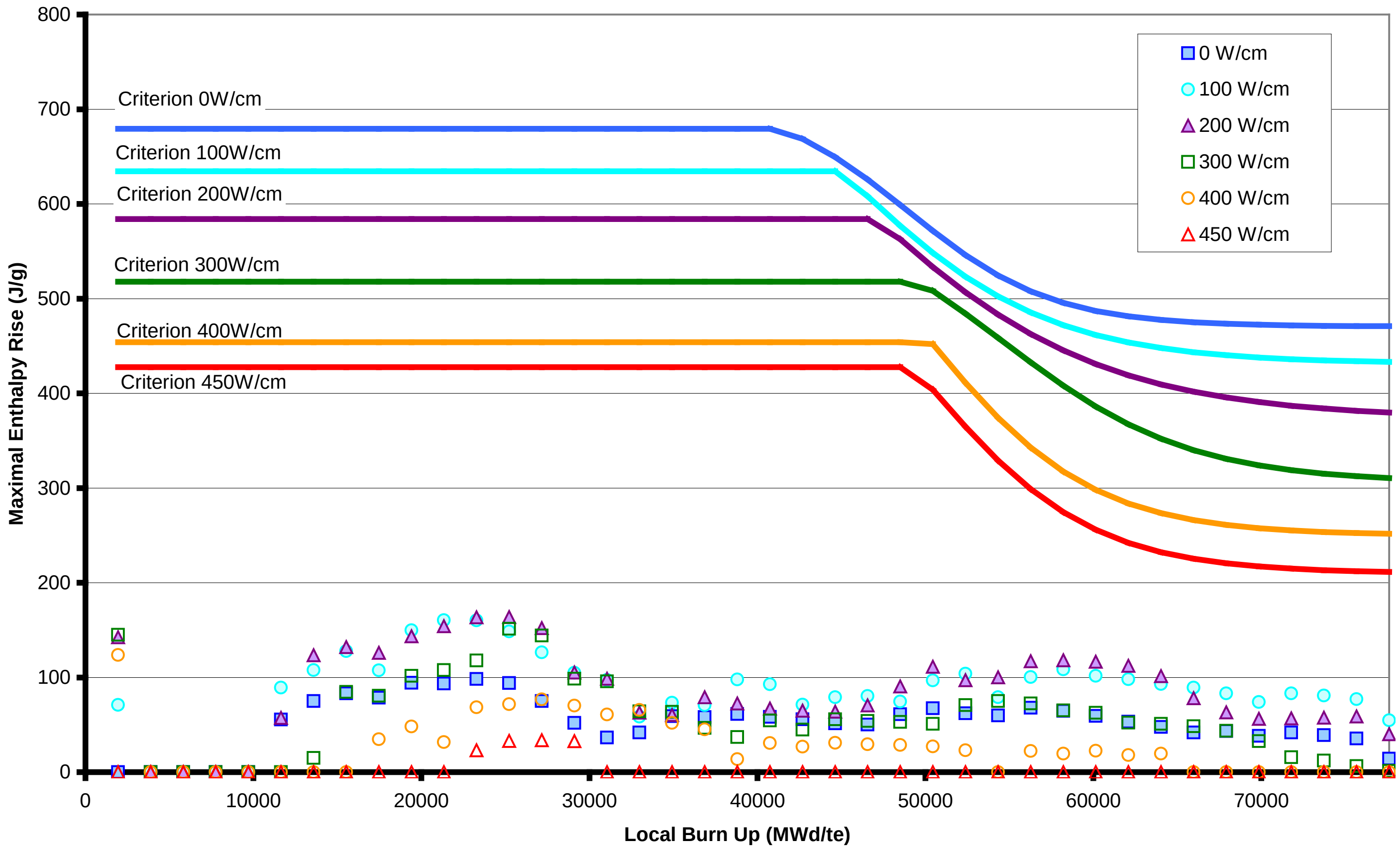
SECTION 14.5.5 - FIGURE 3 (3/3)

Rod Ejection at 42%NP
Maximum Fuel Centreline Temperature



SECTION 14.5.5 - FIGURE 4

Rod Ejection
Maximum Enthalpy Rise versus Local Burn up



6. INTERMEDIATE AND LARGE BREAK LOCA (UP TO SURGE LINE BREAKS – STATES A AND B)

6.1. INTERMEDIATE AND LARGE BREAK LOCA IN STATE A (PCC-4)

6.1.1. Accident Description

6.1.1.1. Definition and classification

The accident analysed in this section is a Loss Of Coolant Accident (LOCA) defined as a rupture of the Reactor Coolant System RCP [RCS] piping or of any line connected to the system before the first isolation valve. Since the main coolant lines are designed according to the break preclusion requirements, guillotine breaks in these pipes are precluded.

The intermediate / large break LOCA (IB/LB-LOCA) is defined as a break of an equivalent diameter:

- larger than 50 mm (equivalent area larger than 20 cm²),

- no larger than the guillotine break of the largest pipe connected to the RCP [RCS] loop. This corresponds to the surge line break on the hot leg side and the safety injection RIS [SIS] line break on the cold leg side. In the case of a rupture of an auxiliary line, the break is considered between the RCP [RCS] and the first isolation valve.

An IB/LB LOCA is classified as a PCC-4 event depending on the break size and the initial plant state. This section only deals with the IB/LB LOCA in state A classified as PCC-4 events, as defined in Sub-chapter 14.0.

6.1.1.2. Typical Sequence of Events

a) From the initiating event to the controlled state

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

A reactor trip (RT) is initiated on a low pressuriser pressure (< MIN2) signal. The RT signal automatically trips the turbine and closes the main feedwater high load lines.

As the secondary side pressure increases, the Main Steam Bypass (GCT [MSB]) valves open allowing steam dump to the condenser. In the case of unavailability of the steam dump to the condenser, typically following LOOP, the Main Steam Relief Trains (VDA [MSRT]) open following a high SG pressure (> MAX1) signal, allowing steam dump to atmosphere.

The steam generators (SGs) are fed by the main feedwater ARE [MFWS] through the low load feed lines. If the ARE [MFWS] is unavailable, the start-up and shutdown pump starts and feeds the SG through the low load feed lines. If the start-up and shutdown system AAD [SSS] is unavailable, the emergency feedwater system ASG [EFWS] is actuated following a low SG level (< MIN2) signal or following a very low pressuriser pressure (< MIN3) signal in the case of LOOP.

A Safety Injection (SI) signal occurs on very low pressuriser pressure (< MIN3). The SI 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. The partial cooldown cools the primary system and reduces the RCP [RCS] pressure.

During the partial cooldown, the RCP [RCS] pressure decreases sufficiently to allow MHSI to the cold legs. The partial cooldown is undertaken by all SGs via steam dump to the condenser or to the atmosphere. This is achieved by automatically decreasing the relevant relief valve setpoints at a rate sufficient to support the constant cooling rate of -250°C/h. This is continued down to a fixed pressure value (according to the availability of the condenser) which is low enough to allow the necessary MHSI and high enough to prevent core re-criticality.

Reactor coolant pump trip occurs following a “ΔP over Reactor Coolant Pumps < MIN1” and RIS [SIS] signal or when the LOOP occurs.

For the smallest intermediate breaks, RCP [RCS] discharge via the break, still in the form of liquid, does not remove sufficient volumetric flow to match the steam production in the core caused by the decay heat. Consequently, the RCP [RCS] depressurisation stops at the end of the partial cooldown.

While the MHSI flow is insufficient to compensate for the break flow, the RCP [RCS] inventory continues to decrease and the void fraction increases in the hot parts of the RCP [RCS]. Subsequently, the break flow rate decreases as the void fraction in the cold legs increases.

Once the break flow changes to single-phase steam, the volumetric RCP [RCS] balance between steam production due to core decay heat and break flow is completely changed and the break size is the dominant parameter in dictating the subsequent depressurisation:

- For the smallest intermediate breaks, some condensation in the SG tubes may occur in conjunction with the direct steam discharge via the break to remove the steam produced in the core. The RCP [RCS] pressure (saturation pressure) remains slightly above the SG pressure.

- Larger breaks discharge sufficient steam to allow further RCP [RCS] depressurisation without steam condensation in the SG tubes. In the longer term the heat transfer reverses between the primary and secondary sides. The RCP [RCS] pressure continues to fall independently of the SG temperature, down to the accumulator actuation pressure and possibly the LHSI pressure setpoint.

The subsequent behaviour of the RCP [RCS] water inventory depends on the balance between total RIS [SIS] flow, MHSI, accumulators and LHSI, and the break flow rate.

The core may uncover before the rate of RIS [SIS] water addition exceeds the discharge of RCP [RCS] coolant via the break. If core uncovery occurs, the fuel cladding temperature will rise above the saturation temperature in the uncovered part of the core. The cladding temperature increase depends principally on the depth and duration of core uncovery. Therefore, dedicated criteria as defined in sub-section 6.1.2 of this sub-chapter must be met in order to prevent any unacceptable core damage and to limit the radiological consequences to the environment.

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

- The core is sub-critical.
- The reactor coolant inventory is stable or increasing.
- The core power is removed via the break, and the SGs if necessary.

b) From the controlled state to the safe shutdown state

The transfer to RIS/RRA [SIS/RHRS] conditions is generally not possible. In these circumstances, there is not enough RIS [SIS] injected flow to compensate for the break flow and hence reflood the hot legs. The exception is for the break sizes close to 50 mm equivalent diameter addressed as SB-LOCA in section 5.1 of Sub-chapter 14.4. Under these conditions, the RCP [RCS] loops cannot be reflooded to permit RIS/RRA [SIS/RHRS] operation. The conditions needed for RIS/RRA [SIS/RHRS] suction in the hot legs are based on ΔT_{sat} and RPVL information.

In these circumstances, the safe shutdown state corresponds to the following conditions:

- The core is sub-critical even after Xenon depletion.
- The break flow is matched by RIS [SIS] flow.
- The decay heat is removed from the core.
- The break flow rate is at a temperature lower than the containment saturation temperature limit defined by the equipment qualification limit in the corresponding section of Sub-chapter 3.4.
- The heat is removed from the containment by the RIS/RRI/SEC [SIS/CCWS/ESWS] cooling chain consistent with the containment, IRWST and RIS [SIS] design conditions.
- The activity release is within the limits of PCC-4 events.

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

- Switchover from LHSI cold leg injection to LHSI hot leg injection. This is required in cases with cold leg breaks:

  This action limits the containment pressure increase in the long term in cases with cold leg breaks. After switchover, the LHSI injection delivered into the hot legs condenses the steam flow at the core outlet, and consequently reduces the amount of steam at the break location, see sub-section 6.2.1 of this sub-chapter.

  The switchover of the LHSI from the cold legs to the hot legs also stops the core boron concentration increase in cases of cold leg breaks. This prevents boron precipitation inside the core and excessive boron dilution inside the IRWST.

  This operator action consists of the opening of the valves located in the LHSI hot leg injection lines and the closing of valves located in the LHSI cold leg injection lines. The MHSI continues injecting to the cold legs.

Note: After the automatic partial cooldown, any further cooldown by the secondary side to reach the safe shutdown state is not required. The only exception is for break sizes close to 50 mm equivalent diameter which are dealt with under the SB-LOCA assessment in section 5.1 of Sub-chapter 14.4. However, this cooldown can be helpful to limit the SG energy release into the containment via reverse SG heat transfer in cases with cold leg breaks. The secondary side cooldown is initiated by the operator decreasing the VDA [MSRT] or GCT [MSB] setpoints using criteria linked to RPVL, ΔT_{sat} (subcooling margin) or high containment pressure.

The F1 systems involved in the achievement of the safe shutdown state are:

- The RIS [SIS] (F1A)
- The RRI/SEC [CCWS/ESWS] (F1A part) as support systems to F1A systems.

6.1.2. Acceptance Criteria

The safety criteria to be met are the dose equivalent limits as addressed in Sub-chapter 3.1.

For LOCA analysis the following decoupling criteria will be met as discussed in Sub-chapter 14.0:

- The peak fuel clad temperature shall remain below 1200°C.
- The maximum fuel clad oxidation shall remain below 17% of the total clad thickness.
- The maximum hydrogen generation shall remain below 1% of the amount that would be generated if all the active part of the fuel clad were to react.
- The core geometry shall remain coolable: calculated changes in core geometry shall be such that the core remains capable of being cooled.
- The long term cooling shall be ensured: the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed.

It must be shown that the two following safe states are reached with the application of the safety analysis rules defined in Sub-chapter 14.0:

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

6.1.3. Methods and assumptions

6.1.3.1. Codes and methods

The CATHARE 2 V2.5 computer code [Ref-1] is used; see Appendix 14A for the code description and section 5.1.3.1 of Sub-chapter 14.4 for discussion of the code adequacy and analysis methodology.

The CATHARE validation and the basic principles for CATHARE application under the LOCA realistic deterministic methodology are described in section 5.1.3.1 of Sub-chapter 14.4.

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

The CATHARE code provides a detailed representation of the primary and secondary systems. For the PCC-LOCA analysis, all the loops are modelled explicitly modelled individually. The break is modelled on a loop distinct from that containing the pressuriser, which is connected to the associated hot leg

The break is conservatively located at the lowest part of the cold leg for all pipe ruptures in the cold leg. For cold leg breaks and surge line break, the RIS [SIS] train delivering into the broken leg is credited.

CATHARE's point-kinetic model is not activated; in a conservative manner, the residual fission power (term A) is provided as an input to the CATHARE code. This term A results from a decoupled conservative RT-simulation.

The transient analyses are performed using the conservative PCC-analysis rules defined in Sub-chapter 14.0.

The codes and methods used for the transient calculation to the controlled state, including the maximum clad temperature calculation are as follows:

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

- A second CATHARE calculation is then performed, called the "hot assembly calculation", which models only the hot core assembly and the hot rod belonging to that assembly, using the core boundary conditions from the "system calculation".

6.1.3.2. Main Assumptions

6.1.3.2.1. Break size and position

An intermediate or a large break LOCA is classified as a PCC-4 event. The break size is limited to the largest RCP [RCS] connecting line as discussed in Sub-chapter 14.0. The surge line is the largest line connected to the RCP [RCS] hot leg and the RIS [SIS] line is the largest RCP [RCS] cold leg connecting line.

The intermediate breaks (IB-LOCA) refer to the following break spectrum:

- Break size > 20 cm² (equivalent diameter > 50 mm).

- Break size ≤ the largest RCP [RCS]-connected line on cold and hot leg sides.

The large breaks (LB-LOCA) refer to the following breaks:

- Double-ended break of the pressuriser surge line, on the hot leg side.

- Double-ended break of the RIS [SIS] line, on the cold leg side.

The IB-LOCA and LB-LOCA analyses in the present section address the core cooling aspects. The objective is to demonstrate that the relevant acceptance criteria are met.

6.1.3.2.2. Assumptions relating to the Control and Safety Systems

The assumptions related to the control and safety systems are listed hereafter and summarised in Section 14.5.6 - Table 2.

a) Reactor Trip

The reactor trip occurs following a low pressuriser pressure signal, with a minimum setpoint value including uncertainty (132 bara) and a maximum delay of 1.2 seconds, composed of 0.9 seconds for the response of the I&C channel (including sensor) and 0.3 seconds for the time between the reactor trip signal and the start of rods drop (Sub-chapter 14.1 – Table 11 (1/2)).

Prior to trip, the reactor is conservatively assumed to operate at full power (namely, 102%NP).

In addition, the control rod with the highest reactivity worth is assumed to be stuck in the fully withdrawn position during the rods drop.

b) Turbine trip

The turbine trip is actuated following a reactor trip check-back signal, and is conservatively assumed to be coincident with the reactor trip.

c) Reactor coolant pumps

The reactor coolant pumps are tripped on the turbine trip signal when a loss of off-site power occurs.

d) ARE [MFWS] system

The ARE [MFW] flow is assumed constant until reactor trip. The ARE [MFWS] high load line is isolated immediately on reactor trip and the ARE [MFWS] low-load line isolation is also conservatively considered coincident with the reactor trip.

The AAD [SSS] is not considered.

e) Safety injection system

The safety injection signal is actuated on a very low pressuriser pressure signal, with a minimum setpoint value of 112.0 bara including uncertainty (115 – 3 bara). The RIS [SIS] is assumed to inject into the RCP [RCS] at full flow after a conservative maximum delay of 40 seconds, including emergency diesel generators start time.

The minimum MHSI and LHSI injection flow rates are used in the analysis in order to pessimise the refilling of the RCP [RCS] (Section 14.5.6 - Figure 1). Furthermore, in order to conservatively model the IRWST heat-up during the accident, the RIS [SIS] injection temperature is set to 70°C, throughout the event.

f) Accumulators

The accumulators discharge into the RCP [RCS] when the primary pressure falls below their initial pressure. In order to delay and minimise the accumulator injection, a bounding set of initial conditions is considered: the temperature, the discharge line resistance and the initial water volume of the accumulators are maximised and the set pressure is minimised. The values of these parameters are as follows:

- Pressure setpoint (minimum value): 45 bara
- Temperature (maximum value): 50°C
- Initial water volume (maximum value in order to delay the discharge of accumulators): 35 m^3
- Line resistance K/A^2 (maximum value): 2500 m^{-4}

g) ASG [EFWS] system

The ASG [EFWS] system is activated following a loss of off-site power signal and safety injection signal or following a low SG level (MIN2) signal. The delay time for achieving full ASG [EFWS] flow is 50 seconds taking into account a LOOP and including emergency diesel generators start time.

The primary to secondary heat transfer is modelled conservatively; the ASG [EFWS] flow rate is minimised and the ASG [EFWS] temperature is maximised (50°C).

The ASG [EFWS] isolation occurs following a high SG level (MAX1) signal.

h) VDA [MSRT] and Partial Cooldown

All VDA [MSRT] (one per SG) are available. The VDA [MSRT] setpoint corresponds to the nominal value plus 1.5 bar uncertainty.

The SI signal initiates the Partial Cooldown: an automatic decrease of SG pressure at a rate equivalent to a cooldown of 250°C/h down to 60 bar (+1.5 bar uncertainty in order to reduce the duration of the cooldown).

If the VDA [MSRT] are already open, the partial cooldown starts immediately; otherwise it starts after 2.0 seconds (maximum delay for VDA [MSRT] opening).

6.1.3.2.3. Operator actions

No operator action is considered before 30 minutes after the reactor trip. After 30 minutes, operator actions are necessary to reach the safe shutdown state.

Following smaller intermediate break sizes for which the transfer to RIS/RRA [SIS/RHRS] operating conditions is possible, the operator actions are those described in section 5.1.3.2 of Sub-chapter 14.4.

For the larger break sizes where RCP [RCS] saturation conditions are reached, the operator must perform the following F1B actions, based on the following F1B information:

- Switchover of LHSI from cold leg injection to hot leg injection, by opening of the LHSI hot leg injection valves and closing the LHSI cold leg injection valves. This action, required for cold leg breaks, is performed at 1.5 hours after RT.
- Maintaining MHSI injection to the cold legs.
- The decision is based on Information on containment pressure, RCP [RCS] pressure, ΔT_{sat} and RPVL.

6.1.4. Limiting break size determination

The analysis covers a break spectrum up to the largest connecting line (surge line). For a given break size, the cold leg location is more limiting in terms of core cooling than the hot leg location or the pressuriser location including breaks in the pressuriser steam space.

Ten break sizes have been studied:

- 45 cm^2 (75 mm equivalent diameter) break in the cold leg,
- 80 cm^2 (100 mm equivalent diameter) break in the cold leg,
- 100 cm^2 (115 mm equivalent diameter) break in the cold leg,
- 120 cm^2 (125 mm equivalent diameter) break in the cold leg,
- 140 cm^2 (135 mm equivalent diameter) break in the cold leg,
- 160 cm^2 (145 mm equivalent diameter) break in the cold leg,
- 180 cm^2 (150 mm equivalent diameter) break in the cold leg,
- 200 cm^2 (160 mm equivalent diameter) break in the cold leg,
- RIS [SIS] line break (390 cm^2 - 225 mm equivalent diameter) located in the cold leg,
- Surge line break (2 x 830 cm^2 - 2 x 325 mm equivalent diameter) located in the hot leg.

The RIS [SIS] line corresponds to the largest nozzle in the cold leg. The RIS [SIS] line break is modelled as the double-ended guillotine break of the RIS [SIS] injection line.

The pressuriser surge line is the largest pipe connected to the RCP [RCS] hot leg. The surge line break is modelled as the double-ended guillotine break of the surge line. The break is conservatively located as close as possible to the hot leg.

The 140 cm^2 break is the most onerous, with a maximum Peak Clad Temperature (PCT) of 692°C.

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

Safety analyses are performed using conservative assumptions.

6.1.5.1. Choice of single failure and preventive maintenance

For the first break spectrum, the most onerous active single failure is the loss of one EDG. One RIS [SIS] train (one MHSI pump and one LHSI/RHR pump) and one ASG [EFWS] pump are therefore unavailable on an intact loop.

For the second break spectrum, a passive single failure on the common RIS [SIS] check-valve is assumed. This leads to the unavailability of an accumulator and of a RIS [SIS] train.

In both cases, the most onerous preventive maintenance is the loss of one EDG. One RIS [SIS] train (one MHSI pump and one LHSI/RHR pump) and one ASG [EFWS] pump are therefore unavailable on a further intact loop.

Note: the unavailability of the ASG [EFWS] train affected by the passive single failure has been considered. This choice of modelling has been made for practical reasons, considering the minimal impact of the ASG [EFWS] on LOCA accidents and bearing in mind that this will lead to a slightly over-conservative result.

6.1.5.2. Initial state

The initial state is summarised in Section 14.5.6 - Table 1. These assumptions are conservative for maximising core heating:

Initial power: the unit is initially operating at full power; 102% of nominal power is considered to maximise the residual power.

Primary loop flow rate: the primary loop flow rate corresponds to the thermal-hydraulic flow rate in order to pessimise core cooling. The assumed flow rate is 27,185 m^3/h/loop.

Vessel average temperature: the vessel average temperature is 315.3°C and corresponds to the nominal value increased by 2.5°C uncertainty in order to maximise the initial fluid and structure stored energy.

Primary pressure: the primary pressure is 157.5 bar and corresponds to the nominal value increased by 2.5 bar uncertainty. The primary pressure is maximised in order to delay the reactor trip and the safety injection signal.

Pressuriser level: the pressuriser level value is 61% and corresponds to the nominal value increased by 5% uncertainty. The pressuriser level is maximised in order to delay the reactor trip and the safety injection signal.

Note that the impact of an increase in the initial pressuriser level is assessed in sub-section .6.1.5.4.2 below.

Steam Generator level: the SG level value is 49% of the narrow range. It corresponds to the nominal level since this parameter does not have a significant effect on the transient.

6.1.5.3. Specific assumptions

a) Core related assumptions

The entire core is modelled by one average core rod in the CATHARE calculation. The axial power shape is shown as curve “Average rod” in Section 14.5.6 - Figure 2. The characteristics of this average core rod are summarised in Section 14.5.6 - Table 3 and are as follows [Ref-1]:

- Axial Offset: AO=+18% (+15% + 3% uncertainty),
- Enthalpy rise factor: $F\Delta H$=1.0,
- Peaking factor: F_Q=1.57 at 2.8 m of the active core height.

The hot assembly and its hot rod are modelled in "hot rod" calculations. In these analyses, the "Hot assembly" curve in Section 14.5.6 - Figure 2 is used as the axial power shape for the average rod of the hot assembly and the "Hot rod" curve in Section 14.5.6 - Figure 2 is used as the axial power shape for the hot rod of the hot assembly.

The characteristics of the average rod of the hot assembly are summarised in Section 14.5.6 - Table 3 and are the following ones:

- Axial Offset: AO= +21% (+18% + 3% uncertainty).
- Enthalpy rise factor: FΔH=1.67
- Peaking factor: FQ=2.61
- Peaking factor elevation: Z_{FQ}=3.5 m

The characteristics of the hot rod of the hot assembly are summarised in Section 14.5.6 – Table 3 and are the following ones:

- Axial Offset: AO= +21% (+18% + 3% uncertainty).
- Enthalpy rise factor: FΔH=1.80
- Peaking factor: FQ=2.82
- Peaking factor elevation: Z_{FQ}=3.5 m

The ratio between the hot rod power and the hot assembly power is 1.08.

These power shapes are chosen because they provide a conservative distribution of power versus core height. The power is maximised in the upper part of the core, which is limiting for IB/LB-LOCA analysis, due to the core uncovery. As the core uncovers, the fuel clad in the upper part of the core heats up. Thus, increasing the linear power in the upper part of the core increases the fuel clad temperature excursion. The clad temperature in the lower part of the core remains close to the saturation temperature.

The bounding curve of residual power used in this study is the 'A+B+C with 2 σ' decay heat law with earthquake impact and assuming a stuck rod for reactor trip [Ref-2].

b) Loss of off-site power

Based on sensitivity studies, the assumption of loss of off-site power is conservative for IB/LB-LOCA analyses under the application of the single failure and preventive maintenance principle. It occurs at the time of turbine trip and causes the reactor coolant pump trip and the activation of the emergency diesels, which delays the actuation of safety injection into the reactor coolant system.

c) Assumptions related to non-F1 systems

Non-classified systems (e.g. control systems) are not credited when they perform a beneficial function.

The pressuriser pressure control system (pressuriser heaters) is modelled as it delays the RT signal. A total heating power of 2592 kW is modelled until the low pressuriser level (12% MR) setpoint is reached.

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

The ARE [MFW] flow is assumed constant until reactor trip. The ARE [MFWS] low-load line is conservatively assumed to be isolated after the reactor trip.

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

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

6.1.5.4. Results

The analysis is described for one break size (140 cm^2).

6.1.5.4.1. Sequence of events

Section 14.5.6 - Table 4 gives the sequence of events for the various break sizes assessed.

6.1.5.4.2. Transients results

6.1.5.4.2.1. First break spectrum

Thermal-hydraulic parameters and core level changes are shown in Section 14.5.6 - Figure 3 to Section 14.5.6 - Figure 42.

Paragraphs hereafter describe the transient for the 140 cm^2 break size. For the other break sizes, the main phenomena that are described below remain valid with a few differences.

At the beginning of the transient, the opening of the break causes the depressurisation of the primary circuit due to the high sub-cooled break flow (Section 14.5.6 - Figure 19).This phase covers the emptying of the pressuriser. As the pressure of the primary circuit reaches the setpoint pressure of 132 bar, the reactor trip signal is actuated. After the turbine trip the secondary pressure increases until the VDA [MSRT] opens.

The primary circuit depressurisation continues and the very low pressuriser pressure signal [MIN3] is actuated so that partial cooldown starts. The actuation of the RIS [SIS] occurs 40 seconds after this signal. The ASG [EFWS] system activated on the SI signal combined with LOOP leads to the SG inventory recovering.

The partial cooldown decreases the RCP [RCS] temperature and pressure allowing MHSI injection 190 seconds after the SI signal.

During the period when the MHSI flow does not compensate for the break flow, the RCP [RCS] inventory continues to fall and core uncovery occurs (Section 14.5.6 - Figure 21). During the uncovery phase, the hot rod clad average temperature reaches a maximum of 692°C. Delivery from the accumulators, MHSI and LHSI pumps restore the primary water inventory.

To conclude, the break size sensitivity study shows that a break area of 140 cm^2 is the limiting case. This break size results in the longest (and one of the deepest) core uncovery. The maximum peak clad temperature is 692°C. Therefore, the criterion of 1200°C for the maximum fuel clad temperature is met. The total oxidation does not exceed 1% for the whole break spectrum.

For the smallest break sizes, less primary fluid mass is discharged via the break and therefore the core uncovery is shallower.

Larger breaks cause a more significant pressure drop. It leads to earlier RIS [SIS] injection with larger flow rates and then to a shorter core uncovery.

a) Cladding temperature

The maximum peak cladding temperature obtained for the worst break (140 cm²) is 692°C.

b) Oxidation rate

As the clad temperature of the worst break is lower than 692°C, the total oxidation during the transient does not exceed 1% for the whole break spectrum.

As a consequence, the core geometry does not change during the transient and thus remains coolable.

c) Hot rod deformation

The hot rod maximum deformation has been calculated on the most onerous case and is shown to be negligible due to the low cladding temperature (see Section 14.5.6 - Figure 75).

As a consequence, the core geometry does not change during the transient and thus remains coolable.

In addition, a sensitivity calculation to the initial pressuriser level has been performed on the most conservative break. It shows that an increase in the pressuriser level of 3.5% (i.e. initial level 56 + 5 + 3.5%) does not impact the peak cladding temperature. Therefore, an increase in initial pressuriser level does not impact the results of this analysis.

6.1.5.4.2.2. Second break spectrum

The analysis of the second break spectrum, performed in the context of the passive single failure analysis (Sub-chapter 14.2), demonstrates that the acceptance criteria are met when considering a passive single failure on the common RIS [SIS] check-valve from the start of the transient. The results are provided in Section 14.5.6 - Table 6 and Section 14.5.6 – Figure 43 to Figure 75.

6.1.6. Description of studied cases from controlled state to safe shutdown state

Depending on the plant status, the transfer to the safe shutdown state is ensured:

- Either by reaching the RIS/RRA [SIS/RHRS] connecting conditions, with at least one LHSI operating in RHR mode (LHSI/RHR).
- Or by reaching the LHSI injection conditions, with all available LHSI operating in SI-mode. In cases with a cold leg breaks, the switchover of LHSI from cold leg injection to hot leg injection is required.

a) Safe shutdown state with heat removal via LHSI in RHR mode

For the smaller break sizes of the PCC-4 break spectrum, the transfer to LHSI/RHR may be possible, as for the PCC-3 SB-LOCA. The connection of LHSI/RHR is possible when the following RCP [RCS] conditions are met:

- RCP [RCS] hot leg pressure below 30 bar
- RCP [RCS] hot leg temperature below 180°C
- ΔT_{sat} and RPVL consistent with LHSI/RHR suction from the hot leg.

For those smaller PCC-4 breaks, the consumption of ASG [EFWS] water before connecting the LHSI/RHR, is lower than for the PCC-3 SB-LOCA. This occurs because a larger break size removes more heat through the break. This decreases the heat transferred to the SG secondary side. The time to reach the RIS/RRA [SIS/RHRS] connecting conditions is also reduced due to the faster RCP [RCS] depressurisation. The ASG [EFWS] tanks water inventory is then sufficient to reach the RIS/RRA [SIS/RHRS] connecting conditions, as is shown in Sub-chapter 14.4 for SB-LOCA.

The safe shutdown state is reached once connection of one LHSI in RHR mode occurs. One LHSI/RHR is sufficient for heat removal. The remaining LHSI operates in SI-mode, possibly complemented by MHSI with the large mini-flow line open:

- At least one LHSI operates in RHR mode.
- At least one LHSI, and two MHSI if needed, operates in SI-mode, with cold leg injection.

This safe shutdown state is the one described for the SB-LOCA in section 5 of Sub-chapter 14.4.

b) Safe shutdown state with heat removal via LHSI in SI-mode

For the larger break sizes of the PCC-4 break spectrum, the degradation of the RPVL and ΔT_{sat} indications does not permit the connection of the LHSI in RHR mode. In such cases, the switchover of LHSI pumps to RCP [RCS] hot leg injection is performed by the operator, while the MHSI pumps are kept injecting to the cold legs.

The switchover of LHSI from the cold legs to the hot legs is needed for cold leg breaks:

- For the containment, to limit steam discharge and prevent excessive pressures and temperatures inside the containment. In the case of a cold leg break, only a part of the RIS [SIS] flow injected into the cold legs enters and boils in the core, the remainder discharging directly via the break. To prevent boiling in the core RIS [SIS] injection in the core outlet is needed. The LHSI flow injected into the hot legs is able to enter the upper plenum.
- For the core, to stop the core boron concentration reaching the precipitation limit. In the case of a cold leg break, should only steam leave the core, removing no boron, the core boron concentration would increase significantly due to the cold leg SI boron injection. To stop the boron concentration increase, it is necessary to promote the discharge of liquid from the core. The LHSI flow injected into the hot legs is able to initiate a liquid flow out from the core and hence reduce the core boron concentration.

The time of switchover will be:

- Short enough to stop boiling before the pressure or temperature limits inside the containment are exceeded.
- Short enough to prevent unacceptable boron dilution of the IRWST, which could lead to the core returning to criticality at initiation of the hot leg injection. The liquid discharge would remove highly borated water from the core and potentially replace it with low boron concentration injection fluid from the IRWST.
- Long enough to ensure that the hot leg injected flow is able to penetrate the RPV upper plenum and the core, with no degradation of the core cooling and preventing an unacceptable core uncovery.

A switchover time of 1.5 hours, after RT, is presently foreseen. After switchover, the RIS [SIS] configuration is:

- At least two LHSI operate in SI-mode, with hot leg injection.
- At least two MHSI operate in SI-mode, with cold leg injection.

For those PCC-4 breaks, the safe shutdown state is achieved after completion of the switchover. For smaller break sizes, the RCP [RCS] cool down via the SG may be needed to reach the LHSI injection conditions (P ≈ 20 bar). The RCP [RCS] cool down capacity shall initiate the LHSI injection, before reaching the core boron precipitation and the IRWST boron dilution limits. The following safety analysis demonstrates that the RIS [SIS] design is sufficient to meet all the safety and acceptance criteria before switchover, from the controlled state to the safe shutdown state, and after switchover, the stable long term safe shutdown state, with use of only F1A and F1B systems.

The acceptance criterion under consideration is the long-term core cooling. To demonstrate long term cooling, the following criteria need to be met:

1. Adequate core cooling before switchover time, with MHSI and LHSI into CL.
2. Adequate core cooling after switchover time, with MHSI into CL and LHSI into HL.
3. Presence of core liquid discharge after switchover time.
4. At switchover time, the core boron concentration is below the precipitation limit,
5. At switchover time, the boron dilution limit in the IRWST is not reached.

This safety analysis is performed as follows:

- The first part of the analysis addresses items 1, 2 and 3, related to the <u>thermal-hydraulic assessment</u>. It consists of the thermal-hydraulic transient calculation of the most onerous IB/LB-LOCA with the coupled CATHARE/CONPATE codes. CATHARE calculates the RCP [RCS] and SG behaviours; CONPATE calculates the containment behaviour, including the IRWST. The objective of the analysis is to show the efficiency of the LHSI switchover at the switchover time of 1.5 hours after RT, i.e. the capability to keep the core covered and to initiate liquid discharge from the core.

- The second part of the analysis addresses items 4 and 5, related to the boron assessment. It consists in a conservative estimation of the minimum switchover times which would have led to either the boron crystallisation limit in the core, the risk of boron precipitation, or the boron dilution limit in the IRWST, the risk of the core becoming critical. This boron assessment relies on simple bounding mass and energy balances, without the use of a code calculation. It utilises appropriately conservative boundary conditions for the core boration and for IRWST dilution. The objective is to show that the switchover time of 1.5 hours is lower than these limiting switchover times, whilst minimising the risks of core boron precipitation and core re-criticality.

All these analyses are performed with the conservative assumptions discussed in sub-section 6.1.5 of this sub-chapter associated with the achievement of the controlled state. The number of LHSI and MHSI pumps available is minimised, considering the worst single failure and preventive maintenance.

As discussed above, the limiting RCP [RCS] breaks with respect to SB LOCA and IB LOCA are the cold leg breaks. For hot leg breaks, there is no need for LHSI switchover, the cold leg injection being the most efficient injection mode as the whole RIS [SIS] flow enters the core. The most onerous PCC-4 IB/LB-LOCA is the largest cold leg break, being the 390 cm^2 RIS [SIS] line break. This is the break considered in the two assessments.

The consequences of the IB/LB-LOCA with respect to the containment are dealt with in Sub-Chapter 6.2 related to "Containment systems".

6.1.6.1. Acceptance criteria

The following values are retained for the boron concentration limits:

The limit for boron solubility in water at 100°C is 27.5% w/o boric acid solution. If a 4% margin for uncertainties is included, this becomes 23.5%. This corresponds to a maximum limit of 41130 ppm for the core boron concentration. [Ref-2]

The minimum boron concentration required at cold shutdown to avoid core re-criticality is 1728 ppm of natural boron [Ref-1]. This value covers all fuel cycles and $U0_2$/MOX fuel managements, including uncertainties, and accounts for all rods inserted except the one with the highest negative reactivity worth. This rod is assumed to be stuck in its fully withdrawn position.

The boron concentration limits are thus:

- Core boron concentration < 41130 ppm, enriched boron, to the prevent crystallisation.
- IRWST boron concentration > 1728 ppm, natural boron, to prevent core re-criticality.

6.1.6.2. Choice of single failure and preventive maintenance

The worst single failure (SF) is the loss of one emergency diesel at the time of LOOP. One RIS [SIS] train, consisting of one MHSI pump and one LHSI pump, and one ASG [EFWS] pump are thus unavailable.

The preventive maintenance (PM) of one emergency diesel is the most conservative configuration because one RIS [SIS] train, comprising one MHSI pump and one LHSI pump, and one ASG [EFWS] pump are thus unavailable.

The assumption of LOOP is conservative for IB and LB-LOCA analyses through application of the SF and PM principles.

6.1.6.3. Specific assumptions

a) Thermal-hydraulic assessment (CATHARE/CONPATE calculation)

- Residual heat

The maximum decay heat curve "MAX 2σ", as described in Sub-chapter 14.1 and used in sub-section 6.1.5.3 of this sub-chapter for the controlled state, is utilised.

- Assumptions concerning the secondary side

Because of the application of the single failure and the preventive maintenance, only two ASG [EFWS] are delivering to the SG. The flow rate per pump is the same as that for the analysis performed of the transient to reach the controlled state discussed in sub-section 6.1.5 of this sub-chapter. The SG level is controlled in order to prevent SG overfill.

The SG pressure is held constant after the end of partial cooldown and remains at 61.5 bar (60 bar + 1.5 bar uncertainty) as at the controlled state as discussed in sub-section 6.1.5 of this sub-chapter***.*** This assumption maximises the reverse heat transfer from the SG to the RCP [RCS], which is conservative for the core cooling, higher steam binding, and for the IRWST water temperature due to higher mass and energy releases into the containment.

- Assumptions concerning the RIS [SIS]

At the controlled state, there is only one RIS [SIS] train comprising one MHSI and one LHSI pump injecting into one RCP [RCS] cold leg. Two RIS [SIS] trains have been lost through the application of the single failure and preventive maintenance assumptions. The one remaining train is delivering direct into the containment through the break and is thus unavailable for RCP [RCS] water injection.

At 1.5 hours after the reactor trip, the operator performs the switchover of all the LHSI pumps from the cold legs to the hot legs. The LHSI pump connected to an intact loop and the LHSI pump whose flow was previously lost through the break are now injecting into two RCP [RCS] hot legs.

The characteristics of the MHSI and LHSI pumps are minimised as described in Sub-chapter 14.1.

The accumulators are conservatively isolated at the beginning of the operator action, 30 minutes after the reactor trip.

The initial IRWST temperature is assumed to be its maximum of 50°C. The initial IRWST water volume is minimised to maximise the IRWST temperature increase. The initial volume is assumed to be 1450 m^3 as discussed in Sub-chapter 14.1***.***

The MHSI and LHSI water temperature profile is also maximised. This assumes the IRWST temperature increase due to degraded containment conditions, limited by the RRI/SEC [CCWS/ESWS] heat removal via the LHSI heat exchanger. This temperature derives from a coupled CATHARE/CONPATE calculation, see Appendix 14A.

The boron calculation in CATHARE is used as qualitative information only. Its purpose is solely to show that the core boron concentration stops increasing following switchover. The initial boron concentration in the IRWST is maximised and bounds both MOX and UO_2 fuel management schemes. The dilution in the IRWST before switchover is not considered in this calculation but is considered in the boron assessment described below.

b) Boron concentration assessment

In order to estimate the minimum switch-over time at which the 41130 ppm core precipitation limit would be reached, the following assumptions are chosen to maximise the core boron concentration:

- Initial IRWST water volume: 1940 m^3 (max value)
- Initial IRWST boron concentration: 1780 ppm (max value)
- Initial accumulator boron concentration: 1780 ppm (max value)
- Initial RCP [RCS] boron concentration: 1520 ppm (max value BOL UO_2 and MOX)
- Initial accumulator water volume: 35 m^3 per accumulator (max value)
- Initial RCP [RCS] water mass: 300 tons

Note: in this calculation, the boron concentrations refer to enriched boron.

In order to estimate the minimum switch-over time at which the 1728 ppm IRWST dilution limit would be reached, the following assumptions are chosen to minimise the IRWST boron concentration:

- Initial IRWST water volume: 1450 m^3 (min value)
- Initial IRWST boron concentration: 2450 / 2700 ppm (min value UO_2/MOX)
- Initial accumulator boron concentration: 2450 / 2700 ppm (min value UO_2/MOX)
- Initial RCP [RCS] boron concentration: 1771 / 1895 ppm (min value BOL UO_2/MOX)
- Initial accumulator water volume: 30 m^3 per accumulator (min value)
- Initial RCP [RCS] water volume: 300 tons

Note: In this calculation, the boron concentrations refer to natural boron.

In both calculations, the maximum decay heat curve "MAX 2σ", as described in Sub-chapter 14.1 and used in sub-section 6.1.5.3 of this sub-chapter for the transient to the controlled state, is considered.

6.1.6.4. Results

a) Thermal-hydraulic assessment (CATHARE/CONPATE calculation)

The CATHARE/CONPATE calculation of the RIS [SIS] line break, of area 390 cm^2 in CL up to the switchover time of 1.5 hours after RT is not performed in the PCSR. The demonstration of adequate core cooling, the thermal-hydraulic aspect, before and after switchover is based on the result of the calculation performed in the BDR-99 for the EPR 4900 MWth. The EPR 4900 MWth analyses show that adequate core cooling is achieved for the RIS [SIS]-line break. Comparison of the relevant characteristics between EPR 4500 MWth and EPR 4900 MWth, shows that the EPR 4900 MWth is conservative compared to the EPR 4500 MWth for showing adequate core cooling.

EPR 4900 MWth accident analysis of BDR-99

The results of the BDR-99 are provided in Appendix 14B. They are summarised below:

- At the switchover time of 1.5 hours after RT, the two LHSI flow-rates injected into the hot legs are sufficient to flow against the steam coming from the core and hence reach the core. The flows consequently reduce the core boron concentration and are sufficient to remove the core residual heat, in the case of the largest cold leg break. The core always remains covered after the controlled state has been reached, and IRWST temperature remains significantly below the design limit for RIS [SIS] pumps operation.

- For a break located in the hot leg there are at least two MHSI pumps delivering into the cold legs at the switchover time of 1.5 hours after RT. This covers both the most conservative single failure and preventive maintenance, assumptions. At this time, the minimum MHSI flow rate injected is sufficient to remove about 3.5 times the core residual heat under saturated steam conditions. The MHSI flow rate alone is able to remove the entire residual heat, with liquid flow leaving the core. The core boron concentration does not increase, and the IRWST boron concentration does not decrease.

These results show that the RIS [SIS] flow injection and heat removal capacities are sufficient to remove the core residual heat, irrespective of the break location, without core uncovery before and after the switchover time of 1.5 hours after RT. For a cold leg break, after switchover the boron concentrations equalise between the core and the IRWST. For a hot leg break, this equalisation occurs without the need for switchover.

Transposition of BDR-99 results to EPR 4500 MWth

When compared to the EPR 4900 MWth characteristics, the EPR 4500 MWth has less onerous requirements for core cooling:

- The power level is 9% lower in the EPR 4500 MWth when compared to the EPR 4900 MWth.

- The MHSI injection flow, in the low pressure range, and the LHSI injection flow are the same for the EPR 4500 MWth design and the EPR 4900 MWth design.

- The IRWST water content is the same in the EPR 4500 MWth and the EPR 4900 MWth, providing a larger heat sink reserve relative to the power level for the EPR 4500 MWth.

Therefore, it can be concluded that at the time of LHSI switchover of 1.5 hours after RT, the benefit from the LHSI flow injected in the hot legs and the benefit from the MHSI flow injected in the cold legs are increased for the EPR 4500 MWth. This is based on the results of BDR-99 and on the less onerous conditions for the EPR 4500 MWth when compared to the EPR 4900 MWth,

Therefore, the RIS [SIS] flow injection and heat removal capacities of EPR 4500 MWth are sufficient to remove the core residual heat, whatever the break location, without core uncovery before and after the switchover time of 1.5 hours after RT. In the case of a cold leg break, the LHSI switchover prevents the boron accumulation within the core and the boron dilution of the IRWST.

b) Boron concentration assessment

To ensure that the heat removal is effective, it must be demonstrated that the core geometry remains coolable with no boron precipitation. It must also be shown that the core return to criticality is prevented with no unacceptable IRWST dilution. The following Boron concentration assessment is performed:

The estimate of the minimum switch-over time for the <u>boron precipitation in the core</u> relies on a simplified conservative mass and energy balance:

- Only steam is assumed to leave the core with no boron removed.
- Steam flow rate assuming the maximum decay heat curve ("MAX 2σ").
- Steam flow rate assumes no subcooling within the RCP [RCS], boiling of 100°C saturated water.
- The volume of liquid in which the boron is concentrating, is minimised. Only the core and lower-plenum volumes are included.

The minimum switchover time, at which the 41130 ppm core boron precipitation limit would be reached, is 20.5 hours after RT. This assumes an initial maximum IRWST boron concentration of 1780 ppm enriched boron. The 20.5 hours calculates bounds the delay for both UO_2 and MOX fuel management schemes. There is consequently no possibility of core boron precipitation for a switchover time of 1.5 hours after reactor trip. Thus the core geometry remains coolable.

The estimation of the minimum switchover time for <u>boron dilution in the IRWST</u> relies on a simplified conservative mass and energy balance:

- Only steam is assumed to be discharged from the core with no boron removed.
- Steam flow rate assuming the maximum decay heat curve ("MAX 2σ").
- Steam flow rate assumes no subcooling within the RCP [RCS], boiling of 100°C saturated water).
- The IRWST volume, in which the boron is diluted, is minimised.

The minimum switchover time for which the 1728 ppm IRWST boron dilution limit would be reached is at least 6.3 hours after RT. This assumes an initial minimum IRWST boron concentration of 2450 ppm natural boron. The 6.3 hours calculated bounds the delay for both UO_2 and MOX fuel management schemes. Consequently, no possibility of core re-criticality exists for a switchover time of 1.5 hours after RT.

As a result, there is no risk of boron crystallisation in the core and no risk of unacceptable dilution in the IRWST, for a switchover time of 1.5 hours after RT:

- Switchover time considered: 1.5 hours after RT.
- Maximum switchover time without risk of core boron crystallisation: 20.5 hours after RT
- Maximum switchover time without risk of core re-criticality: 6.3 hours after RT [1]

c) Conclusion

Under appropriate operation, the RIS [SIS] design:

- Ensures adequate core heat removal for all IB/LB-LOCA break sizes and locations
- Prevents the possibility of boron precipitation in the core, and prevents unacceptable dilution in the IRWST

The appropriate operation relies on the switch-over of LHSI from CL-injection to HL-injection, actuated by the operator at a minimum time of 1.5 hours after RT. This is required for cold leg breaks large enough to result in RCP [RCS] saturation. This ensures safe long-term core cooling is achieved.

The analysis shows that despite the worst single failure and consideration of the most onerous preventive maintenance, the safe shutdown state can be reached meeting all the safety criteria, and consistent with the following F1 methods:

- During the transfer from the controlled state to the safe shutdown state:
  - The VDA [MSRT] and ASG [EFWS] pump and tank capacities for core heat removal if needed.
  - The LHSI/RHR heat exchange capacity for IRWST heat removal.
  - The MHSI, accumulators and LHSI if needed, cold leg injection capacities from the IRWST for RCP [RCS] water inventory.
  - The MHSI and LHSI boron injection capacities for core sub-criticality.
- At the switchover time of 1.5 hours after RT, corresponding to the safe shutdown state for larger cold leg breaks:
  - The LHSI/RHR heat exchange capacity for IRWST heat removal
  - The LHSI hot leg injection capacity for RCP [RCS] water inventory
  - The MHSI cold leg injection capacity for RCP [RCS] water inventory
  - The MHSI and LHSI boron injection capacities for core sub-criticality

[1] The RCP [RCS] cooldown capacity is sufficient to reach the LHSI injection (P ≈ 20 bar) before 6.3 hours after reactor trip, for the break sizes which result in only steam discharge from the core.

6.1.7. Conclusion

Significant results are given in Section 14.5.6 - Table 7. The limiting break is the 140 cm^2 cold leg break.

The present analysis shows that the safety criteria of IB/LB LOCA are met with significant margins:

- The most limiting peak cladding temperature is 763°C and is obtained for the 140 cm^2 break for the case with the failure on the common RIS check-valve.
- The total oxidation does not exceed 1% for the whole break spectrum.
- The maximum hydrogen generation remains below 1%
- The core geometry remains coolable since it is demonstrated that the maximum hot rod cladding deformation remains negligible during the whole transient.
- The long term cooling is ensured.

As a conclusion, the cladding temperatures following PCC-4 Intermediate (IB) and Large Break (LB) LOCA faults are far below the limiting criterion.

The PCC-4 acceptance criteria are met with significant margins.

It is also demonstrated, in the context of the passive single failure analysis (Sub-chapter 14.2), that the acceptance criteria are met when considering a passive single failure on the common RIS [SIS] check-valve from the start of the transient.

The safe shutdown state can be reached and the safety criteria met, consistent with:

- RIS [SIS] design conditions, the IRWST temperature, defined in section 2 of Sub-chapter 6.3,
- Containment design pressure and temperature defined in section 1 of Sub-chapter 6.2,
- Containment equipment qualification pressure and temperature limits defined in Sub-chapter 3.4.

6.2. INTERMEDIATE AND LARGE BREAK LOCA IN SHUTDOWN STATE, RIS/RRA [SIS/RHRS] NOT CONNECTED (PCC-4)

6.2.1. Accident definition

The initiating event is a non-isolatable break or leak on the RCP [RCS].

It is defined as a break size of equivalent diameter greater than 50 mm, equivalent area greater than 20 cm^2, up to the largest break areas corresponding to the rupture of the largest RCP [RCS] connecting nozzle. This is 390 cm^2 for the cold leg, corresponding to the RIS [SIS] nozzle rupture and 830 cm^2 for the hot leg, corresponding to the surge line nozzle break [Ref-1]. As in state A, the intermediate and Large Break LOCAs are classified as a PCC-4 event in state B (intermediate shutdown state).

The shutdown state with RIS/RRA [SIS/RHRS] not connected (state B) covers all shutdown states during normal plant operation where primary heat is removed by the SG. In this state, some F1A classified I&C signals have been changed compared to power operation i.e. they are different to state A. It extends from 130 bar where deactivation of some F1A signals occurs, to 30 bar/120°C, where conditions for the connection of the LHSI/RHR RCP [RCS] are reached.

The IB/LBLOCA analysis in state B primarily addresses the core cooling degradation aspect. The objective is to demonstrate that the relevant safety and acceptance criteria are met, including demonstrating the capability of the F1 systems is sufficient to ensure the safe shutdown is reached.

State B introduces the following differences in terms of F1 mitigation methods when compared to state A:

- The change of SI signal.
- The unavailability of accumulators below an RCP [RCS] pressure of 70 bar.

6.2.2. Typical sequence of events

6.2.2.1. From initiator up to controlled state

The main consequence of the RCP [RCS] break is a loss of RCP [RCS] water inventory.

The following mitigation actions are assumed to compensate for the loss of RCP [RCS] water inventory.

The SI signal on "Low-low pressuriser pressure" available in state A is deactivated in state B. This is to avoid spurious actuation at reduced RCP [RCS] pressure conditions during the normal cooldown. Consequently, a different SI signal is utilised. It relies on the ΔP_{sat} [2] measurement (F1A classified), calculated from the hot leg temperature measurement, to define the saturation pressure, and the hot leg pressure measurement. This signal actuates the RIS [SIS] trains while the RCP [RCS] hot legs are still subcooled.

Matching the break flow: The RCP [RCS] water inventory stops decreasing when the RIS [SIS] flow injected by the MHSI, accumulators (if available) and LHSI (if needed) is sufficient to match the break flow rate.

For core heat removal: Whilst the break flow is insufficient to remove the decay heat, secondary side heat removal is necessary. This utilises the ASG [EFWS] and VDA [MSRT], both F1A classified if the ARE [MFWS], AAD [SSS], and GCT [MSB] are unavailable, all non F1 classified. Once the break flow combined with the RIS [SIS] is sufficient to remove the full decay heat, the SG heat removal is no longer required and the RCP [RCS] pressure falls below the SG pressure.

For containment heat removal: As the break flow temperature is higher than 100°C, flashing occurs at the break. This results in steam discharge to the containment and in an increase in the IRWST temperature. The IRWST heat removal via the RIS/RRI [SIS/CCWS] heat exchangers limits the IRWST temperature increase, and performs the containment heat removal.

The controlled state is defined as a state, where:

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

- The reactivity is under control.
- The core power is removed via the SG which are in standby prior to the accident.
- The reactor coolant inventory is stable or increasing, with the safety injection flow at least matching the break flow.

6.2.2.2. From the controlled state to the safe shutdown state

The safe shutdown is the same as that defined for state A.

As there is no core heat-up a switchover of LHSI from cold to hot leg injection, required in state A, is not necessary.

6.2.3. Safety criteria

The safety criteria and acceptance criteria to be met are the same as those described for IB/LB LOCA in reactor state A.

6.2.4. Definition of studied cases

The following two limiting break sizes and locations are analysed using the CATHARE code for the range of state B where the accumulators are isolated and therefore unavailable:

- Break of RIS [SIS] line on the cold leg side, size $390cm^2$.
- Break of surge line on the hot leg side, size 2 x $830cm^2$.

Unlike state A, in the most onerous state B2, with very low power and low secondary pressure, there is no possible “gap” between MHSI and accumulator injection for intermediate break sizes. Thus, the largest possible break sizes lead to the maximum RCP [RCS] discharge and the largest potential core heat up.

The cases studied consider:

- The most onerous single failure
- The onerous preventive maintenance on the most significant equipment

In state B, there is no consideration of LOOP coincident with the event.

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

The demonstration of acceptable conditions is performed making the same conservative assumptions as those made for the analysis of the event in state A.

6.2.5.1. Choice of single failure and preventive maintenance

The most onerous single failure is the loss of one MHSI pump at the RIS [SIS] actuation. Thus one MHSI pump is assumed to be unavailable.

The preventive maintenance on one division, affecting one MHSI, one LHSI and of one ASG [EFWS] pumps is the most onerous assumption. Thus, in addition one MHSI pump, one LHSI pump and one ASG [EFWS] pump are unavailable.

6.2.5.2. Initial state

The reactor state B "Intermediate shutdown state", starting at the earliest 5 hours after RT, is defined as follows:

State B starts at the hot shutdown state where some F1A I&C signals are inhibited compared to power operation. It therefore starts at an RCP [RCS] pressure below 130 bar and RCP [RCS] temperature below 300°C. It extends to the cold shutdown state with LHSI/RHR connected at an RCP [RCS] pressure below 30 bar and RCP [RCS] temperature below 120°C (from Sub-chapter 14.0)]

Dependent on the availability of the accumulators, two sub-states named B1 and B2 are considered in the safety assessment of IB/LB-LOCA in state B:

- In State B1: all accumulators are available, i.e. not isolated.
    - RCP [RCS] pressure ranges between 70 bar [3] and 130 bar.
    - RCP [RCS] temperature ranges between 245°C and 303°C.
- In State B2: all accumulators are isolated.
    - RCP [RCS] pressure ranges between 30 bar and 70 bar.
    - RCP [RCS] temperature ranges between 120°C and 245°C.

For the RCP [RCS] water inventory and core cooling concerns, the higher the RCP [RCS] pressure and the RCP [RCS] temperature at the time of the RCP [RCS] break, the more severe the consequences:

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

Consequently, the following two limiting initial states have been considered in the safety assessment:

- State B1:
    - RCP [RCS] pressure: 132.5 bar, (including 2.5 bar uncertainty).

[3] During the normal RCP [RCS] cooldown to cold shutdown, accumulators are isolated at an RCP [RCS] pressure of 70 bar (above their operating pressure range of 45 to 50 bar), and vice-versa are de-isolated at 70 bar during the normal RCP [RCS] heat-up to hot shutdown. At an RCP [RCS] pressure of 70 bar, the maximum RCP [RCS] temperature is 245°C.

  - o RCP [RCS] temperature: 308°C, (including 5°C for uncertainty and dead band).

- State B2:
  - o RCP [RCS] pressure: 72.5 bar, (including 2.5 bar uncertainty).
  - o RCP [RCS] temperature: 250°C, (including 5°C for uncertainty and dead band).

For state B1 the analysis in state A is bounding as the same equipment and capacities are considered but a lower core power occurs in state B. Therefore code analysis is performed for state B2 only.

The core power for state B2 is conservatively assumed to be maintained at a constant 50 MW. This assumption bounds the decay power 5 hours after RT at the start of state B of 0.9% of the nominal power. [Ref-1]

6.2.5.3. Specific assumptions

a) Assumptions related to non-F1 systems

As in state A, the non-classified systems are not claimed when their operation would be beneficial.

b) Assumptions related to F1 systems

Safety Injection (F1A): SI signal is automatically actuated on a “RCP [RCS] subcooling margin ΔP_{sat} [4] “ signal. This has a setpoint in the range of 10 to 5 bar. To address the uncertainty on the setpoint, including the degraded containment conditions, the setpoint is assumed to be zero resulting in a maximum delay for the start up of the RIS [SIS] pumps, and the beginning of partial cooldown.

The conservative assumptions associated with the resulting actions, are listed below [Ref-1]:

- 20 seconds maximum delay for RIS [SIS] pumps start-up (incl. signal delay).
- Minimum characteristics for RIS [SIS] pumps
- 50°C for maximum initial IRWST temperature and injection flow temperature.

The RIS [SIS] train injecting to the broken loop is assumed to spill directly into the containment with no contribution to the RCP [RCS] injection.

Based on the single failure and preventive maintenance assumptions, the RIS [SIS] components performing the RIS [SIS] cold leg injection for the two limiting cases RIS [SIS] line and surge line break are the following:

RIS [SIS] component	RIS [SIS] line break	Surge line break
MHSI pumps	1	2
Accumulators	3	4
LHSI pumps	2	3

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

VDA [MSRT] (FA): VDA [MSRT] setpoints are values corresponding to saturated conditions in the RCP [RCS] plus 1.5 bar uncertainty including the pressure measurement uncertainty in a conservative manner.

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

The SI signal initiates the Partial Cooldown, as in state A. The automatic decrease of SG pressure at a rate of 100°C/h down to 60 bar (± 1.5 bar uncertainty) is however only effective in the high RCP [RCS] pressure range of state B1 where the initial SG pressure is higher than 60 bar, i.e. the RCP [RCS] temperature is higher than 275°C.

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

Three trains are available.

The conservative assumptions for the characteristics of the ASG [EFWS] pump and the ASG [EFWS] water temperature are similar to the ones described in state A – related analyses.

RCP [RCS] pumps trip (FA): RCP [RCS] pumps trip is automatically actuated on a reduced Reactor Coolant Pump pressure drop. The setpoint is 80 ± 5% of the Reactor Coolant Pump nominal pressure drop. This is the same as in state A.

6.2.6. Other assumptions

Pressuriser water level/volume corresponds to the minimum water level at zero power level, including the uncertainties in a conservative manner. This results in an initial value of 21.5 m^3, including an uncertainty of 5% MR.

6.2.7. Results

The safety assessment relies on a qualitative argument for reactor state B1. This state is bounded by state A. However, for state B2, without availability of the accumulators, explicit CATHARE (version V1.3L) code calculations are performed.

State B1: accumulators available (not isolated)

For any given break size, the consequences of an IB/LB-LOCA on the RCP [RCS] water inventory and core cooling in state B1 are less severe than those calculated in the CATHARE accident analyses of state A, since:

- All F1A mitigation measures from state A are available in state B1, MHSI, LHSI and accumulators for SI injection, VDA [MSRT] and ASG [EFWS] for heat removal.
- The SI signal occurs before RCP [RCS] saturation with the RCP [RCS] still full of subcooled water, except the pressuriser. The RIS [SIS] pumps are quickly operable in state B1 as no delay for diesel reloading sequence is required, as LOOP is not assumed.

- Initial SG pressure[5] is lower in state B1 than in state A, resulting in:
  - o Lower break flow rate due to a lower RCP [RCS] pressure once RCP [RCS] saturation conditions are reached.
  - o Earlier MHSI injection arising from a more efficient Partial Cooldown[6]. The lower initial SG pressure results in the MHSI delivery pressure, 85 bar/300°C, and completion of Partial Cooldown, 60 bar/275°C, occurring earlier.
  - o Higher SI injection flow rate, a consequence of the lower RCP [RCS] pressure.
  - o Core decay heat is lower in state B1, allowing a faster RCP [RCS] depressurisation.

Consequently, the lower RCP [RCS] break flow, and the earlier actuation and higher efficiency of the RIS [SIS] injection in state B1 are beneficial for the RCP [RCS] water inventory. Thus the inventory reduction is lower compared to state A.

The CATHARE analyses results of the PCC-4 IB/LB-LOCA in state A are thus bounding for the PCC-4 IB/LB-LOCA in state B. Therefore, the controlled state is reached, with all safety and decoupling criteria met, utilising only F1A systems.

The consequences of IB/LB-LOCA transient in state B1 up to the controlled state are bounded by the consequences following the event occurring in state A.

State B2: accumulators unavailable (isolated)

The transient was not re-calculated for EPR 4500 MWth. The ability to reach the final state and satisfy the acceptance criteria are deduced from the results obtained for EPR 4250 MWth.

The initial conditions for the analyses of the RIS [SIS] and surge line break, given in Section 14.5.6 - Table 8, are conservatively chosen, especially the power level as discussed in sub-section 6.2.5.2 of this sub-chapter.

<u>RIS [SIS] line break</u>

The sequence of events is given in Section 14.5.6 - Table 9.

The most representative parameters are presented in the following figures:

- Section 14.5.6 - Figure 76: Reactor and SG power
  RCP [RCS] and SG secondary side pressures
- Section 14.5.6 - Figure 77: Break mass flow rate (steam, liquid and total)
  Break flow velocities (steam and liquid)
- Section 14.5.6 - Figure 78: Swell levels in RPV
  Core void fraction

[5] Maximum SG pressure 90 + 1.5 bar in state B1, instead of 95.5 + 1.5 bar in state A.
[6] With SG initial pressure of 90 bar in state B1 (saturation 303°C), instead of 95.5 bar in state A (saturation 307.5°C), the MHSI injection into RCP [RCS] (RCP [RCS] saturation 85 bar/300°C) can be effective a few minutes earlier in state B1 compared to state A (partial cooldown cooling rate being 100°C/h).

- Section 14.5.6 - Figure 79: Coolant liquid and steam temperatures in RPV and core
  Fuel Clad temperatures

- Section 14.5.6 - Figure 80: Integrated flow rates of RIS [SIS]
  Coolant inventory on primary and secondary side

- Section 14.5.6 - Figure 81: Totals of SI and break flow rates
  Integrated SI and break flow rates

The detailed description of the parameters is given in Section 14.5.6 - Table 11.

The basic automatic countermeasure consists of the response to the SI signal ΔP_{sat} which initiates the MHSI and LHSI pumps. The MHSI and LHSI injection provides an early compensation of the break flow. The lack of any accumulator injection is not significant for the event mitigation. About 400 seconds after event initiation the LHSI delivery becomes effective. The RCP [RCS] inventory starts increasing from this point however before the LHSI begins injecting, the inventory loss is almost halted by the MHSI alone.

The core remains covered by at least a two-phase mixture throughout the event. Thus, there is no core heat-up and the acceptance criterion of 1200°C in the hot rod is clearly fulfilled.

After about 700 seconds, the main RCP [RCS] and SG parameters have stabilised.

<u>Surge line break</u>

The sequence of events is given in Section 14.5.6 - Table 10.

The most representative parameters are presented in the following figures:

- Section 14.5.6 - Figure 82: Reactor and SG power
  RCP [RCS] and SG secondary side pressures

- Section 14.5.6 - Figure 83: Break mass flow (steam, liquid and total)
  Break flow velocities (steam and liquid)

- Section 14.5.6 - Figure 84: Swell levels in RPV
  Core void fraction

- Section 14.5.6 - Figure 85: Coolant liquid and steam temperatures in RPV and core
  Fuel Clad temperatures

- Section 14.5.6 - Figure 86: Integrated flow rates of RIS [SIS]
  Coolant inventory on primary and secondary side

- Section 14.5.6 - Figure 87: Totals of SI and break flow rates
  Integrated SI and break flow rates

The detailed description of the parameters is given in Section 14.5.6 - Table 11.

The basic automatic countermeasure consists of the response to the SI signal ΔP_{sat} which initiates the MHSI and LHSI pumps. This results in early matching of the break flow. The lack of any accumulator injection is not significant for mitigation of the event. About 250 seconds after initiation of the event, shortly after the LHSI delivery becomes effective, the RCP [RCS] inventory begins increasing. However, before the LHSI starts the inventory loss is almost halted by the MHSI alone.

The core remains covered by at least two-phase mixture throughout the event. Therefore, there is no core heat-up and the decoupling criterion of 1200°C in the hot rod is clearly fulfilled.

After about 700 seconds, the main RCP [RCS] and SG data are stabilised.

EPR at 4500 MWth

The calculations have not been repeated for EPR 4500 MWth because at the new power the results are not significantly different for the following reasons:

The analysis for EPR 4250 MWth was performed for a power of 50 MW. This power bounds the power of the EPR 4500 MWth, 5 hours after RT, of 0.9% of the nominal power. [Ref-1]

The analysis for EPR 4250 MWth has sufficient conservative assumptions to bound operation at 4500 MWth

The initial conditions for EPR 4500 MWth are relatively unchanged compared to EPR 4250 MWth. The decrease in the pressuriser level of 3% is not sufficient to change the conclusions of the analysis.

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

The plant status, power level and RCP [RCS] water inventory, at the controlled state is less onerous in state B than in state A.

All F1A and F1B mitigation systems used in state A for the transfer of the plant from the controlled state to the safe shutdown state are also available in state B. However, in this state, the switch-over of LHSI to hot leg injection is not needed.

It can be concluded that, despite the most onerous single failure and consideration of the most onerous preventive maintenance, the safe shutdown is reached with all safety and decoupling criteria met, utilising only F1A and F1B means.

6.2.9. Conclusion

When compared to IB/LB-LOCA in state A, a IB/LB-LOCA in state B has the following main differences:

- Change of SI signal to "ΔP_{sat} low" in state B, instead of "low pressuriser pressure" in state A.
- More favourable plant initial conditions
- Accumulators isolated below RCP [RCS] pressure of 70 bar in state B2.

The conclusions reached for the PCC-4 IB/LB-LOCA in reactor state A apply to the PCC-4 IB/LB-LOCA in reactor state B. They are listed below.

Despite of the worst single failure and assuming the most onerous preventive maintenance:

- All the safety criteria are met:
    - The maximum fuel clad temperature remains below the acceptance criterion of 1200°C.
    - The maximum percentage of total fuel clad thickness oxidised at the hot spot is less than the limit of 17%.
    - There is no clad rupture.
    - Integrity of the core geometry is maintained.
    - The long-term core cooling is assured.
- The controlled state is reached, relying only on F1A classified systems:
    - RIS [SIS], MHSI, accumulators if available, LHSI if needed.
- The safe shutdown state is reached, relying only on F1A and F1B classified systems:
    - The VDA [MSRT] and ASG [EFWS] pump & tank capacities for core heat removal.
    - The LHSI heat exchange capacity for IRWST heat removal.
    - The MHSI, accumulators if available, and LHSI if needed, injection capacities from the IRWST for RCP [RCS] water inventory
    - The MHSI boron injection capacity for core sub-criticality.

6.3. SYSTEM SIZING

RIS [SIS] Requirements

- IB-LOCA/LB- LOCA 80 cm^2 (less than 100 mm equivalent diameter) in power state A: cladding temperature less than 800/900°C defines the MHSI flow rate at approximately 50 bar.
- LOCA in power state A: P&T requirement with Containment pressure less than Design Pressure and IRWST temperature less than MHSI pump design pressure defines the LHSI flow rate for hot leg injection.
- LOCA in shutdown state: no core uncovery accounting for one MHSI pump defines the MHSI flow rate with large mini-flow line open at 10 bar.

SECTION 14.5.6 - TABLE 1

SUMMARY OF INITIAL CONDITIONS

PARAMETER		VALUE		VALUE USED
		Nominal	**Uncertainty**	
PRIMARY SYSTEM				
Core Power	Maximum	4500 MWth	+ 2%	**4590 MWth**
Vessel average temperature	Maximum	312.8°C	+ 2.5°C	**315.3°C**
Pressuriser pressure	Maximum	155 bara	+ 2.5 bar	**157.5 bara**
Pressuriser level	Maximum	56%	+ 5%	**61%**
Primary loop flow rate	Minimum	Thermal hydraulic		**27,185 m^3/h**
SECONDARY SYSTEM				
SG tube plugging		0%		
SG level	Nominal	49% NR	-	**49% NR**
ARE [MFW] temperature	Nominal	230°C	-	**230°C**

SECTION 14.5.6 - TABLE 2

REACTOR PROTECTION

REACTOR PROTECTION	SETPOINT	NOMINAL VALUE	UNCERTAINTY	VALUE USED
Reactor trip	Low pressuriser pressure [MIN2] (bara)	135	-3	132
Turbine Trip	On RT signal	-	-	-
Reactor Coolant Pumps trip	On RT signal	-	-	-
ARE [MFW] pumps loss	On RT signal	-	-	-
SI signal	Very low pressuriser pressure [MIN3] (bara)	115	-3	112
	Delay for full flow rate with LOOP(s)	40	-	40
ASG [EFWS] actuation	On SI signal + LOOP	-	-	-
	On low SG level [MIN2]	40	-5	35
	Delay for full flow rate with LOOP (s)	50	-	50
ASG [EFWS] isolation	On high SG level [MAX1]	89	-5	84
VDA [MSRT] opening	On high SG pressure [MAX1]	95.5	+1.5	97
	Delay (s) Dead time and opening time	1.5 0.5		2.0
Partial cooldown	On SI signal	-	-	-

SECTION 14.5.6 - TABLE 3

AXIAL POWER SHAPES CHARACTERISTICS

AXIAL POWER SHAPE	FQ (peaking factor)	Z_{FQ} (FQ elevation)	FΔH (enthalpy rise factor)	AO (axial offset)
Average rod	1.57	2.8 m	1.00	18%
Hot assembly average rod	2.61	3.5 m	1.67	21%
Hot rod of the hot assembly	2.82	3.5 m	1.80	21%

SECTION 14.5.6 - TABLE 4

FIRST BREAK SPECTRUM: SEQUENCE OF EVENTS 1/2

Break size	45 cm²	80 cm²	100 cm²	120 cm²	140 cm²
Break opening	0.0 s	0.0 s	0.0 s	0.0 s	0.0 s
Reactor trip signal	44.3 s	24.9 s	20.2 s	17.4 s	15.4 s
Turbine trip, ARE [MFWS] isolation and loss of Reactor Coolant Pumps	44.3 s	24.9 s	20.2 s	17.4 s	15.4 s
Beginning of the rod drop	44.6 s	25.2 s	20.5 s	17.7 s	15.7 s
Opening of the VDA [MSRT]	53.3 s	34.2 s	29.5 s	26.7 s	24.6 s
Pressuriser heaters trip	59.5 s	39.4 s	34.3 s	31.0 s	28.7 s
SI signal	82.9 s	47.3 s	39.2 s	35.4 s	32.6 s
Beginning of Partial Cooldown at 250°C/h	82.9 s	47.3 s	39.2 s	35.4 s	32.6 s
EFWS [ASG] actuation on LOOP+ SI signal	132.9 s	97.3 s	89.2 s	85.4 s	82.6 s
Start of MHSI injection	268.9 s	234.6 s	227.7 s	225.4 s	223.0 s
End of partial cooldown	536.4 s	500.9 s	492.8 s	488.9 s	486.1 s
Secondary side no more needed (RCP [RCS] pressure < SG pressure)	1192.9 s	960.0 s	589.8 s	519.7 s	497.2 s
Beginning of core uncovery	-	-	-	879.4 s	588.4 s
Beginning of accumulators injection	-	1430.5 s	1141.6 s	928.9 s	768.4 s
PCT occurs at	-	-	-	1058.0 s	893.0 s
End of core uncovery	-	-	-	1495.1 s	1491.4 s
EFWS [ASG] isolation	2133.5 s	1825.2 s	1716.5 s	1610.0 s	1507.5 s
Start of LHSI injection	-	-	-	-	2575.2 s
End of calculation	3000.0 s	3000.0 s	3000.0 s	3000.0 s	3000.0 s

SECTION 14.5.6 - TABLE 4

FIRST BREAK SPECTRUM: SEQUENCE OF EVENTS 2/2

Break size	160 cm²	180 cm²	200 cm²	390 cm²	2x830 cm²
Break opening	0.0 s	0.0 s	0.0 s	0.0 s	0.0 s
Reactor trip signal	13.5 s	12.5 s	11.8 s	9.3 s	5.0 s
Turbine trip, ARE [MFWS] isolation and loss of Reactor Coolant Pumps	13.5 s	12.5 s	11.8 s	9.3 s	5.0 s
Beginning of the rod drop	13.8 s	12.8 s	12.1 s	9.6 s	5.3 s
Opening of the VDA [MSRT]	22.5 s	21.3 s	20.9 s	17.9 s	-
Pressuriser heaters trip	26.7 s	25.4 s	24.4 s	19.2 s	7.6 s
SI signal	30.6 s	28.8 s	28.0 s	21.8 s	8.5 s
Beginning of Partial Cooldown at 250°C/h	30.6 s	28.8 s	28.0 s	21.8 s	10.5 s
EFWS [ASG] actuation on LOOP+ SI signal	80.6 s	78.8 s	78.0 s	71.8 s	58.5 s
Start of MHSI injection	222.7 s	216.8 s	210.6 s	150.9 s	74.5 s
End of partial cooldown	484.2 s	482.4 s	481.6 s	475.4 s	464.1 s
Secondary side no more needed (RCP [RCS] pressure < SG pressure)	408.2 s	357.5 s	315.2 s	141.7 s	10.9 s
Beginning of core uncovery	578.2 s	504.6 s	453.7 s	220.5 s	107.9 s
Beginning of accumulators injection	673.8 s	589.1 s	525.1 s	260.4 s	120.2 s
PCT occurs at	829.0 s	686.0 s	618.0 s	303.0 s	132.0 s
End of core uncovery	1200.3 s	917.9 s	774.8 s	317.1 s	138.3 s
EFWS [ASG] isolation	1438.1 s	1388.5 s	1346.2 s	964.2 s	928.2 s
Start of LHSI injection	1972.0 s	1279.2 s	1062.2 s	369.1 s	139.9 s
End of calculation	3000.0 s	3000.0 s	3000.0 s	3000.0 s	3000.0 s

SECTION 14.5.6 - TABLE 5

FIRST BREAK SPECTRUM RESULTS

Break size	45 cm²	80 cm²	100 cm²	120 cm²	140 cm²
Minimum primary water mass	86 t	70 t	65 t	59 t	55 t
Minimum core level	No core uncovery	4.09 m	3.83 m	3.73 m	2.86 m
Peak Clad Temperature (Hot rod)	-	406°C	406°C	430°C	692°C
Location	-	3.4 m	3.4 m	3.9 m	3.7 m
Hot spot clad oxidation rate	-	negligible	negligible	negligible	negligible
Percentage of clad rupture	0%	0%	0%	0%	0%

Break size	160 cm²	180 cm²	200 cm²	390 cm²	2x830 cm²
Minimum primary water mass	52 t	49 t	48 t	43 t	45 t
Minimum core level	2.9 m	2.8 m	2.8 m	0.9 m	1.2 m
Peak Clad Temperature (Hot rod)	662°C	632°C	637°C	623°C	530°C
Location	2.9 m	2.9 m	2.8 m	1.3 m	1.2 m
Hot spot clad oxidation rate	negligible	negligible	negligible	negligible	negligible
Percentage of clad rupture	0%	0%	0%	0%	0%

SECTION 14.5.6 - TABLE 6

SECOND BREAK SPECTRUM: SEQUENCE OF EVENTS 1/2

Break size	120 cm²	140 cm²	160 cm²	180 cm²	200 cm²
Break opening	0	0	0	0	0
Reactor trip signal	14	13	11	11	10
Turbine trip, ARE [MFWS] isolation and loss of Reactor Coolant Pumps	15	14	12	12	11
Beginning of the rod drop	15	14	12	12	11
Opening of the VDA [MSRT]	26	24	22	21	21
Pressuriser heaters trip	29	27	26	24	24
SI signal	32	30	28	27	27
Beginning of Partial Cooldown at 250°C/h	33	31	29	27	27
ASG actuation on LOOP+ SI signal	83	81	79	77	77
Start of MHSI injection	223	222	220	216	210
End of partial cooldown	486	484	482	481	480
Secondary side no more needed (RCP [RCS] pressure < SG pressure)	518	496	407	357	314
Beginning of core uncovery	877	591	581	509	453
Beginning of accumulators injection	929	770	674	590	526
PCT occurs at	1313	983	891	795	736
End of core uncovery	1859	1732	1616	1392	1322
EFWS [ASG] isolation	1591	1486	1424	1392	1345
Start of LHSI injection	*	2539	1985	1547	1326
End of calculation	3000	3000	3000	3000	3000

SECTION 14.5.6 - TABLE 6

SECOND BREAK SPECTRUM: SEQUENCE OF EVENTS 2/2

Break size	220 cm²	240 cm²	390 cm²
Break opening	0	0	0
Reactor trip signal	10	9	8
Turbine trip, ARE [MFWS] isolation and loss of Reactor Coolant Pumps	11	10	9
Beginning of the rod drop	11	10	9
Opening of the VDA [MSRT]	21	20	18
Pressuriser heaters trip	23	22	19
SI signal	26	25	21
Beginning of Partial Cooldown at 250°C/h	26	25	21
EFWS [ASG] actuation on LOOP+ SI signal	76	75	71
Start of MHSI injection	208	203	151
End of partial cooldown	480	479	475
Secondary side no more needed (RCP [RCS] pressure < SG pressure)	265	241	145
Beginning of core uncovery	433	395	219
Beginning of accumulators injection	471	431	260
PCT occurs at	758	738	311
End of core uncovery	1130	976	417
EFWS [ASG] isolation	1292	1267	1047
Start of LHSI injection	1093	920	417
End of calculation	3000	3000	3000

SECTION 14.5.6 - TABLE 7

SECOND BREAK SPECTRUM: RESULTS

Break size	120 cm²	140 cm²	160 cm²	180 cm²
Minimum primary water mass	58 t	54 t	51 t	48 t
Minimum core level	3.4 m	2.9 m	2.9 m	2.9 m
Peak Clad Temperature (Hot rod)	562°C	763°C	730°C	736°C
Location	3.4 m	2.9 m	2.9 m	2.9 m
Hot spot clad oxidation rate	negligible	negligible	negligible	negligible
Percentage of clad rupture	0%	0%	0%	0%

Break size	200 cm²	220 cm²	240 cm²	390 cm²
Minimum primary water mass	48 t	50 t	50 t	43 t
Minimum core level	2.8 m	3.1 m	3.1 m	2.4 m
Peak Clad Temperature (Hot rod)	732°C	671°C	656°C	689°C
Location	2.8 m	3.1 m	1.8 m	1.4 m
Hot spot clad oxidation rate	negligible	negligible	negligible	negligible
Percentage of clad rupture	0%	0%	0%	0%

SECTION 14.5.6 - TABLE 8

INITIAL CONDITIONS, IB- AND LB-LOCA IN STATE B2 (PCC-4)

Parameters	Values
Reactor coolant system	
Initial reactor power (% of nominal power)	1.11% or 50 MW
Initial average RCP [RCS] temperature (°C)	245 + 5 = 250
Initial pressuriser pressure (bar)	70 + 2.5 = 72.5
RCP [RCS] loop flow rate (m^3/h)	27180
Pressuriser water volume (m^3)	19.6 [(1)]
Steam generators	
Initial steam pressure (bar)	40 (according to 250°C)
Initial water temperature (°C)	130
Initial water level (m)	15.69

[(1)] 24.5 m^3 in the calculation, no impact on reaching the controlled state and acceptance safety criteria, RT a little bit earlier.

SECTION 14.5.6 - TABLE 9 [Ref-1]

SEQUENCE OF EVENTS RELATED TO CONTROLLED STATE (GLOBAL TIME VALUES),

RIS [SIS] LINE BREAK (390 CM^2 - 225 MM EQUIVALENT DIAMETER) IN STATE B2 (PCC-4)

TIME (S)	EVENT
0	Break opening
0	Immediate Reactor Coolant Pump trip from "ΔP over RCP [RCS]< 80%" because of rapid depressurisation of the cold leg
30	RCP [RCS] pressure < 45 bar, but no accumulator injection (isolated)
60	Safety Injection signal on ΔP_{sat}=0
80	Start of safety injection by MHSI RCP [RCS] pressure at secondary side level of 40 bar
220	RCP [RCS] pressure drops below secondary pressure
380	Start of safety injection by LHSI (RCP [RCS] pressure < 20 bar)
400	Minimum RCP [RCS] inventory, minimum level around mid-loop Thereafter continuous refilling of RCP [RCS].
700	New stable conditions reached with RCP [RCS] pressure somewhat < 10 bar. End of calculation

SECTION 14.5.6 - TABLE 10 [Ref-1]

SEQUENCE OF EVENTS RELATED TO CONTROLLED STATE (GLOBAL TIME VALUES),

SURGE LINE BREAK (2 X 830 CM^2 - 2 X 325 MM EQUIVALENT DIAMETER) IN STATE B2

(PCC-4)

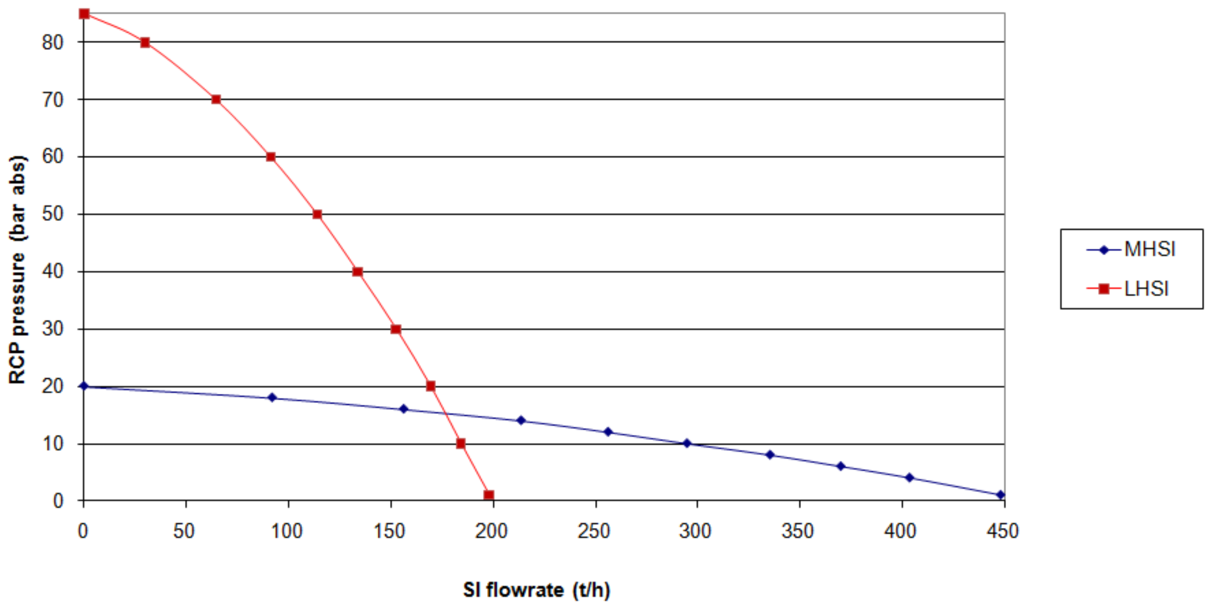
TIME (S)	EVENT
0	Break opening, prompt drop of RCP [RCS] pressure to about 45 bar, but no accumulator injection (isolated)
0	Immediate Reactor Coolant Pump trip from "ΔP over RCP[RCS] < 80%" because of rapid depressurisation of the cold leg
30	Safety Injection signal on ΔP_{sat}=0
50	Start of safety injection by MHSI
70	RCP [RCS] pressure drops below secondary side pressure
230	Start of safety injection by LHSI (RCP [RCS] pressure < 20 bar)
250	Minimum RCP [RCS] inventory, min. level around lower edge of loop. Thereafter continuous refilling of RCP [RCS]
700	New stable conditions reached with RCP [RCS] pressure around 5 bar. End of calculation

SECTION 14.5.6 - TABLE 11

LIST OF FIGURES: 390 CM^2 RIS [SIS] LINE (FIGURE 76 TO 81) AND 2 X 830 CM^2 SURGE LINE BREAK (FIGURE 82 TO 87) IN STATE B2

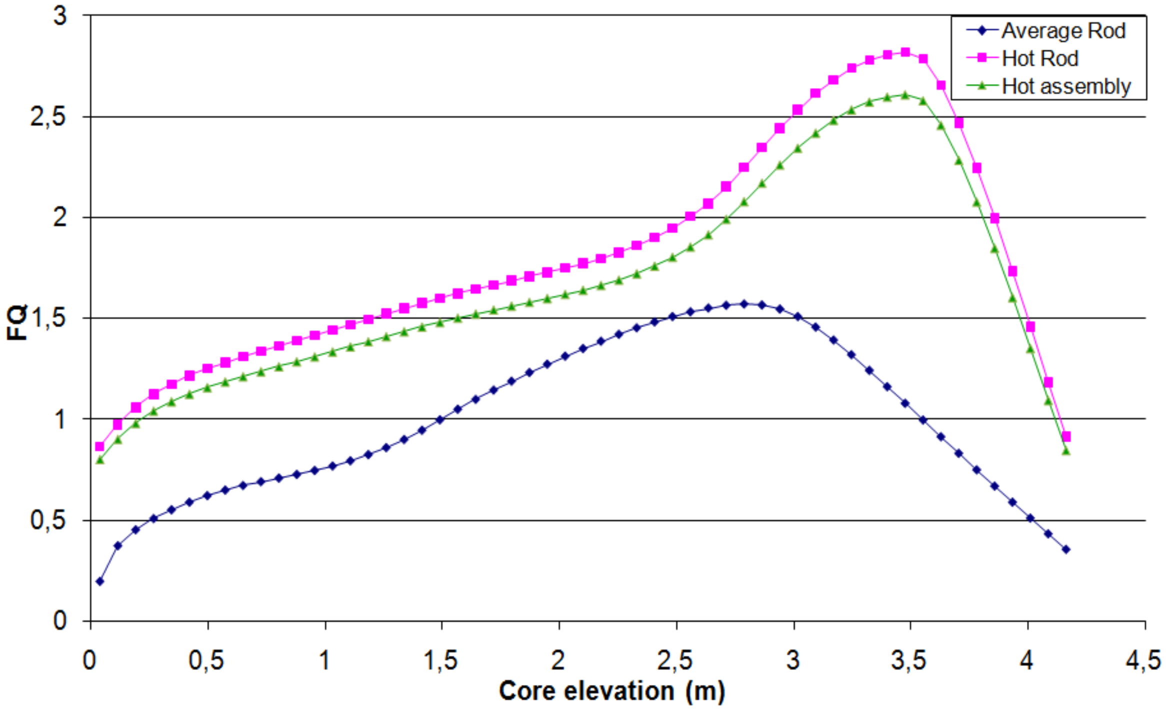
SECTION 14.5.6 - FIGURE 1

SAFETY INJECTION FLOWRATE



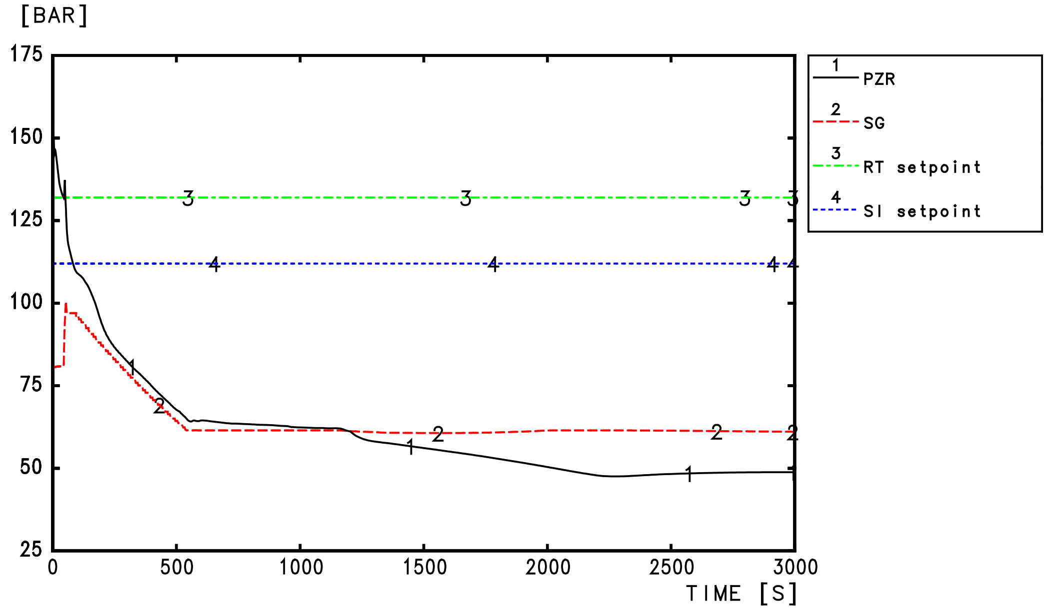
SECTION 14.5.6 - FIGURE 2

AXIAL POWER SHAPE CURVES

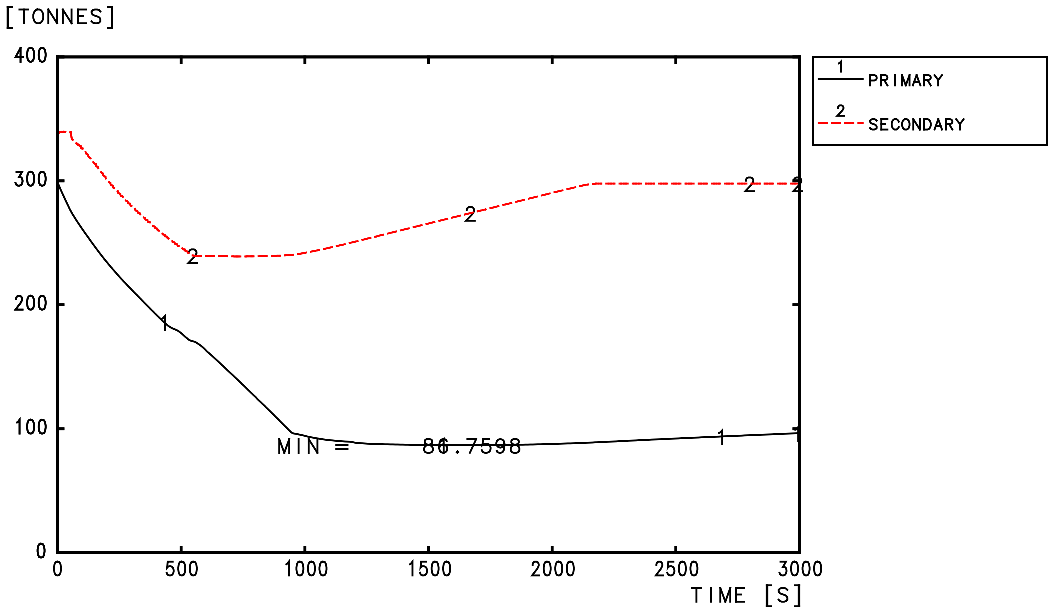


SECTION 14.5.6 - FIGURE 3

FIRST BREAK SPECTRUM: 45 CM² BREAK
PRIMARY AND SECONDARY PRESSURES
PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



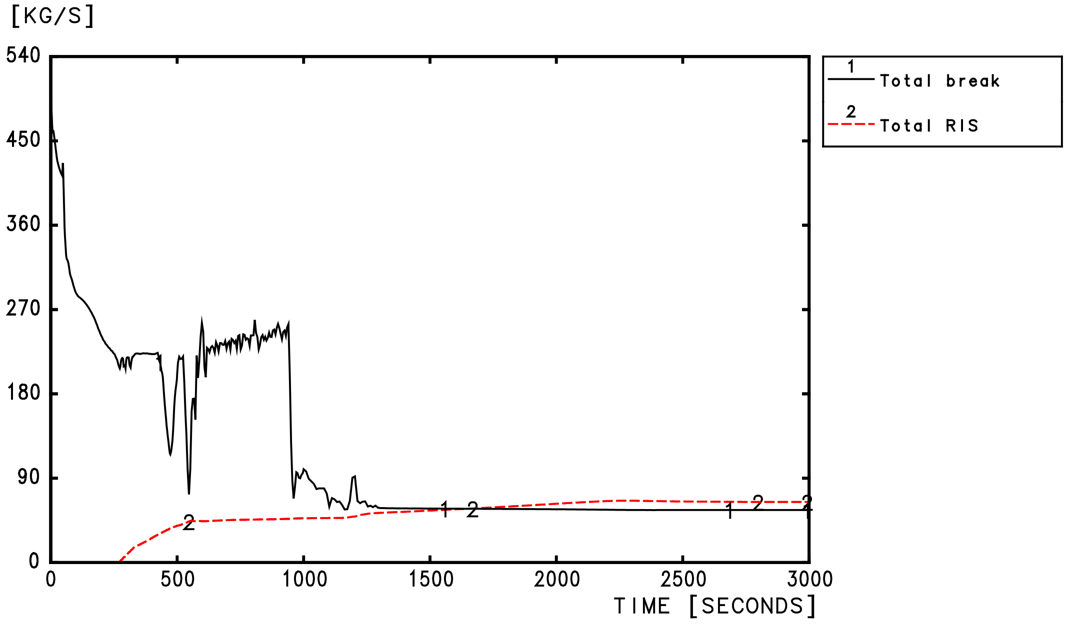
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 4

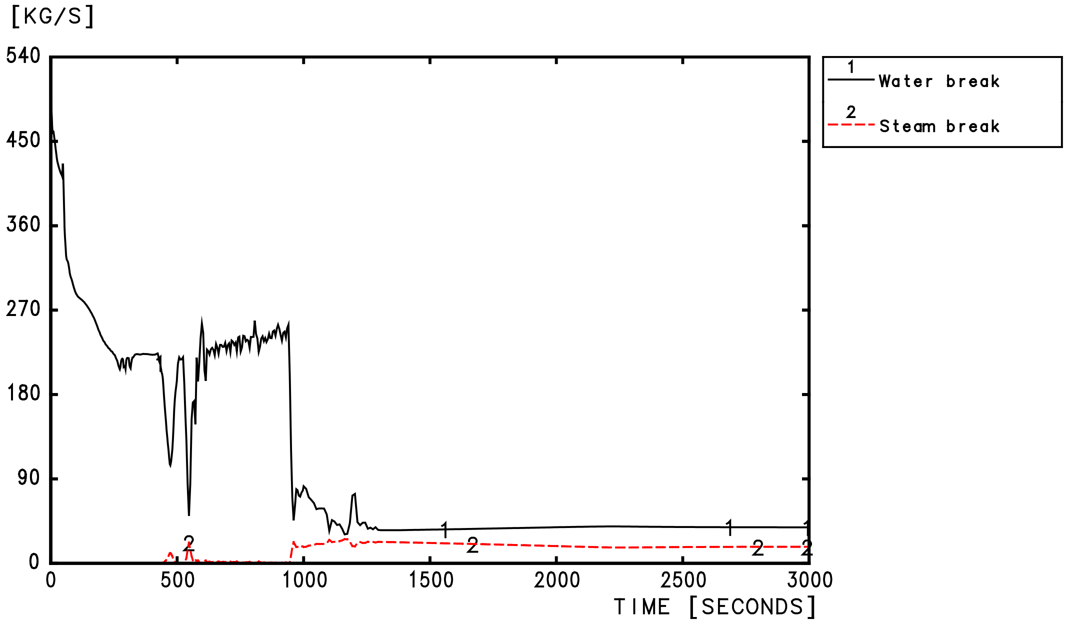
FIRST BREAK SPECTRUM: 45 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



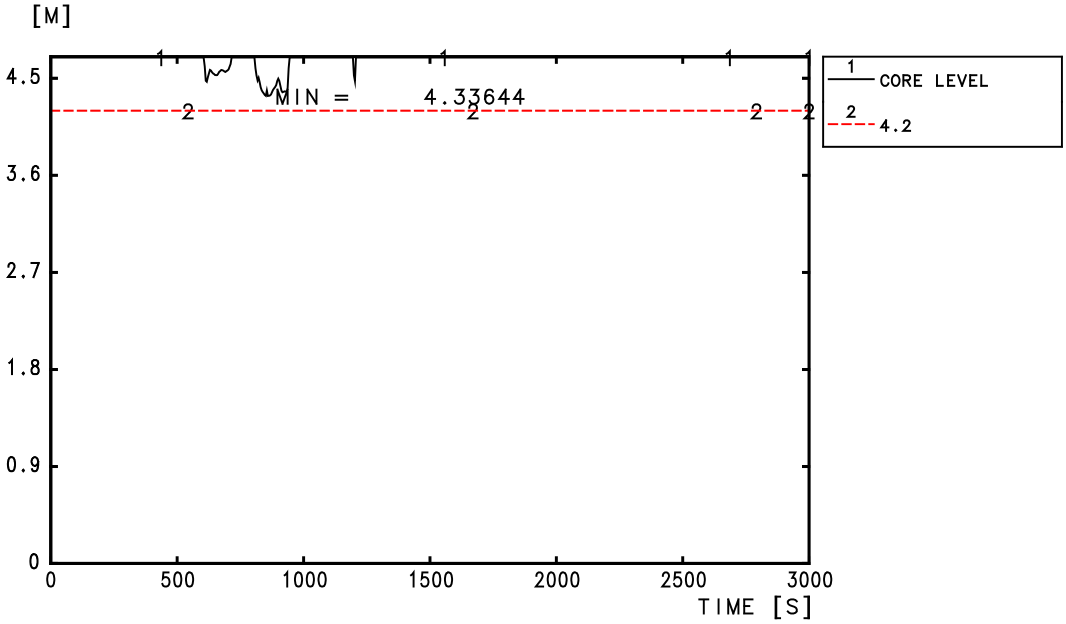
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 5

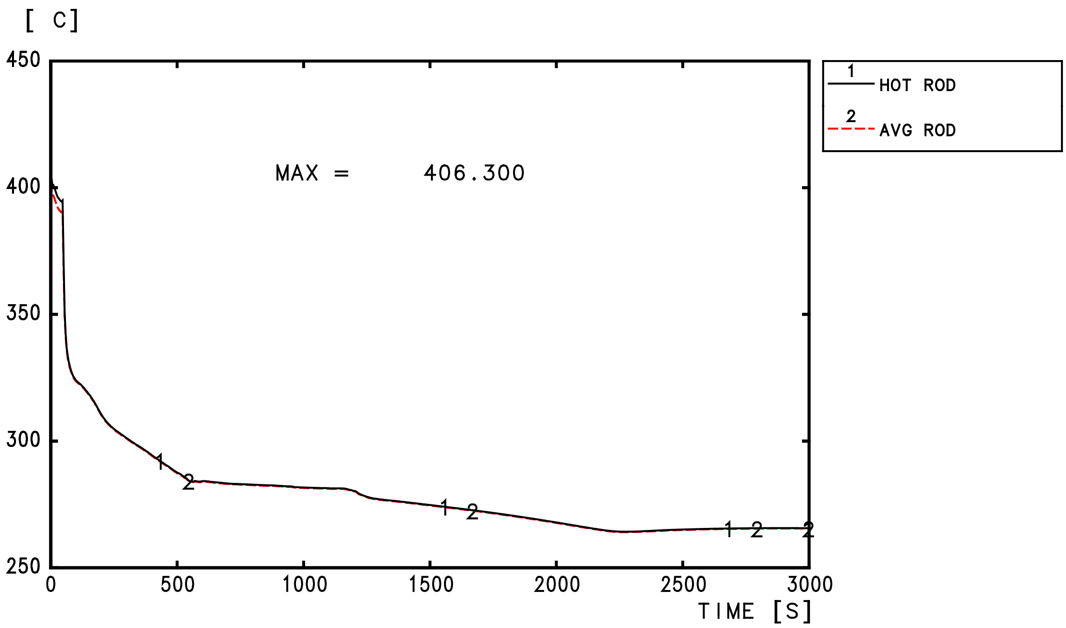
FIRST BREAK SPECTRUM: 45 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

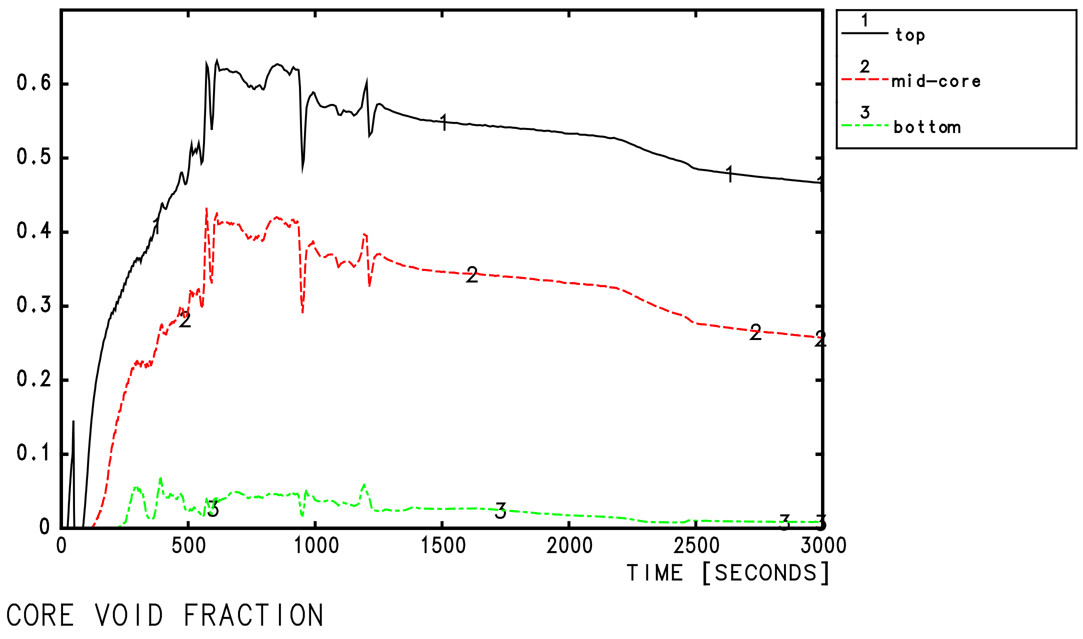


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 6

FIRST BREAK SPECTRUM: 45 CM² BREAK

CORE VOID FRACTION

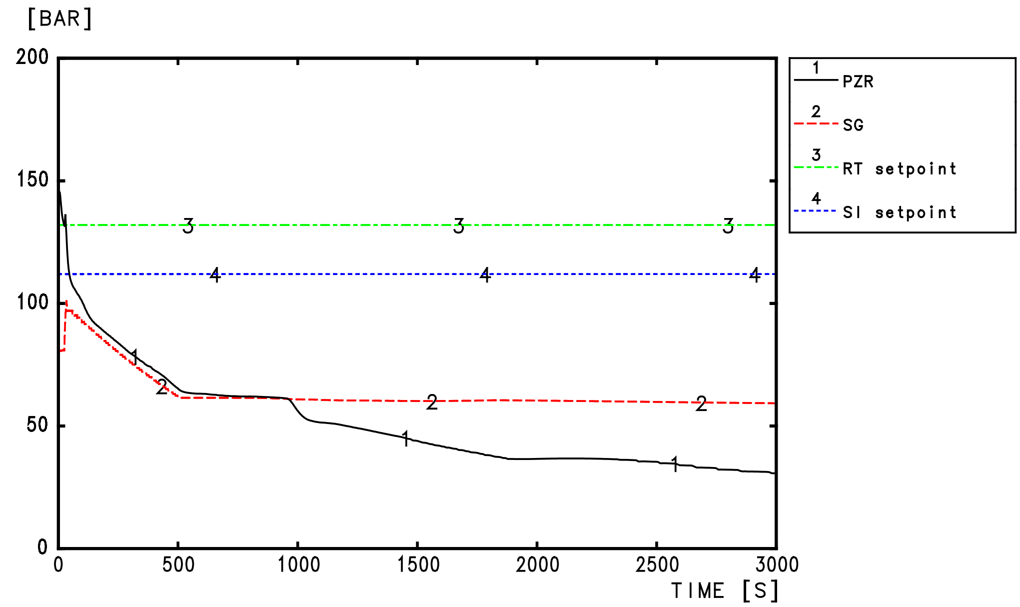


SECTION 14.5.6 - FIGURE 7

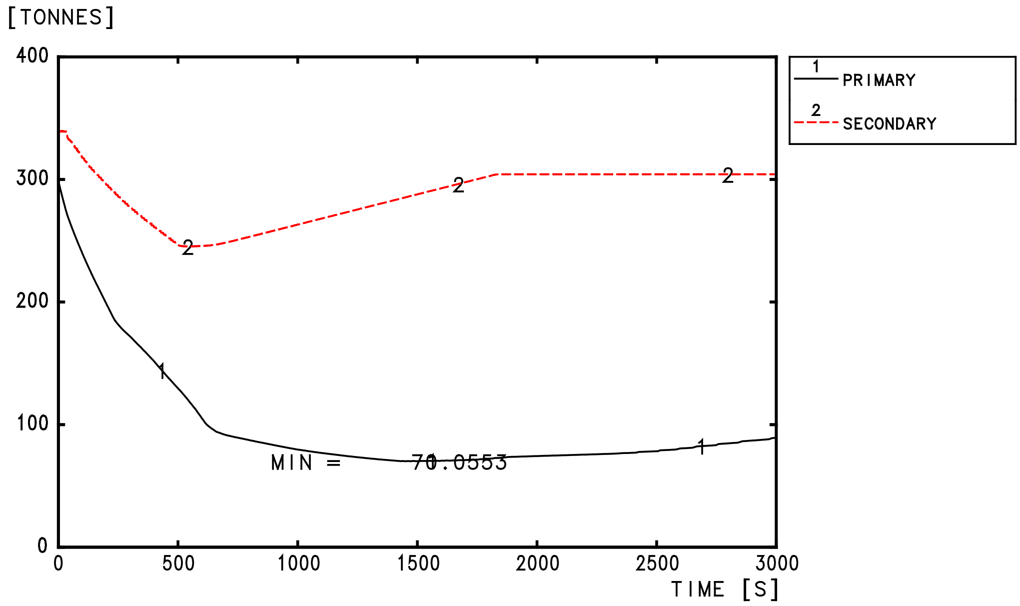
FIRST BREAK SPECTRUM: 80 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



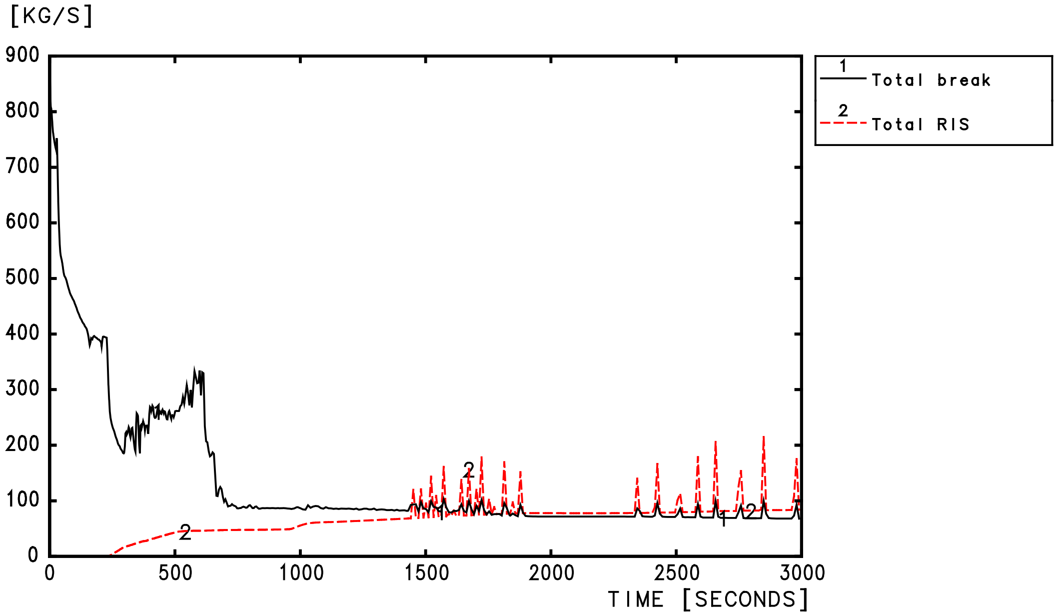
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 8

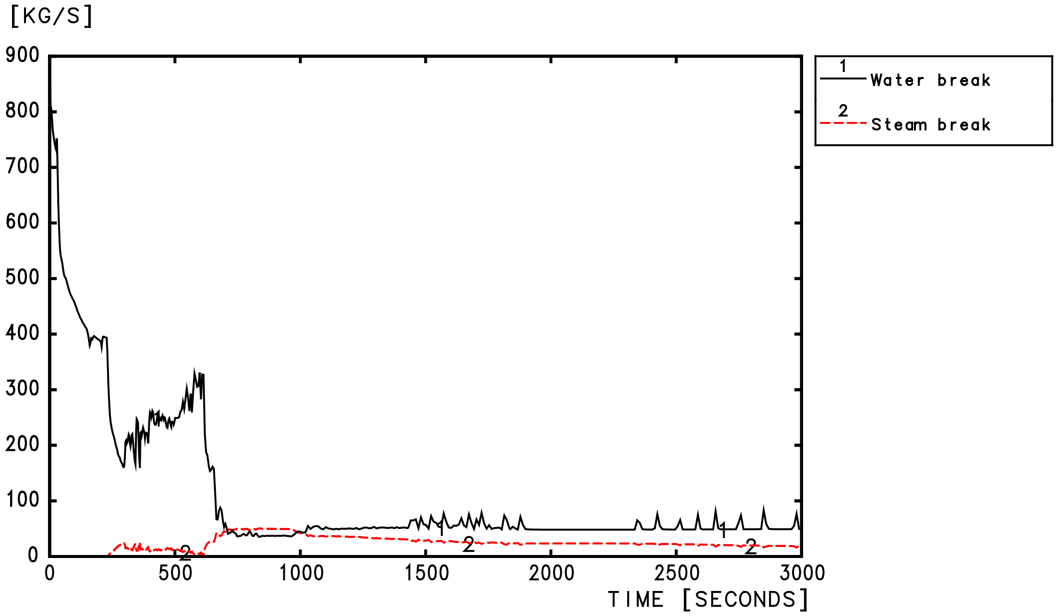
FIRST BREAK SPECTRUM: 80 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATE

WATER AND STEAM BREAK FLOW RATES



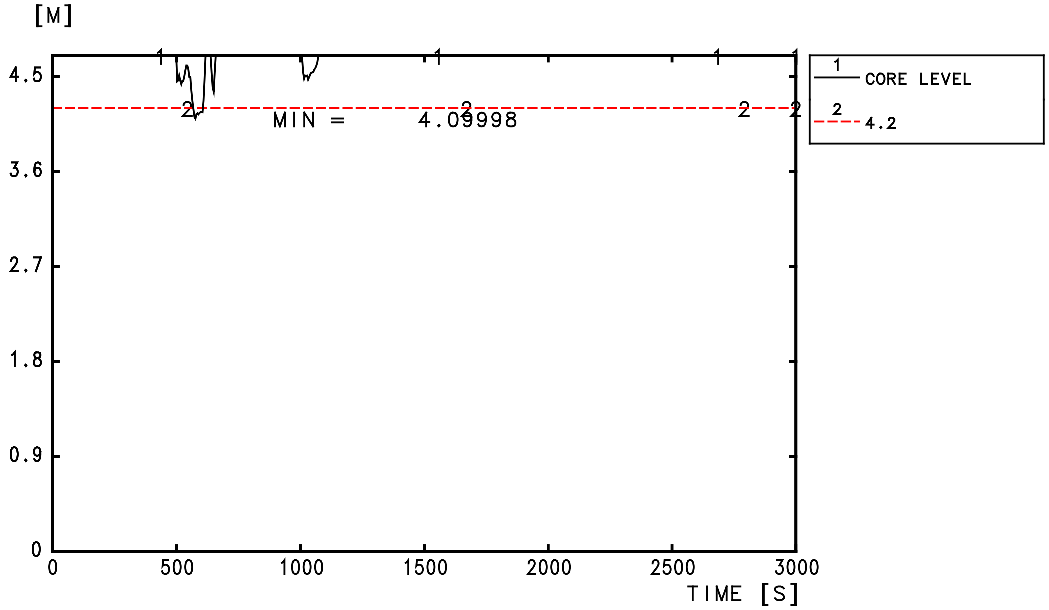
Break flowrate and total RIS injection rate



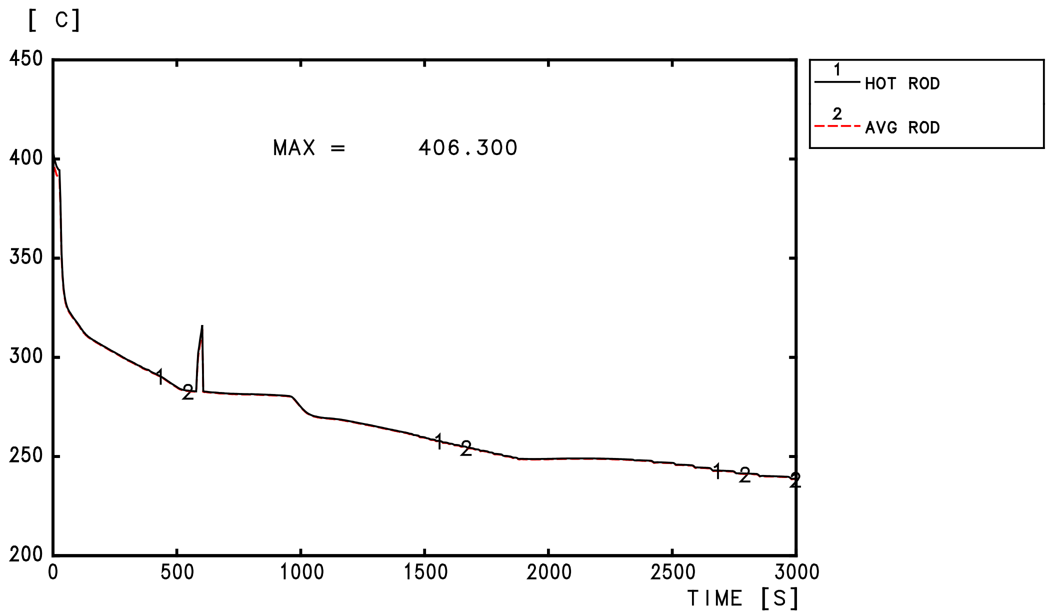
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 9

FIRST BREAK SPECTRUM: 80 CM² BREAK
CORE LEVEL
MAXIMUM CLADDING TEMPERATURE



Core level

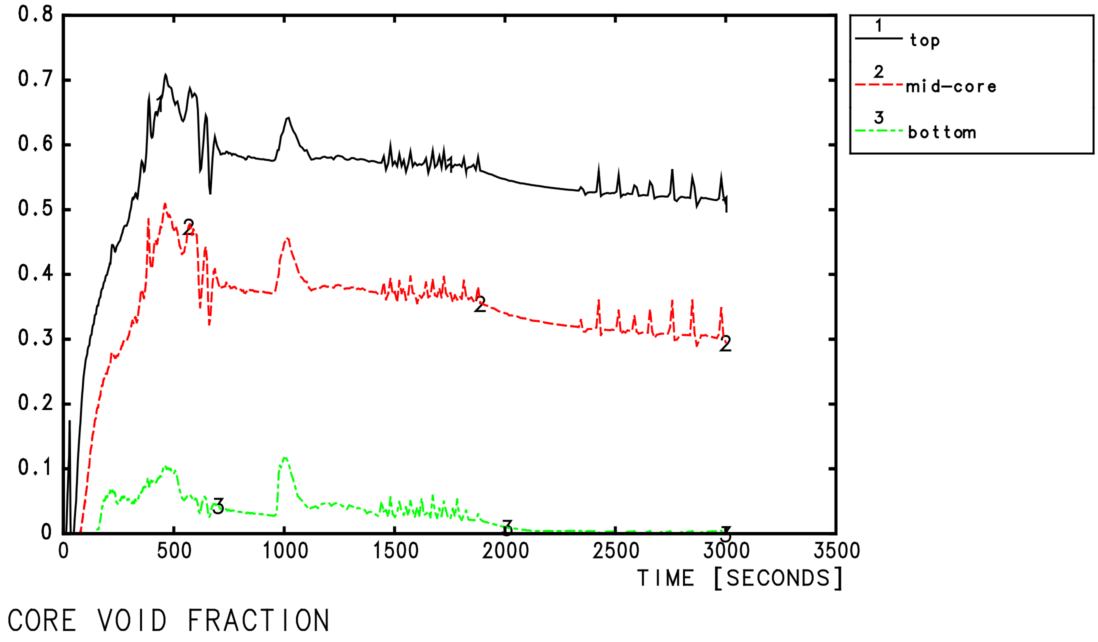


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 10

FIRST BREAK SPECTRUM: 80 CM² BREAK

CORE VOID FRACTION

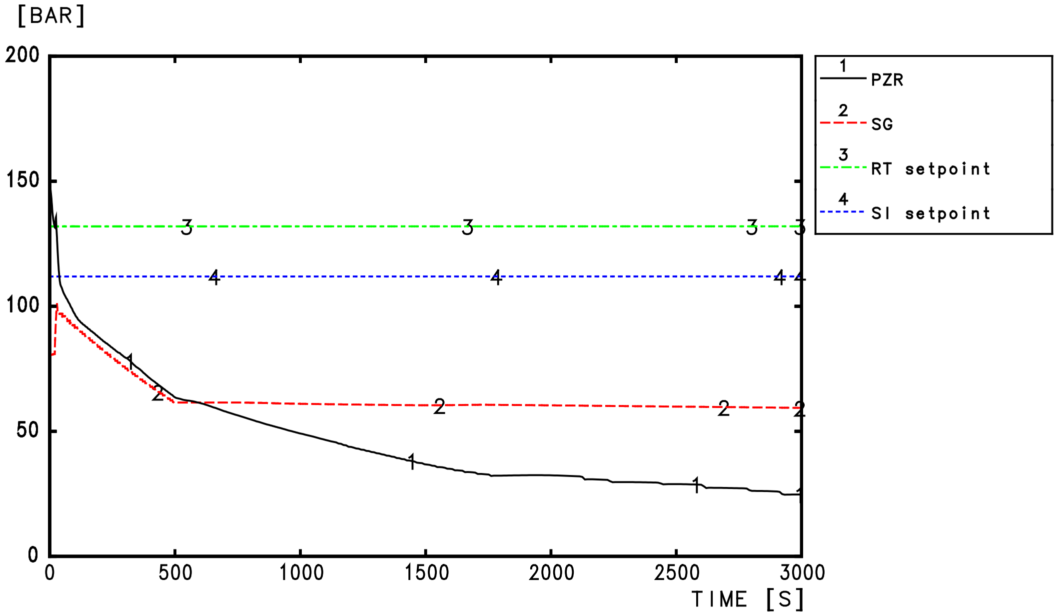


SECTION 14.5.6 - FIGURE 11

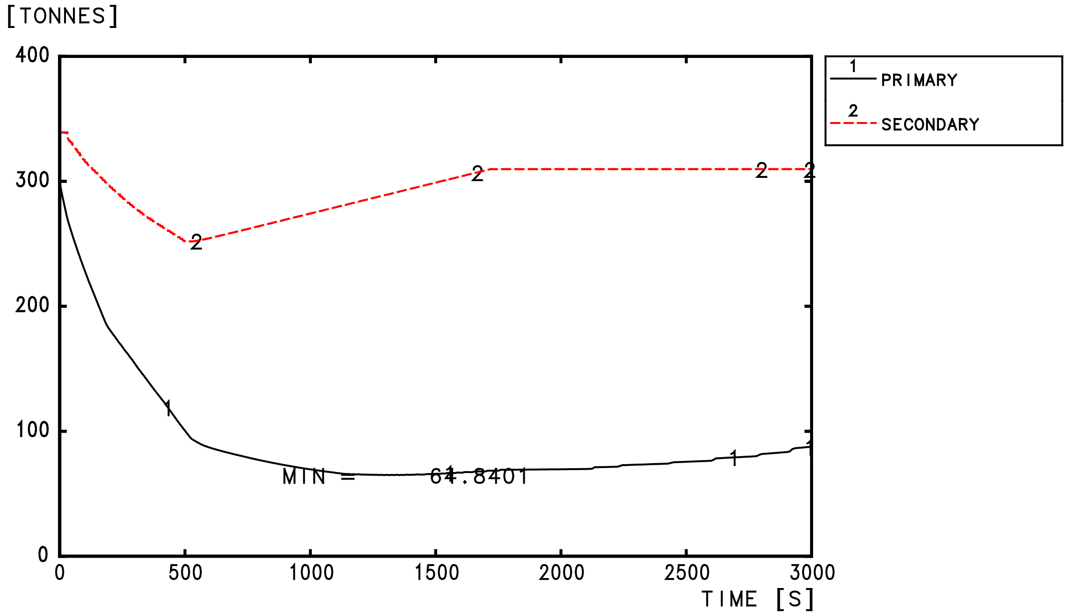
FIRST BREAK SPECTRUM: 100 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



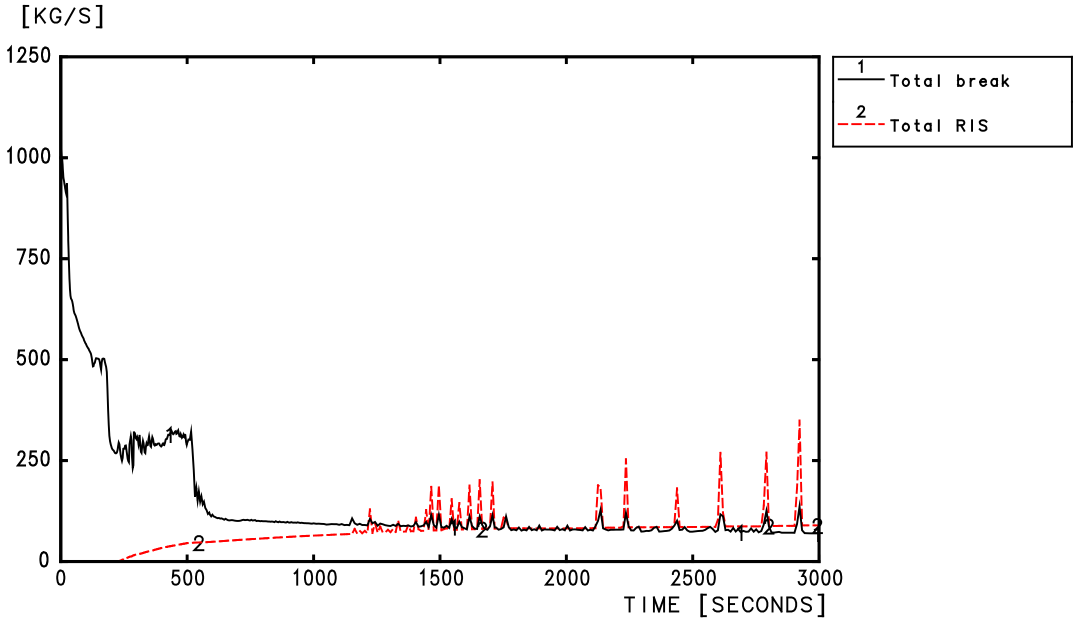
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 12

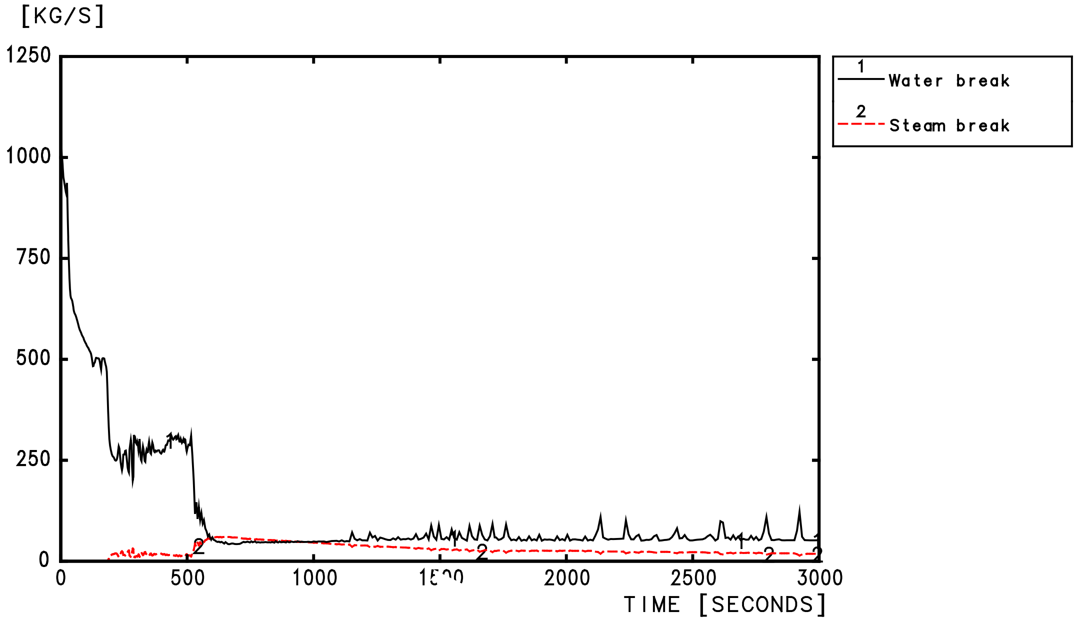
FIRST BREAK SPECTRUM: 100 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



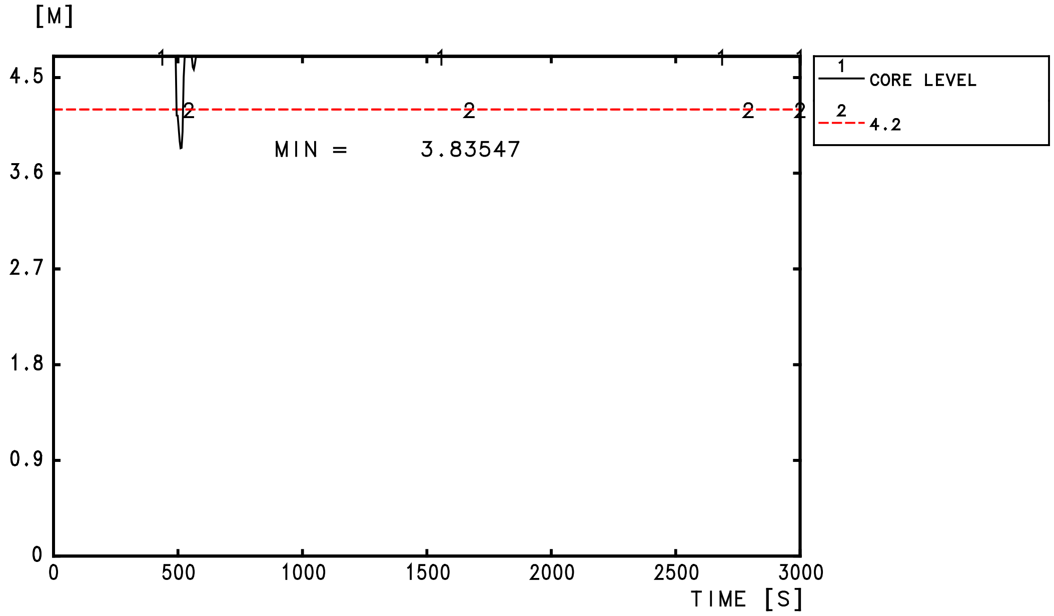
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 13

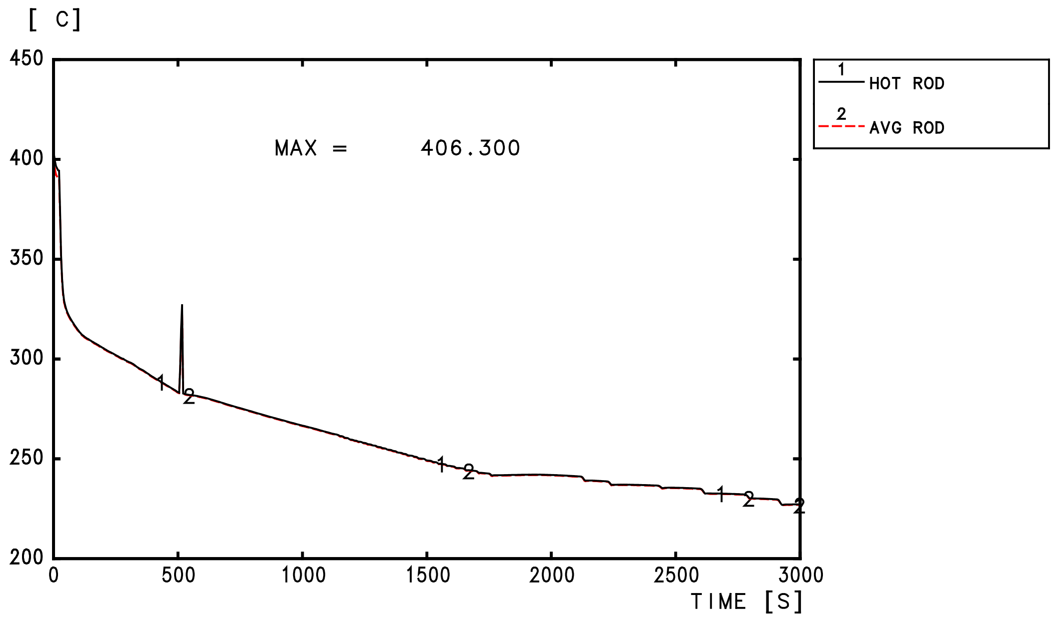
FIRST BREAK SPECTRUM: 100 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

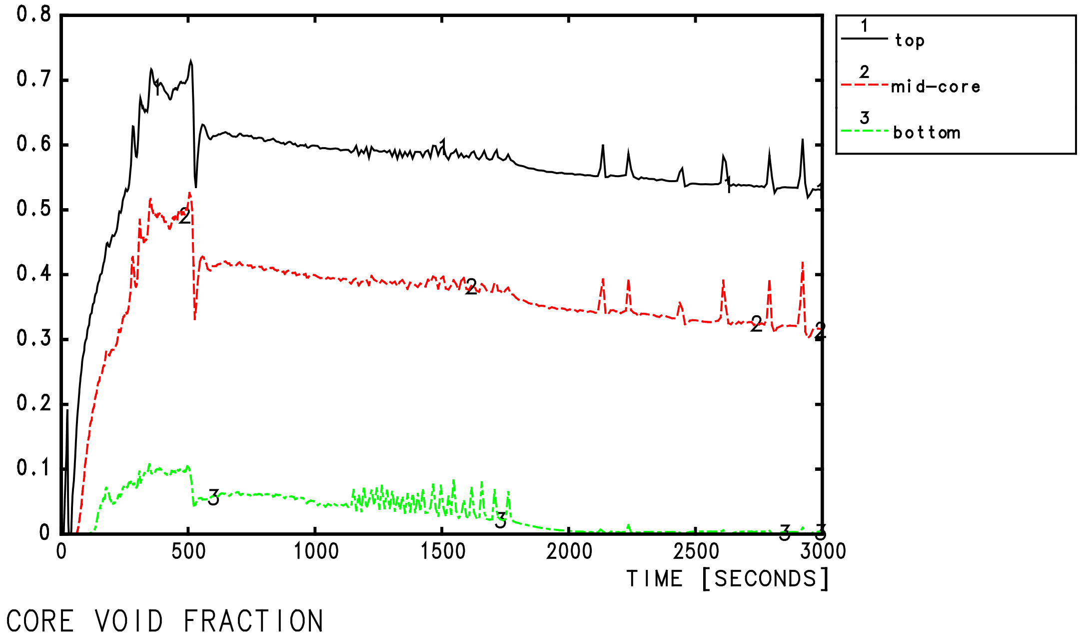


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 14

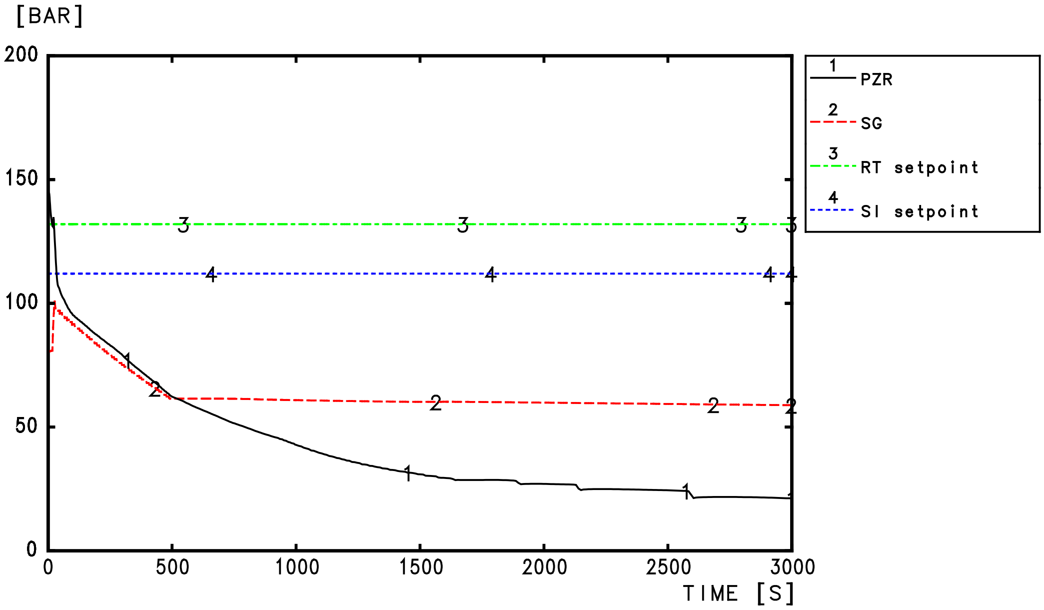
FIRST BREAK SPECTRUM: 100 CM² BREAK

CORE VOID FRACTION

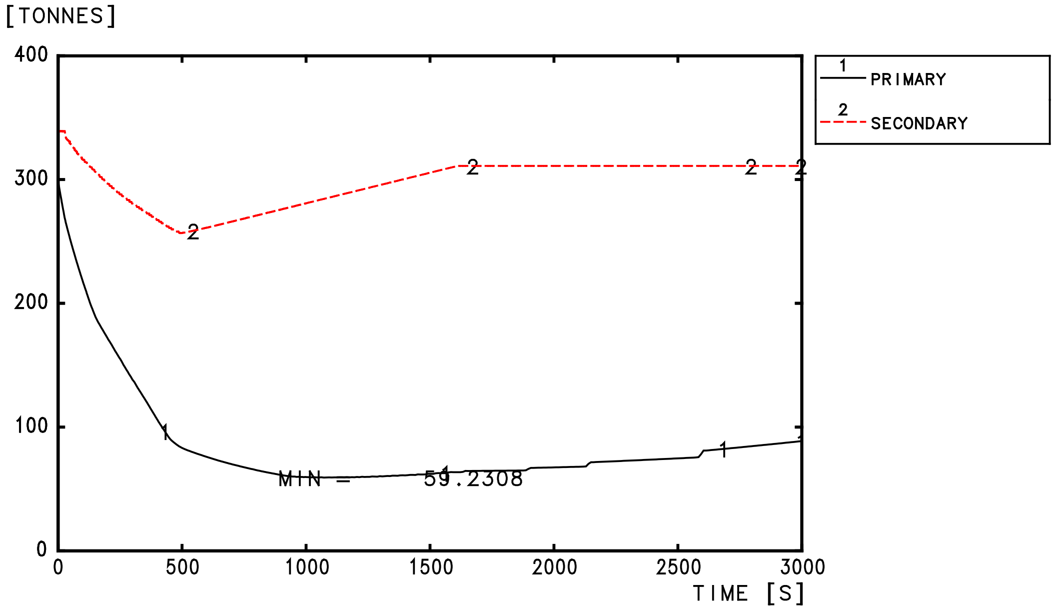


SECTION 14.5.6 - FIGURE 15

FIRST BREAK SPECTRUM: 120 CM² BREAK
PRIMARY AND SECONDARY PRESSURES
PRIMARY AND SECONDARY SIDE WATER INVENTORIES



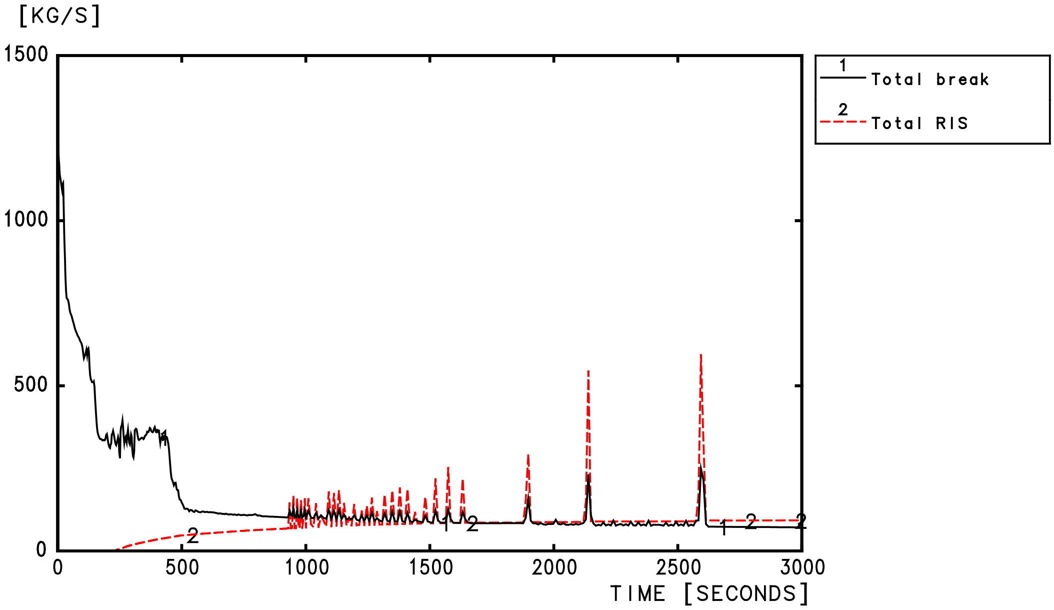
Primary and secondary pressures



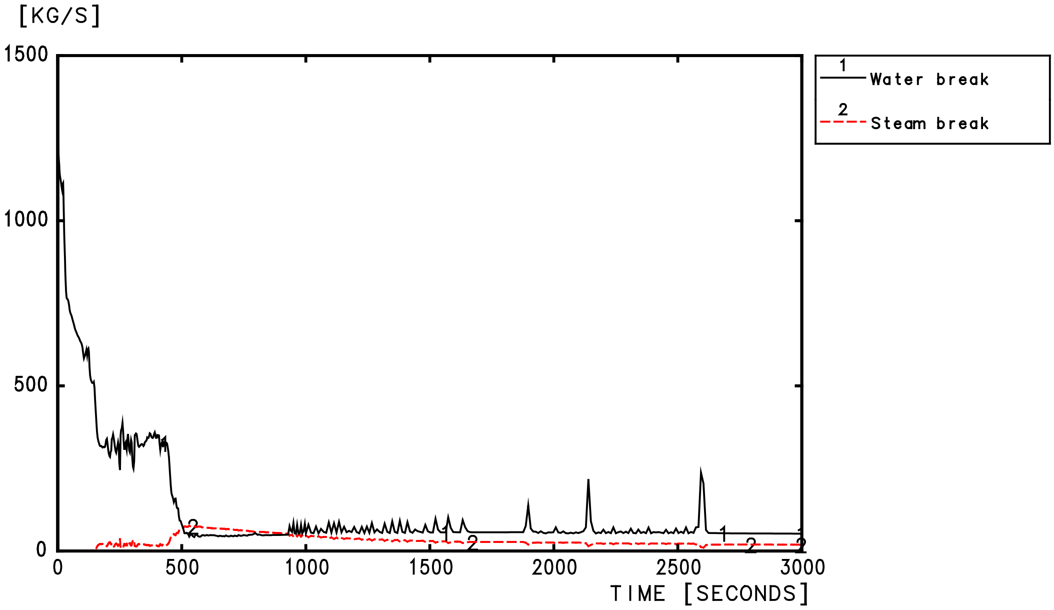
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 16

FIRST BREAK SPECTRUM: 120 CM² BREAK
TOTAL BREAK AND RIS [SIS] FLOW RATES
WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



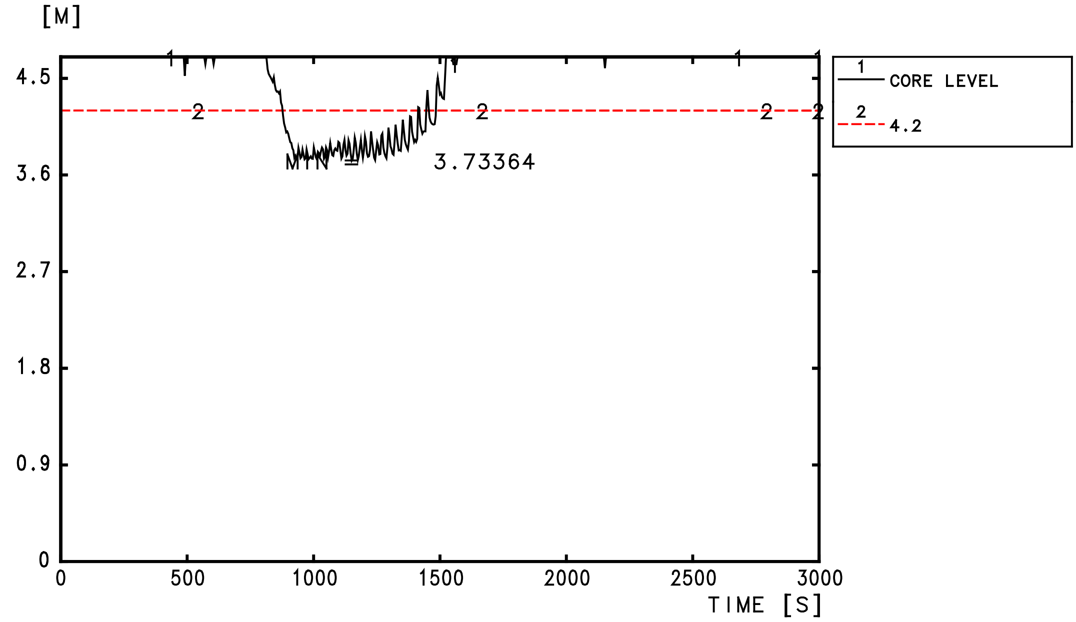
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 17

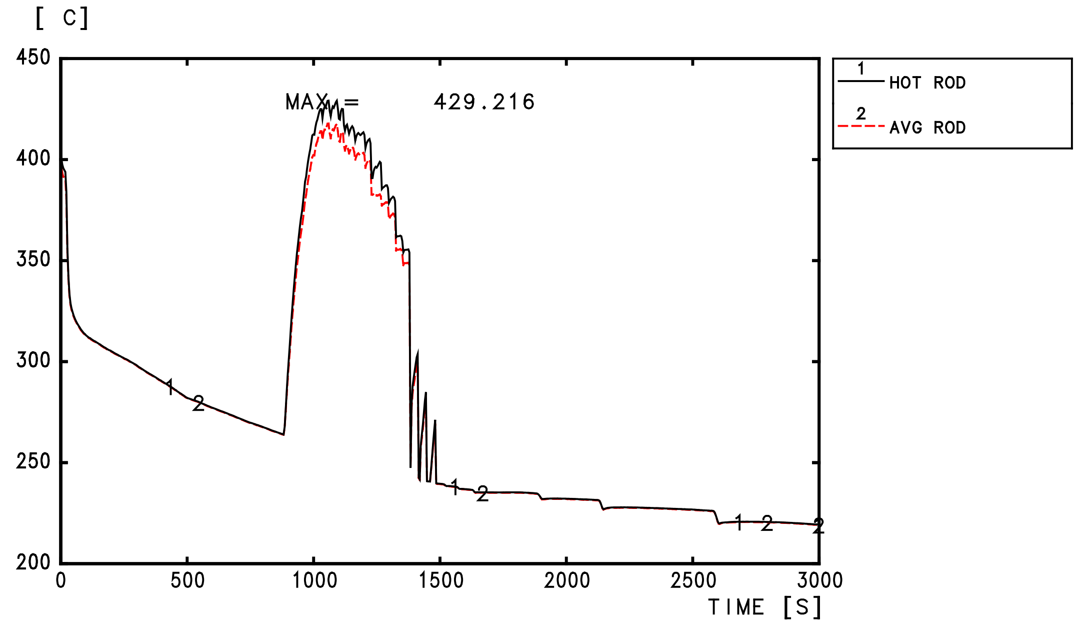
FIRST BREAK SPECTRUM: 120 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

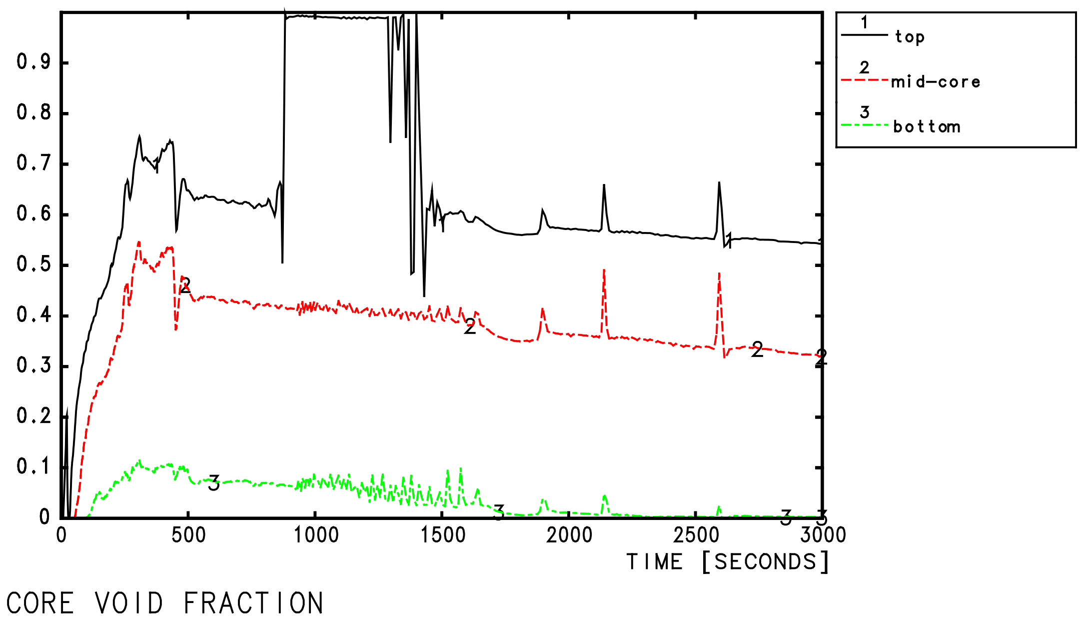


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 18

FIRST BREAK SPECTRUM: 120 CM² BREAK

CORE VOID FRACTION

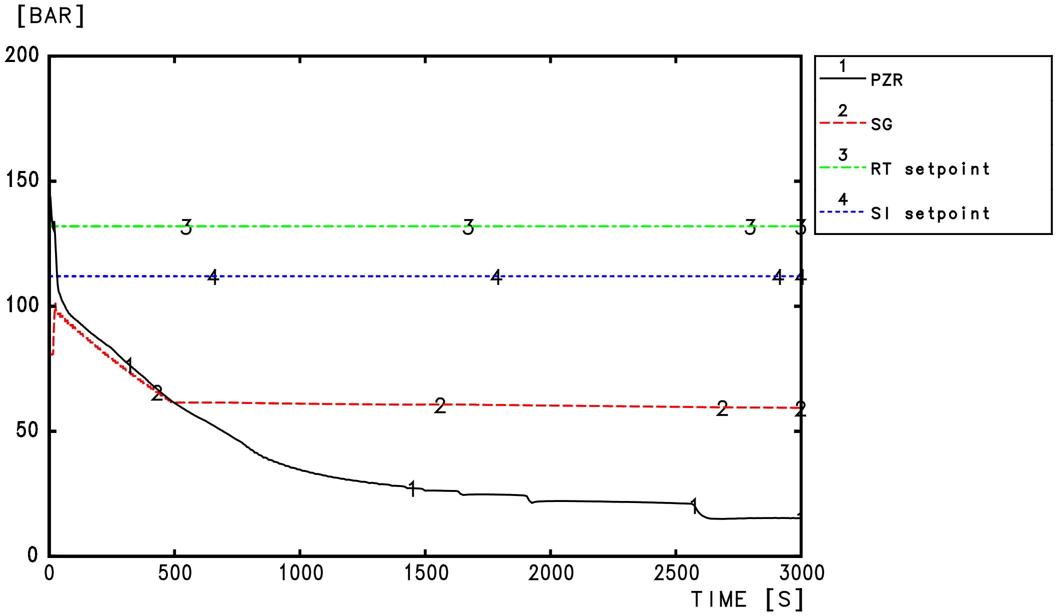


SECTION 14.5.6 - FIGURE 19

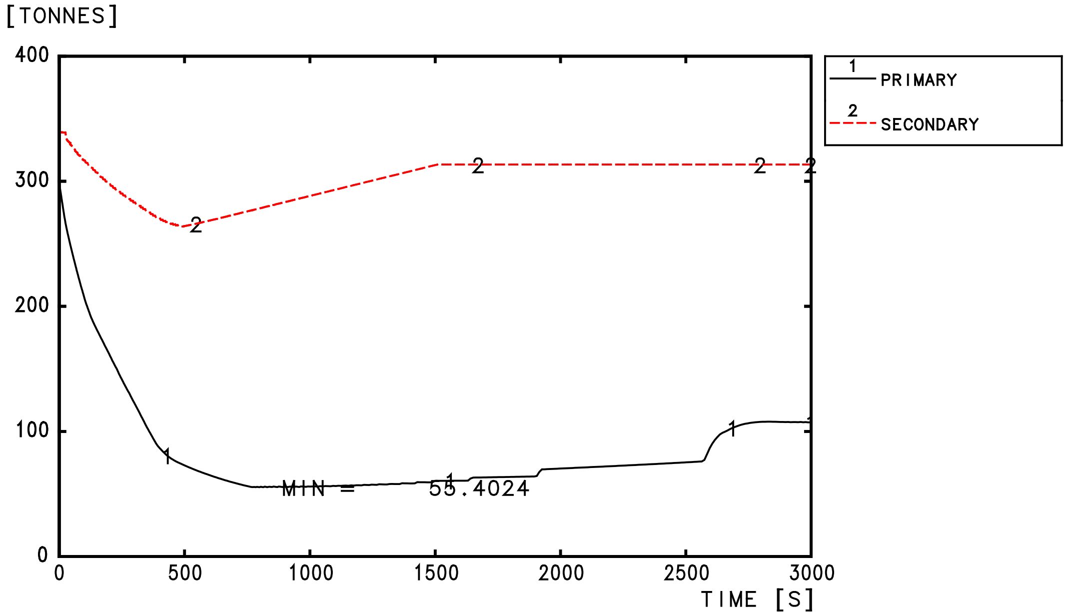
FIRST BREAK SPECTRUM: 140 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



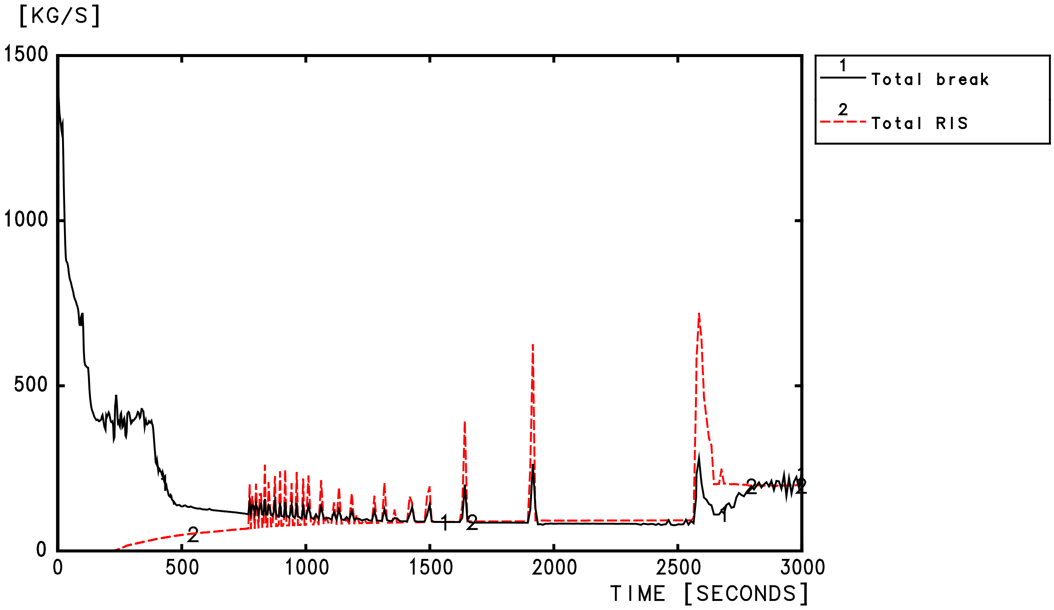
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 20

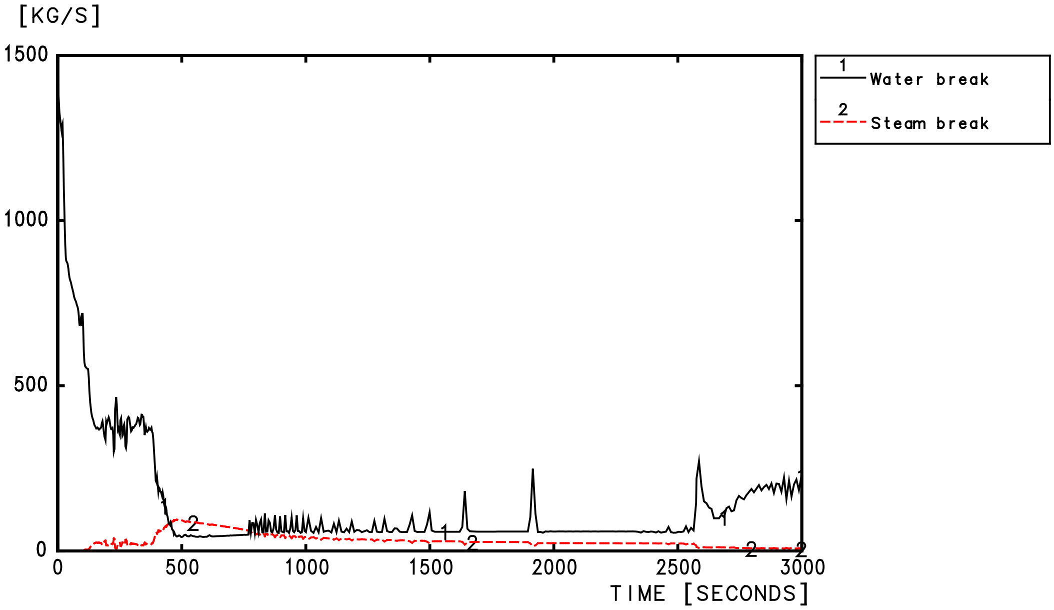
FIRST BREAK SPECTRUM: 140 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



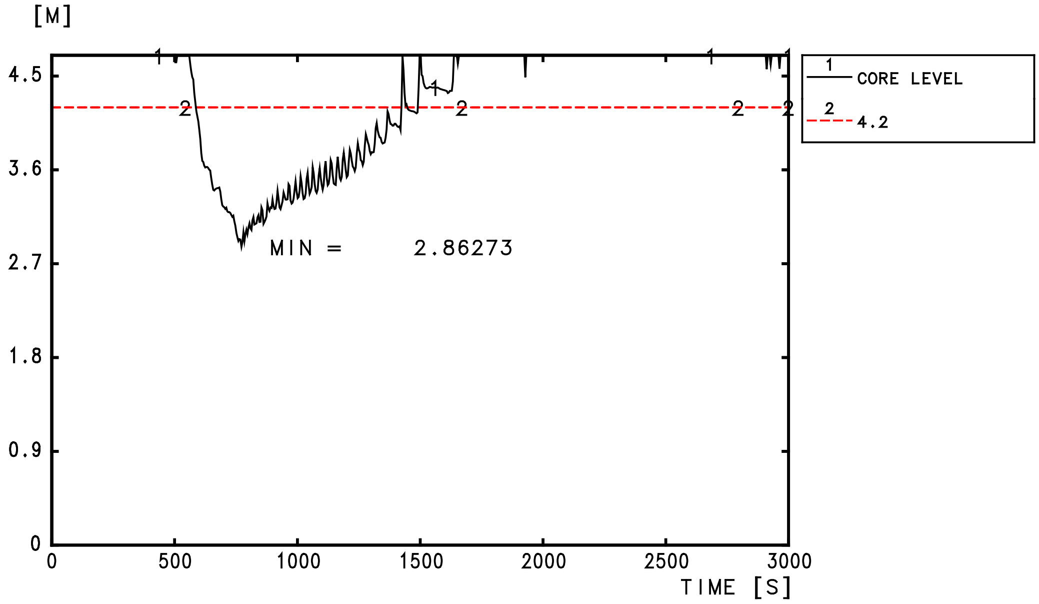
Break flowrate and total RIS injection rate



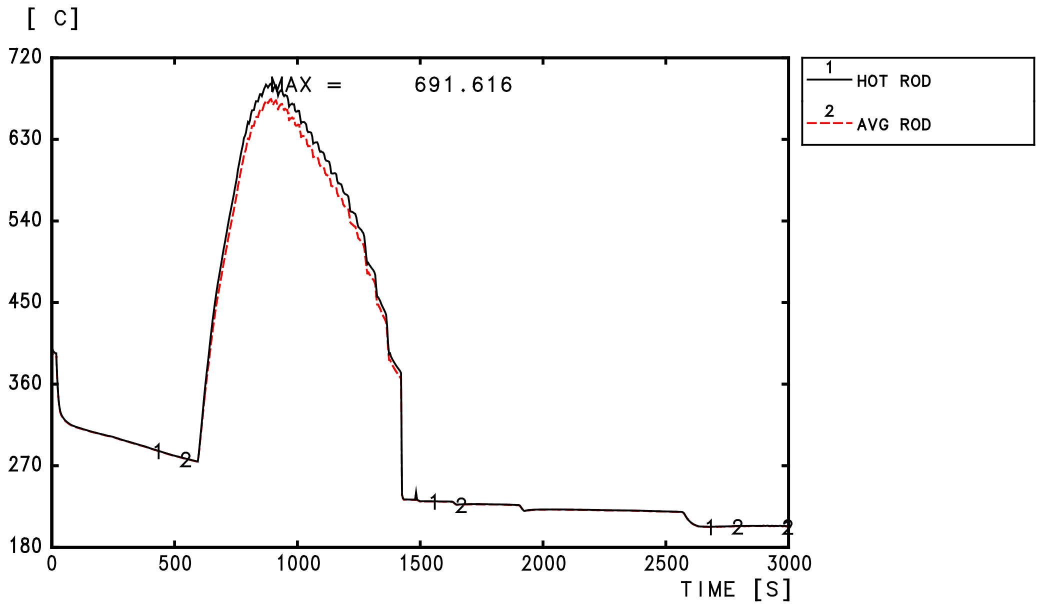
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 21

FIRST BREAK SPECTRUM: 140 CM² BREAK
CORE LEVEL
MAXIMUM CLADDING TEMPERATURE



Core level

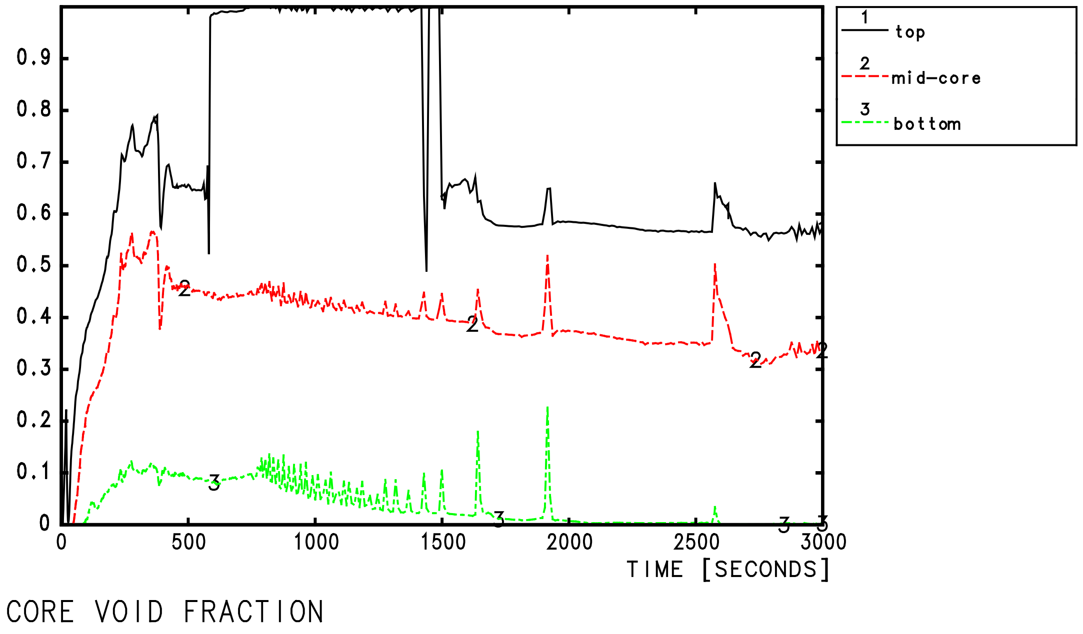


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 22

FIRST BREAK SPECTRUM: 140 CM² BREAK

CORE VOID FRACTION

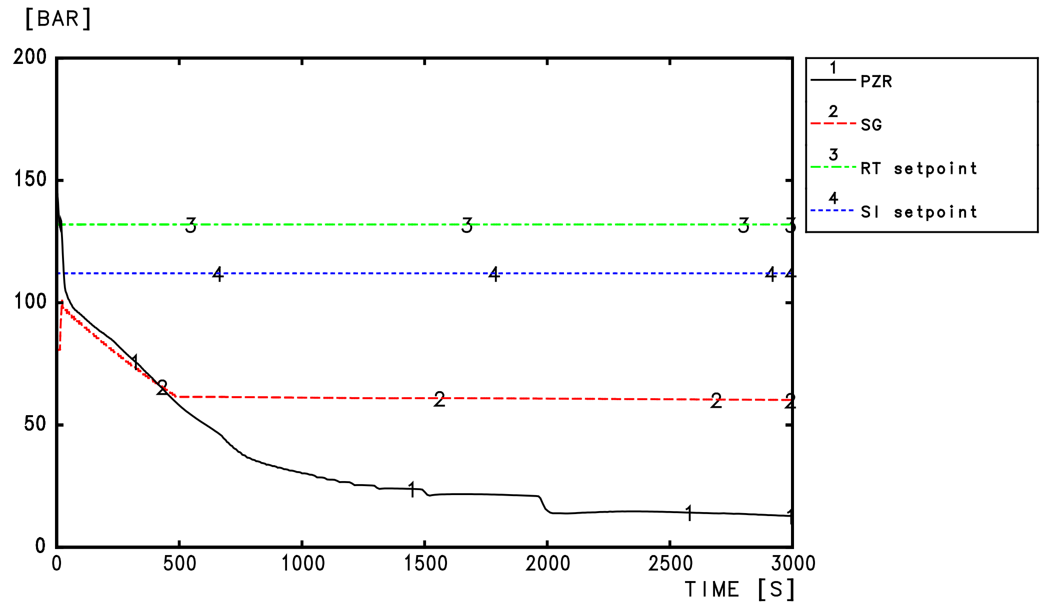


SECTION 14.5.6 - FIGURE 23

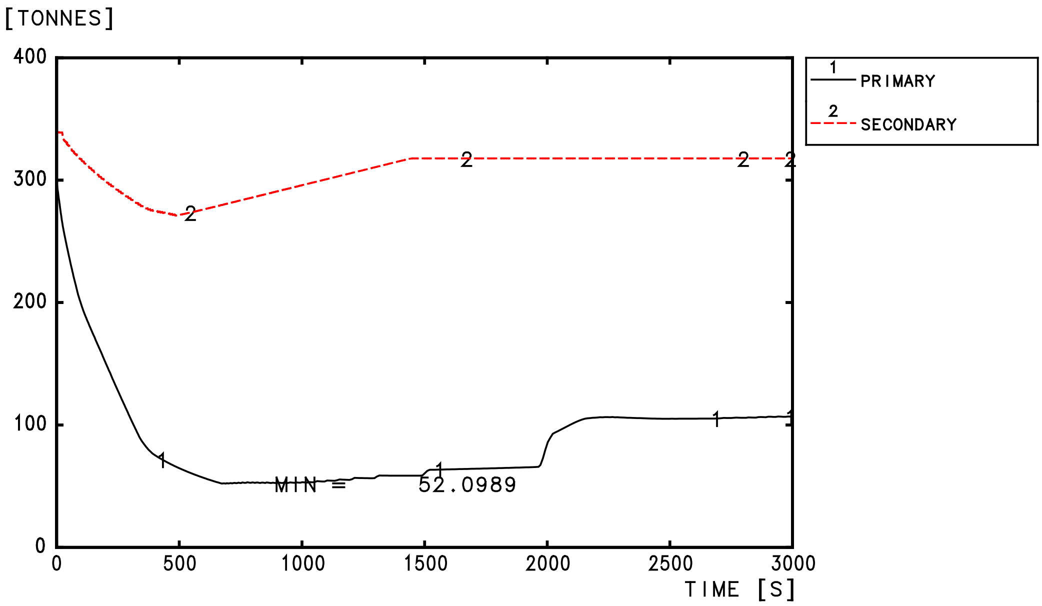
FIRST BREAK SPECTRUM: 160 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



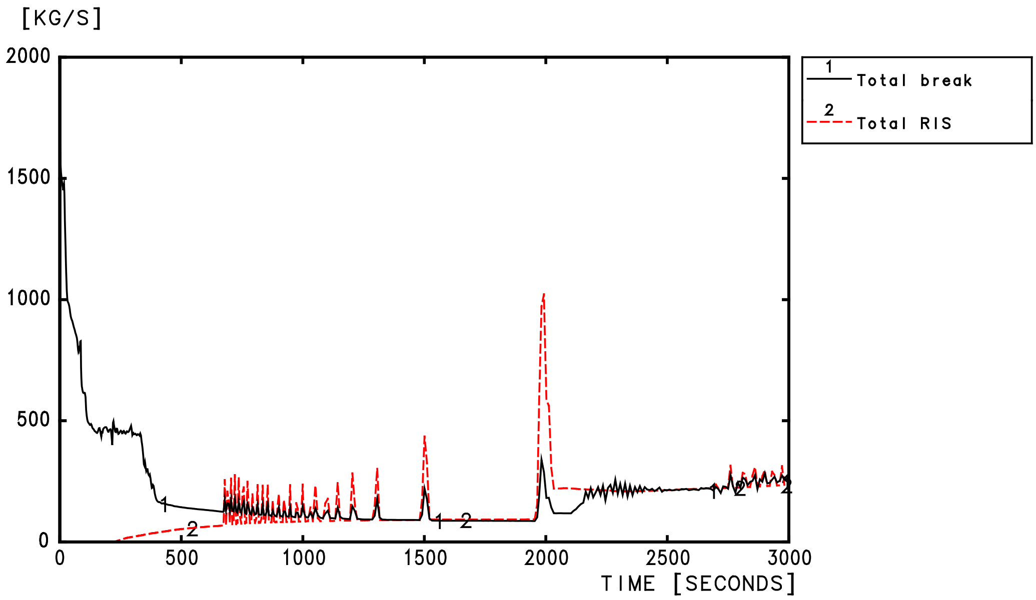
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 24

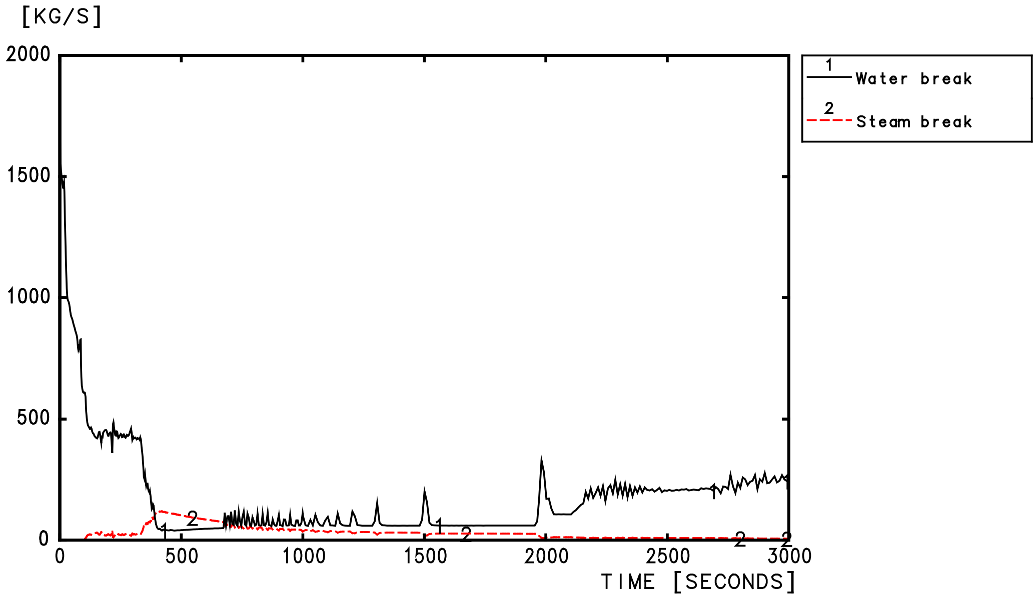
FIRST BREAK SPECTRUM: 160 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



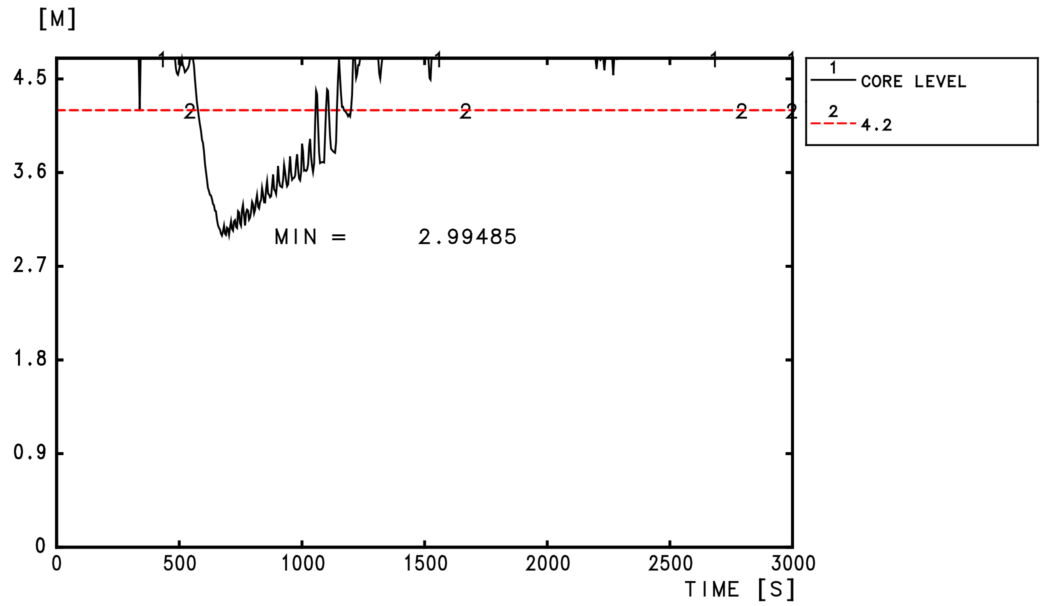
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 25

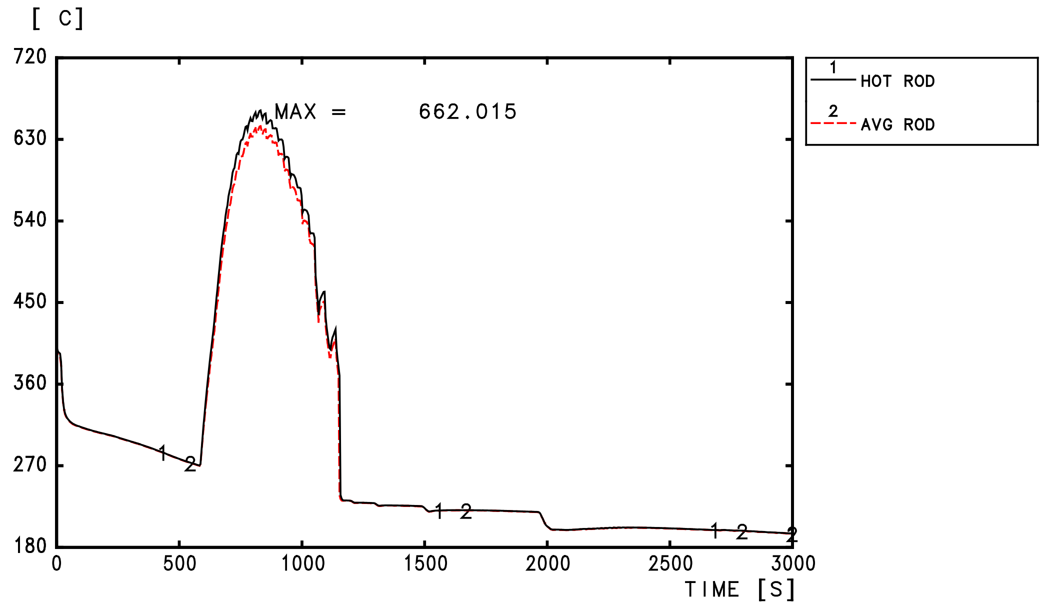
FIRST BREAK SPECTRUM: 160 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

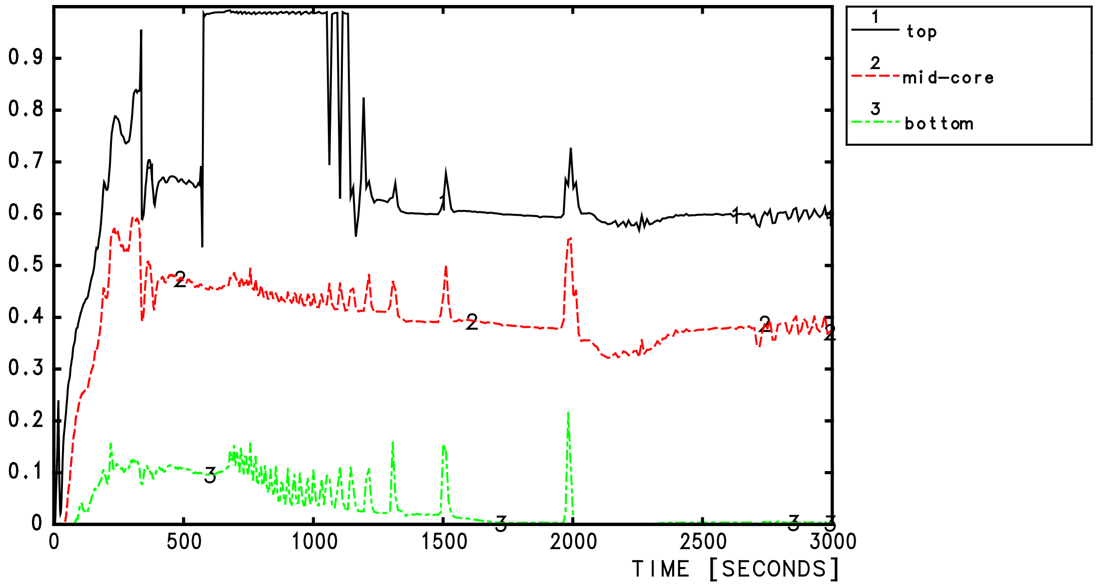


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 26

FIRST BREAK SPECTRUM: 160 CM² BREAK

CORE VOID FRACTION



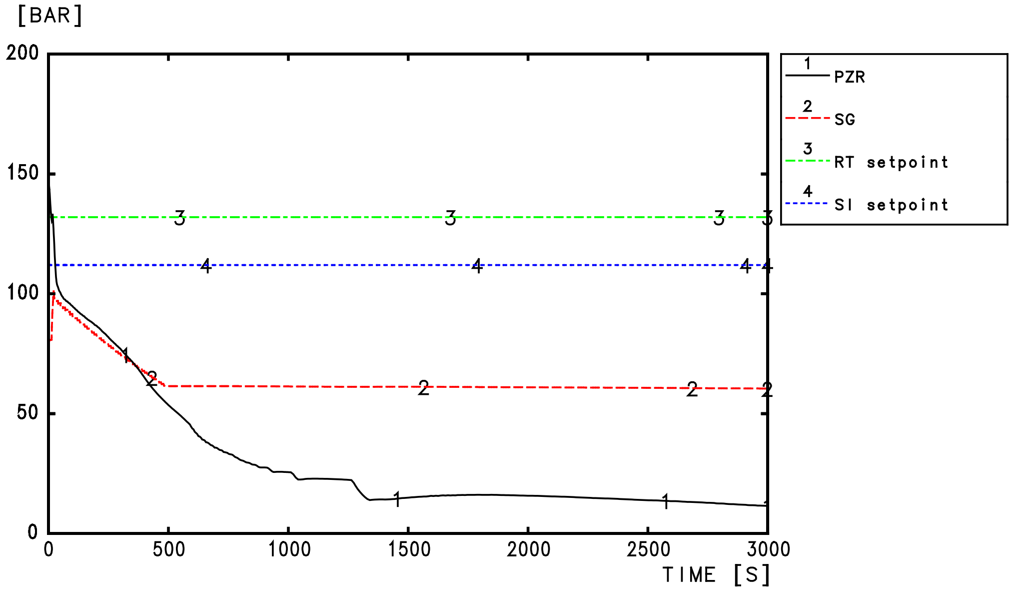
CORE VOID FRACTION

SECTION 14.5.6 - FIGURE 27

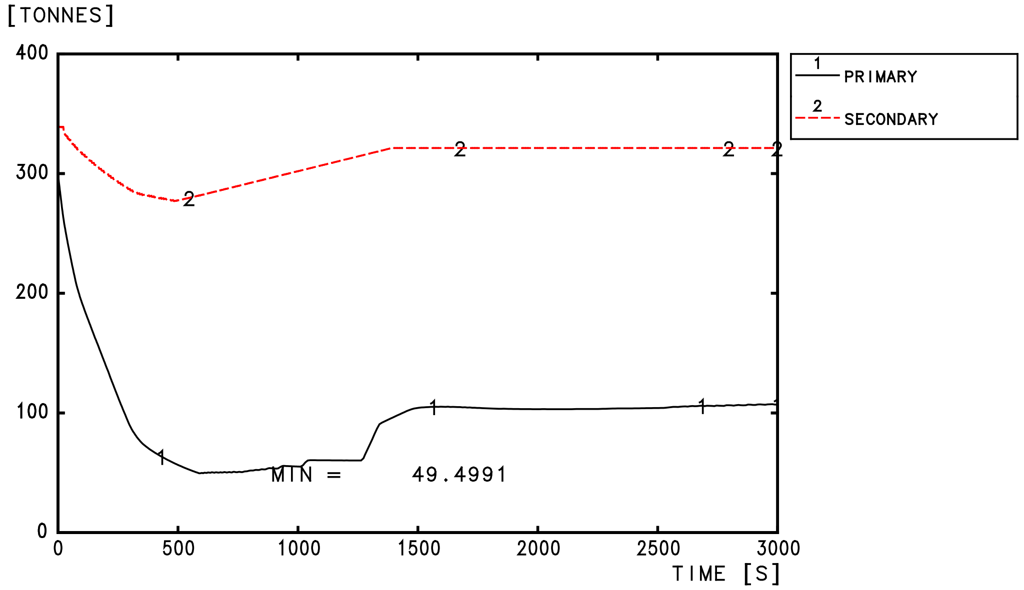
FIRST BREAK SPECTRUM: 180 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



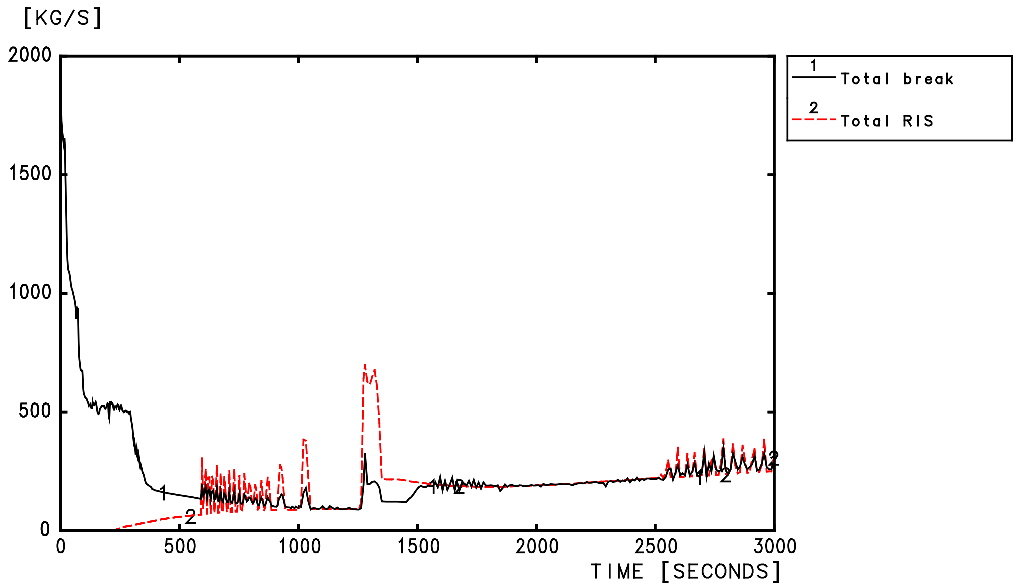
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 28

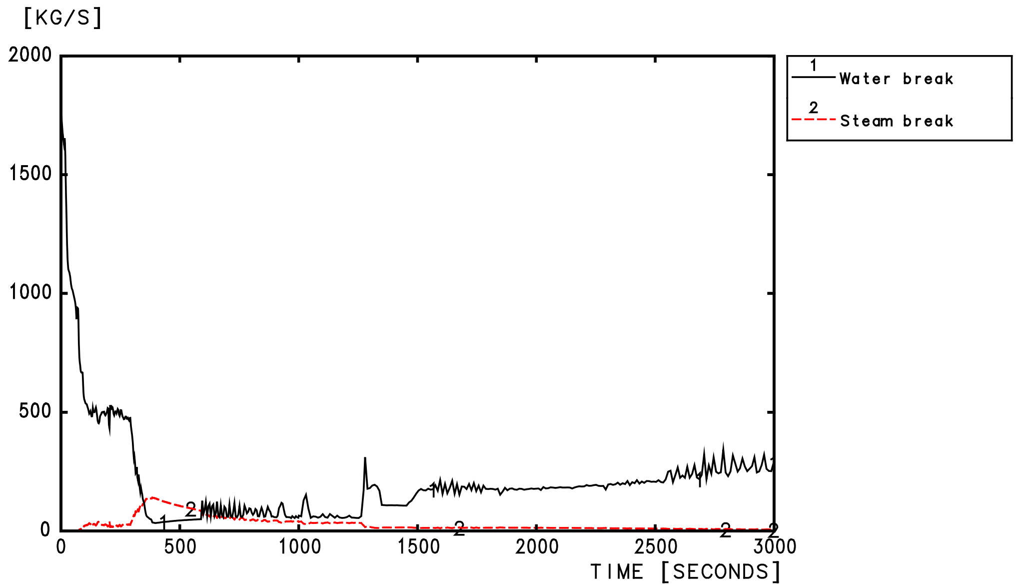
FIRST BREAK SPECTRUM: 180 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



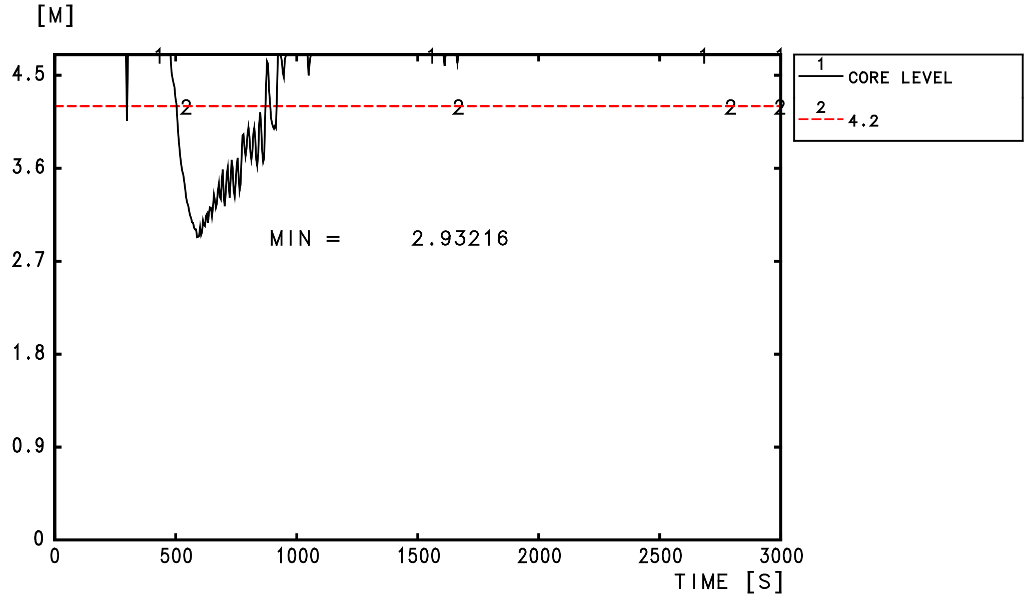
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 29

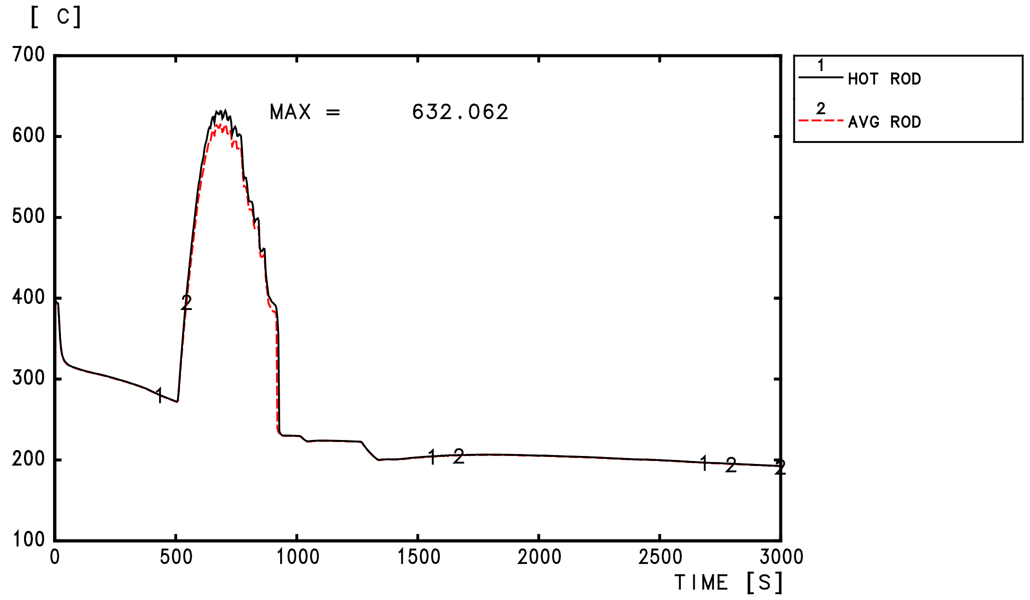
FIRST BREAK SPECTRUM: 180 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

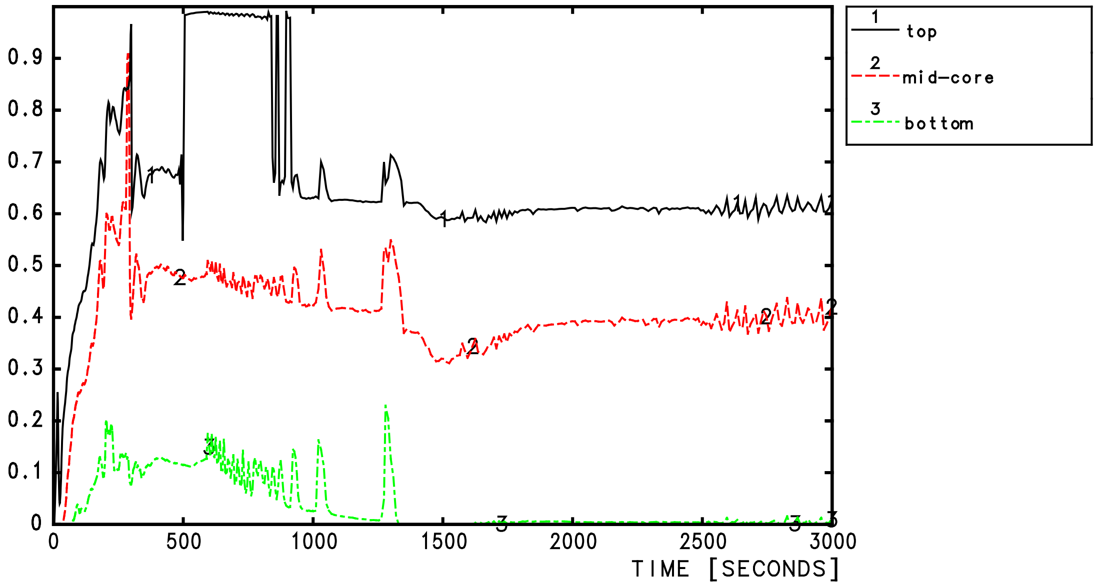


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 30

FIRST BREAK SPECTRUM: 180 CM² BREAK

CORE VOID FRACTION



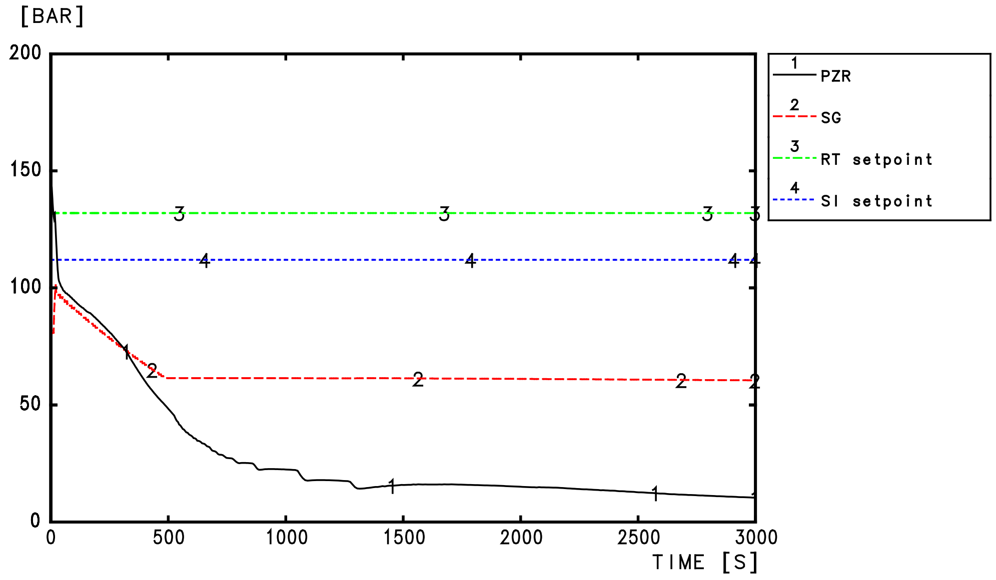
CORE VOID FRACTION

SECTION 14.5.6 - FIGURE 31

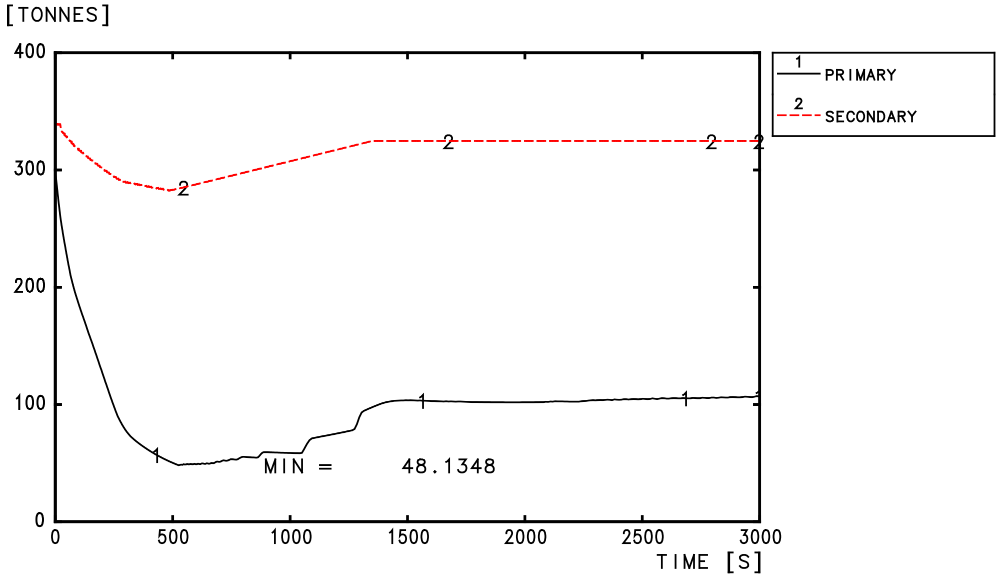
FIRST BREAK SPECTRUM: 200 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



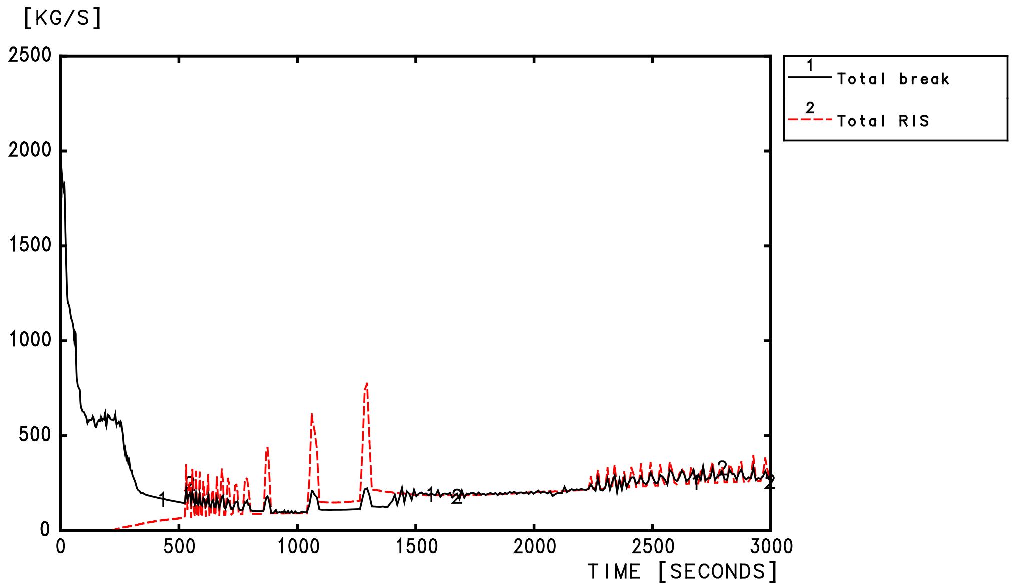
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 32

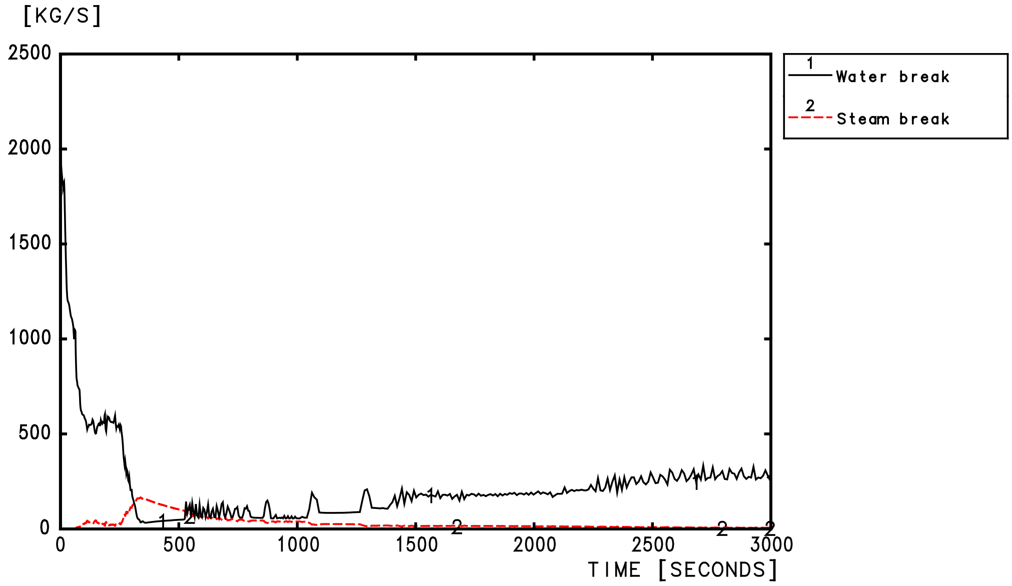
FIRST BREAK SPECTRUM: 200 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



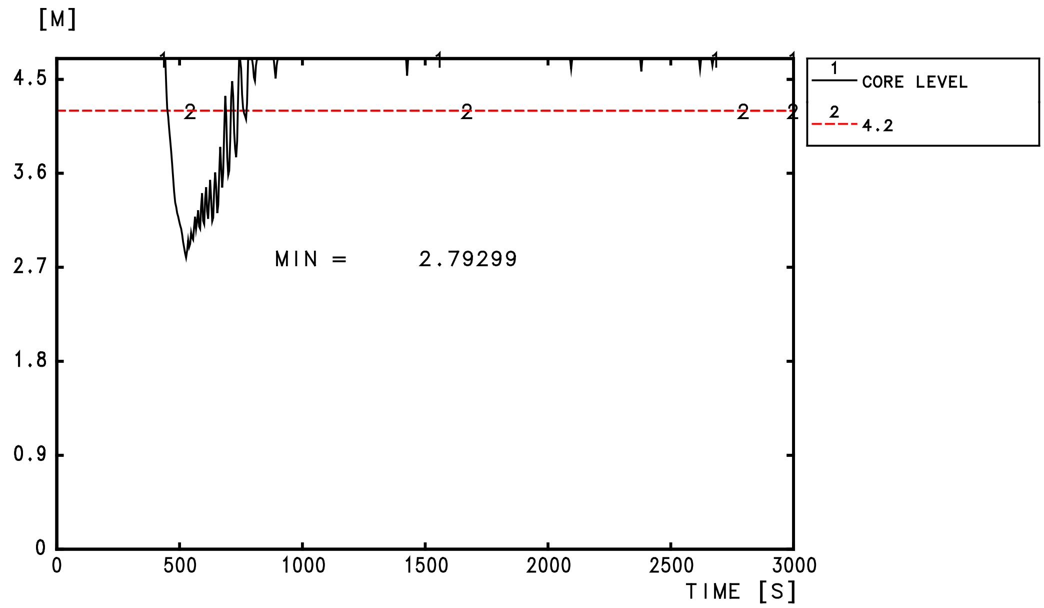
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 33

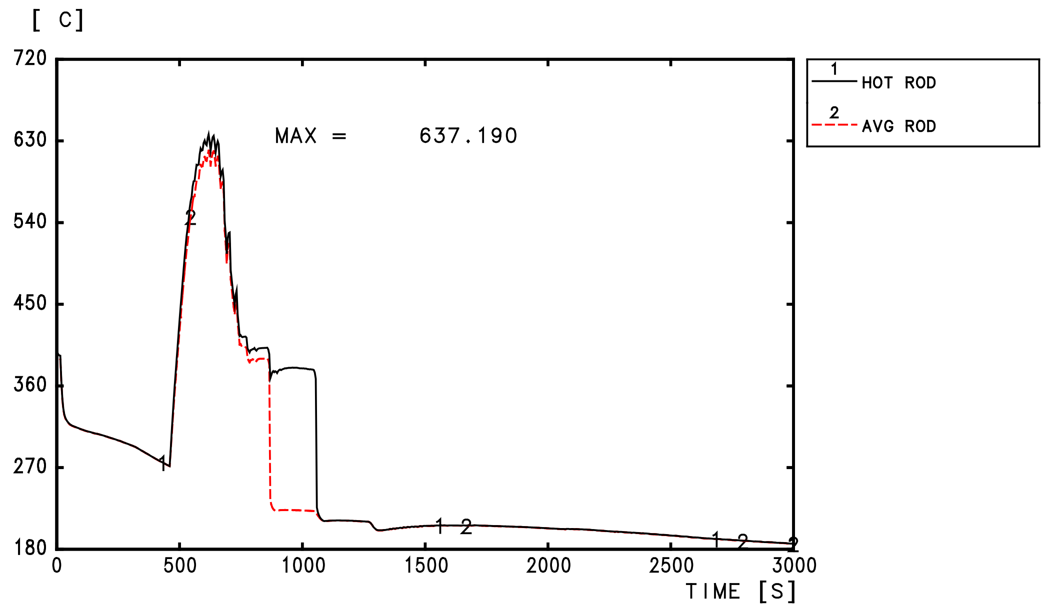
FIRST BREAK SPECTRUM: 200 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

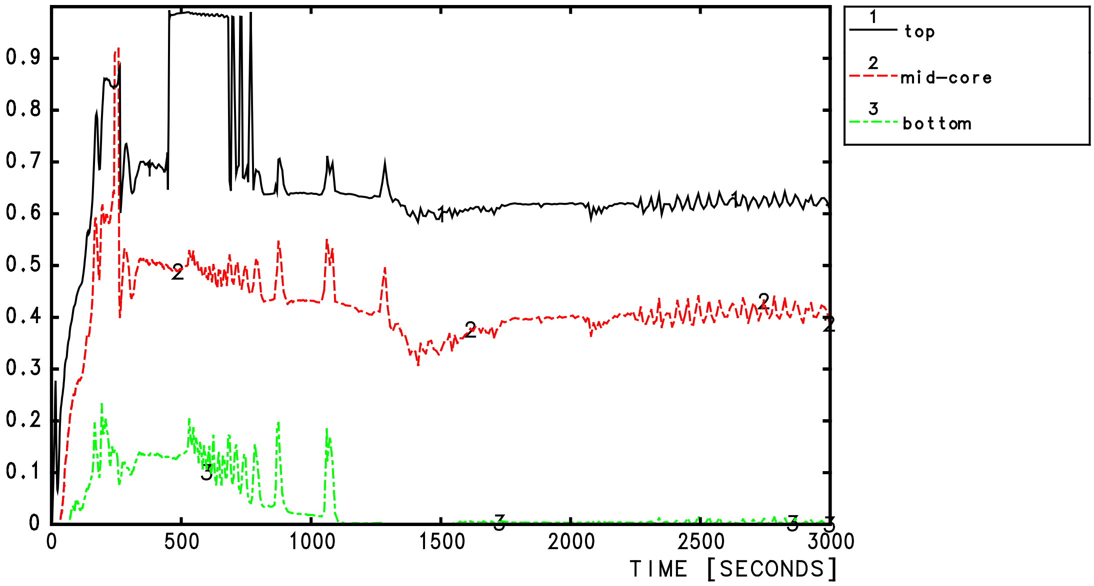


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 34

FIRST BREAK SPECTRUM: 200 CM² BREAK

CORE VOID FRACTION



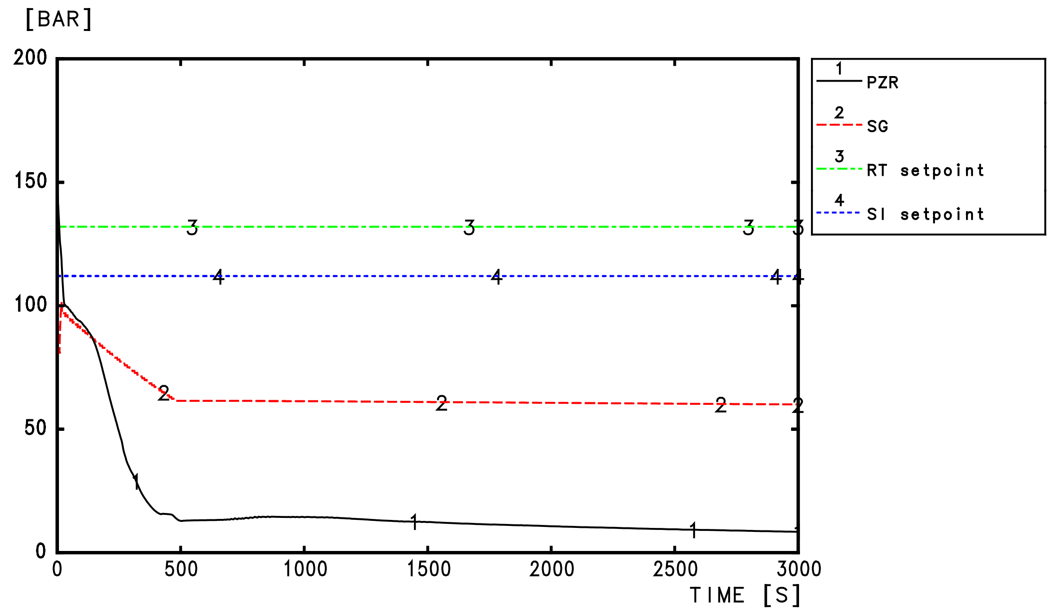
CORE VOID FRACTION

SECTION 14.5.6 - FIGURE 35

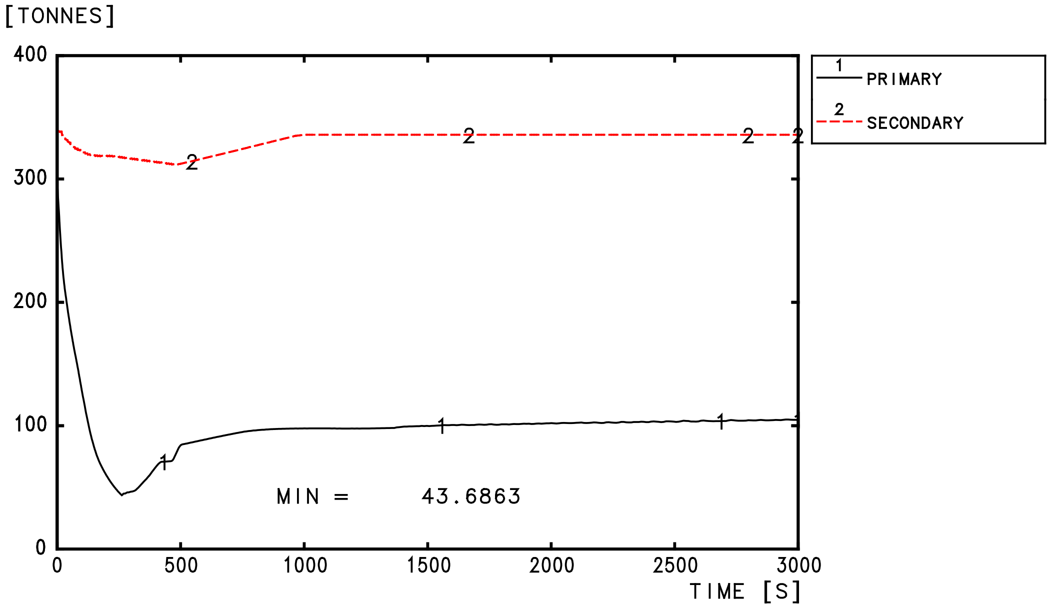
FIRST BREAK SPECTRUM: 390 CM² BREAK (RIS)

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



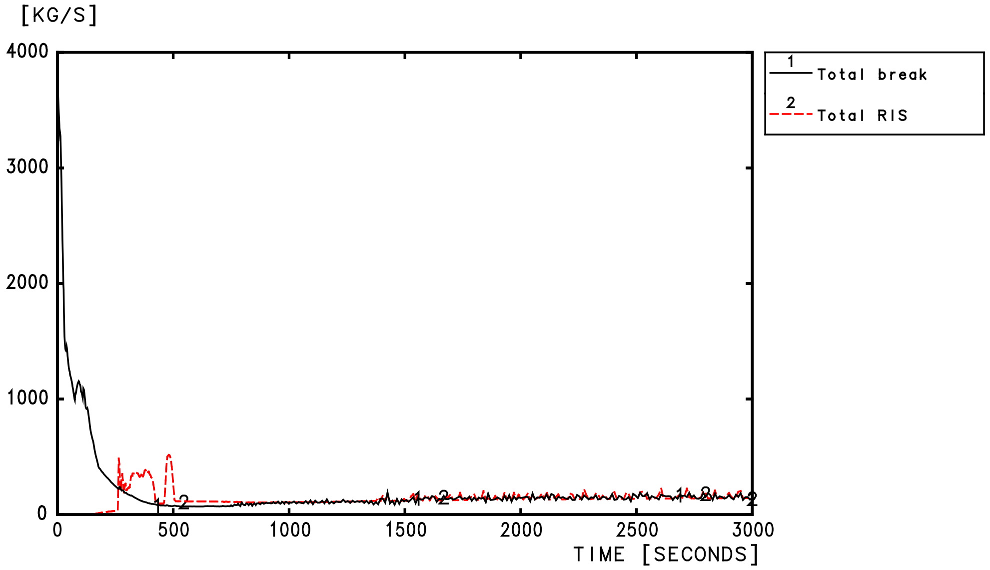
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 36

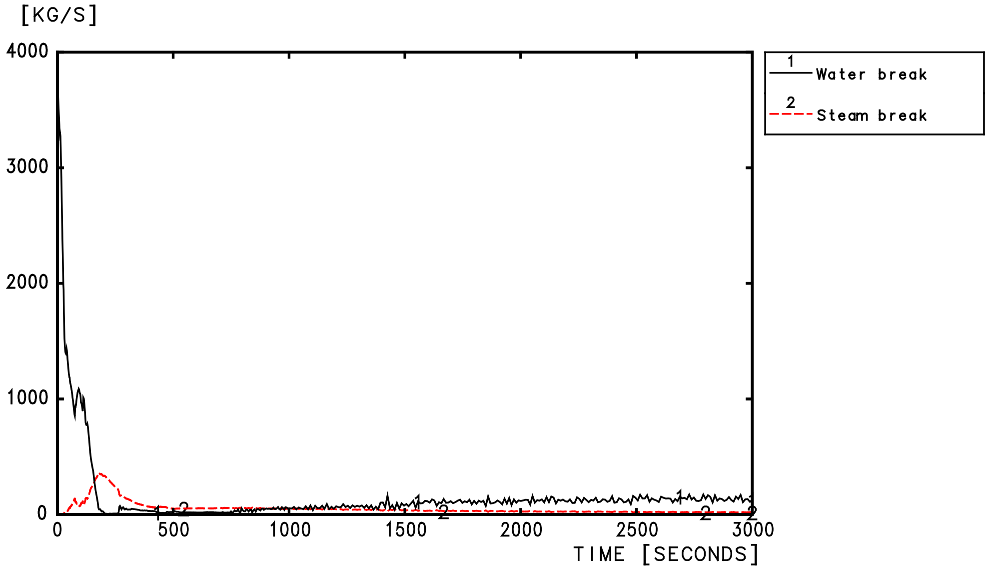
FIRST BREAK SPECTRUM: 390 CM² BREAK (RIS)

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



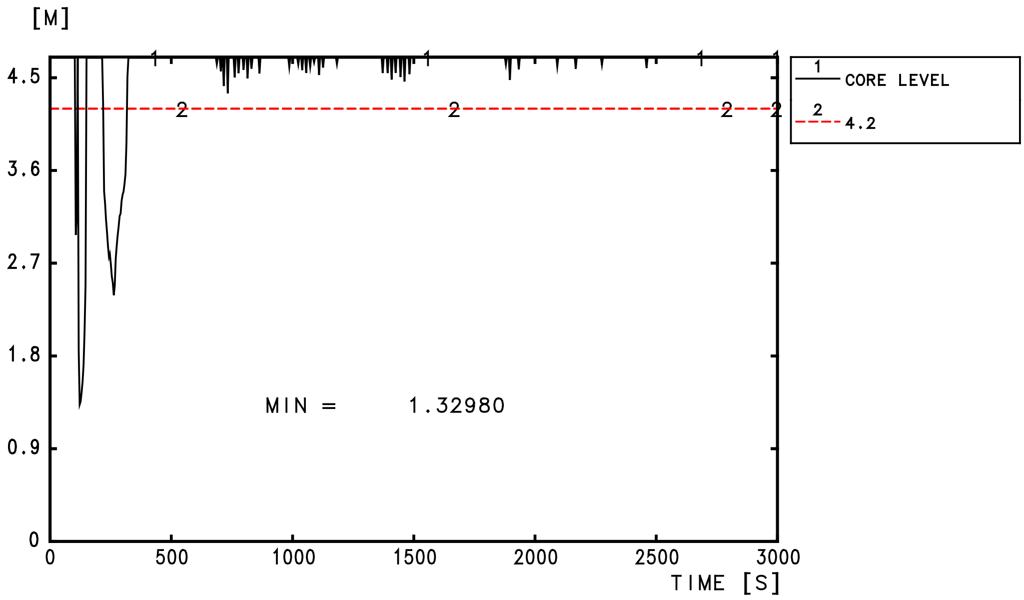
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 37

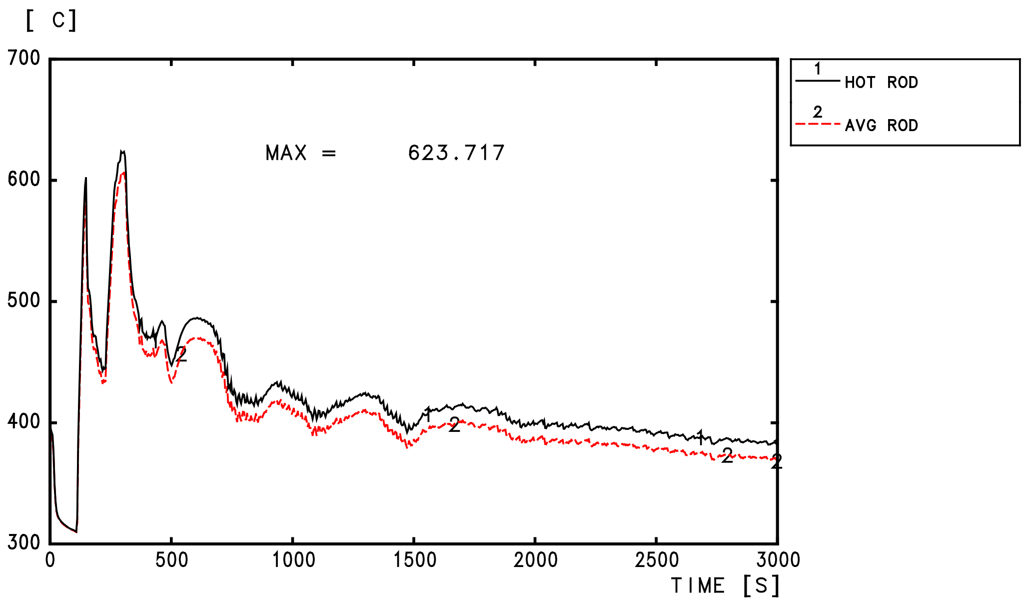
FIRST BREAK SPECTRUM: 390 CM² BREAK (RIS)

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

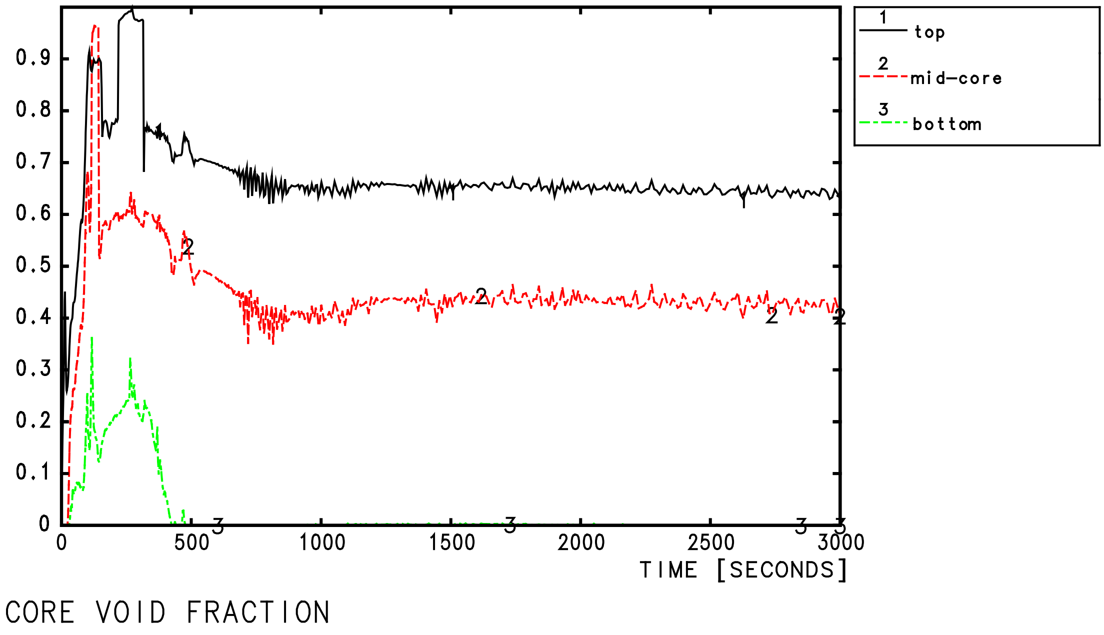


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 38

FIRST BREAK SPECTRUM: 390 CM² BREAK (RIS)

CORE VOID FRACTION

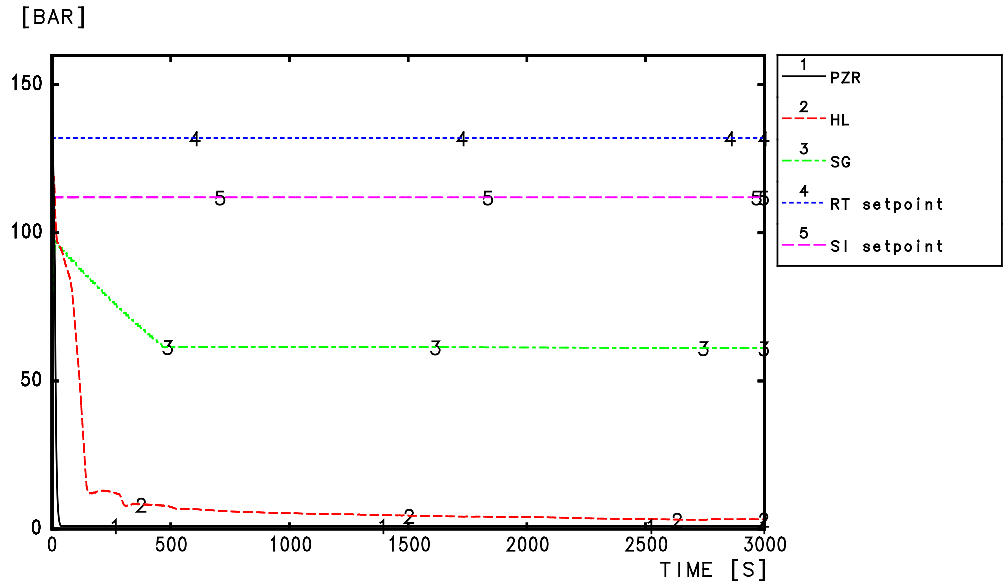


SECTION 14.5.6 - FIGURE 39

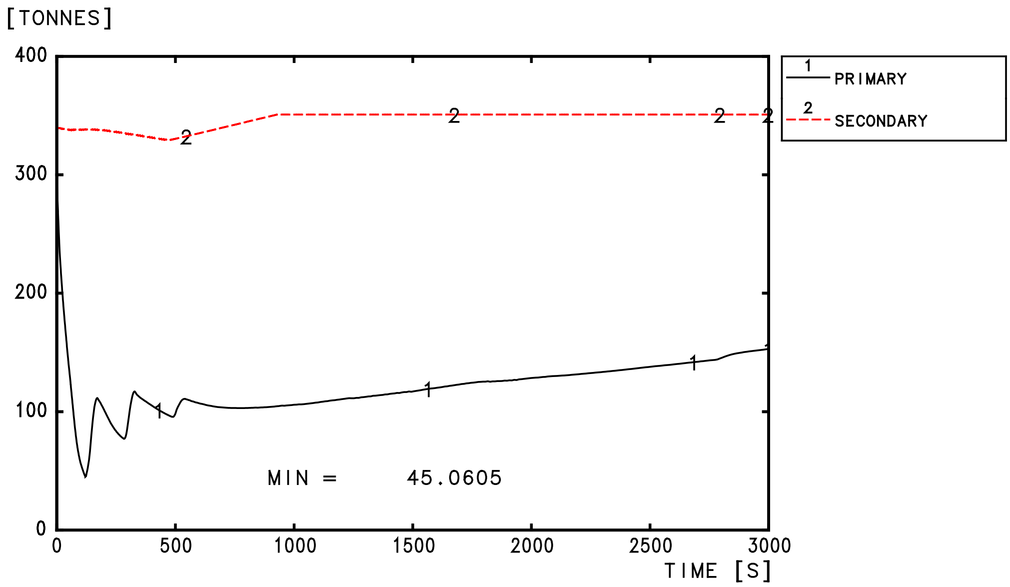
FIRST BREAK SPECTRUM: SURGE LINE BREAK (2X830 CM²)

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



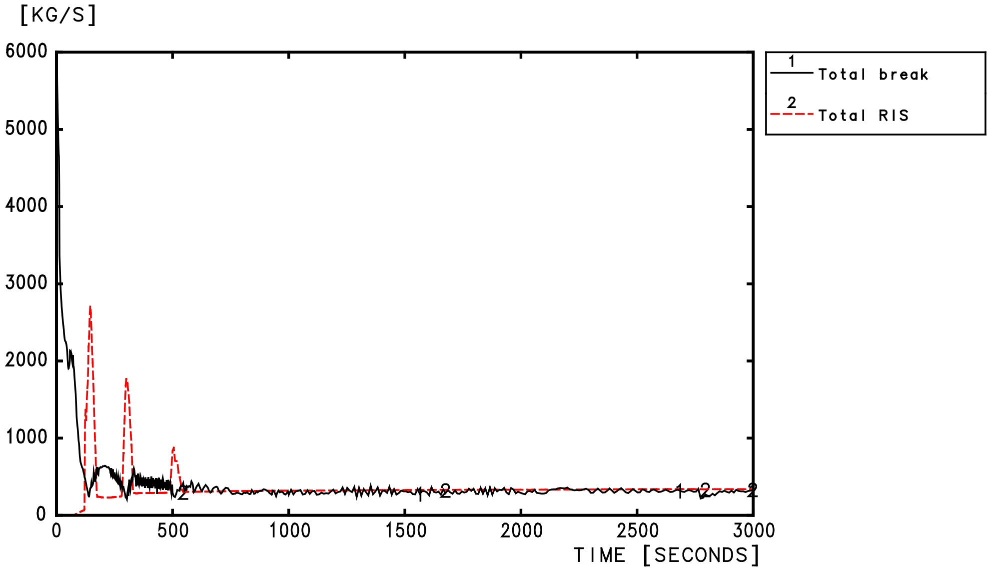
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 40

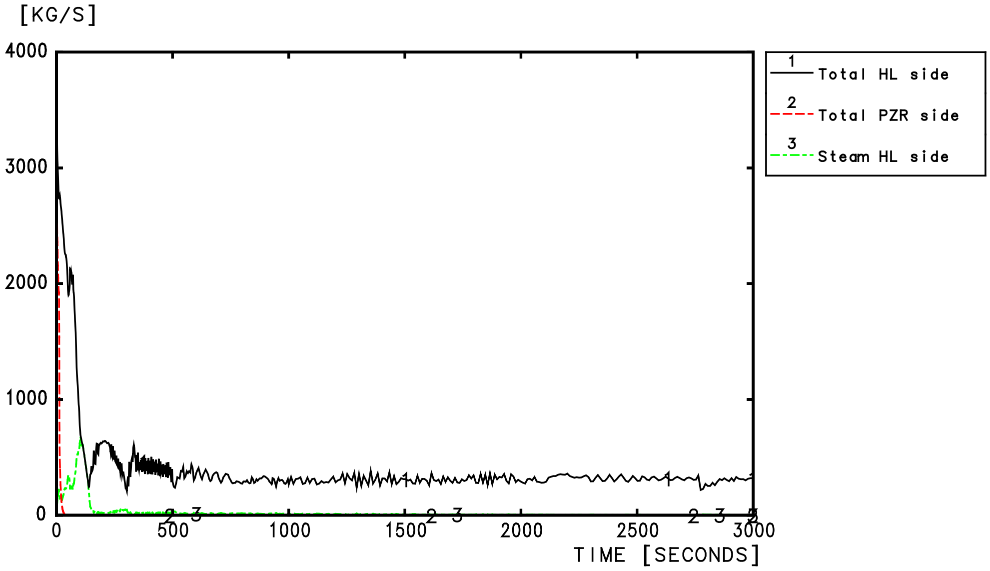
FIRST BREAK SPECTRUM: SURGE LINE BREAK (2X830 CM²)

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



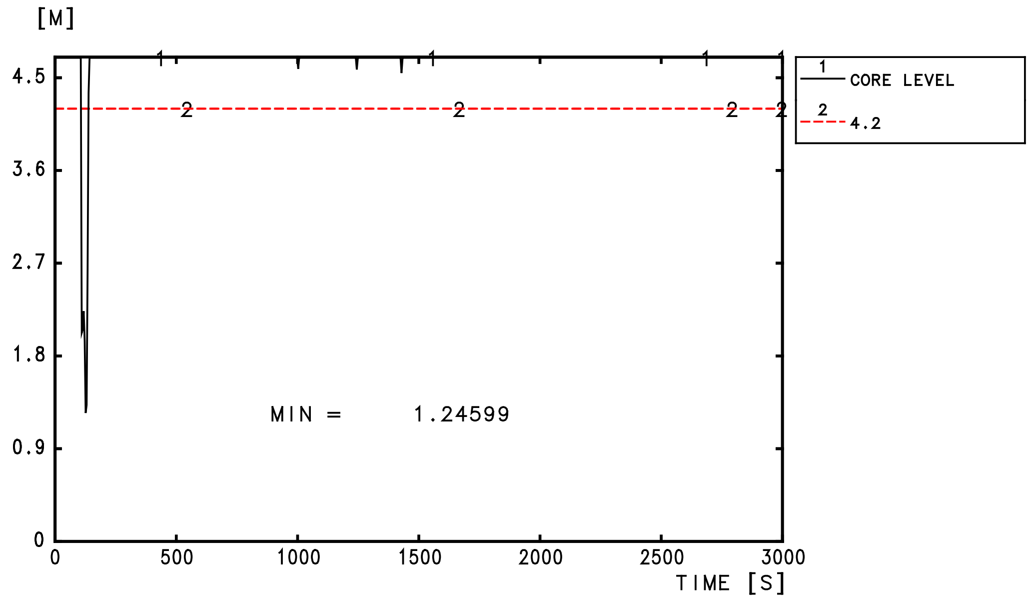
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 41

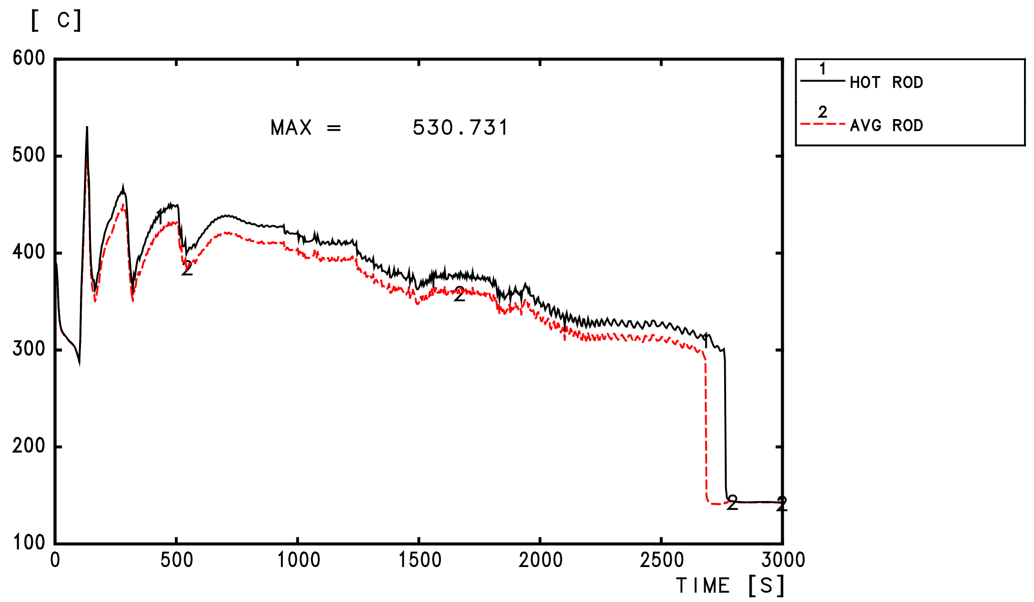
FIRST BREAK SPECTRUM: SURGE LINE BREAK (2X830 CM²)

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

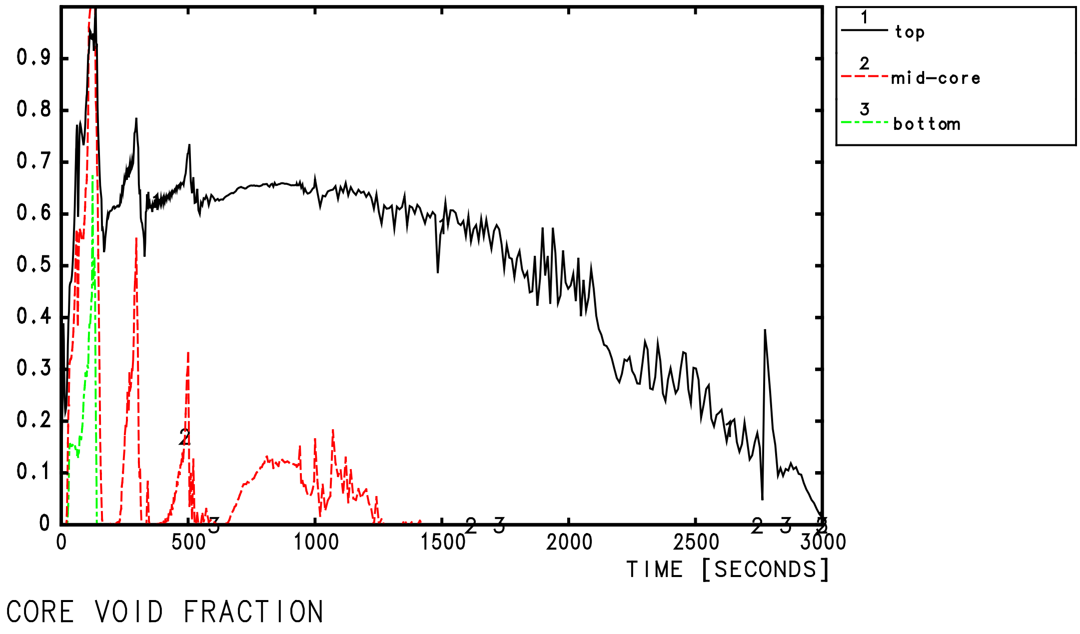


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 42

FIRST BREAK SPECTRUM: SURGE LINE BREAK (2X830 CM²)

CORE VOID FRACTION

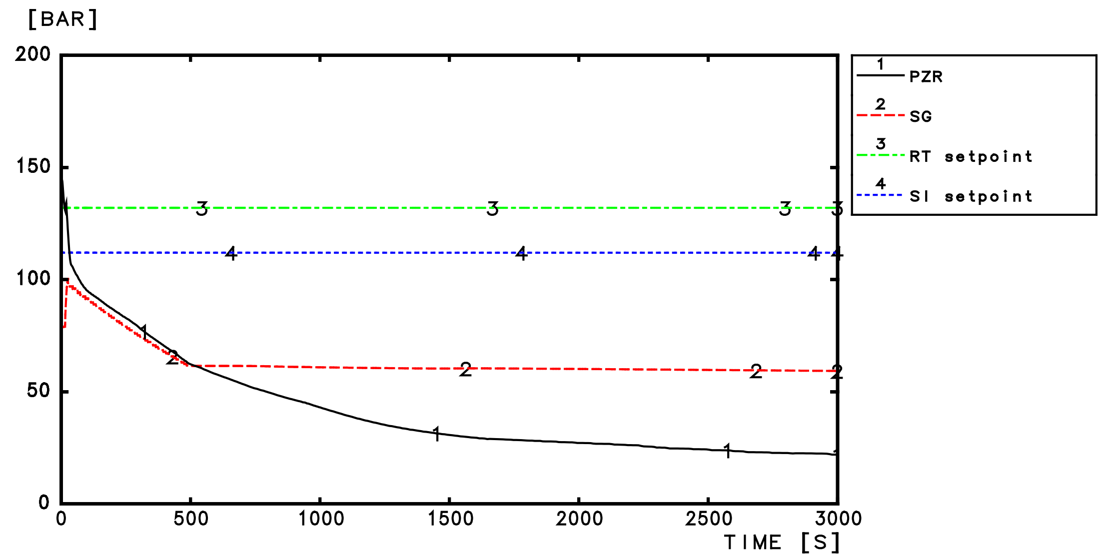


SECTION 14.5.6 - FIGURE 43

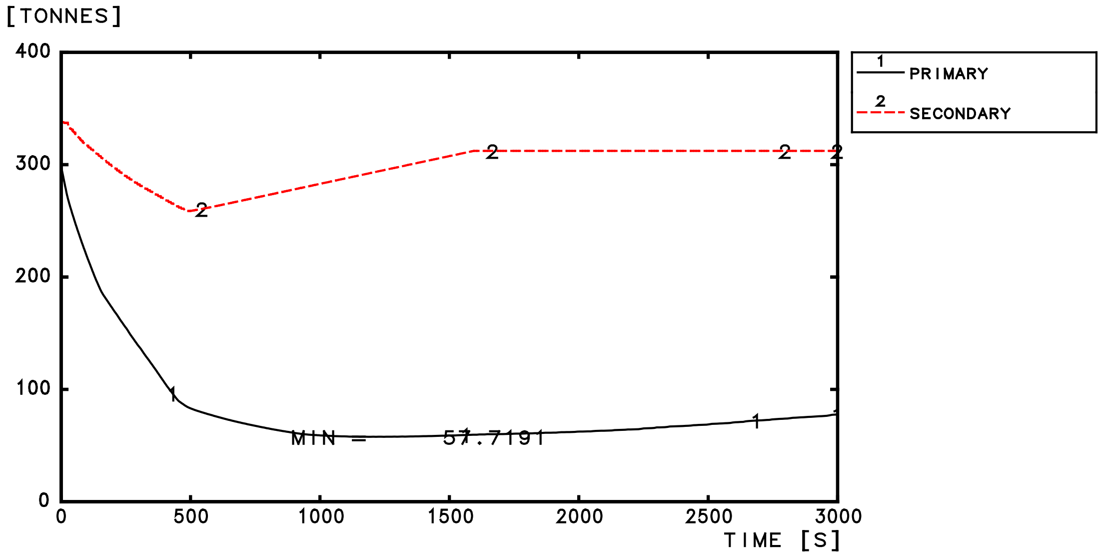
SECOND BREAK SPECTRUM: 120 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



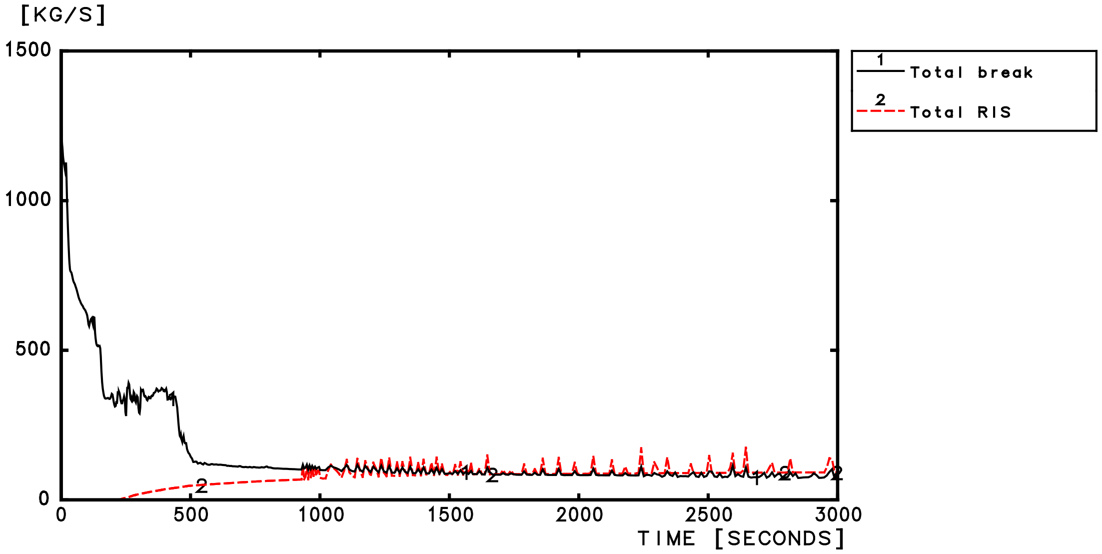
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 44

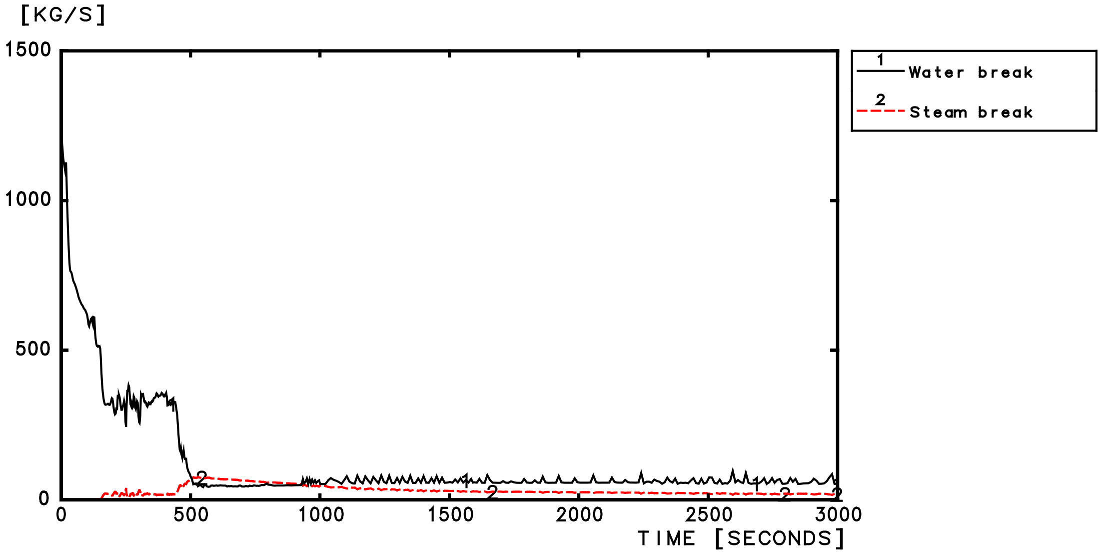
SECOND BREAK SPECTRUM: 120 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



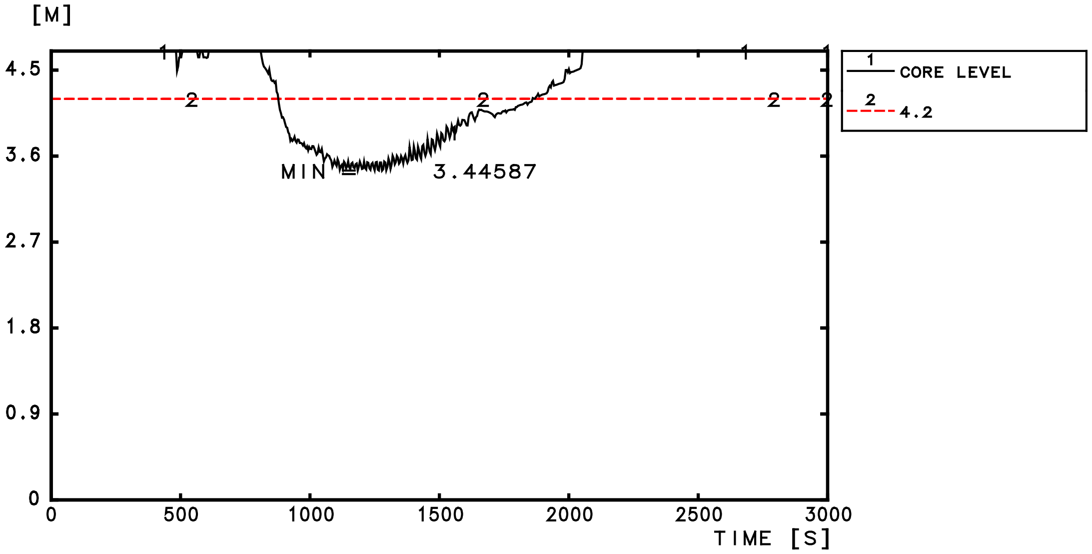
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 45

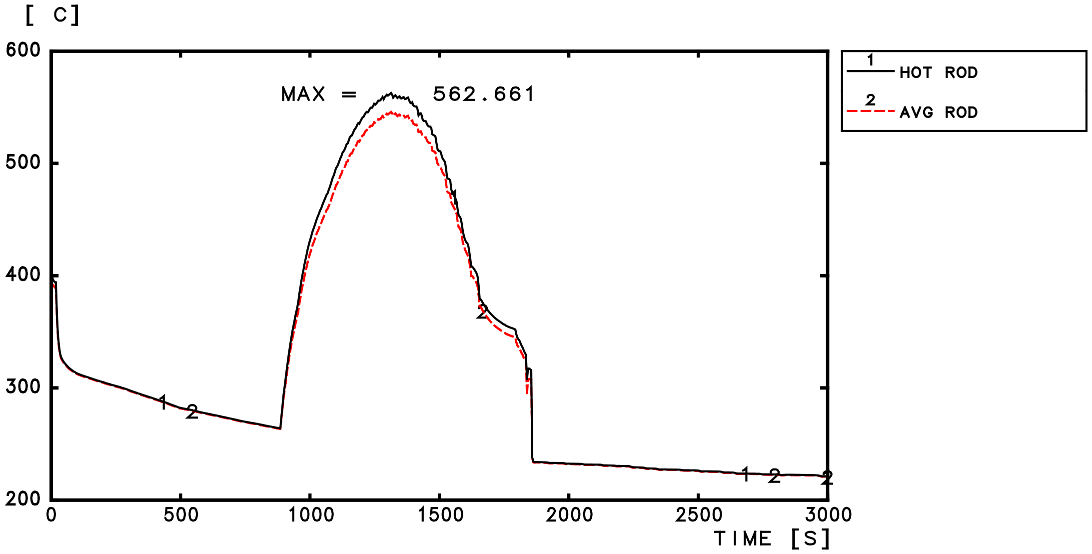
SECOND BREAK SPECTRUM: 120 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

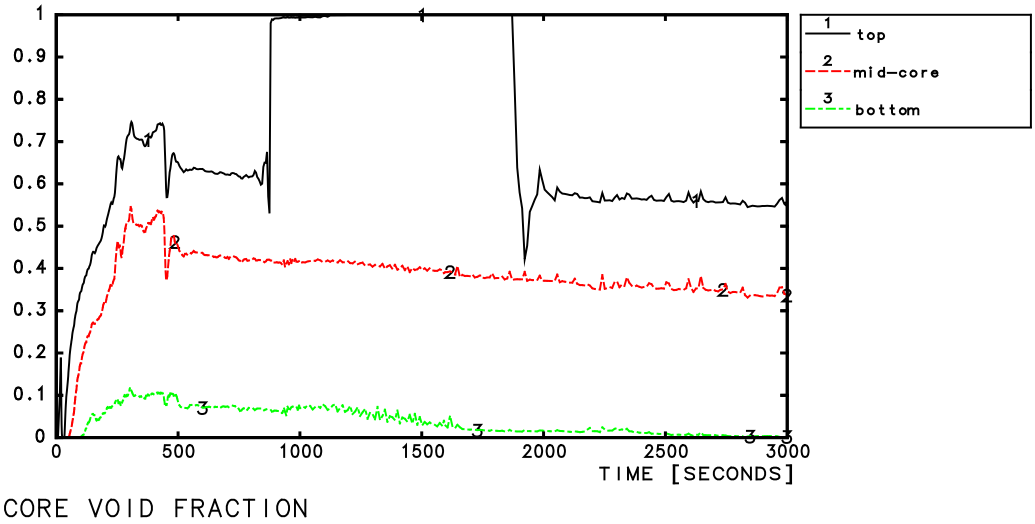


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 46

SECOND BREAK SPECTRUM: 120 CM² BREAK

CORE VOID FRACTION

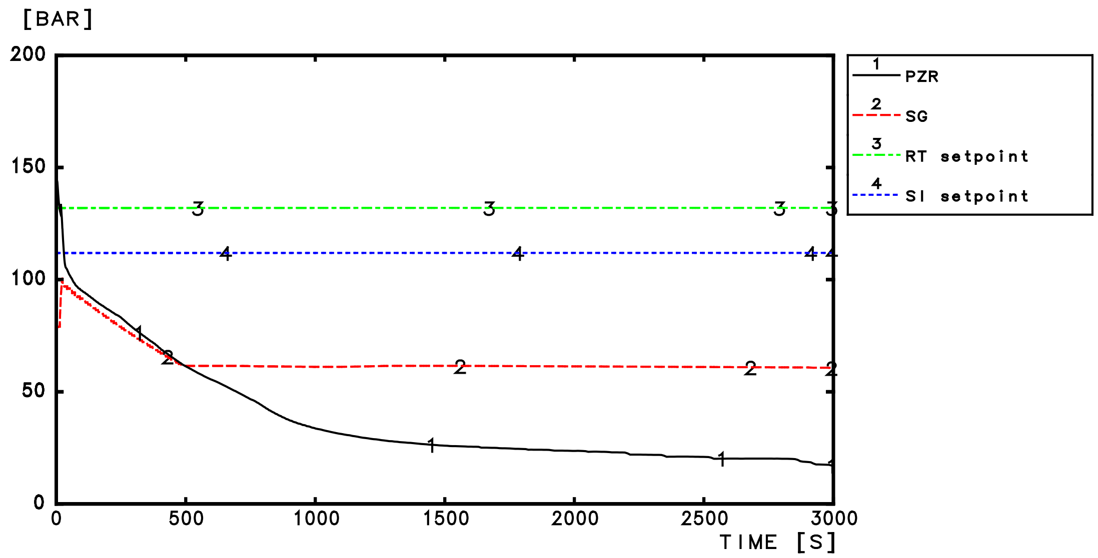


SECTION 14.5.6 - FIGURE 47

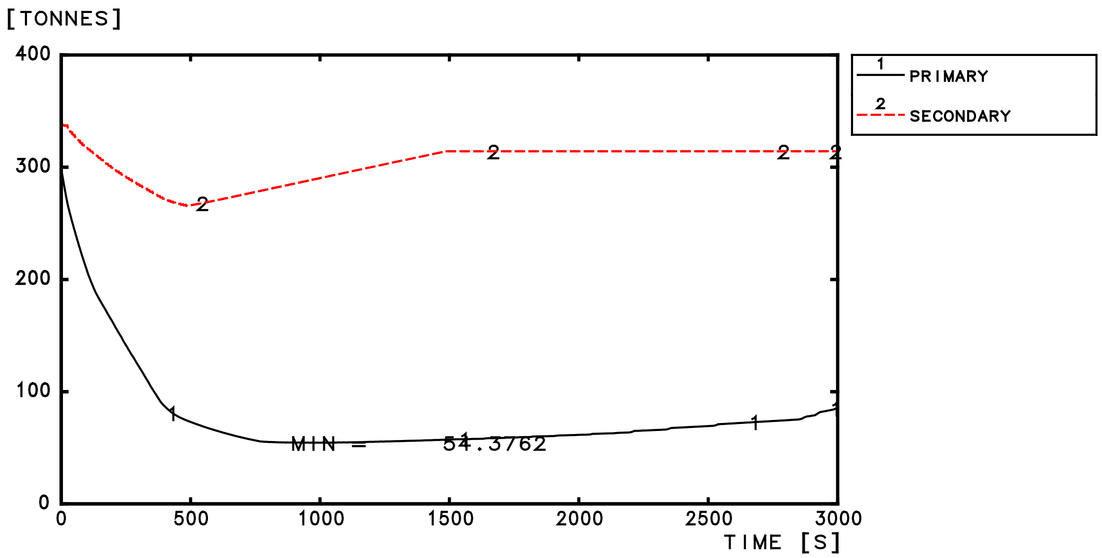
SECOND BREAK SPECTRUM: 140 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



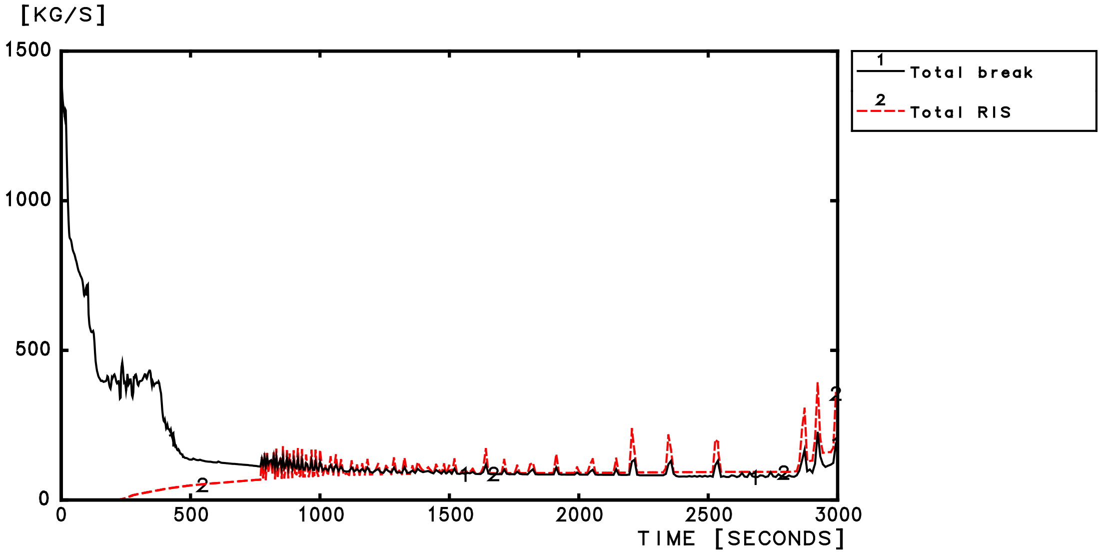
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 48

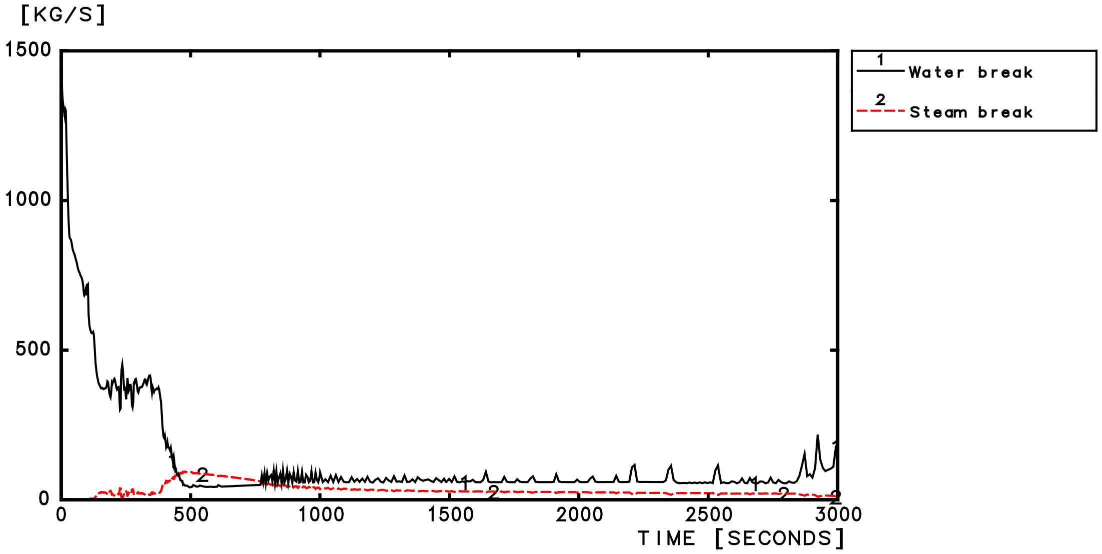
SECOND BREAK SPECTRUM: 140 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



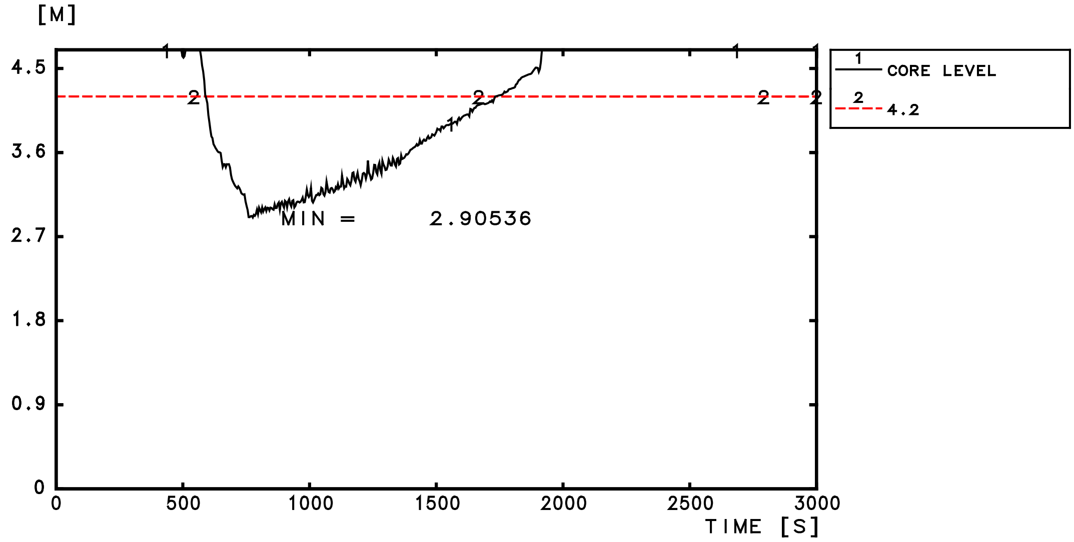
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 49

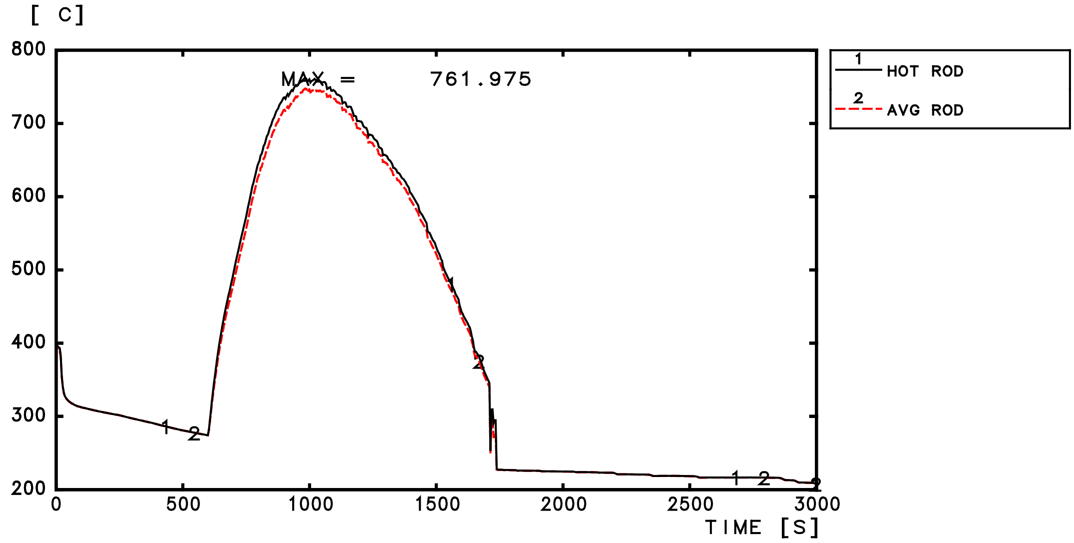
SECOND BREAK SPECTRUM: 140 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

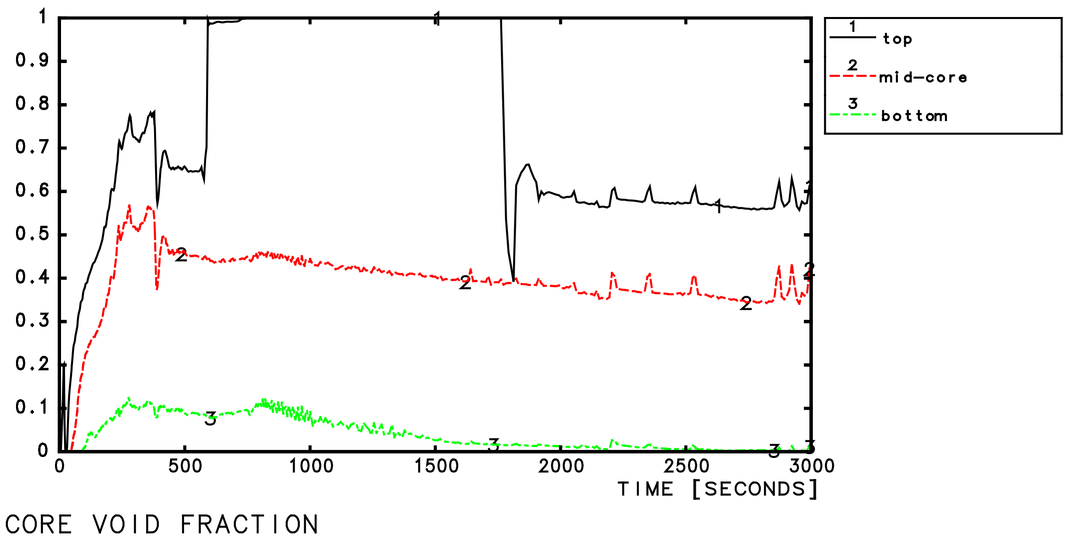


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 50

SECOND BREAK SPECTRUM: 140 CM² BREAK

CORE VOID FRACTION

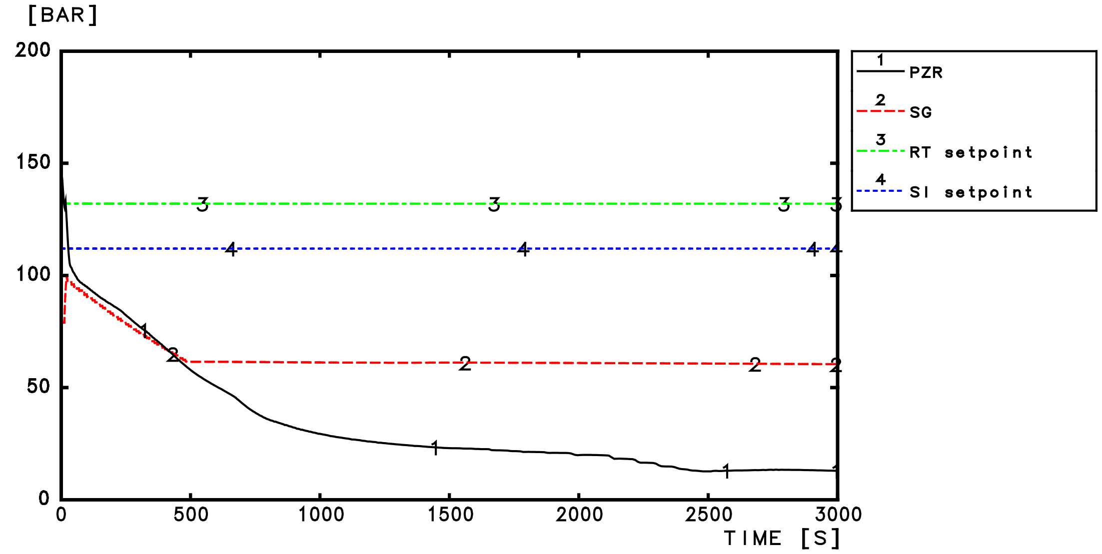


SECTION 14.5.6 - FIGURE 51

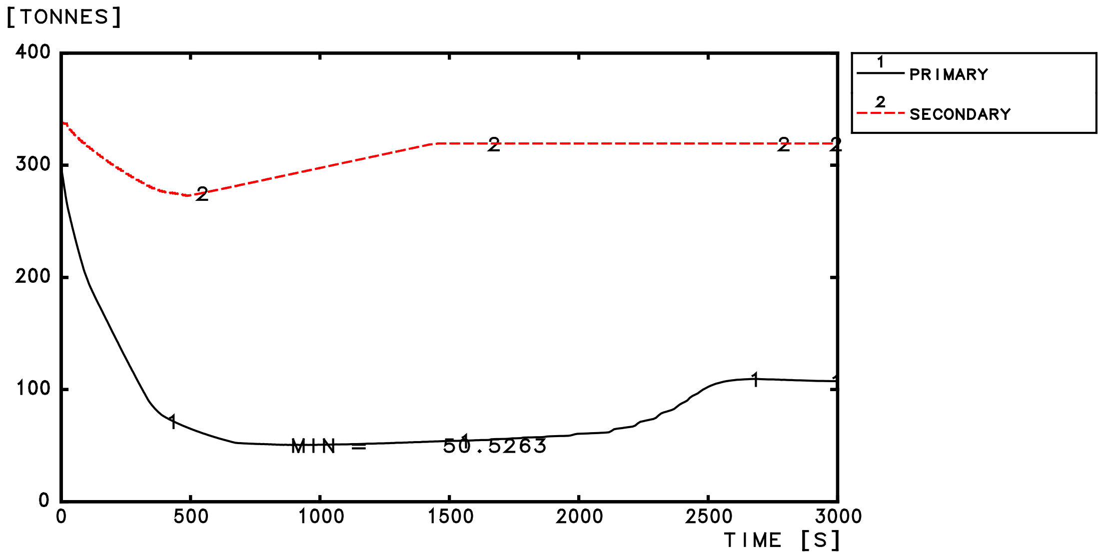
SECOND BREAK SPECTRUM: 160 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



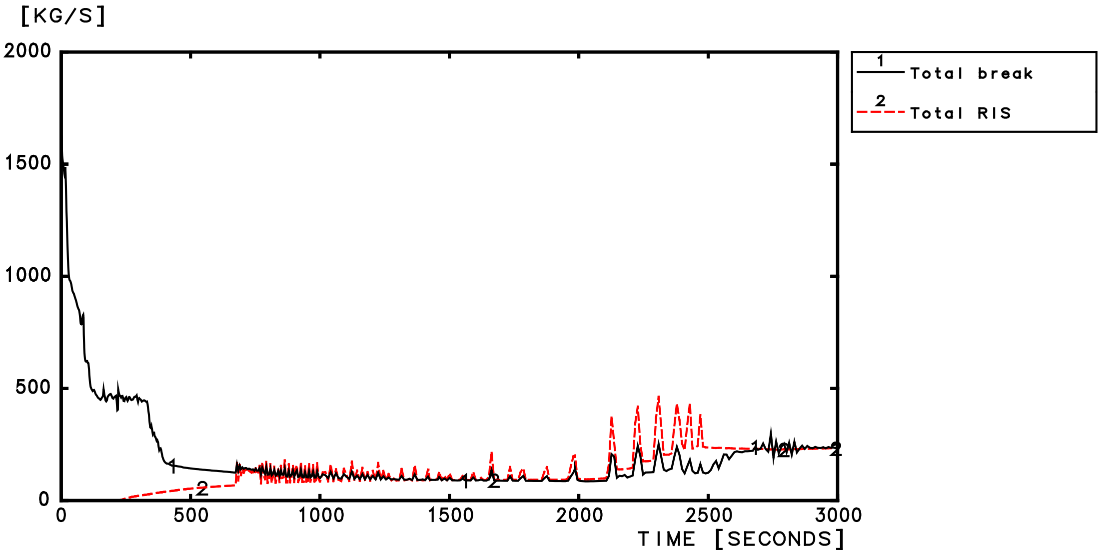
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 52

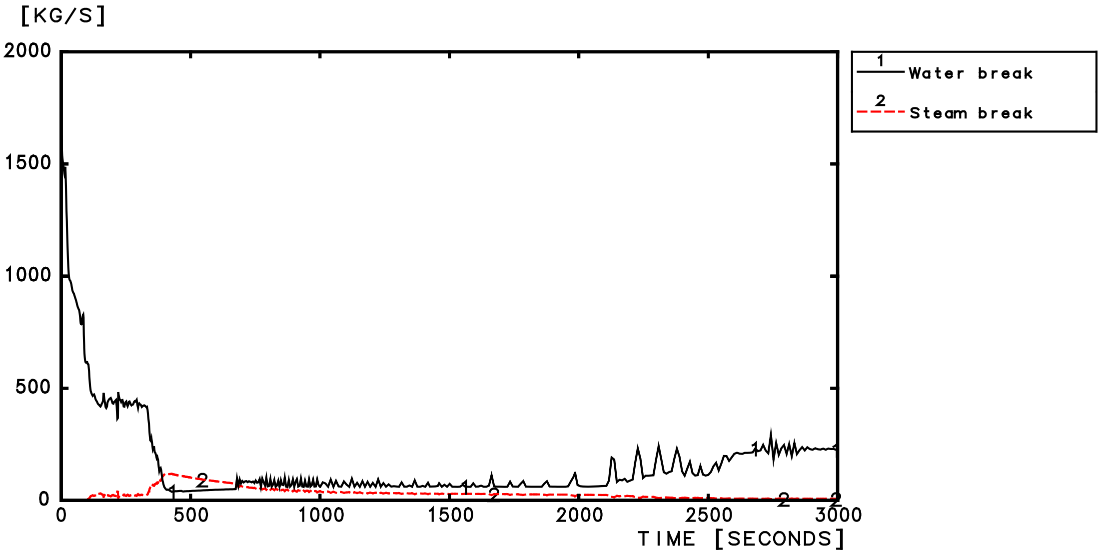
SECOND BREAK SPECTRUM: 160 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



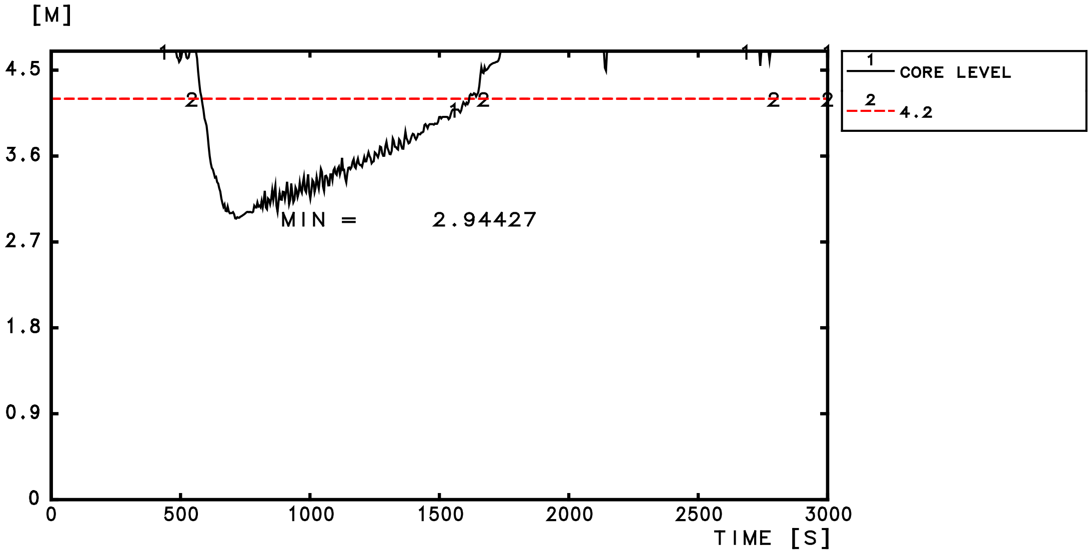
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 53

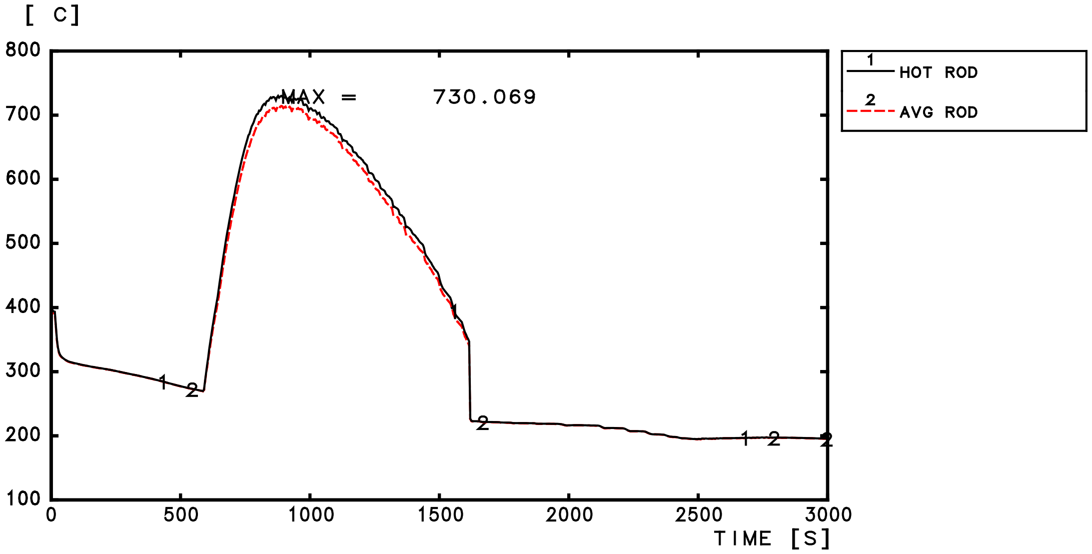
SECOND BREAK SPECTRUM: 160 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

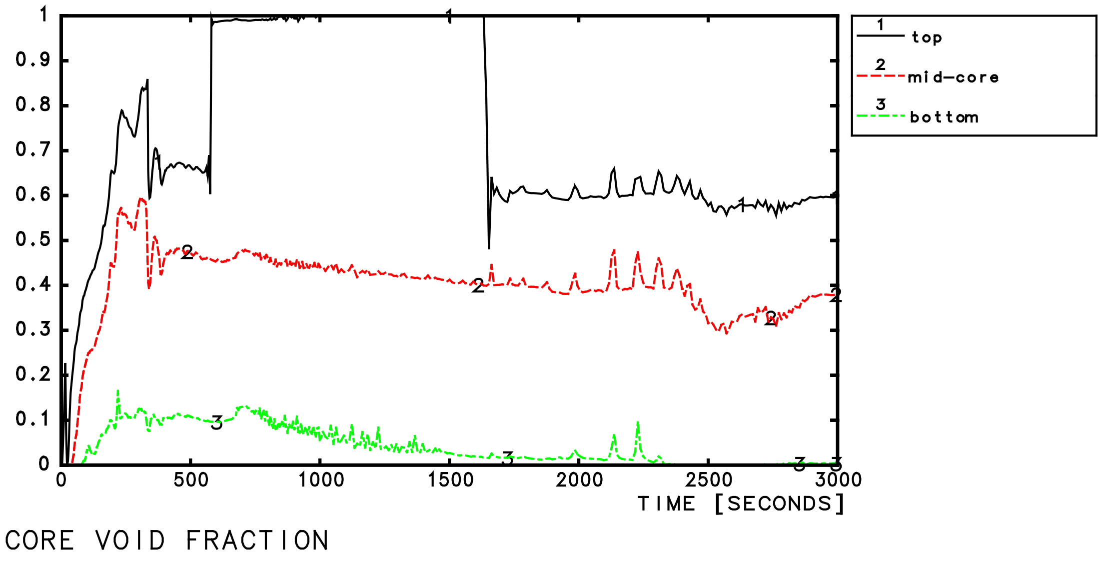


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 54

SECOND BREAK SPECTRUM: 160 CM² BREAK

CORE VOID FRACTION

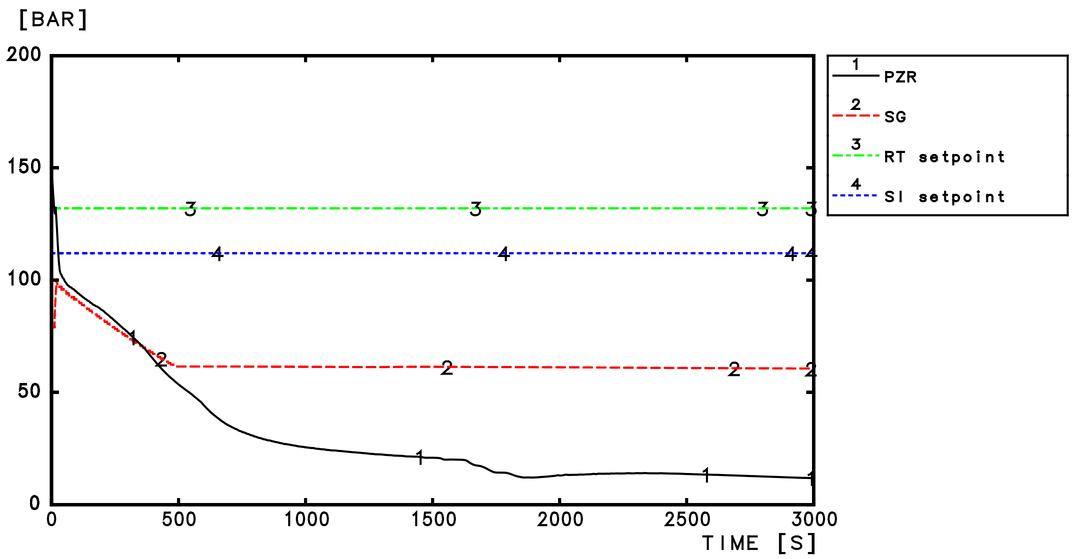


SECTION 14.5.6 - FIGURE 55

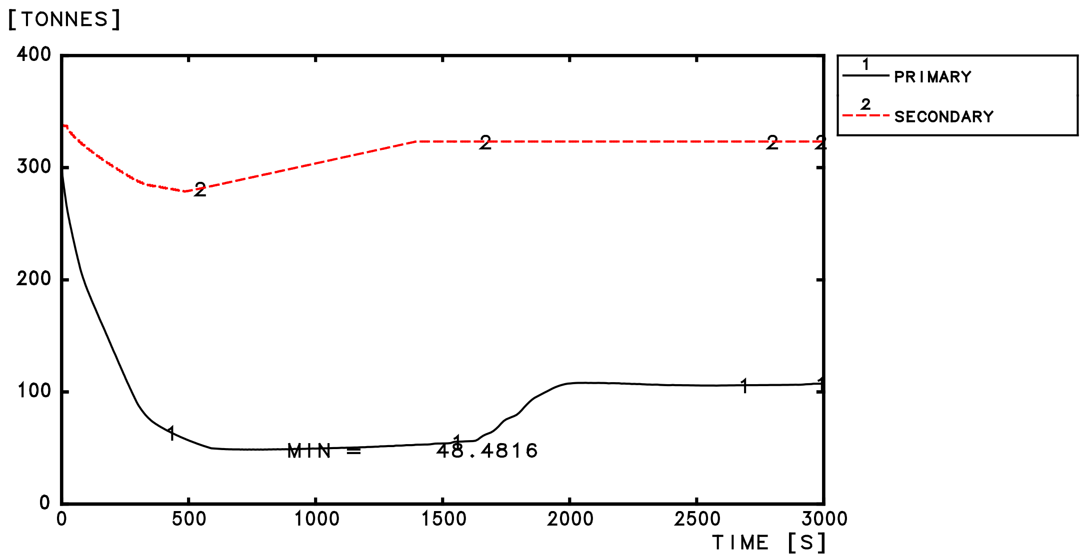
SECOND BREAK SPECTRUM: 180 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



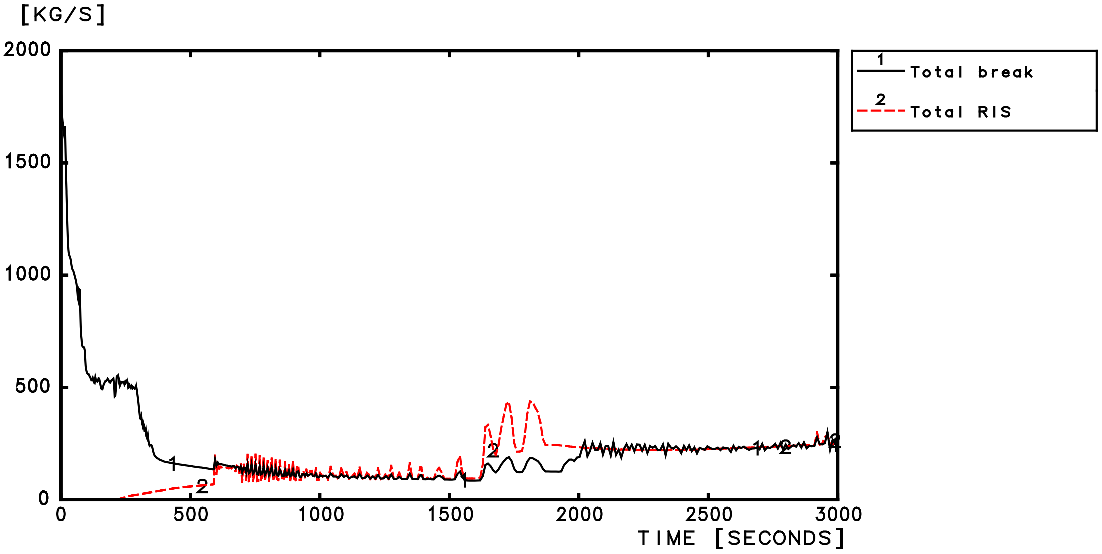
Primary and secondary Masses

SECTION 14.5.6 - FIGURE 56

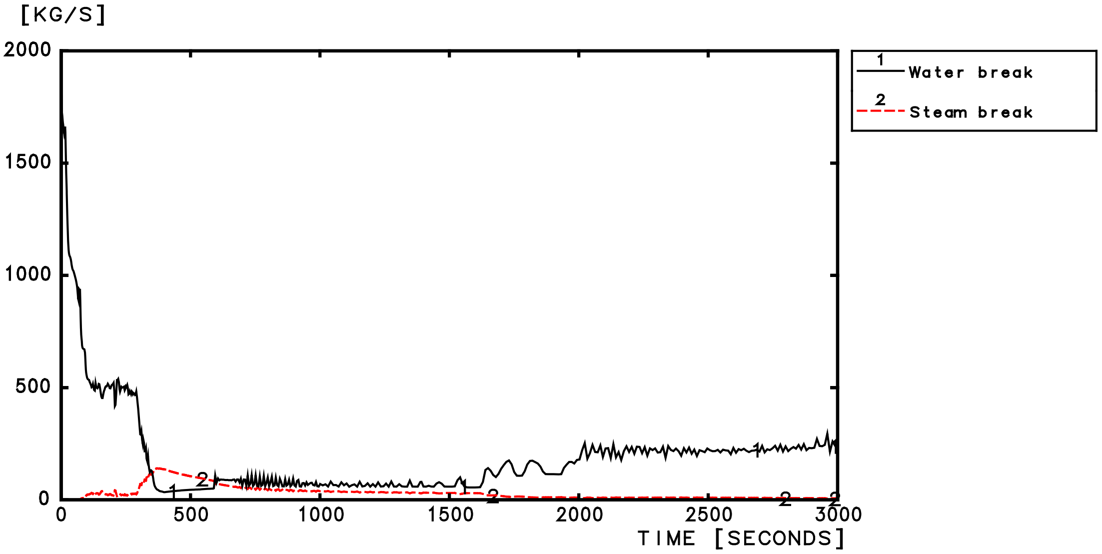
SECOND BREAK SPECTRUM: 180 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



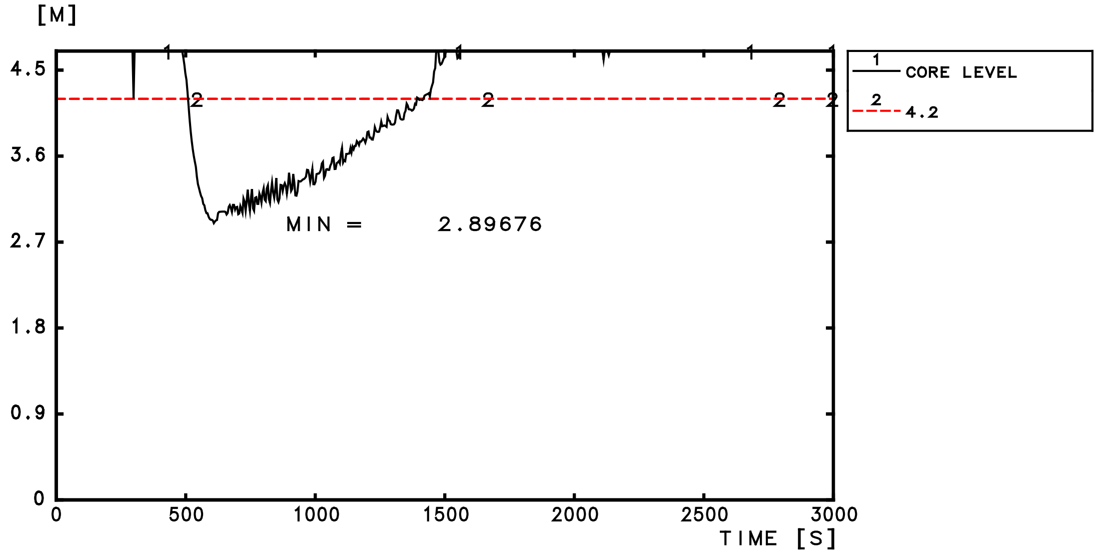
Water and steam break flowrate

SECTION 14.5.6 - FIGURE 57

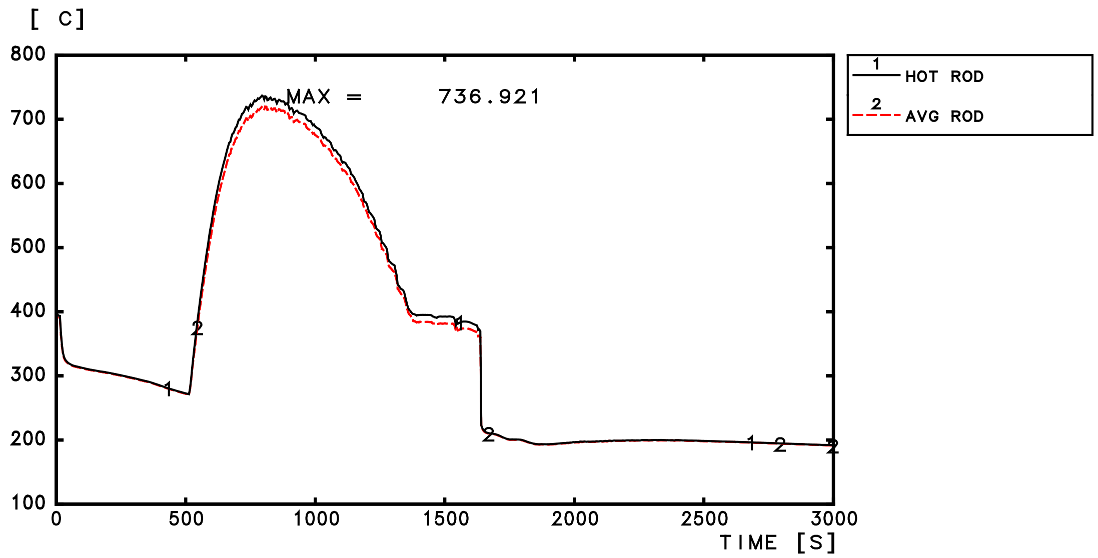
SECOND BREAK SPECTRUM: 180 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

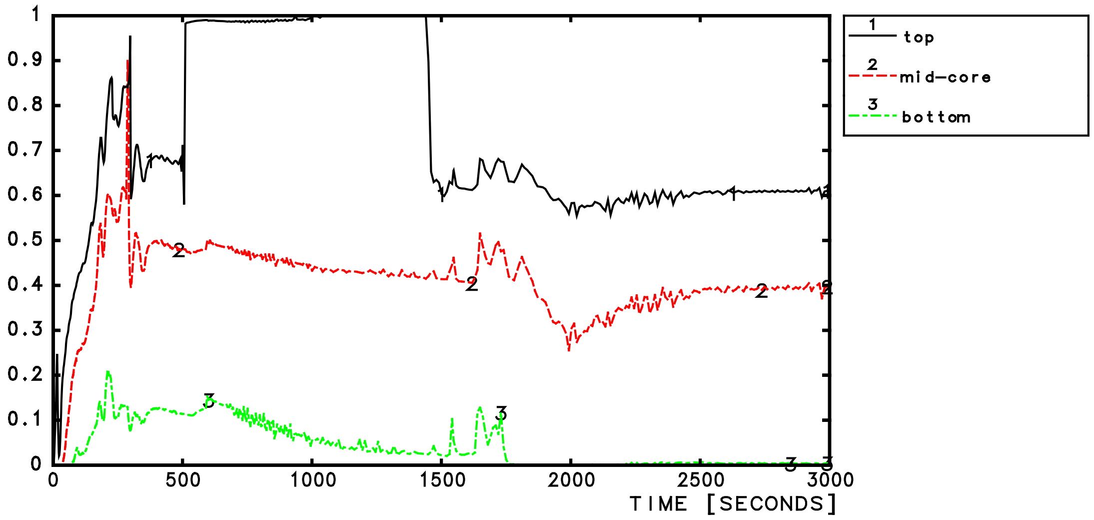


Maximum clad temperatures

SECTION 14.5.6 - FIGURE 58

SECOND BREAK SPECTRUM: 180 CM² BREAK

CORE VOID FRACTION



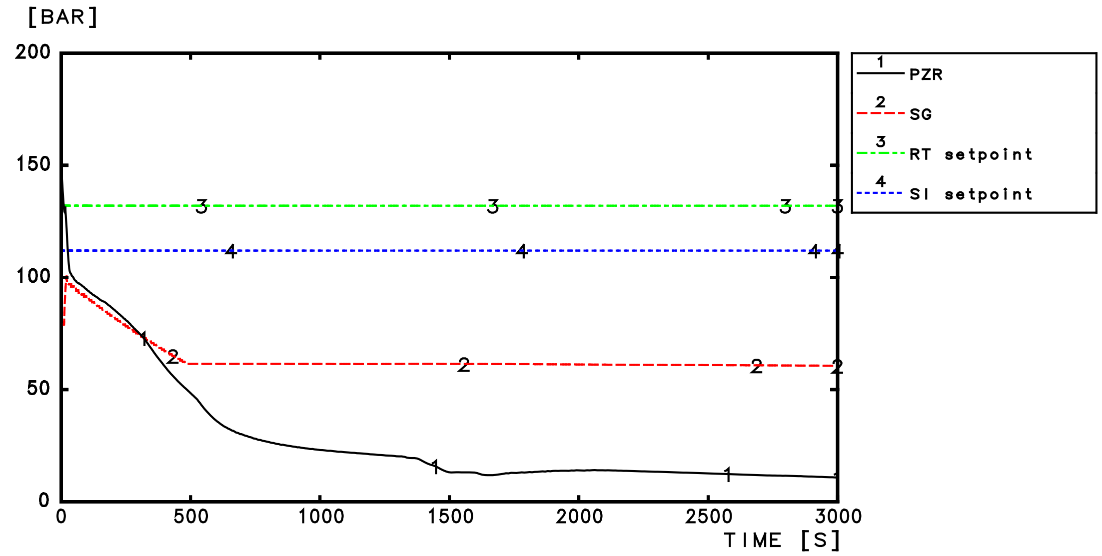
CORE VOID FRACTION

SECTION 14.5.6 - FIGURE 59

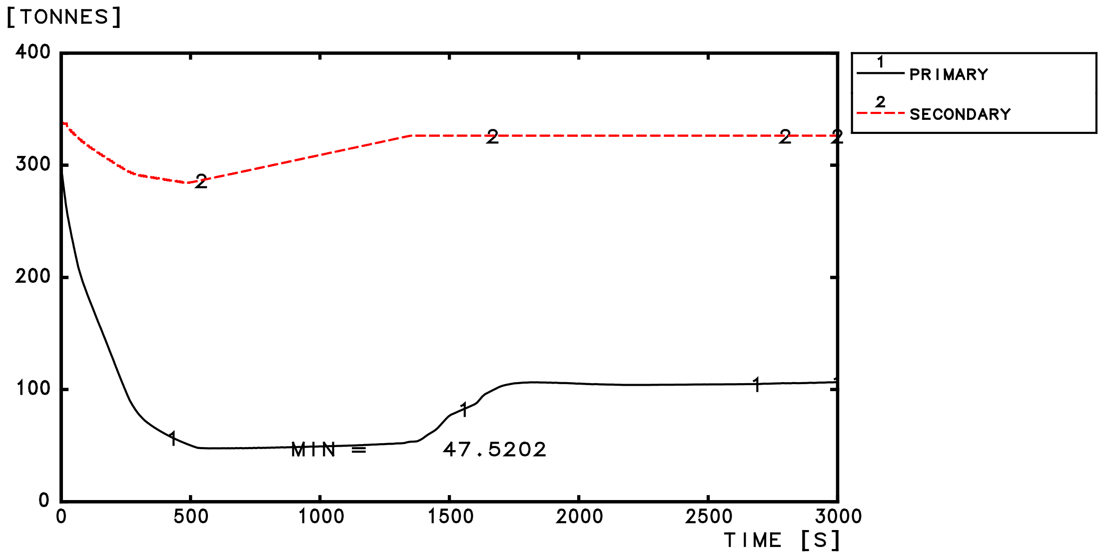
SECOND BREAK SPECTRUM: 200 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



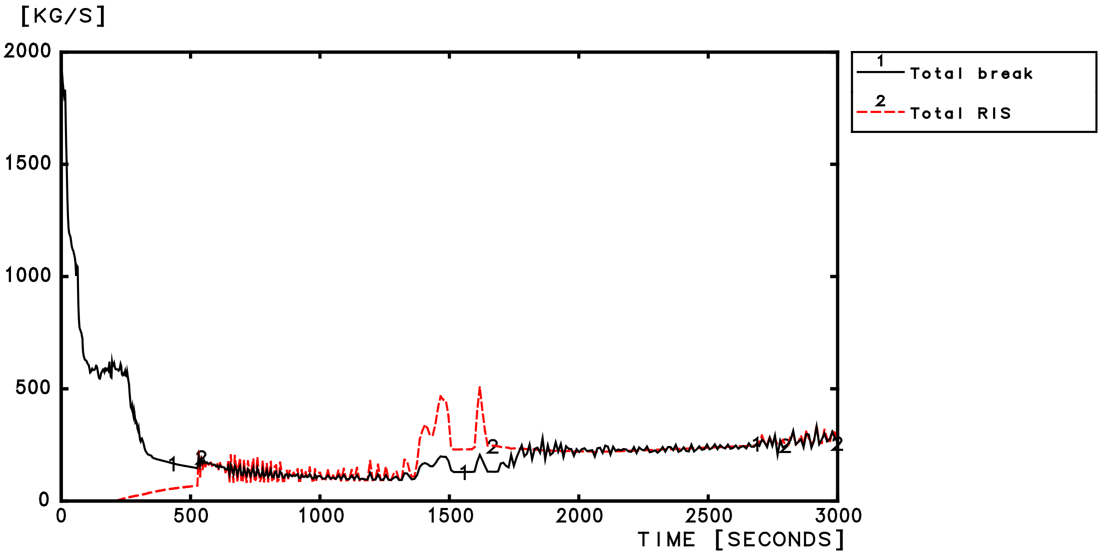
Primary and secondary Masses

SECTION 14.5.6 – FIGURE 60

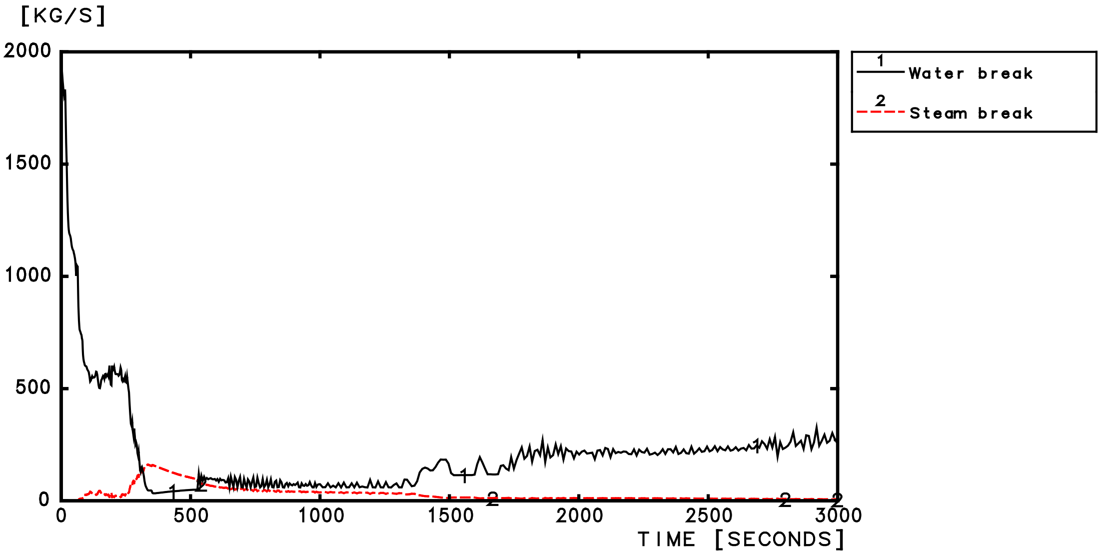
SECOND BREAK SPECTRUM: 200 CM² BREAK

- TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



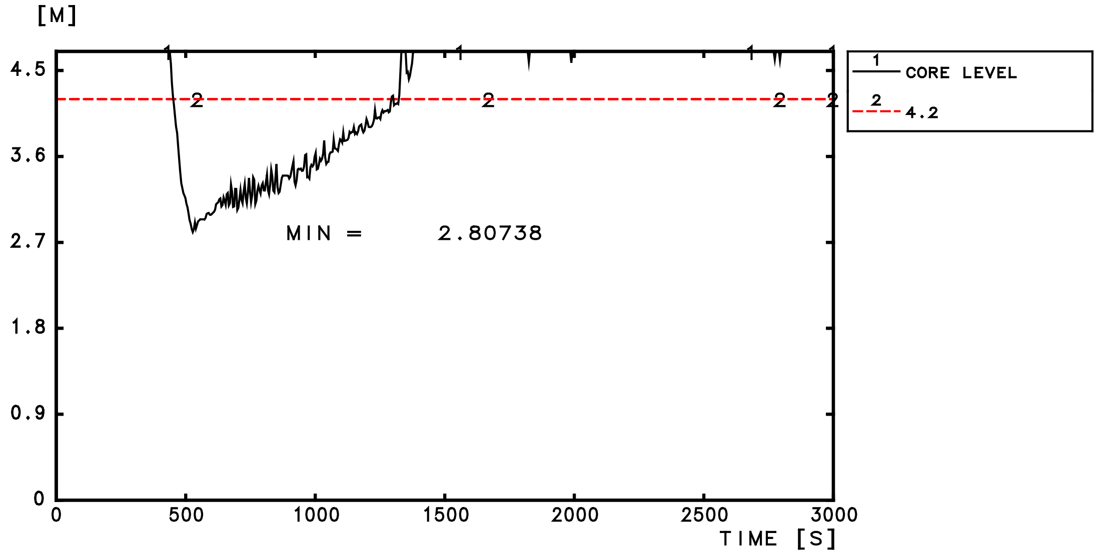
Water and steam break flowrate

SECTION 14.5.6 – FIGURE 61

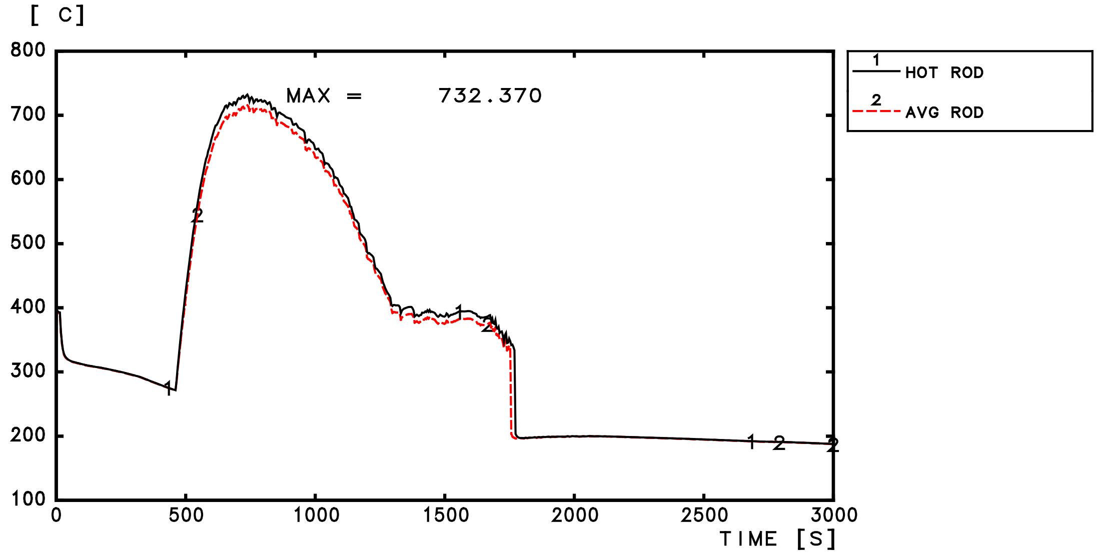
SECOND BREAK SPECTRUM: 200 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

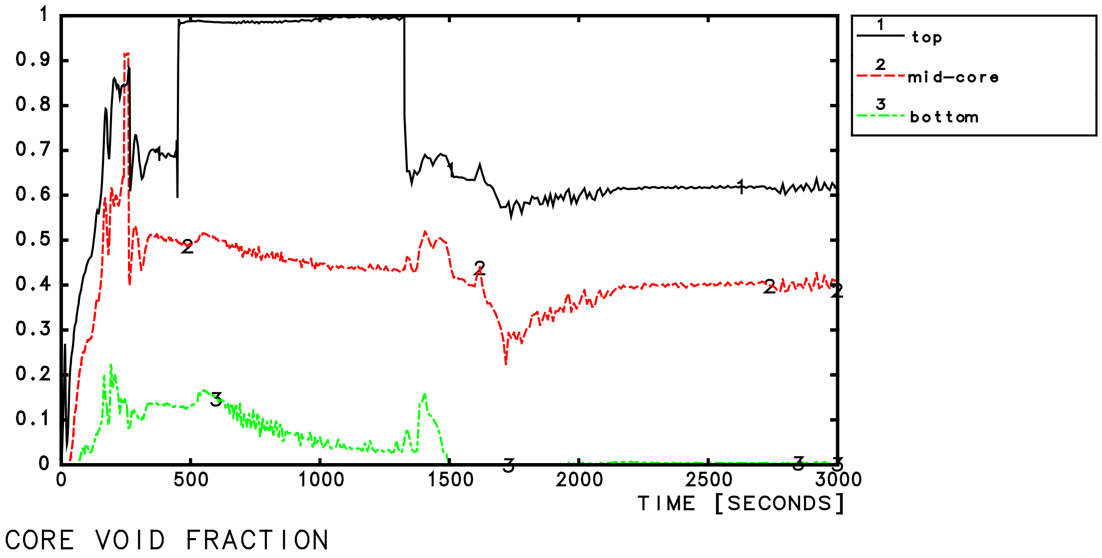


Maximum clad temperatures

SECTION 14.5.6 – FIGURE 62

SECOND BREAK SPECTRUM: 200 CM² BREAK

CORE VOID FRACTION

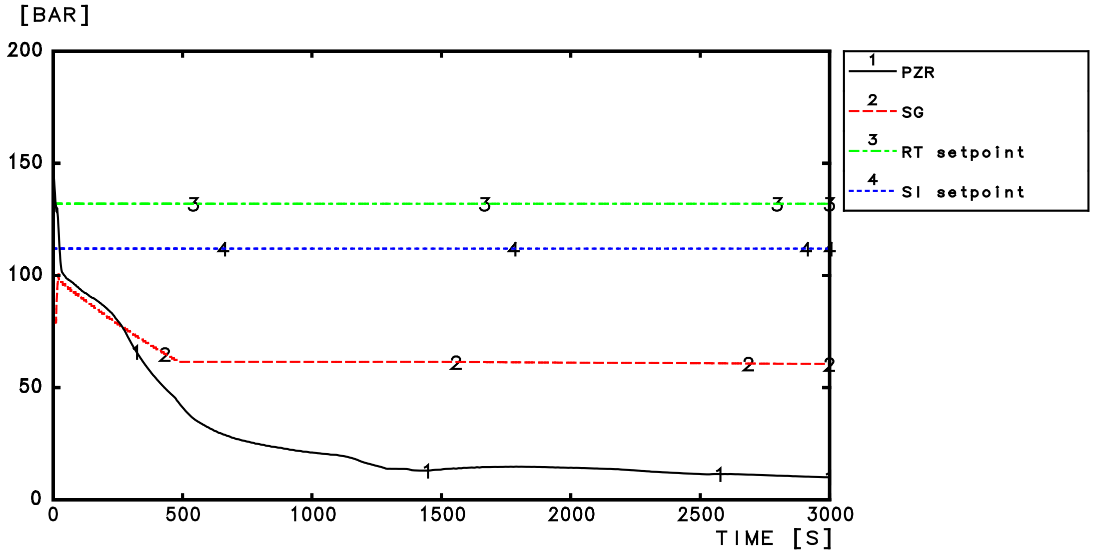


SECTION 14.5.6 – FIGURE 63

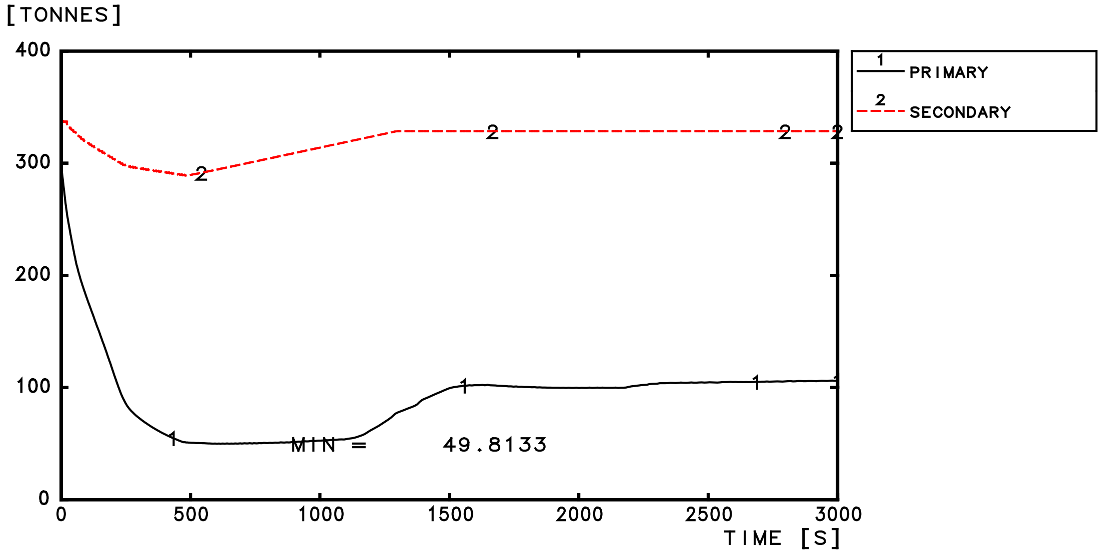
SECOND BREAK SPECTRUM: 220 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



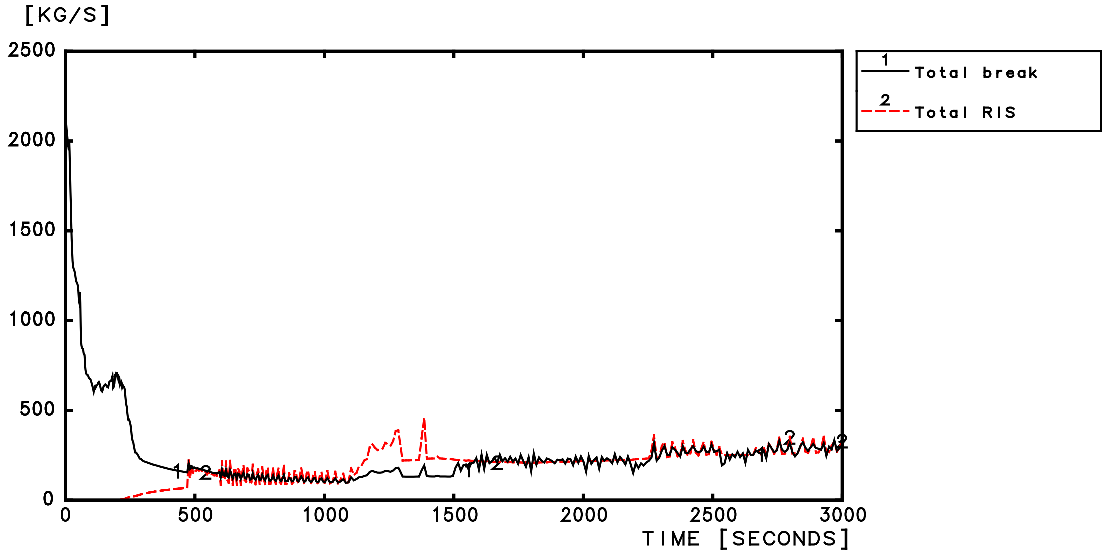
Primary and secondary Masses

SECTION 14.5.6 – FIGURE 64

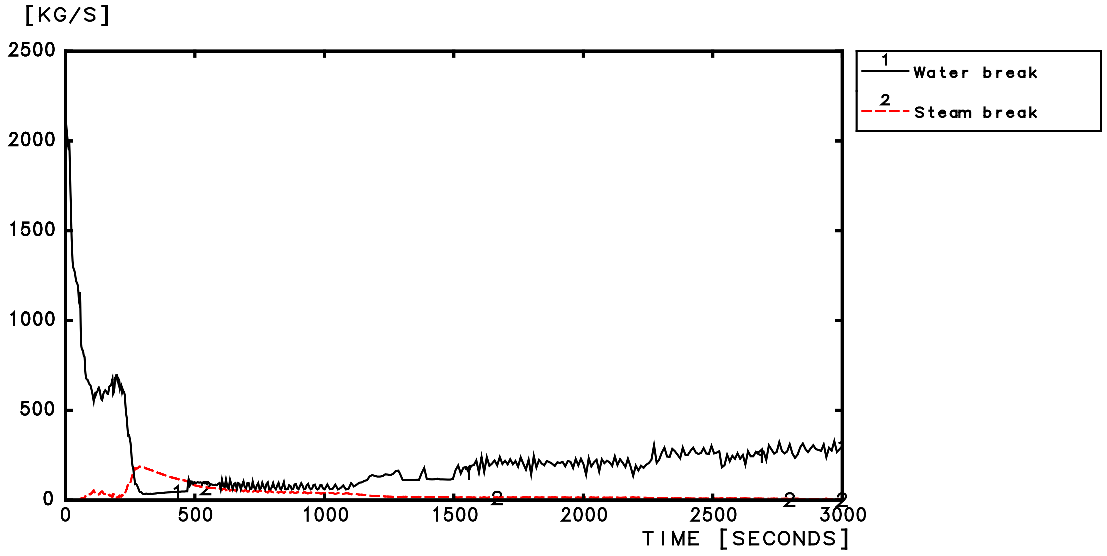
SECOND BREAK SPECTRUM: 220 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



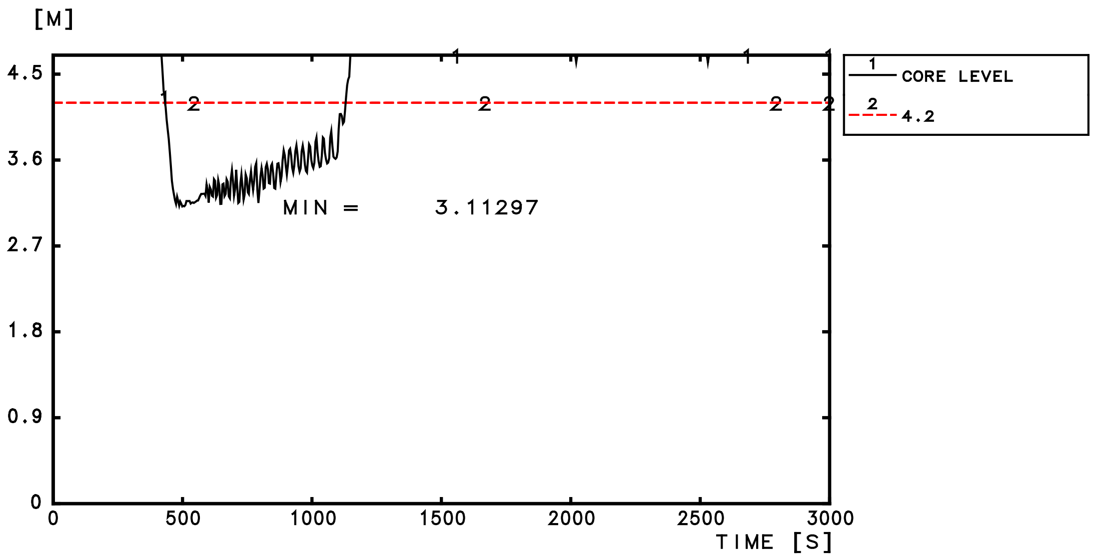
Water and steam break flowrate

SECTION 14.5.6 – FIGURE 65

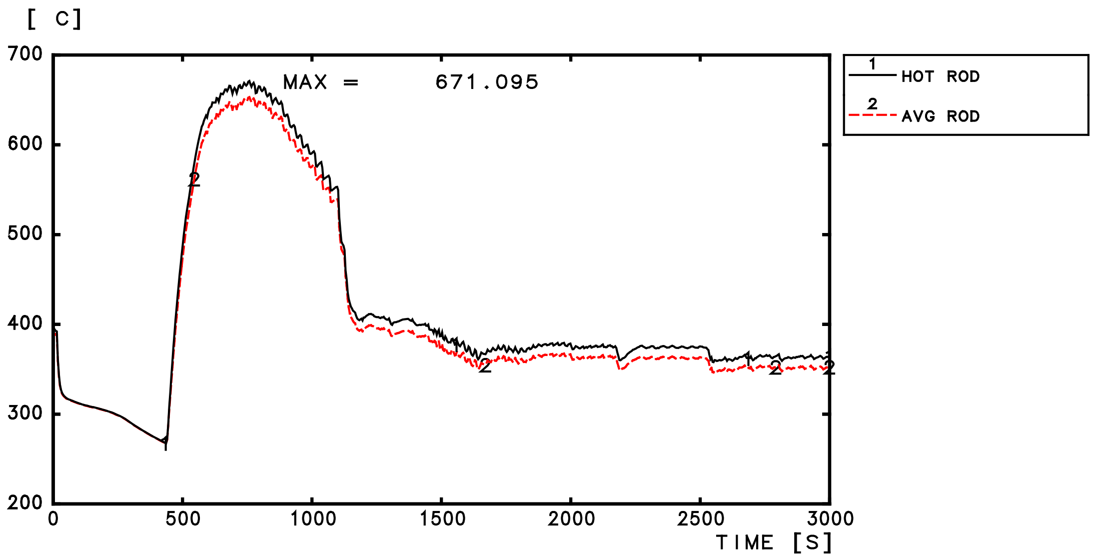
SECOND BREAK SPECTRUM: 220 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

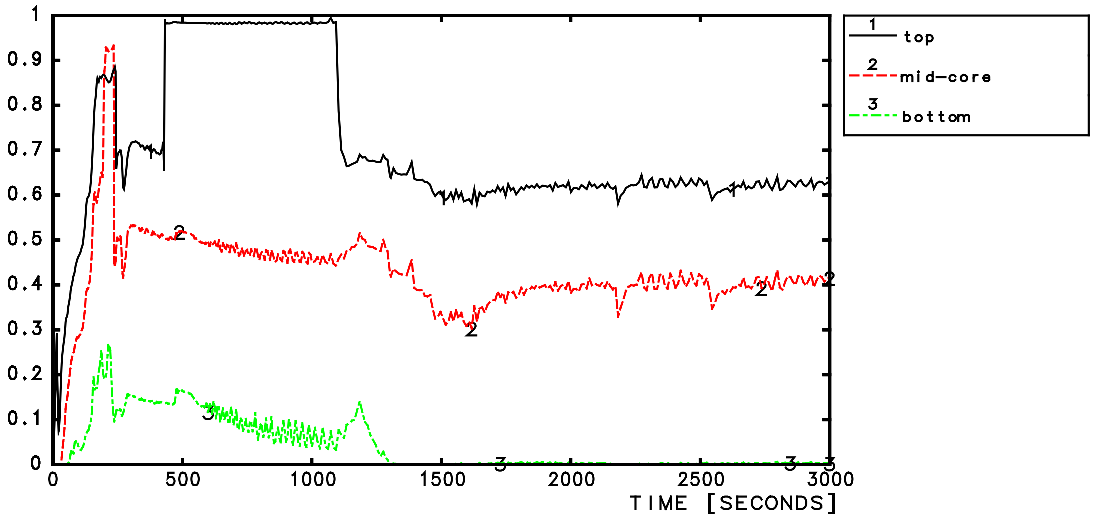


Maximum clad temperatures

SECTION 14.5.6 – FIGURE 66

SECOND BREAK SPECTRUM: 220 CM² BREAK

CORE VOID FRACTION



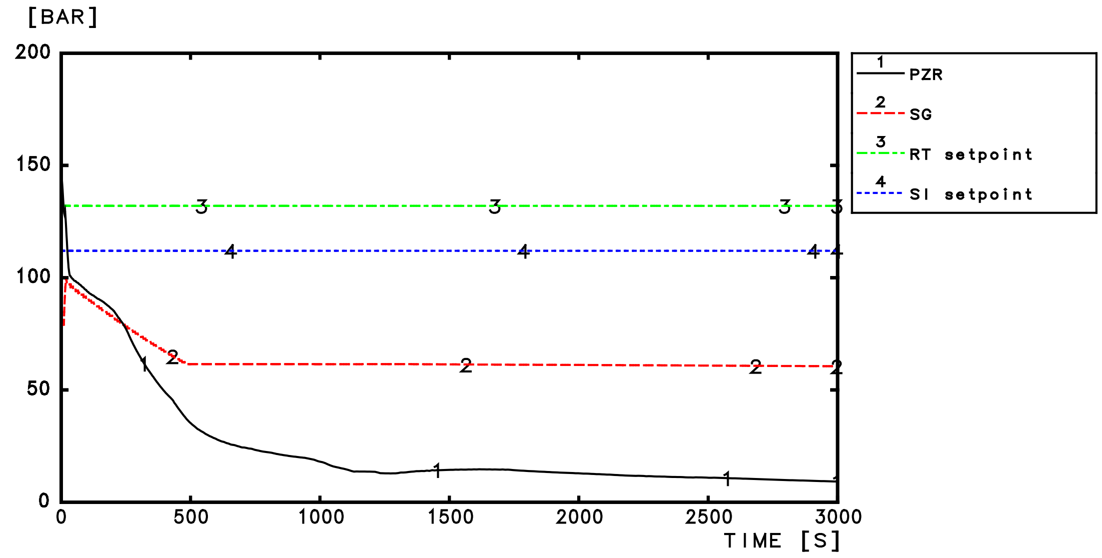
CORE VOID FRACTION

SECTION 14.5.6 – FIGURE 67

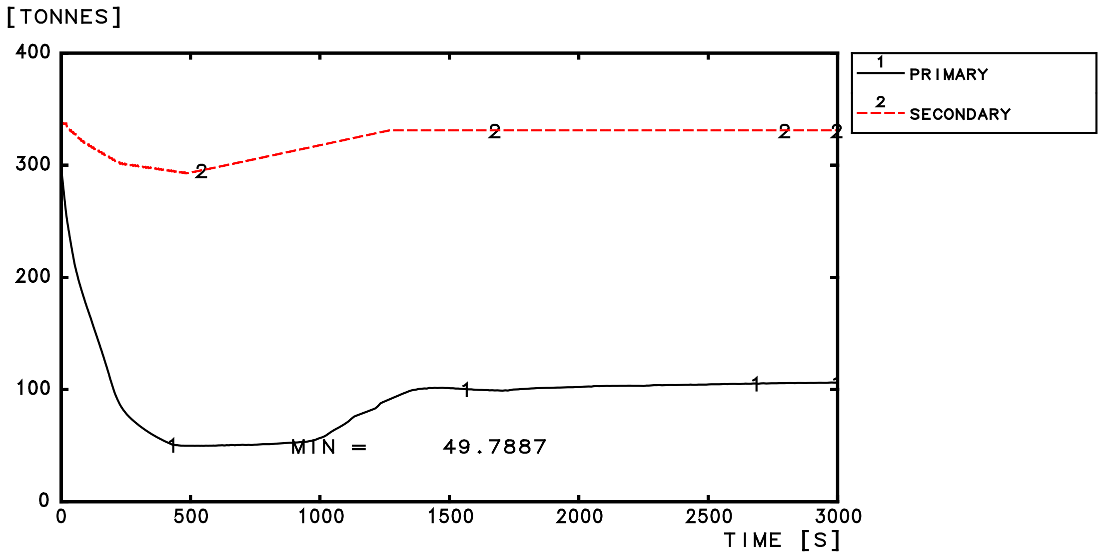
SECOND BREAK SPECTRUM: 240 CM² BREAK

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



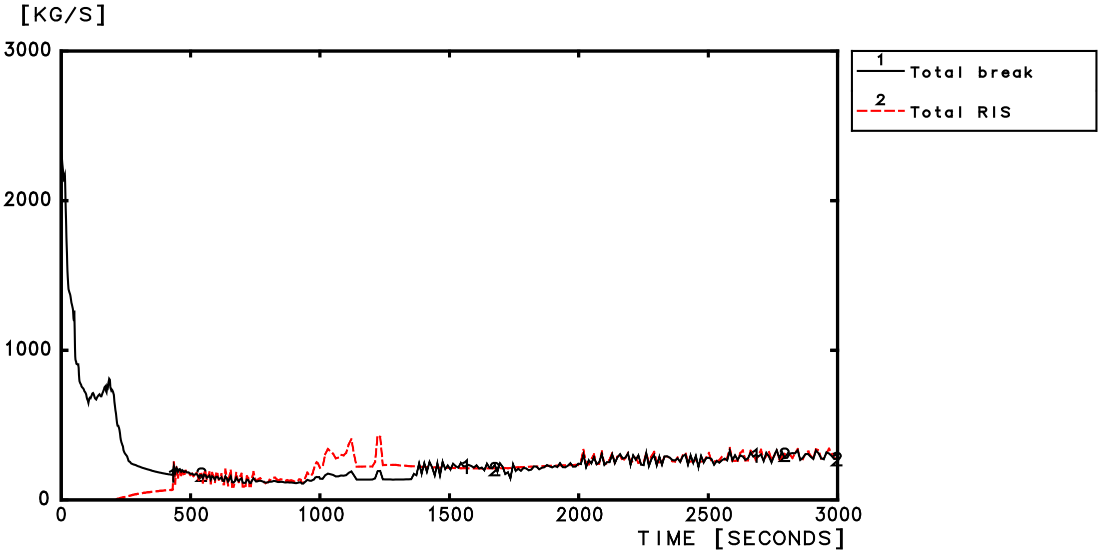
Primary and secondary Masses

SECTION 14.5.6 – FIGURE 68

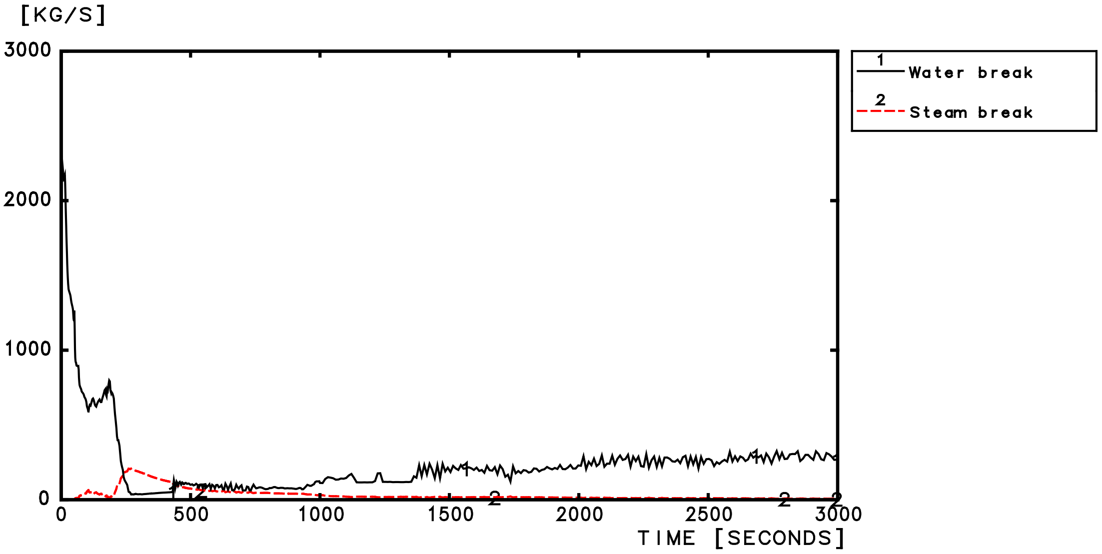
SECOND BREAK SPECTRUM: 240 CM² BREAK

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



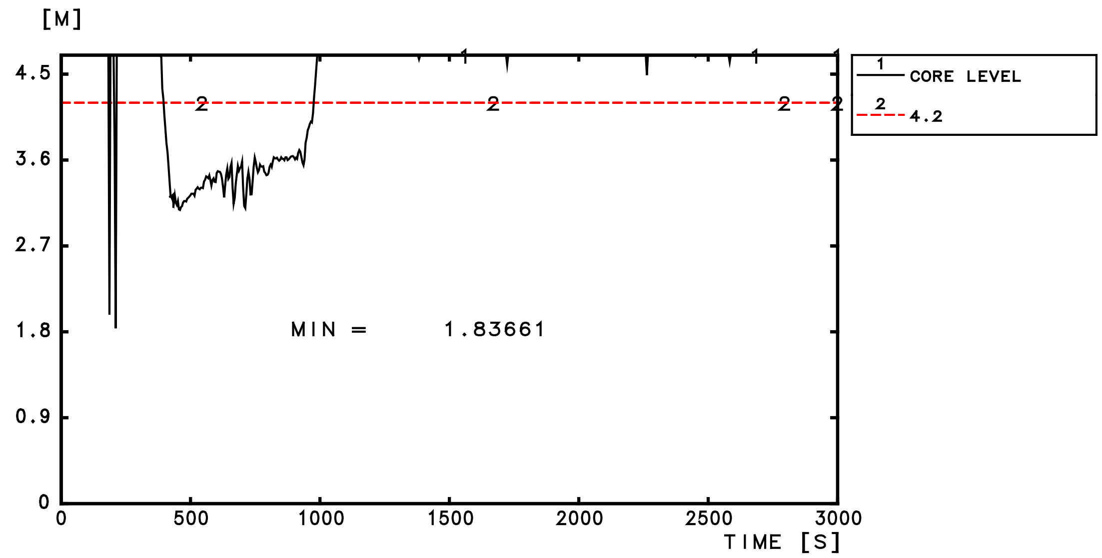
Water and steam break flowrate

SECTION 14.5.6 – FIGURE 69

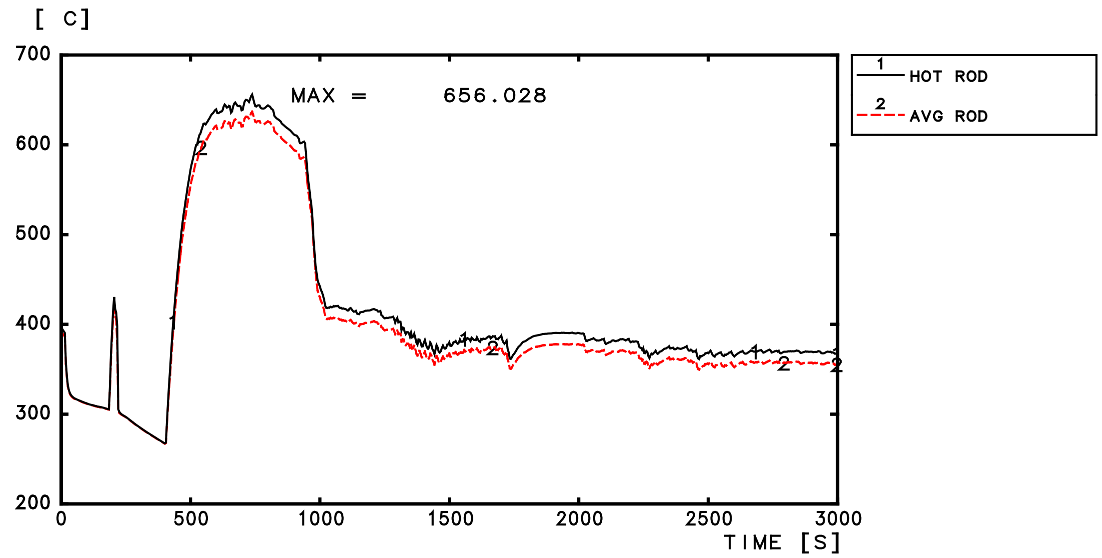
SECOND BREAK SPECTRUM: 240 CM² BREAK

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

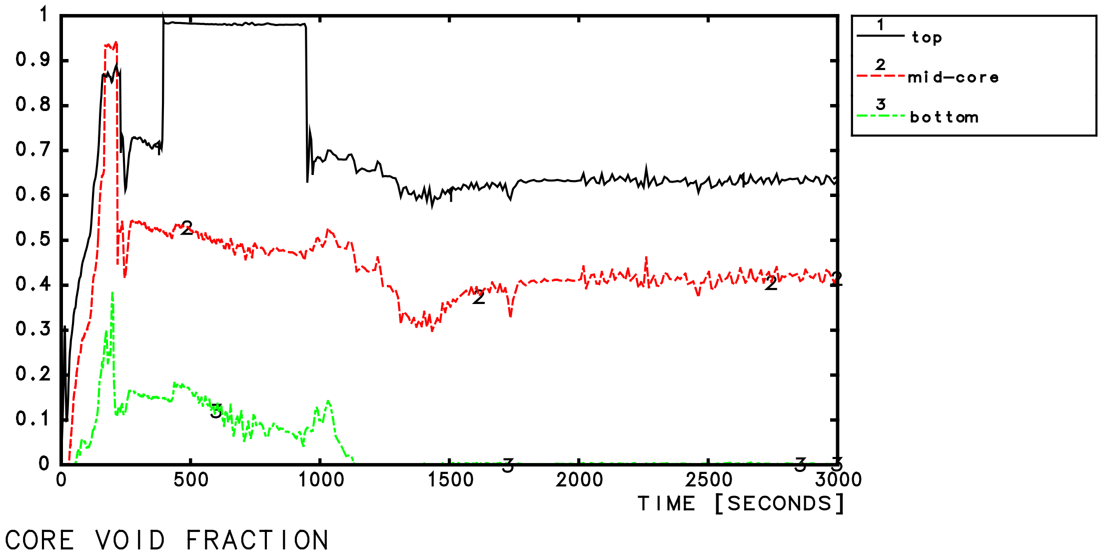


Maximum clad temperatures

SECTION 14.5.6 – FIGURE 70

SECOND BREAK SPECTRUM: 240 CM² BREAK

CORE VOID FRACTION

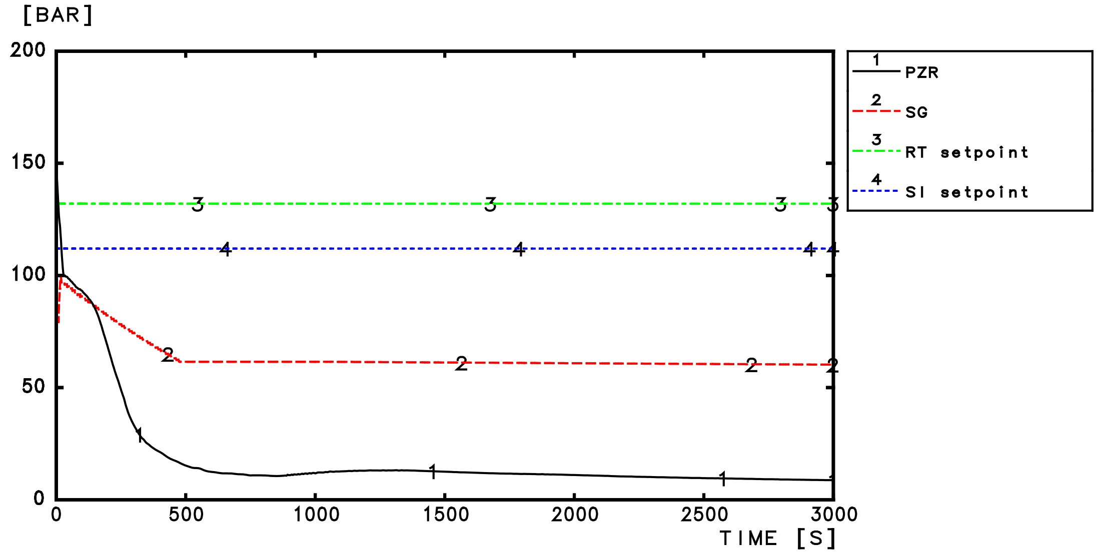


SECTION 14.5.6 – FIGURE 71

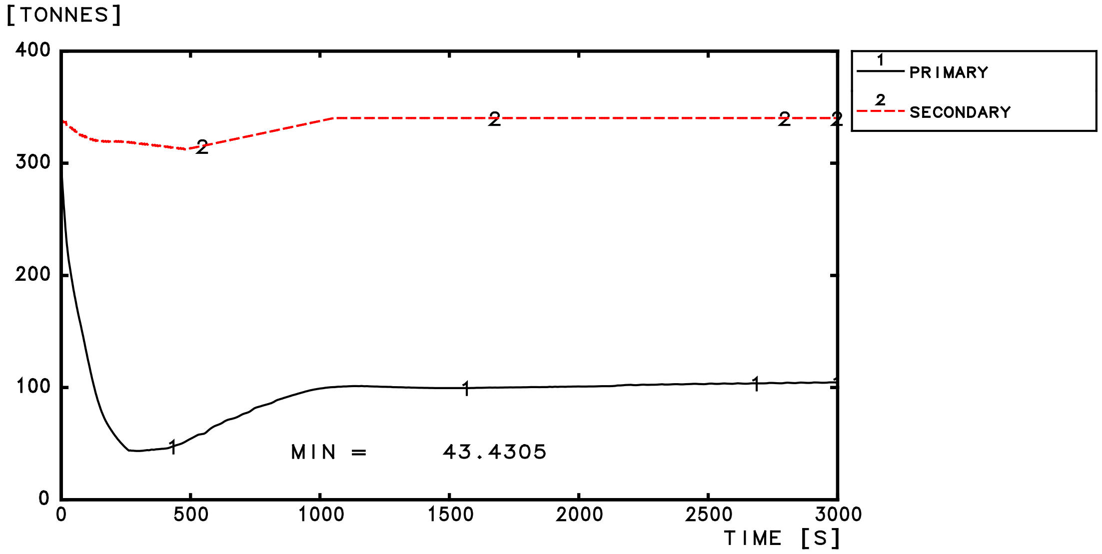
SECOND BREAK SPECTRUM: 390 CM² BREAK RIS [SIS]

PRIMARY AND SECONDARY PRESSURES

PRIMARY AND SECONDARY SIDE WATER INVENTORIES



Primary and secondary pressures



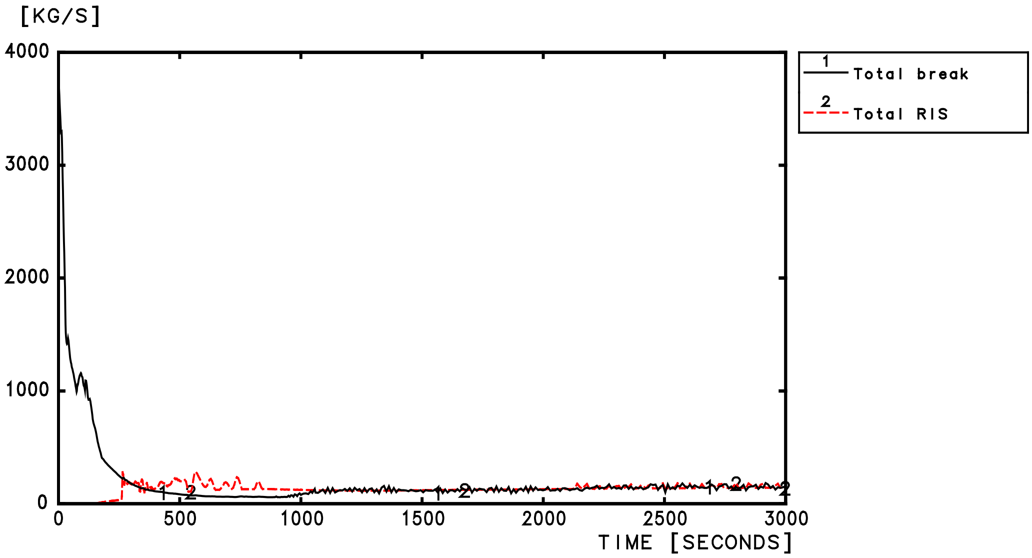
Primary and secondary Masses

SECTION 14.5.6 – FIGURE 72

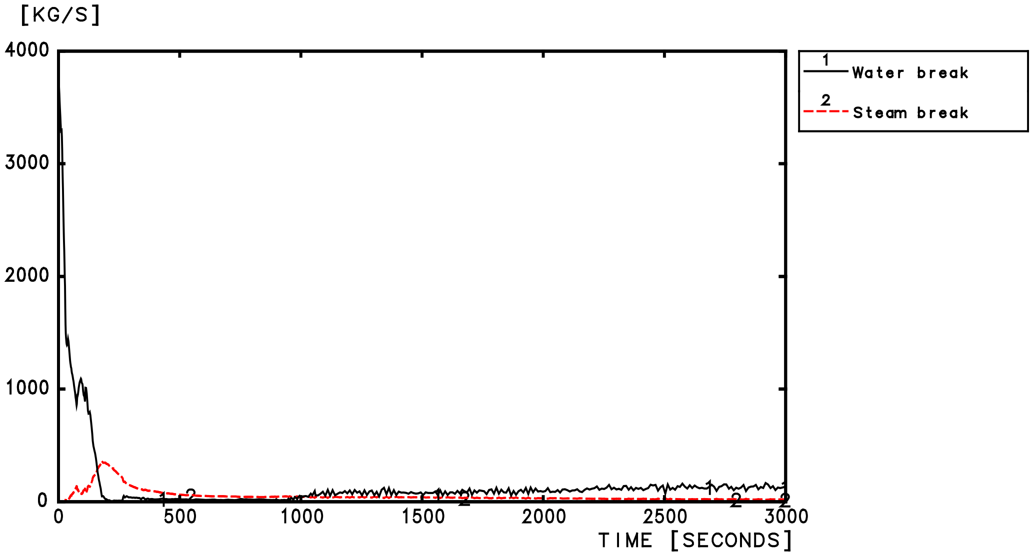
SECOND BREAK SPECTRUM: 390 CM² BREAK RIS [SIS]

TOTAL BREAK AND RIS [SIS] FLOW RATES

WATER AND STEAM BREAK FLOW RATES



Break flowrate and total RIS injection rate



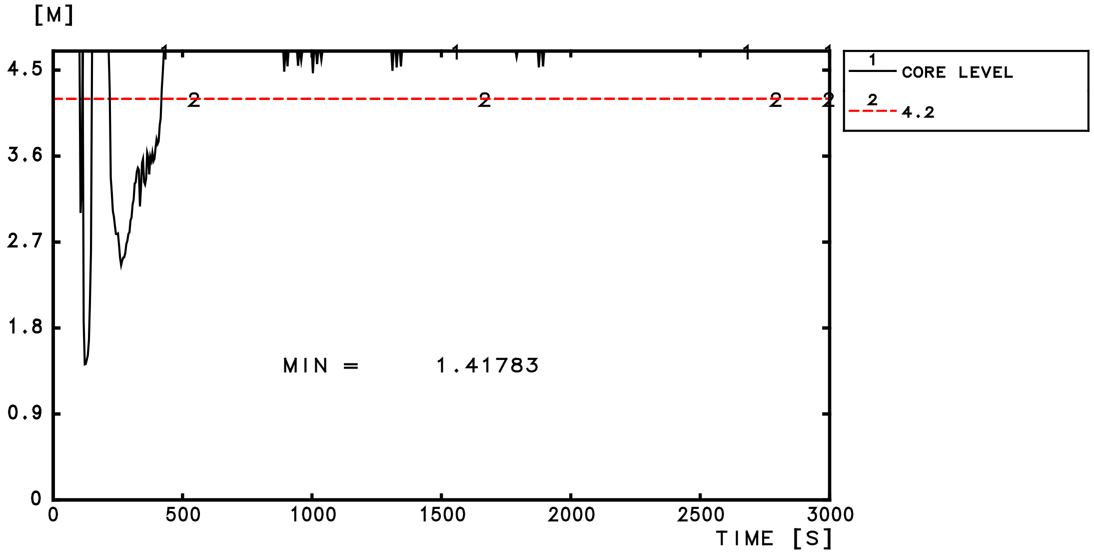
Water and steam break flowrate

SECTION 14.5.6 – FIGURE 73

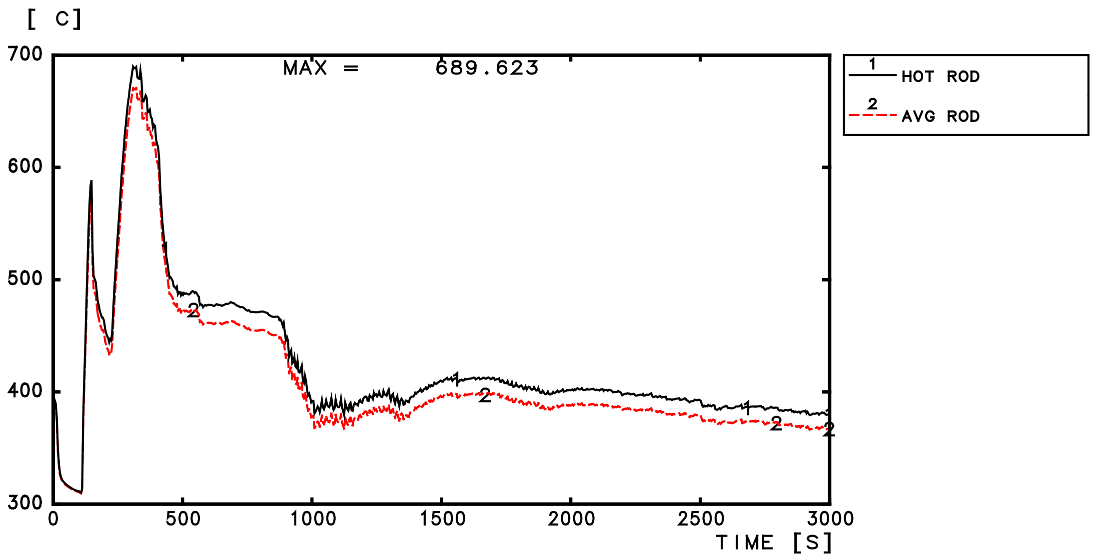
SECOND BREAK SPECTRUM: 390 CM² BREAK RIS [SIS]

CORE LEVEL

MAXIMUM CLADDING TEMPERATURE



Core level

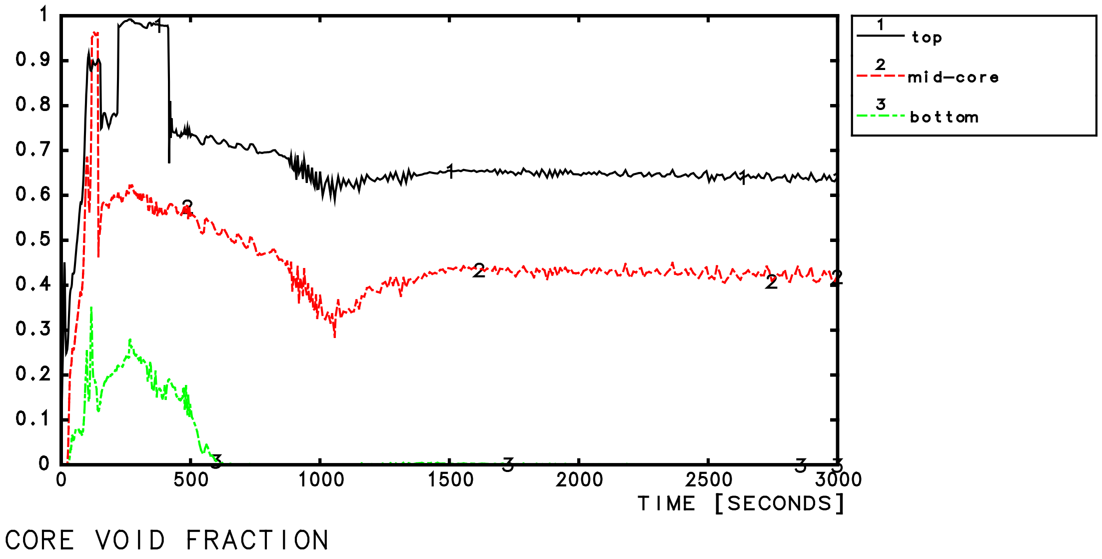


Maximum clad temperatures

SECTION 14.5.6 – FIGURE 74

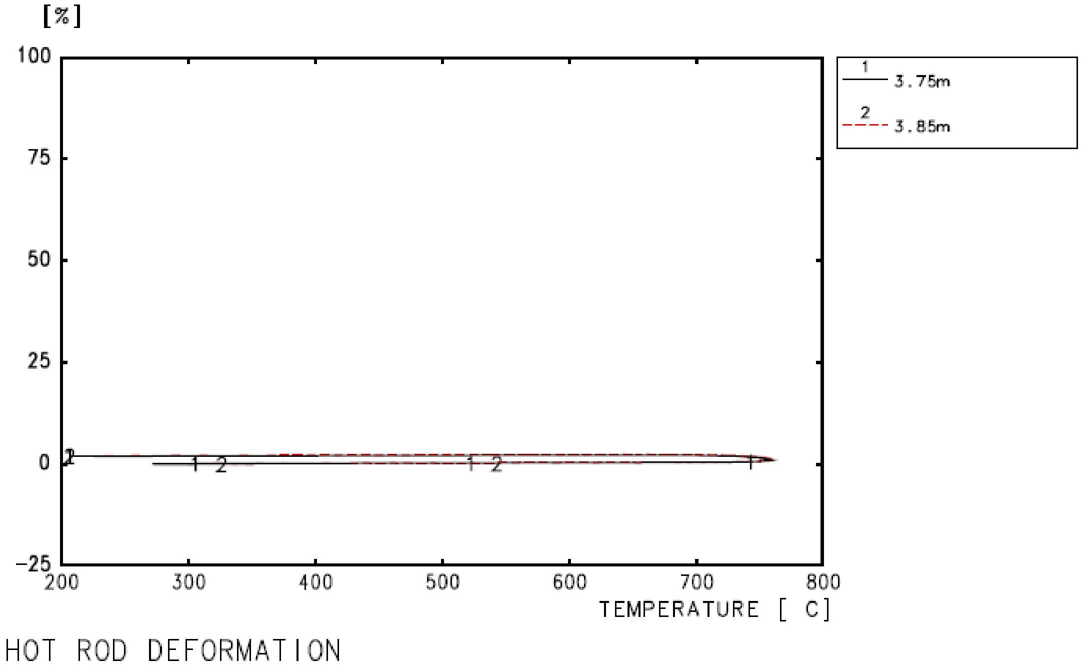
SECOND BREAK SPECTRUM: 390 CM² BREAK RIS [SIS]

CORE VOID FRACTION



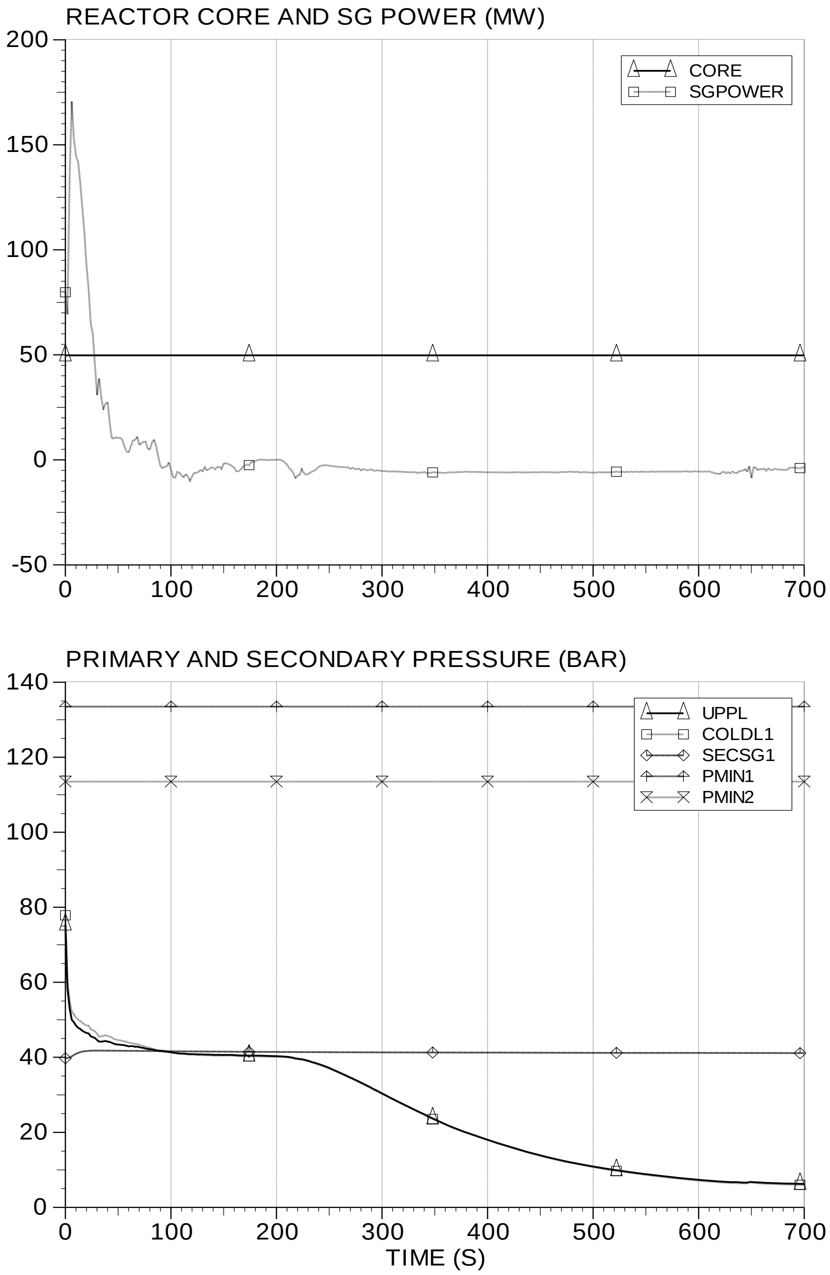
SECTION 14.5.6 – FIGURE 75

SECOND BREAK SPECTRUM: MAXIMUM HOT ROD DEFORMATION FOR THE PENALISING BREAK SIZE



SECTION 14.5.6 - FIGURE 76 [Ref-1]

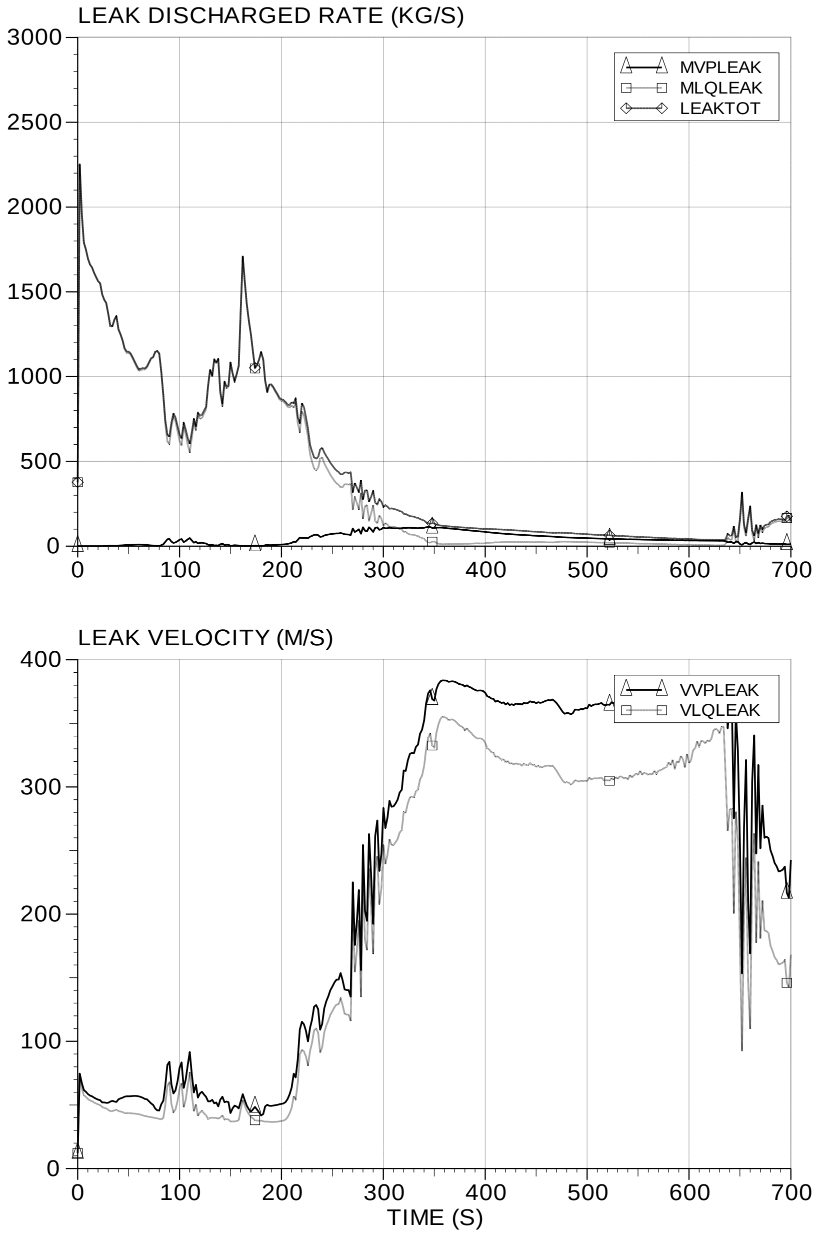
390 CM^2 RIS [SIS] LINE BREAK – STATE B2 CORE POWER AND TOTAL HEAT EXCHANGE IN STEAM GENERATOR PRIMARY AND SECONDARY SYSTEM PRESSURE



SECTION 14.5.6 - FIGURE 77 [Ref-1]

390 CM² RIS [SIS] LINE BREAK – STATE B2

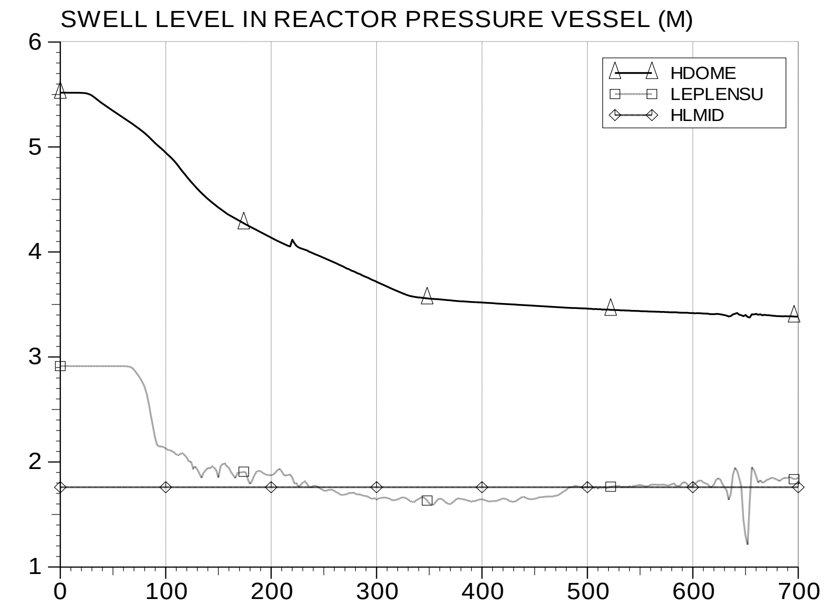
VAPOUR AND LIQUID MASS FLOW AT THE LEAK; VAPOUR AND LIQUID VELOCITY AT LEAK

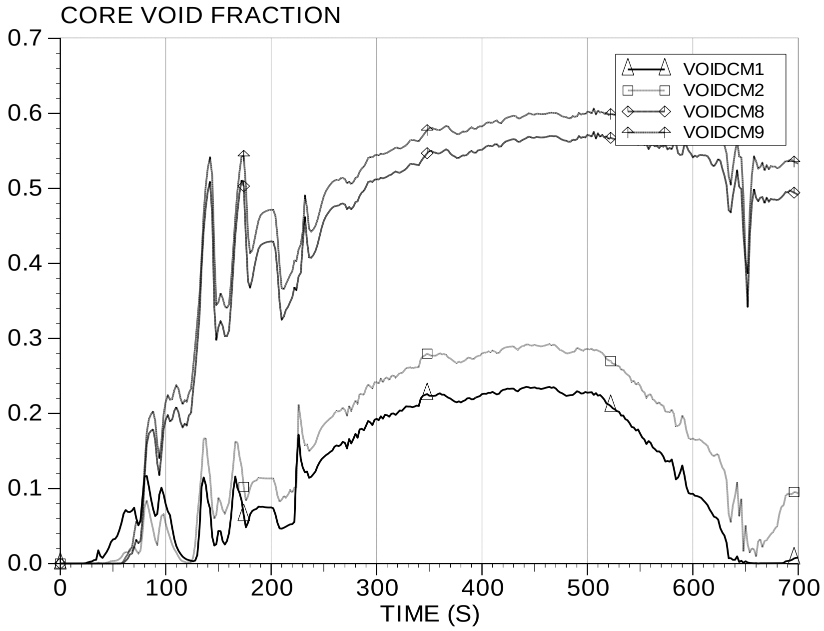


SECTION 14.5.6 - FIGURE 78 [Ref-1]

390 CM^2 RIS [SIS] LINE BREAK – STATE B2;

TWO-PHASE LEVEL IN REACTOR PRESSURE VESSEL; VOID FRACTION IN THE CORE



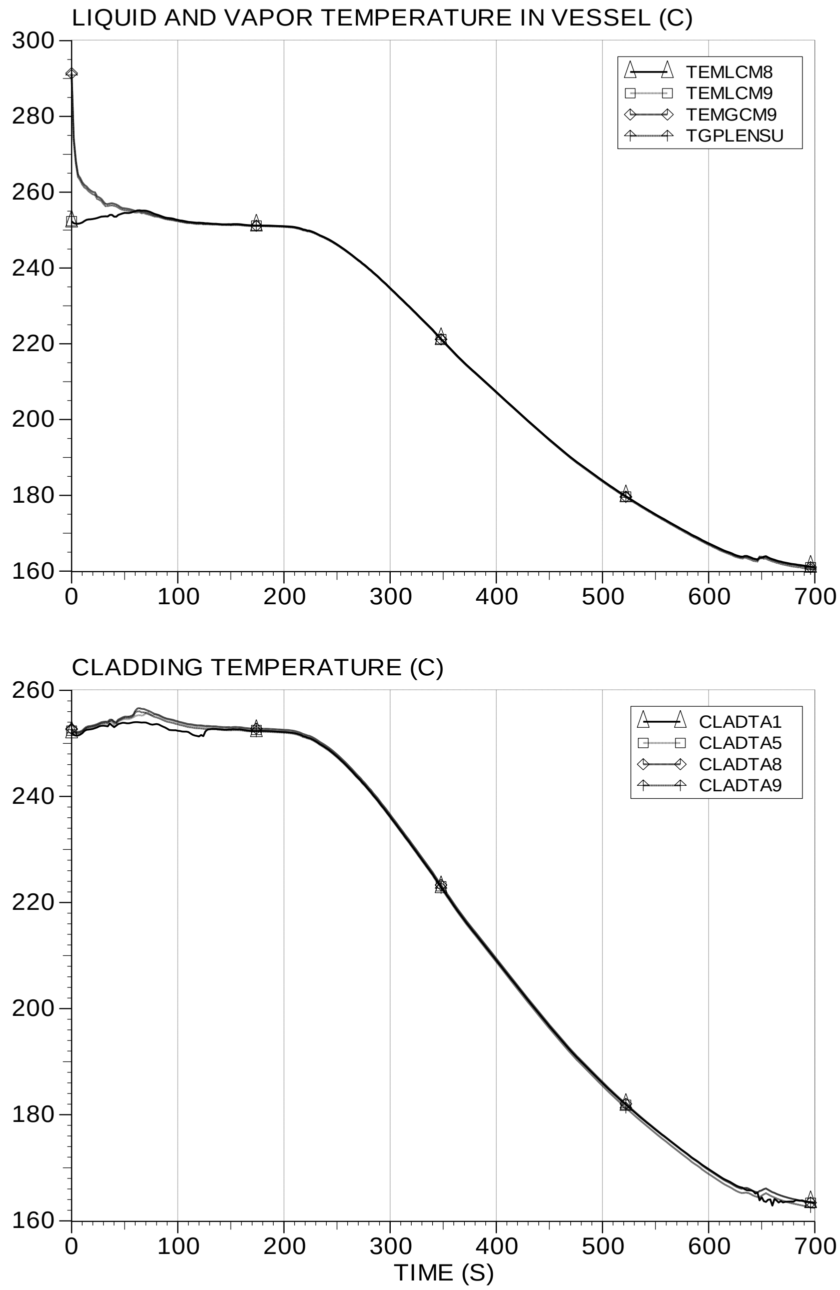


SECTION 14.5.6 - FIGURE 79 [Ref-1]

390 CM^2 RIS [SIS] LINE BREAK – STATE B2

LIQUID AND VAPOUR TEMPERATURE IN REACTOR PRESSURE VESSEL

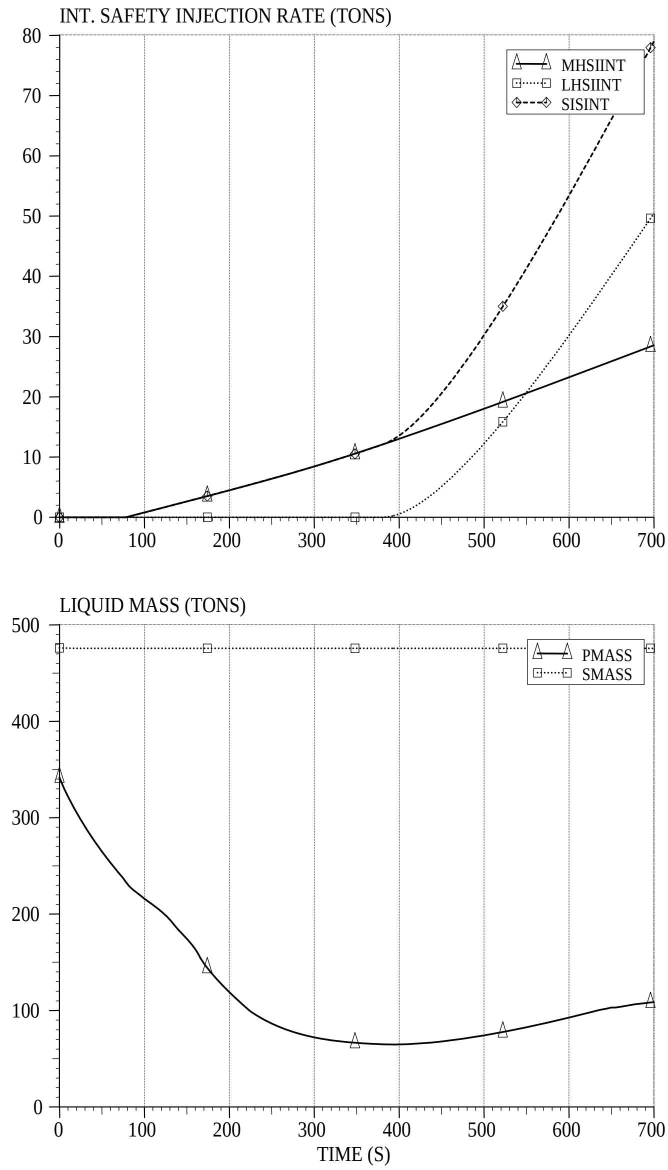
CLADDING TEMPERATURE OF THE AVERAGE RODS



SECTION 14.5.6 - FIGURE 80 [Ref-1]

390 CM^2 RIS [SIS] LINE BREAK – STATE B2 INTEGRAL SAFETY INJECTION RATE

WATER INVENTORY IN THE PRIMARY AND SECONDARY SYSTEM

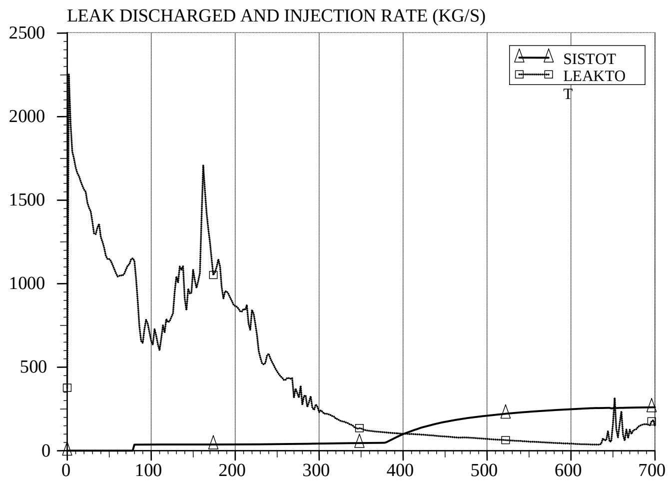


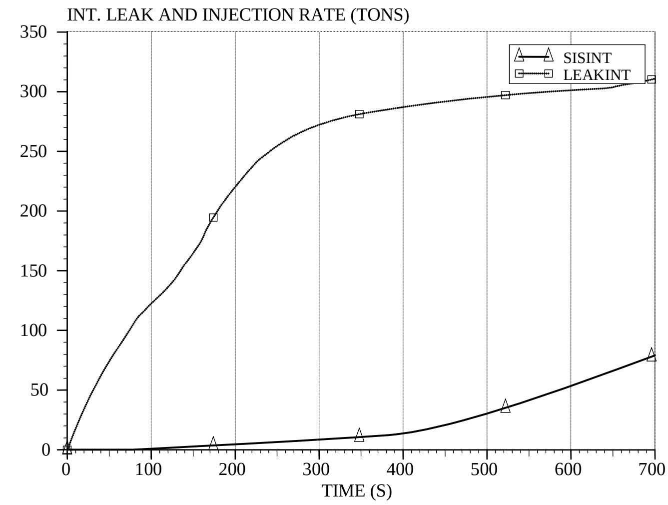
SECTION 14.5.6 - FIGURE 81 [Ref-1]

390 CM² RIS [SIS] LINE BREAK – STATE B2

SAFETY INJECTION AND LEAK DISCHARGED RATE

INTEGRAL SAFETY INJECTION AND LEAK DISCHARGED RATE



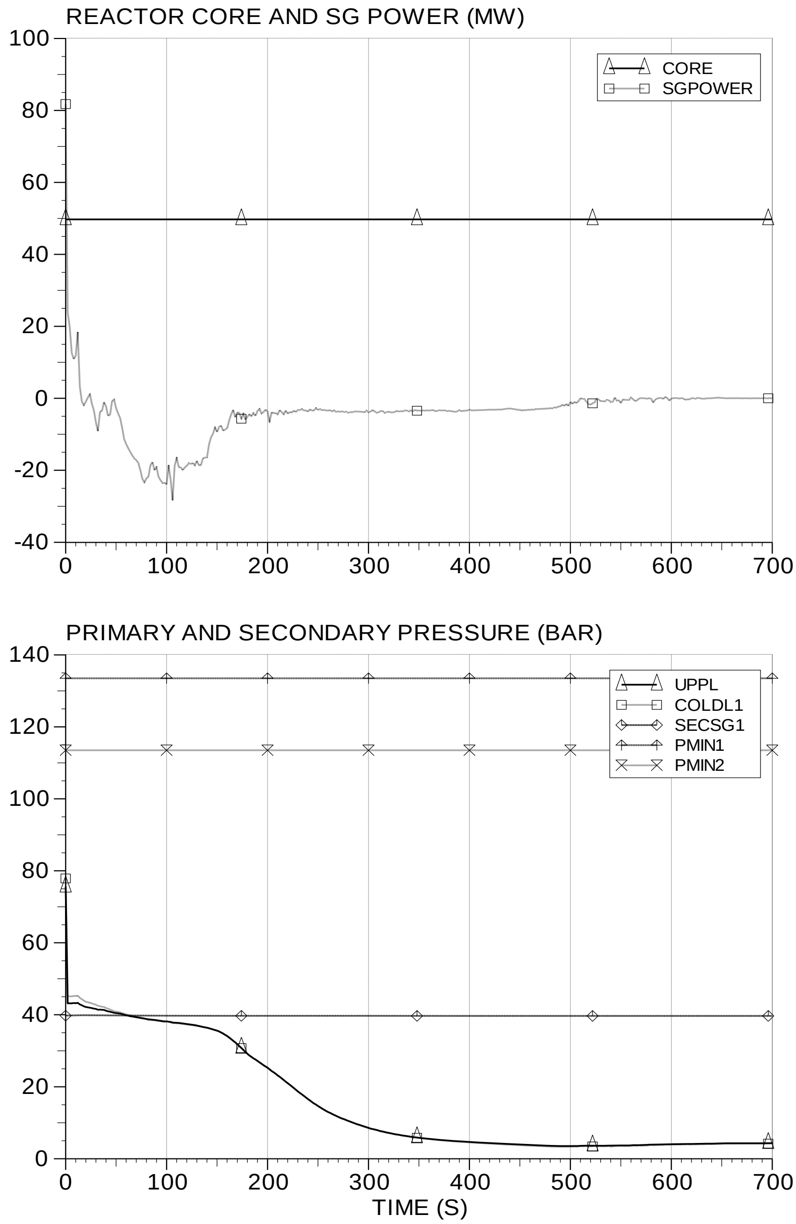


SECTION 14.5.6 - FIGURE 82 [Ref-1]

2 X 830 CM^2 SURGE LINE BREAK – STATE B2

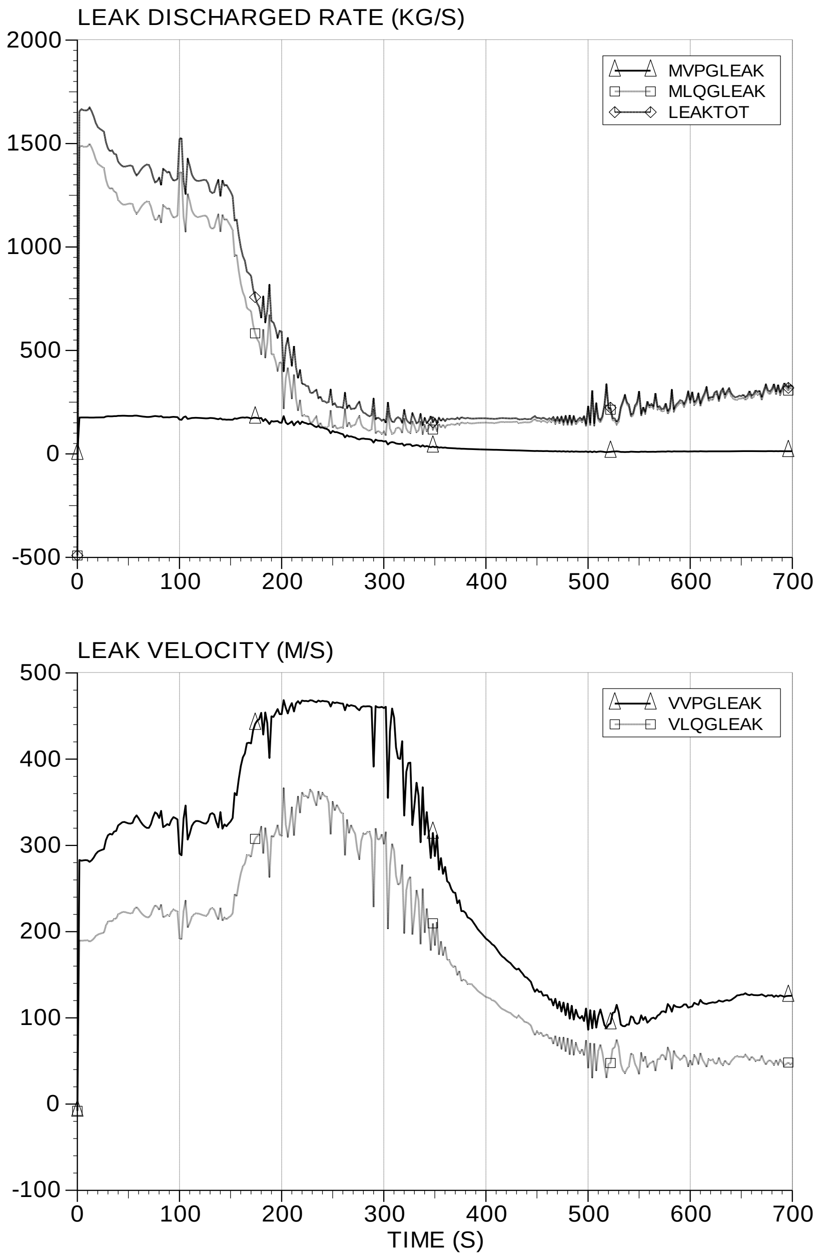
CORE POWER AND TOTAL HEAT EXCHANGE IN STEAM GENERATOR

PRIMARY AND SECONDARY SYSTEM PRESSURE



SECTION 14.5.6 - FIGURE 83 [Ref-1]

2 X 830 CM^2 SURGE LINE BREAK – STATE B2 VAPOUR AND LIQUID MASS FLOW AT THE BREAK VAPOUR AND LIQUID VELOCITY AT THE BREAK

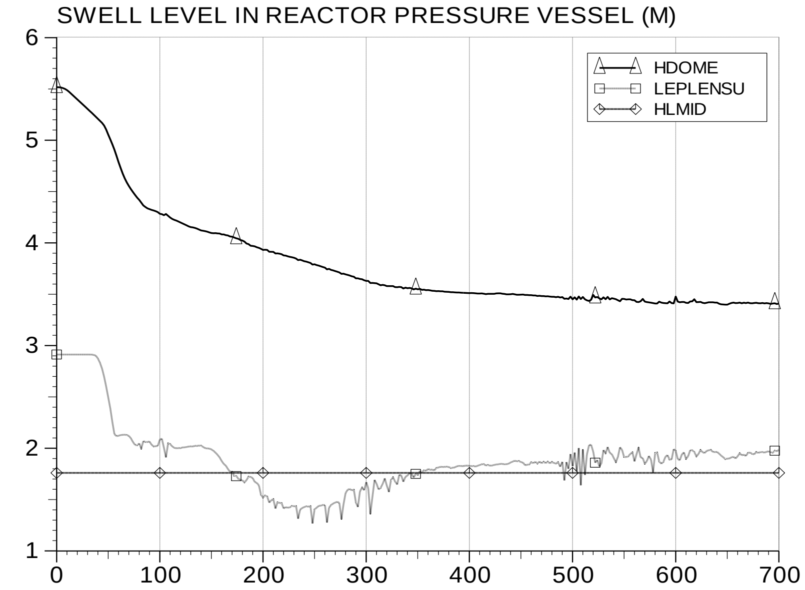


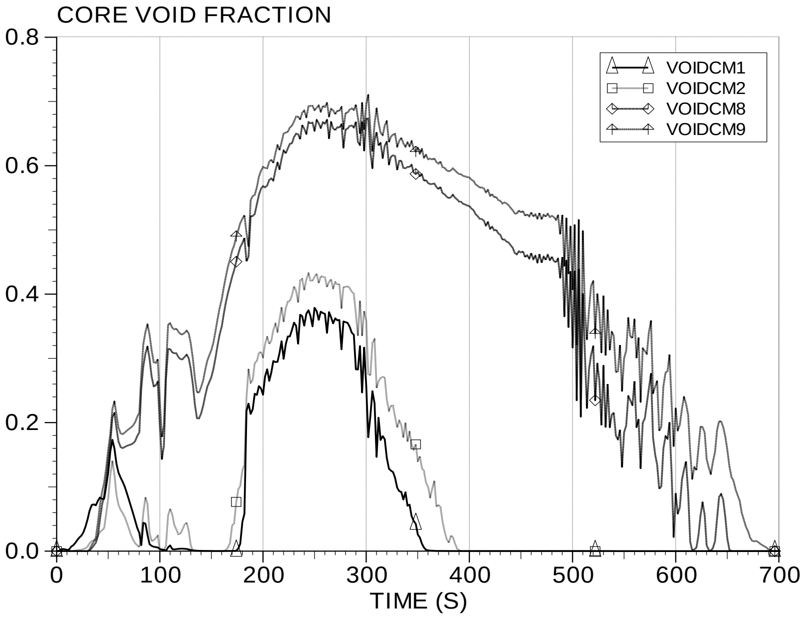
SECTION 14.5.6 - FIGURE 84 [Ref-1]

2 X 830 CM^2 SURGE LINE BREAK – STATE B2

TWO-PHASE LEVEL IN REACTOR PRESSURE VESSEL

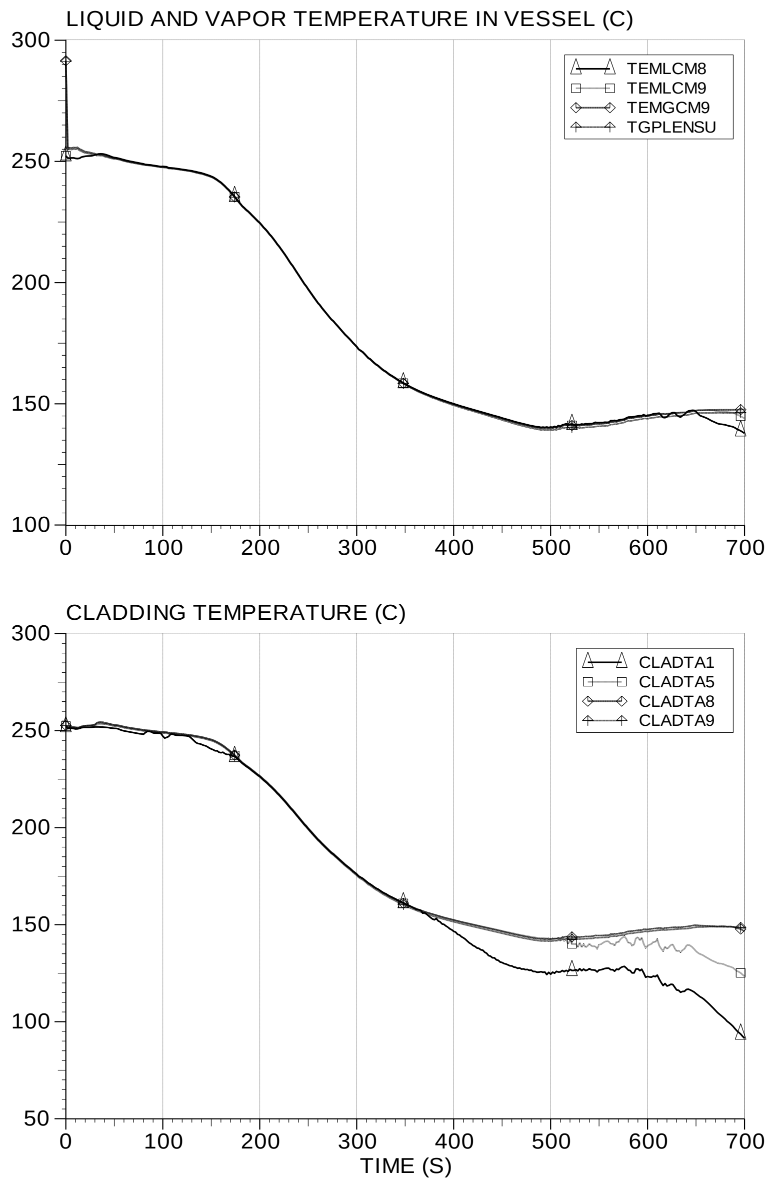
VOID FRACTION IN THE CORE





SECTION 14.5.6 - FIGURE 85 [Ref-1]

2 X 830 CM^2 SURGE LINE BREAK – STATE B2 TEMPERATURE IN REACTOR PRESSURE VESSEL; CLADDING TEMPERATURE OF THE AVERAGE RODS

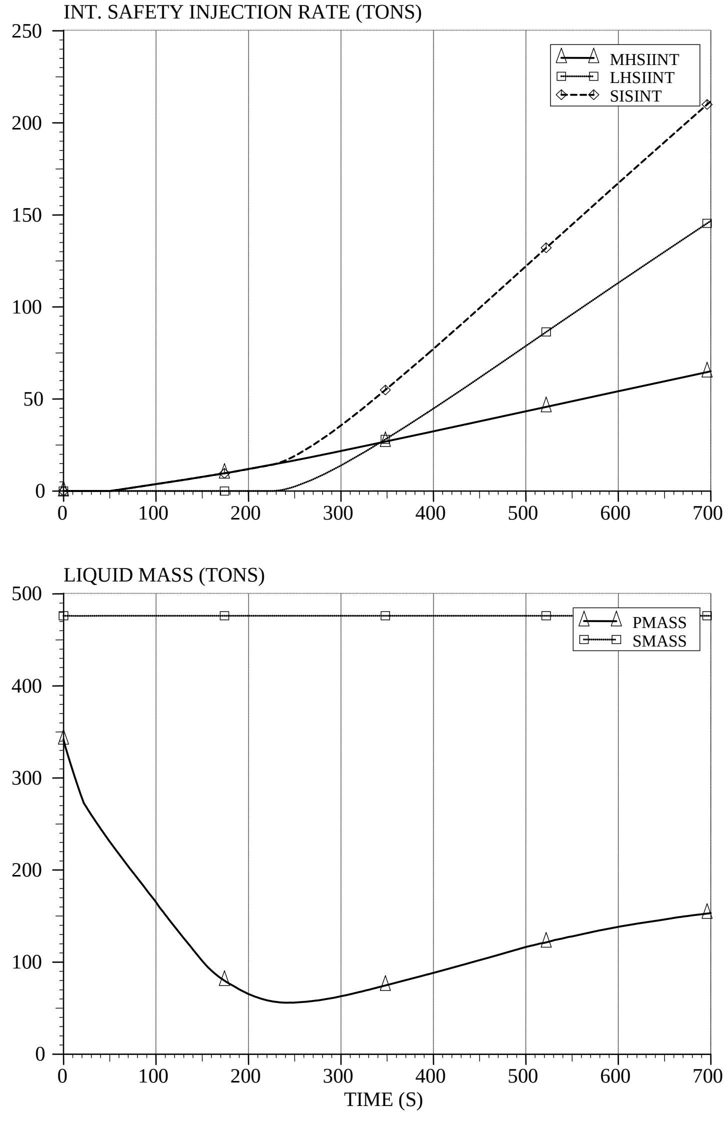


SECTION 14.5.6 - FIGURE 86 [Ref-1]

2 X 830 CM^2 SURGE LINE BREAK – STATE B2

INTEGRAL SAFETY INJECTION RATE

WATER INVENTORY IN THE PRIMARY AND SECONDARY SYSTEM

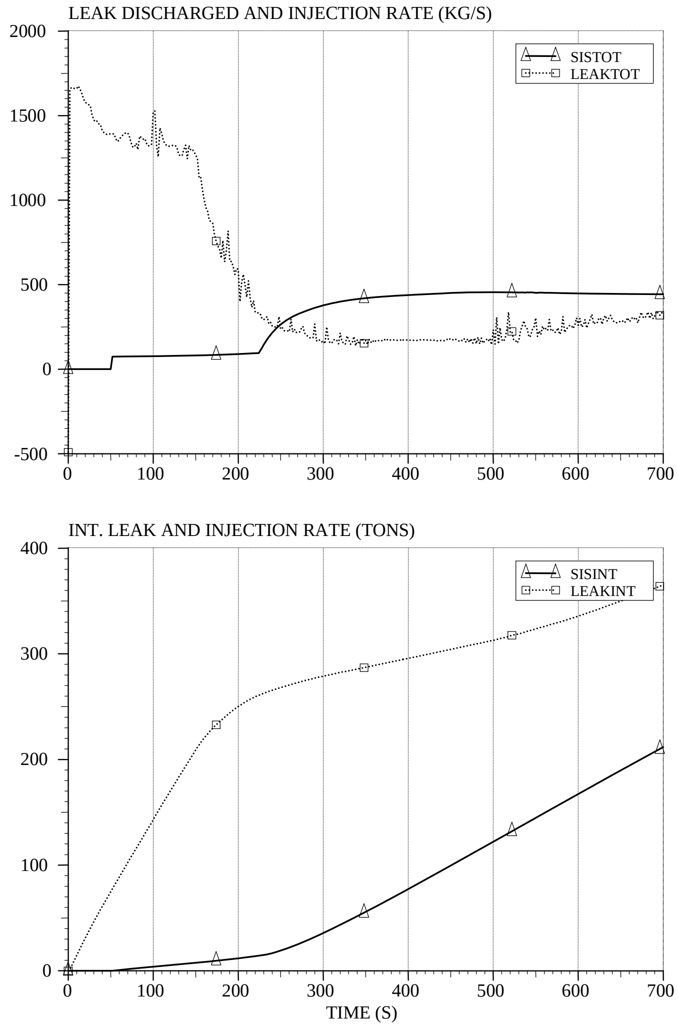


SECTION 14.5.6 - FIGURE 87 [Ref-1]

2 X 830 CM[2] SURGE LINE BREAK – STATE B2

SAFETY INJECTION AND BREAK DISCHARGED RATE

INTEGRAL SAFETY INJECTION AND BREAK DISCHARGED RATE



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

The SB-LOCA is defined as a break of equivalent diameter smaller than or equal to 50 mm (~20 cm^2), located in the RCP [RCS] pressure boundary.

The SB-LOCA in state C or D is classified as a PCC-4 event

Section 7.1 deals with SB-LOCA in state C (LHSI/RHR on, RCP [RCS] closed).

Section 7.2 deals with SB-LOCA in state D (LHSI/RHR on, RCP [RCS] open, fuel in RPV).

7.1. SMALL BREAK LOCA IN STATE C (PCC-4)

7.1.1. Accident definition

The initiating event is a non-isolatable break or leak, located either on the RCP [RCS] or on a RCP [RCS] connecting line, located between RCP [RCS] and RCP [RCS] isolation valves.

It includes a break or leak in the RIS/RRA [SIS/RHRS] lines inside the containment, between RCP [RCS] and the second isolation valve in the hot leg suction line and between RCP [RCS] and the check-valve in the cold leg injection line. The consequences of a break or leak located on an LHSI/RHR train outside the containment are discussed in the section related to ‘Residual heat removal system break outside containment’ (section 13 of this sub-chapter).

Non-isolatable LOCA dealt with in the present section are called ‘LOCA on RCP [RCS]’ or LOCA. Isolatable LOCA dealt with in section 13 of this sub-chapter are called ‘LOCA on LHSI/RHR’.

State C covers all shutdown states during normal plant operation, with heat removal performed by LHSI pumps operating in RHR-mode (LHSI/RHR) and the RCP [RCS] closed. It extends from 30 bar/120°C, when connection of the LHSI/RHR during transfer from hot to cold shutdown occurs, to 1 bar/55°C when the RCP [RCS] can be opened. As described in Sub-chapter 14.0, the RCP [RCS] is closed or can be rapidly reclosed by vent lines isolation, such that at least two SG can be used for decay heat removal, if required. The RCP [RCS] is full of water for temperatures above 55°C or at 3/4 loop operation level for temperatures of 55°C and below.

The consequences of such an event are analysed with the same objective as in state A as discussed in section 5.1 of Sub-chapter 14.4.

The SB-LOCA analysis in state C addresses the core cooling requirements. The aim of the analysis is to demonstrate that the relevant safety and acceptance criteria are met. This includes demonstrating the capability of the F1 systems to transfer the plant to the controlled state and the safe shutdown state.

State C introduces the following differences to the F1 mitigation measures when compared to those for the fault in state B discussed in section 5.2 of Sub-chapter 14.4:

- The change of safety injection (SI) signal at 3/4 loop operation, when all reactor coolant pumps are off.

- The non actuation of the LHSI in SI mode following an SI signal. In this state the LHSI/RHR trains are either in RHR-mode, or in stand-by without automatic actuation.

- The lower injection capacity of MHSI. In this plant state the large mini-flow line has been opened before LHSI/RHR connection. This decreases the MHSI delivery pressure to in the range of 40 bar, and prevents PSV opening following MHSI actuation.

The consequence of a SB-LOCA in reactor state A is discussed in section 5.1 of Sub-chapter 14.4.

The consequence of a SB-LOCA in reactor state B is discussed in section 5.2 of Sub-chapter 14.4.

The consequence of a SB-LOCA in reactor state D is discussed in section 7.2 of this sub-chapter.

7.1.2. Typical sequence of events

7.1.2.1. From initiating event to the controlled state

Reactor state C covers RCP [RCS] temperatures from 120°C to 55°C. Three sub-states named C1, C2, and C3 can be considered, depending on the operating status of the reactor coolant pumps and the RIS [SIS] pumps as defined in Section 14.5.7 - Table 6:

- State C1:
  - two reactor coolant pumps on.
  - two LHSI/RHR trains on.
  - two LHSI/RHR trains in stand-by (no automatic actuation).
  - three MHSI trains available for SI (automatic actuation on SI signal).

Note: three out of four MHSI trains are assumed available, allowing for the unavailability of one MHSI train.

  - RCP [RCS] pressure ≈ 30 bar.
  - RCP [RCS] temperature from 120°C (two LHSI/RHR on) to 100°C (four LHSI/RHR on).
  - RCP [RCS] water content – PZR level at zero-load set point.

- State C2:
  - two or one reactor coolant pumps on,
  - four LHSI/RHR trains on,
  - three MHSI trains available for SI (automatic actuation on SI signal).

Note: three out of four MHSI trains are assumed available, allowing for the unavailability of one MHSI train.

- RCP [RCS] pressure ≈ 30 bar.
- RCP [RCS] temperature from 100°C (four LHSI/RHR on) to 55°C (last reactor coolant pump shut down).
- RCP [RCS] water content – PZR level at zero-load set point.

- State C3:
  - All reactor coolant pumps off.
  - three LHSI/RHR trains on.
  - three MHSI trains available for SI (automatic actuation on SI signal).

Note: three out of four MHSI trains are assumed available, allowing for the unavailability of one MHSI train.

- RCP [RCS] pressure from 30 bar to 1 bar.
- RCP [RCS] temperature ≈ 55°C.
- RCP [RCS] water content – from PZR level to 3/4 loop level.

Whilst the RCP [RCS] temperature remains higher than 100°C, only two LHSI/RHR trains are used in RHR-mode for heat removal, with suction from hot leg and injection into the cold leg of the same loop. The two other LHSI trains are kept in stand-by for manual actuation if needed in either SI-mode or RHR-mode. Below 100°C, all four LHSI/RHR trains are used in RHR-mode operation.

In reactor state C, the RCP [RCS] break results in a loss of RCP [RCS] water inventory. To compensate for the loss of RCP [RCS] water inventory, different levels of mitigation measure are provided:

- The RCP [RCS] water inventory control, a non-F1 classified system, will intervene as the first level of mitigation. This will use the allowable imbalance between the RCV [CVCS] charging flow and letdown flow to provide a net addition of fluid. In the different reactor states, the control is:
  - States C1, C2:

    The PZR level control.
  - State C3 (including 3/4 loop operation):

    The RCP [RCS] loop level control defined below.
- The RCV [CVCS] letdown line isolation a non F1 classified action, is the second level of mitigation. This action limits the loss of coolant, and is actuated on:

- States C1, C2:

  PZR level low.

- State C3 (including 3/4 loop operation):

  RCP [RCS] loop level low.

- The third level of mitigation is the SI signal, F1A classified, which actuates the MHSI pumps in their appropriate configuration. The delivery head has previously been reduced to about 40 bar by opening of each MHSI large mini-flow line. This is performed before LHSI/RHR connection:

  - States C1, C2:

    Provided at least one reactor coolant pump is running (RCP [RCS] > 55 °C), the SI signal utilises the ΔP_{sat} [1] measurement used in state B, as described in section 5.2 of Sub-chapter 14.4.

  - State C3 (including 3/4 loop operation):

    When all reactor coolant pumps are shut down (RCP [RCS] ≈ 55°C), the SI signal utilises the RCP [RCS] loop level measurement. The RCP [RCS] loop level measurement consists of a ΔP measurement between the bottom and the top of each hot leg, with two out of four logic using one measurement per hot leg.

The LHSI pumps are not actuated in SI-mode by the automatic SI signal. They are only available following manual actuation by the operator, if necessary. Thus, only the MHSI pumps are used for automatic injection in reactor state C.

The automatic actuation of the MHSI pumps does not prevent the LHSI/RHR trains in operation continuing to provide the main heat removal function. Thus changes in hot leg thermal-hydraulic conditions, the RCP [RCS] loop level, HL pressure and temperature, do not prevent the sustained operation of the LHSI trains operating initially in RHR-mode.

In all states C1, C2, and C3, three out of four MHSI trains are assumed to be available, allowing for the unavailability of one MHSI train. The four accumulators are isolated before these states are reached.

In the safety analysis it is conservatively assumed that the SB-LOCA occurs on a loop where the LHSI train is initially operating in RHR-mode. In addition the break causes the complete loss of the LHSI/RHR train and the MHSI train connected to the broken loop. They are assumed to be unavailable for both heat removal and for safety injection.

After the break or leak occurs, the sequence of events is as follows:

- Loss of one LHSI/RHR train for RCP [RCS] heat removal:

  As indicated previously, the safety analysis assumes conservatively that the SB-LOCA causes the loss of the LHSI/RHR train connected to the broken loop. Thus, the following systems remain for LHSI/RHR heat removal following a SB-LOCA,

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

- o In state C1 (100°C<T<120°C):
  one of the two LHSI/RHR trains which were in operation before the event.

- o In states C2 (55°C<T<100°C):
  three of the four LHSI/RHR trains which were in operation before the event.

- o In states C3 (T ≈ 55°C, including 3/4 loop operation):
  two of the three LHSI/RHR trains which were in operation before the event.

- Additional RCP [RCS] heat removal can be provided by the two SG, if necessary. This is not required in the safety analysis of PCC-4 'SB-LOCA on RCP [RCS] in state C':

  - o Should the remaining LHSI/RHR train(s) not be sufficient to remove all the primary heat, the RCP [RCS] temperature would increase. As the RCP [RCS] temperature increases, the heat removal capacity of the remaining LHSI/RHR train(s) increases. If the RCP [RCS] temperature reaches the saturation temperature for the pressure setpoint of the two SGs maintained in stand-by, those SGs will contribute to the RCP [RCS] heat removal.

  - o In normal operation, the cooling of the SG is performed in parallel with the LHSI/RHR cooling below 120°C. However, in the safety analysis it is assumed that the SG cooling has been stopped at the LHSI/RHR connecting temperature of 120°C. As a result, two SG are maintained in stand-by at a temperature of 120°C and the pressure setpoint above 2 bar (Psat 120°C). 'In stand-by' means that these two SG are ready to operate, with feed from the ASG [EFWS], F1A classified, automatically started on a "SG-level MIN2" signal, and steam discharge from the VDA [MSRT], F1A classified. This is automatically actuated when the SG pressure reaches the pressure setpoint. The two other SGs are assumed unavailable for steam removal, with the ARE/AAD/ASG [MFW/SSS/EFWS] and VIV/VDA [MSIV/MSRT] isolated.

- Break flow compensation:

  As stated previously, the safety analysis assumes conservatively that the SB-LOCA causes the loss of the MHSI train connected to the broken loop. Consequently, if the loss of this MHSI train, the initial unavailability of one MHSI train, and the single failure of one MHSI train are assumed, only one MHSI train remains for RIS [SIS] injection when the SI signal occurs.

  The SI signal automatically actuates MHSI injection. The RCP [RCS] water inventory is quickly restored by the MHSI pump available for RCP [RCS] injection. For the largest SB-LOCA break size of 20 cm^2 the break flow is compensated for in liquid form by one MHSI. This occurs at a RCP [RCS] pressure above the hot leg saturation pressure as shown by the transient analysis results in section 7.1.5.3.

- Containment heat removal:

  The break flow temperature is higher than the containment temperature. Consequently there is flashing at the break and steam is released into the containment. This results in an increase in the containment pressure and temperature and the IRWST temperature.

The controlled state is defined as a state where:

- The core is sub-critical.
- The core power is removed via LHSI/RHR, with additional heat removal via the SG in stand-by mode prior to the accident if required.
- The reactor coolant inventory is stable or increasing, with the MHSI flow at least matching the break flow.

7.1.2.2. From the controlled state to the safe shutdown state

Additional F1B information is needed by the operator when compared to operation following the fault in states A and B. The RPV level (RPVL) and the ΔT_{sat} measurements used in states A and B to perform the diagnosis of the plant status may not be available in state C. The electrical cables are disconnected before RPV-head removal. Thus these F1B functions and are replaced by other F1B measurements as listed below.

For RCP [RCS] temperatures from 120°C to 100°C (state C1):

- Control of RCP [RCS] water inventory: - RPVL when the indication is available, RCP [RCS] loop level.
- Control of RCP [RCS] pressure and temperature: - hot leg pressure and temperature.
- Control of core sub-criticality:- intermediate range flux channels.

For RCP [RCS] temperatures from 100°C to 55°C (states C2, C3):

- Control of RCP [RCS] water inventory: - RPVL when the indication is available, RCP [RCS] loop level.
- Control of RCP [RCS] pressure and temperature: hot leg pressure and temperature.
- Control of core sub-criticality:- source range flux channels.

In addition, the following F1B information available in states A and B remains available in state C if needed:

- PZR level (cold calibration).
- SG level.
- SG pressure.

The safe shutdown state is defined as a state where:

- The core is sub-critical.
- The reactor coolant inventory is stable or increasing.
- The decay heat is removed by the cooling chain LHSI/RRI/SEC [SIS/CCWS/ESWS] used in RHR-mode.

- The activity release is within the limits for PCC-4 events.

The main actions necessary to reach the safe shutdown state are as follows:

- Control of core sub-criticality:

  RCP [RCS] boration is unnecessary. However, the RBS [EBS], an F1A classified system, is available for boration. This is in addition to the boration provided by the MHSI. Neither is claimed in the safety analysis of the PCC-4 'SB-LOCA on RCP [RCS] in state C'.

- Control of RCP [RCS] water inventory:

  The available MHSI pump(s) provide leakage make-up and maintain adequate LHSI/RHR operating conditions.

  For a SB-LOCA occurring in state C1, above 100°C, the LHSI pumps which were in stand-by before the accident can be manually actuated to provide an additional water supply, if needed. This is not modelled in the safety analysis of the PCC-4 'SB-LOCA on RCP [RCS] in state C'.

  The LHSI/RHR heat removal maintains the IRWST temperature below the temperature limit for adequate RIS [SIS] pump operation.

- Control of RCP [RCS] heat removal:

  If the SG were contributing to the RCP [RCS] heat removal when the controlled state was reached, the operator would transfer the heat removal function to the available LHSI/RHR train(s) before the ASG [EFWS] tanks empty.

  For SB-LOCA occurring in state C1, above 100°C, the LHSI pumps, which were in stand-by before the accident, can be manually actuated to provide an additional heat removal capacity, if needed. This is not modelled in the safety analysis of the PCC-4 'SB-LOCA on RCP [RCS] in state C'.

- Control of containment:

  The operator re-closes the containment, if open, as soon as possible. This limits the radiological release from the presence of steam inside the containment.

7.1.3. Safety criteria

The safety criteria and acceptance criteria to be met are the same as those described in sub-section 5.1.2 of Sub-chapter 14.4 for SB-LOCA in reactor state A.

7.1.4. Definition of cases studied

The safety justification utilises a SB-LOCA transient calculation performed with the CATHARE V1.3L computer code. Appendix 14A provides the code description, and section 5.1.3.1 of Sub-chapter 14.4 discusses the code adequacy and analysis methodology for this type of calculation.

One transient calculation is performed, covering all the SB-LOCA cases in state C. As in states A and B, the worst break location is on the RCP [RCS] cold leg, and the largest SB-LOCA break size is assumed. This has an equivalent diameter of 50 mm, an equivalent area 20 cm^2. The break is assumed to occur at the start of state C1 operation with the RCP [RCS] at 30 bar and 120°C. This is the most limiting initial condition for the loss of RCP [RCS] water inventory.

The accident mitigation in each of states C1, C2 and C3 is described below. In addition, the initial plant conditions and the accident mitigation measures are also provided below. The data for states C1, C2 and C3, as well as for the bounding CATHARE calculation performed for state C1 are shown to clearly show how the different plant states are bounded by the calculation.

The safety demonstration shows the controlled state and the safe shutdown state can be reached, whilst meeting the associated safety and acceptance criteria.

As required by the rules defined for safety analyses in Sub-chapter 14.0, the controlled state is reached using only F1A classified systems. The safe shutdown state is reached using only F1A and F1B classified systems.

As required by the accident analysis rules:

- The transient calculation is performed assuming the worst single failure.
- In state C, there is no preventive maintenance on F1 systems, except on two SG and associated ASG [EFWS] and VDA [MSRT] trains. However, the unavailability of one MHSI train is assumed during the plant normal operation in state C. As a consequence of this assumption, only three MHSI trains are available at the beginning of the accident.
- In state C, LOOP is assumed to not occur coincident with the event.
- No operator action is credited before 30 minutes after the SI-signal actuation.

7.1.5. Description of cases studied

The safety demonstration is provided making the same conservative assumptions as in states A and B.

The analyses are performed for EPR_{4250}. The application of the analysis results to EPR_{4500} is discussed in this section.

7.1.5.1. Choice of single failure

The worst single failure is the loss of one MHSI train.

7.1.5.2. Initial state

The initial conditions are chosen to delay the SI signal actuation and to maximise the break flow rate. This minimises the RCP [RCS] water inventory and the pessimises the thermal-hydraulic conditions in the hot legs for adequate LHSI/RHR pump operation:

- The initial reactor power is maximised and corresponds to the maximal residual heat load following continuous reactor operation at 102% NP prior to the accident. It assumes the minimum time for LHSI connection in RHR-mode during the plant cooldown from hot to cold shutdown and represents the start of state C.
- The initial RCP [RCS] pressure and temperature are maximised.

The resulting bounding initial conditions for each reactor state are (the pressure and temperature values are given without uncertainties):

- State C1:
  - The maximum RCP [RCS] pressure is 30 bar
  - The maximum RCP [RCS] temperature is 120°C.
  - The RCP [RCS] temperature of 120°C is reached no earlier than 6 hours after reactor shutdown.
- State C2:
  - The maximum RCP [RCS] pressure is 30 bar.
  - The maximum RCP [RCS] temperature is 100°C.
- State C3 ‘3/4 loop operation’:
  - The maximum RCP [RCS] pressure is 1 bar.
  - The maximum RCP [RCS] temperature is 55°C.

The initial conditions in the bounding CATHARE calculation are nominal values at beginning of state C1, pessimised by the following uncertainties[2]:

- The RCP [RCS] pressure is 32.5 bar (30 + 2.5 bar).
- The RCP [RCS] temperature is 122.5°C (120 + 2.5°C).

These conditions provide the highest RCP [RCS] initial pressure and saturation pressure, which maximises the break flow rate and minimises the MHSI injection flow rate.

The initial water inventory of the pressuriser and the SG are given in Section 14.5.7 - Table 4.

As in states A and B, the worst location for the break is on the RCP [RCS] cold leg, or on the RIS [SIS] injection line, close to the RCP [RCS] cold leg. The CATHARE calculation assumes that the break is located in the cold leg. This assumption bounds breaks in a RIS [SIS] line with the failures assumed as a consequence of the break, the loss of MHSI and LHSI/RHR trains, assumed in the broken loop.

Four loops are explicitly modelled in the CATHARE calculation, with the break located in loop four, and the pressuriser in loop two.

[2] Uncertainties credited at present are those defined for initial operation in state A (see Section 14.1).

The RCP [RCS] plant alignment prior to and after the break is as presented in Section 14.5.7 - Table 1.

The following assumptions refer mainly to the CATHARE calculation of the fault, initiated at conditions for the start of state C1. Information related to states C2 and C3 is added in the note.

7.1.5.2.1. *Decay heat*

The maximum residual heat is assumed and corresponds to the terms B+C, fission products + actinides, presented for the maximum decay-heat curve of Sub-chapter 14.1 (‘MAX+2σ’ [3]).

Under the plant outage schedule, the beginning of state C occurs 10 hours after reactor shutdown. This corresponds to the connection of LHSI in RHR-mode. In the present SB-LOCA analysis, the decay-heat level assumed represents a point 6 hours after RT. This significantly over-predicts the core power level.

At 6 hours after reactor trip, the maximum core decay heat is 0.91% [4]. This assumes continued power operation prior to reactor shutdown at the maximum core power of 102% NP. Therefore the maximum core power is 39.5 MW. This assumption is used for the bounding CATHARE calculation.

7.1.5.2.2. *Assumptions related to F1 systems*

<u>LHSI/RHR trains (F1B):</u>

The LHSI trains assigned to loops 1 and 4 are initially in operation in RHR-mode [5].

The LHSI train in loop 4 is assumed to be lost when the break on loop 4 occurs. This assumption is over-conservative and it may not be retained in subsequent safety analyses.

After the break occurs, only the LHSI train in loop 1 remains in operation, performing the entire heat removal [6].

The opening of the LHSI/RHR control valve is maintained constant during the transient calculation, being equal to the initial opening. One LHSI/RHR train removes half of core residual heat plus one reactor coolant pump power during normal operation.

The LHSI trains assigned to loops 2 and 3 are in stand-by mode, and may be manually actuated either in SI-mode or in RHR-mode. This is assumed to occur no earlier than 30 minutes after the SI signal, if needed. Their actuation is not modelled in the CATHARE calculation[7].

The characteristics of each LHSI/RRI/SEC [SIS/CCWS/ESWS] cooling chain are pessimised to minimise the heat removal capacity. The following data is related to one chain [Ref-1]:

- LHSI/RRI [LHSI/CCWS] heat exchanger capacity (min): 1.1 MW/°C.

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[3] "MAX+1.645σ" can be considered since LOCA occurs in the long term phase. "MAX+2σ" is presently credited, being over-pessimistic.

[4] At RT + 10 hours, "MAX+2σ" decay heat is 0.805 % which gives a maximum core power of 35 MW

[5] In states C2, C3: LHSI trains 1, 2, 3 are initially in operation in RHR-mode.

[6] In states C2, C3: LHSI trains 1, 2 remain in operation in RHR-mode.

[7] In states C2, C3: no LHSI train in stand-by.

- RRI/ SEC [CCWS/ESWS] heat exchanger capacity (min): 2.5 MW/°C.
- LHSI flow rate in RHR-mode (min): 150 kg/s
- RRI [CCWS] flow rate (min): 500 kg/s
- SEC [ESWS] flow rate (min): 750 kg/s
- Typical power (max): 9.75 MW
- Fraction RRI [CCWS] flow towards commons (max): 40%.
- Fraction RRI [CCWS] flow towards LHSI/CCWS HX (min): 60%.
- SEC [ESWS] suction temperature (max): 30°C.

<u>Safety Injection (F1A):</u>

An SI signal is automatically actuated on RCP [RCS] subcooling margin ΔP_{sat}, with a setpoint in the range of 10 bar[8].

The setpoint claimed in the CATHARE calculation is a minimum value of 5 bar. This value accounts for uncertainty, including the degraded containment conditions. This delays the start-up of the MHSI pump.

The consequences of the conservative assumptions are listed below:

- 25 seconds maximum delay for MHSI pump start-up (including signal delay).
- Minimum MHSI injection curve with the large mini-flow line open. This limits the MHSI delivery pressure at approximately 40 bar in RHR-mode This is discussed in Sub-Chapter 14.1: minimum MHSI characteristics in states C, D.
- 50°C for maximum initial IRWST temperature and injection flow temperature.

The status of the MHSI trains is as follows:

- The MHSI train assigned to RCP [RCS] loop 1 is assumed unavailable in normal plant operation in state C.
- The MHSI train assigned to RCP [RCS] loop 3 is assumed to fail to operate on demand following the SI signal. This is the assumed single failure.
- The MHSI train assigned to the RCP [RCS] broken, loop 4, is assumed to spill directly into the containment with no contribution to the RCP [RCS] injection. Such an assumption is very conservative and it may not be retained in subsequent analyses.
- Consequently, the MHSI train assigned to loop 2 is the only one assumed available for RIS [SIS] injection into the RCP [RCS].

The accumulators, which are initially isolated, are not claimed in the CATHARE calculation.

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[8] in state C3 (all RCP off, including 'mid-loop operation') : the SI signal is automatically actuated on 'low RCS loop level'.

VDA [MSRT] (F1A):

Two SG are initially available in stand-by.

Two VDA [MSRT] of the two SGs in stand-by are automatically actuated train by train using an SG specific signal when SG pressure rises to above the setpoint. The setpoint assumed in the CATHARE calculation is 5.15 bar. This is 3 bar above the initial RCP [RCS] saturation pressure at 120 + 2.5°C, which is the SG temperature level when the LHSI/RHR is connected.

The conservative characteristics are identical to those used in state A, as discussed in Sub-chapter 14.1.

However, VDA [MSRT] are not actuated in the CATHARE calculation.

ASG [EFWS] (F1A):

Each of the two SG in stand-by is automatically fed by its ASG [EFWS] pump when required. The two other SG are fully isolated: ARE/AAD/ASG [MFW/SSS/EFWS] via feed isolation, and the VIV/VDA [MSIV/MSRT] via steam isolation, with no possibility for steam removal.

The preventive maintenance on the SG has no impact on the availability of the ASG [EFWS] assigned to the two SG in stand-by. Any maintenance will be performed on one of the two divisions related to the unavailable SG, not those placed in stand-by.

The ASG [EFWS] is automatically actuated train by train using a SG specific measurement, a "SG level MIN2" signal.

The conservative characteristics of minimum flow and maximum temperature are identical to those used in states A and B as discussed in Sub-Chapter 14.1.

However, the ASG [EFWS] is not actuated in the CATHARE calculation.

VIV [MSIV] (F1A):

The four VIV [MSIV] are closed in state C.

Reactor coolant pumps trip (F1A):

The reactor coolant pumps initially operating are those in loops 2 and 3[9].

There is no LOOP assumed for the fault occurring in reactor state C. The reactor coolant pumps are automatically tripped on low reactor coolant pump pressure drop (pressure drop < MIN1), as in the case in states A and B.

In the CATHARE calculation, it is conservatively assumed, that all reactor coolant pumps trip at the start of the event. The early reactor coolant pump trip is conservative, compared to any reactor coolant pumps trip occurring before the reactor coolant pumps trip signal. It contributes to an additional loss of RCP [RCS] inventory by keeping the primary pressure high in the early part of the transient.

[9] in state C2 :'two reactor coolant pumps on' above about 70°C, 'one reactor coolant pump on' between 55°C and 70°C.
in state C3 : all reactor coolant pumps off.

7.1.5.2.3. *Other assumptions*

The non-F1 classified systems are not claimed if their operation would be beneficial or have no impact on the accident mitigation. For example, the RCV [CVCS] injection actuated by I&C control/limitation is not modelled

7.1.5.3. Results

The typical sequence of events produced by a CATHARE V1.3 L calculation can be found in Section 14.5.7 - Table 5. [Ref-1]

The most representative parameters are presented in the following figures:

- Section 14.5.7 - Figure 2: RCP [RCS] and secondary side water inventories, RCP [RCS] and secondary side pressures.
- Section 14.5.7 - Figure 3: RCP [RCS] average temperature, Break and total RIS [SIS] flow rates.
- Section 14.5.7 - Figure 4: RCP [RCS] subcooled margin (in upper plenum), RPV upper plenum water level (collapsed level).

Typical transient behaviour is described below.

The SI signal is reached at 207 seconds, about 60 seconds after the pressuriser empties.

- The flow injected by one MHSI matches the break flow soon after MHSI start up. The decrease of the RCP [RCS] water inventory is halted with a loss of water of around 30 tons.
- The decay heat removal is partly via the break, but mostly by the one cooling chain RIS/RRI/SEC [SIS/CCWS/ESWS] remaining in operation. Credit is taken for the subcooling of the MHSI injected water.
- The primary pressure stabilises below 4 bar and the hot leg thermal-hydraulic conditions are suitable for the continued operation of the LHSI/RHR pump. Sufficient subcooled margin and a RCP [RCS] loop level at the top of the hot leg are maintained. The controlled state has therefore been reached.

About 2000 seconds after SB-LOCA occurrence, all the parameters are stabilised and the core residual heat is fully removed by one RIS/RRI/SEC [SIS/CCWS/ESWS] cooling chain:

- A small primary to secondary heat transfer temporarily occurs when the RCP [RCS] temperature increases above 122.5°C before 1800 seconds. However, the VDA [MSRT] remains closed, without SG boiling, with the SG pressure remaining below the VDA [MSRT] setpoint of 5 bar. After 2000 seconds, the primary to secondary heat transfer is zero. The SGs do not participate significantly in the RCP [RCS] heat removal.

- In the CATHARE calculation, the LHSI/RHR operates with a constant opening of the LHSI/RHR control valve. The initial opening is maintained during the transient. For this reason the RCP [RCS] temperature decreases below 122.5°C. This assumes the MHSI injection flow subcooling contributes to the heat removal. The LHSI/RHR train is able to remove the entire decay heat at a temperature below 110°C. With one LHSI train operating, the 110°C IRWST temperature limit for MHSI operation is not reached.

- The operator re-closes the containment (if open) as soon as possible to limit the radiological release. Once that action has been performed, the safe shutdown state is reached.

Following the SB-LOCA, the RCP [RCS] water inventory decrease has been limited to the draining of the pressuriser. The core remains covered at all times during the transient, with no clad heat-up. All the safety and acceptance criteria are met, using only on F1 measures. The criteria are met assuming the worst single failure and the unavailability of one MHSI train in normal operation during state C.

The following assumptions, considered in the CATHARE calculation, ensure a high level of conservatism for the SB-LOCA transient in reactor state C1:

- The loss of the LHSI/RHR pump connected to the broken loop at the start of SB-LOCA.

- The MHSI pump assigned to the broken loop is assumed not to inject. In the CATHARE calculation, only one MHSI pump is modelled. However, two pumps are available despite the single failure assumption and the unavailability of one MHSI train.

Note: In the CATHARE calculation, the break is located in a cold leg and the MHSI injecting in the broken loop is not claimed for RCP [RCS] water injection as it is assumed to be completely lost via the break. If the break location were assumed at the cold leg injection nozzle of one LHSI/RHR line, the injection of the MHSI pump into the same loop, injecting directly to the break, represents about 75% of the break flow rate at the worst subcooled conditions of 10°C minimum IRWST temperature and 4 bar CATHARE RCP [RCS] pressure. The other available MHSI pump would thus provide more than sufficient RCP [RCS] water supply to match the remaining break flow. The injection to the RCP [RCS] would represent three times the remaining 25% of the break flow. This would result in a higher RCP [RCS] water inventory. The CATHARE assumption of not claiming the MHSI assigned to the broken loop for SBLOCA is overly conservative.

- Additional conservatism is introduced into the calculation by the assumed non-actuation of the remaining LHSI pump in stand-by. This pump can be manually actuated by the operator 30 minutes after the SI signal, and would greatly increase the RCP [RCS] water injection.

The CATHARE calculation in state C1 bounds all the SB-LOCA in state C2 and C3, since:

- The amount of residual heat for removal is lower in states C2 and C3.

- The LHSI/RHR trains remaining in operation are more numerous than in state C1 (three instead of one), which supports the RCP [RCS] heat removal at a lower temperature.

- The initial RCP [RCS] temperature is lower in states C2 and C3, with nominal values between 100°C and 55°C, rather than 120°C. This results in a lower saturation pressure. The break flow rate is reduced and is easily compensated by the available MHSI pump as the injection capacity increases with the lower RCP [RCS] pressure.

Note: In state C3, the RCP [RCS] loop level initiating the SI-signal is defined such that the MHSI effective injection occurs while the LHSI/RHR operating conditions are maintained.

Applicability of EPR_{4250} analysis results to EPR_{4500}

The characteristics of the EPR_{4250} are similar to the EPR_{4500}, from the point of view of RCP [RCS] injection and decay heat removal.

- The flow rate of MHSI at low pressure (P > 40 bar) and the flow rate of LHSI at low pressure (P > 20 bar) are identical for the two power levels
- The characteristics of the RIS/RRI/SEC [SIS/CCWS/ESWS] cooling chain are identical for EPR_{4250} and EPR_{4500}.
- The flow rate of the ASG [EFWS] and VDA [MSRT] are identical for EPR_{4250} and EPR_{4500}.
- The ASG [EFWS] tank inventory is the same in EPR_{4500} as in $EPR_{4250.}$
- The capacity of SG to remove decay heat is identical.

The analysis for EPR_{4250} was performed for an initial power of 39.5 MW. This value bounds the power of EPR_{4500} 10 hours after RT of 36 MW. [Ref-2]

In addition, the initial primary pressures for the EPR_{4250} and the EPR_{4500} are identical at 32.5 bar. After the break initiation, due to the identical heat removal capabilities of the SG, the primary pressure drops and reaches a plateau which is identical for EPR_{4500} and EPR_{4250} at about 4 bar. This pressure corresponds to that at which the break flow equals the MHSI injection. In effect, as the break flow is liquid, it only depends on the initial primary pressure. Hence the pressure is identical to that for EPR_{4250}.

The MHSI compensates for the break flow in the same way as for EPR_{4250}.

Further, the residual power is lower for EPR_{4500} than that modelled for EPR_{4250} and, the margin to saturation is larger than for EPR_{4250}. These are sufficient to ensure the continued operation of the LHSI/RHR pumps.

The similarities between EPR_{4500} and EPR_{4250} with respect to RCP [RCS] injection and cooling are evident. It can be concluded, without specific analysis, that the acceptance criteria are satisfied and the final state can be reached for the EPR_{4500}. This is based on the results presented for EPR_{4250} which show a significant margin to core uncovery and clad heat-up due to a lower residual power for EPR_{4500} than that modelled for EPR_{4250}.

7.1.6. Conclusion

The current analysis of the SB-LOCA in reactor state C shows that despite assuming the worst single failure and the unavailability of one MHSI train during normal operation:

- All the safety and acceptance criteria are met,
- The controlled state is reached relying only on F1A classified systems:
    - MHSI injection for RCP [RCS] water inventory.
    - Remaining LHSI/RHR train for RCP [RCS] is sufficient for heat removal. This is F1B classified, already in service and unaffected by the accident.
    - The safe shutdown state is reached relying only on F1A and F1B classified systems:
    - MHSI injection for RCP [RCS] water inventory.
    - Remaining LHSI/RHR train for RCP [RCS] and containment/IRWST heat removal.

7.2. SMALL BREAK LOCA (DN50 IN STATE D)

7.2.1. Accident definition

The initiating event is a non-isolatable break or leak, located either on the RCP [RCS] or on an LHSI/RHR line inside the containment.

It is defined as a break size of equivalent diameter less than 50 mm (~20 cm^2). This small break LOCA is classified as a PCC-4 event in state D, the cold shutdown state.

State D is a cold shutdown state with the RCP [RCS] open. In these conditions the SG cannot be used for decay heat removal. The RCP [RCS] pressure is 1 bar, the RCP [RCS] temperature is lower than 55°C. All the reactor coolant pumps have been shutdown and the RCP [RCS] is at the 3/4 loop operation level. During 3/4 loop operation preventive maintenance on the RIS/RRA [SIS/RHRS] trains does not occur. Three LHSI pumps are used in RHR mode to provide the heat removal, with one LHSI/RHR on stand-by.

The consequences of the event are analysed with the same objectives as in state A.

The objective of the analysis in state D is to show that the core remains adequately cooled. The aim is to demonstrate that the relevant safety and acceptance criteria are met, including the demonstration of the capability of the F1 systems to provide the safe shutdown.

Compared to state C3 ‘3/4 loop with RCP [RCS] closed’, the difference in terms of F1 mitigation measures are:

- All SG are unavailable for heat removal as the RCP [RCS] is open.

7.2.2. Typical sequence of events

7.2.2.1. From initiation to controlled state

In reactor state D, the RCP [RCS] break causes a loss of RCP [RCS] water inventory. Dedicated mitigation measures compensate for this loss.

The main mitigation for this accident is via the SI signal which is F1A classified. This signal actuates the MHSI pumps in the appropriate configuration. The shut-off head has previously been reduced to about 40 bar by opening each MHSI high capacity mini-flow line. This action is undertaken before entering LHSI/RHR mode. The SI signal utilises the RCP [RCS] loop level measurement, as in state C3 with 3/4 loop operation.

Thus, as in state C, only the MHSI pumps are available make-up in reactor state D.

The automatic actuation of the MHSI pumps prevents the loss of the main heat removal function performed by the systems in operation. The change in hot leg thermal-hydraulic conditions, RCP [RCS] loop level, HL temperature is such that the operation of LHSI/RHR trains is not impaired.

In the safety analysis it is conservatively assumed that the LHSI/RHR train already in operation in the broken loop before the occurrence of the event fails. This covers the possibility that adequate operating conditions may be lost in this train before effective MHSI injection is established.

The sequence of events following the event is as follows:

- Loss of one LHSI/RHR train for RCP [RCS] heat removal. As indicated above, the safety analysis conservatively assumes that the SB-LOCA causes the loss of the LHSI/RHR train connected to the broken loop. Therefore, two LHSI/RHR trains remain available for heat removal following the SB-LOCA.

- Break flow balance: the automatic MHSI start-up is sufficient to quickly balance the RCP [RCS] water inventory loss. For the largest break size (20 cm^2), the break flow is compensated by the MHSI.

7.2.2.2. From the controlled state to the safe shutdown state

The main actions needed to reach the safe shutdown state are as follows:

- Control of core sub-criticality. RCP [RCS] boration is unnecessary. However, the RBS [EBS], manually actuated, is available for boration in addition to the MHSI.

- Control of RCP [RCS] water inventory:

    - The available MHSI pumps provide sufficient make-up and support adequate LHSI/RHR operating conditions. The LHSI pump on stand-by before the accident can be manually actuated to provide an additional water supply, if required.

    - The LHSI/RHR heat removal, and the LHSI-SI if actuated, limits the IRWST temperature to below the temperature limit for adequate SI pump operation.

- Control of RCP [RCS] heat removal. The two remaining LHSI/RHR trains remove the core decay heat. The LHSI pump on stand-by before the accident can be manually actuated to provide additional heat removal capacity, if required.

- Containment integrity. The containment is closed when the reactor is in state D.

7.2.3. Safety criteria

The safety and acceptance criteria to be met are the same as those given in section 5.1.2 of Sub-chapter 14.4 for SB-LOCA, in reactor state A.

7.2.4. Definition of cases studied

In states A, B and C, the most limiting location for the break is the cold leg. It is assumed the break is either on a RCP [RCS] cold leg, or on an LHSI/RHR connecting line close to the cold leg of the RCP [RCS]. The small break LOCA studied corresponds to the largest PCC-4 break size of equivalent diameter 50 mm (~20 cm^2).

This case is analysed assuming the worst single failure.

There is no preventive maintenance in state D.

There is no requirement to consider LOOP in coincidence with the event in state D.

There is no need to perform a specific code calculation of the SB-LOCA transient to demonstrate adequate core cooling in state D. This demonstration is undertaken using simple mass and energy balance calculations showing the appropriate design of F1 systems for the heat removal (LHSI/RHR train) and RCP [RCS] injection (MHSI) functions.

The SB-LOCA is assumed to occur at the beginning of state D, which is the most limiting initial condition for the assessment of the possible core heat-up. It has the maximum decay heat for this plant state.

The safety demonstration shows the controlled state and safe shutdown can be reached and the fulfilment of the associated safety and acceptance criteria.

As required by the rules for safety analyses, the controlled state is reached using only F1A classified systems. The safe shutdown state is reached using only F1A or F1B classified systems.

No operator action is considered earlier than 30 minutes after the SI signal generation.

7.2.5. Initial and Boundary Conditions

7.2.5.1. Choice of single failure

Possible worst single failures are:

- The failure of one MHSI pump to start on demand following an SI-signal. This limits the RCP [RCS] water injection.
- The loss of the LHSI/RHR train, initially in stand-by. This degrades the heat removal function.
- Of these two possibilities the loss of one MHSI train is the worst single failure for this event.

7.2.5.2. Initial state

The initial conditions are chosen to minimise the RCP [RCS] water inventory at the time of the event, and to maximise the consequences of the event for the challenge to core cooling:

- The RCP [RCS] is at 3/4 loop operation with a minimum RCP [RCS] inventory.
- 3/4 loop operation with the RCP [RCS] open does not begin before than 30 hours after RT. This includes the presence of non-condensable gases. This is the minimum time after RT corresponding to the maximum decay heat level for this state.
- The RCP [RCS] pressure is 1 bar, the maximum value for 3/4 loop operation.
- The RCP [RCS] temperature is 55°C, the maximum value for 3/4 loop operation.

The status of LHSI and MHSI trains prior to and after the break is presented in Section 14.5.7 - Table 2.

7.2.5.3. Specific assumptions

7.2.5.3.1. *Decay heat*

The maximum residual heat is considered and corresponds to the term B + C (fission products + actinides) for the maximum decay heat curve.

30 hours after RT, the maximum core decay heat is 0.6 %. Assuming continuous power operation prior to reactor shutdown at the maximum core power of 102% NP, the maximum core power is 27.3 MW at the beginning of reactor state D. [Ref-1]

7.2.5.3.2. *Assumptions related to MHSI and LHSI/RHR (F1A)*

Prior to 3/4 loop operation, the SI signal on low RCP [RCS] loop level is automatically enabled.

The conservative assumptions are listed below: [Ref-1]

- 25 seconds maximum delay for MHSI pumps start-up, including signal delay.
- Restriction of the MHSI shutoff head to 40 bar in LHSI/RHR mode operation.
- 50°C maximum initial IRWST temperature and injection flow temperature.

In the safety demonstration, the MHSI pump injecting into the RCP [RCS] loop affected by the break is assumed to spill directly into the containment with no contribution to the RCP [RCS] injection. In addition, the LHSI/RHR train connected to the broken loop is conservatively assumed to be lost when the break occurs. This assumption is highly conservative and may not be retained in future safety analyses.

Based on the above conservative assumptions, the available RIS/RRA [SIS/RHRS] equipment is:

- two LHSI/RHR trains remain in operation following the break.
- The LHSI/RHR in stand-by may be manually actuated either in SI-mode or in RHR-mode. This is not assumed to occur before 30 minutes after the SI signal, if required. This actuation is not claimed in this study.
- one MHSI pump is available for SI, the three other trains are lost to the break, by application of the single failure (SF) and due to an additional unavailability.

The accumulators, which are isolated below a system pressure of 70 bar, are therefore not available.

Minimum characteristics for the LHSI/RHR and the SEC/RRI [ESWS/CCWS] heat exchangers are assumed to reduce the heat removal capacity of the LHSI trains available.

For more details please refer to the SB-LOCA in the state C study discussed in section 7.1 of this sub-chapter.

7.2.5.3.3. *Other assumptions*

During 3/4 loop operation, all reactor coolant pumps are stopped.

Non-F1 classified systems are not claimed e.g. RCV [CVCS] injection actuated by I&C limitation, since they may improve the outcome or have no impact on the accident mitigation.

7.2.6. Results and Conclusions

The safety demonstration relies on a quantitative assessment, based on mass and energy balances, of the capability of the F1A measures providing the required functions.

The analysis is performed for EPR_{4250}.

The loss of RCP [RCS] water inventory between the generation of the SI-signal on "low RCP [RCS] loop level", and the effective MHSI injection is less than 1 te, with no significant impact on the RCP [RCS] loop level (25 s × 30 kg/s [10] break flow rate = 0.75 te).

For the RCP [RCS] loop level, one MHSI pump is sufficient to compensate for the leak flow rate. As a consequence, the RCP [RCS] loop water level is maintained.

For the RCP [RCS] temperature, only a slight RCP [RCS] heat-up of about 10°C occurs due to the loss of the LHSI/RHR train assigned to the broken loop:

Residual Power at 30 hours:	0.6% NP
Maximum temperature rise in the RCP [RCS],until the residual power is completely removed by the two RIS/RRA [SIS/RHRS] trains	9°C
Stable RCP [RCS] Temperature	About 64°C

The conditions requires for LHSI/RHR operation are maintained throughout the transient.

After actuation of the MHSI and stabilisation of the RCP [RCS] temperature, the controlled state is reached, corresponding to the safe shutdown state.

Following the SB-LOCA, the decrease in RCP [RCS] water inventory is negligible. The core remains covered throughout the transient, with no fuel clad heat-up. All the safety and acceptance criteria are met, using only F1 measures. These requirements are met despite the assumption of the worst single failure.

The analysis of the SB-LOCA in reactor state D shows that even with the assumption of the worst single failure and the assumption of an additional unavailability, all the safety and acceptance criteria are met and the safe shutdown state is reached using only F1A and F1B classified systems.

EPR at 4500 MWth

The analyses have not been repeated for EPR_{4500} as the results would not be significantly changed, for the following reasons:

[10] 30 kg/s is a conservative value for break flow rate and refers to 1 bar and 50°C. Even if the IRWST water temperature were 10°C (minimum value), the 30 kg/s leak flow rate would still be conservative.

- One pump MHSI is capable of matching the break flow. Thus, the RCP [RCS] loop level is maintained at the same magnitude for EPR_{4500}.

- The loss of one RIS/RHR train injecting into the broken loop is more important for the primary heat up,. The residual power, 30 hours after RT is equal to 27.3 MW for EPR_{4500} compared to 25.5 MW for EPR_{4250}. This is an increase of 7%, as shown in Section 14.5.7 - Table 3. However, the RCP [RCS] heat-up will remain of the same order of magnitude at about 10°C. Thus the RIS/RRA [SIS/RHRS] operation is not impaired. [Ref-1]

7.3. SYSTEM REQUIREMENTS

RIS [SIS]

SB-LOCA 20 cm^2 (∅ < 50 mm) in shutdown state: no loss of LHSI/RHRS function	MHSI flow rate with large mini-flow line open at 6 bar
LOCA in shutdown state: no core uncovery accounting for one MHSI pump	MHSI flow rate with large mini-flow line open at 10 bar

SECTION 14.5.7 - TABLE 1

Initial State
Prior to Occurrence of the Break

	Reactor coolant pumps	LHSI	MHSI
Loop 1	OFF	ON in RHR-mode	Unavailable [11]
Loop 2	ON	OFF in stand-by	Available
Loop 3	ON	OFF in stand-by	Available
Loop 4	OFF	ON in RHR-mode	Available

After Occurrence of the Break

	Reactor coolant pumps	LHSI	MHSI
Loop 1	OFF	ON in RHR-mode	OFF unavailable before SB-LOCA
Loop 2	OFF conservative reactor coolant pump cut-off see section 7.1.5.3.2	OFF in stand-by (not used in the PCC analysis)	ON after SI-signal
Loop 3	OFF conservative reactor coolant pump cut-off see section 7.1.5.3.2	OFF in stand-by (not used in the PCC analysis)	OFF single failure to operate after SI-signal
Loop 4 20 cm^2 CL break	OFF	LOST AT BREAK bounding conservative assumption see section 7.1.5.3.2	LOST AT BREAK bounding conservative assumption see section 7.1.5.3.2

[11] Other equivalent configuration could be MHSI unavailable in loop 3, and single failure on MHSI in loop 1 following the SI signal.

SECTION 14.5.7 - TABLE 2

Initial State

Prior to occurrence of the break		
	LHSI	**MHSI**
Loop 1	RHR-mode	Unavailable
Loop 2	RHR-mode	Available
Loop 3	RHR-mode	Available
Loop 4	In stand-by	Available

After occurrence of the break		
	LHSI	**MHSI**
Loop 1	RHR-mode	Unavailable
Loop 2	RHR-mode	Available
Loop 3	Lost at break	Lost at break
Loop 4	In stand-by	SF

SECTION 14.5.7 - TABLE 3 [Ref-1]

Results and Conclusion

Decay heat at 30 hours:	**0.6% NP**
Maximum RCP [RCS] heat-up, until the residual decay heat is fully removed by the two remaining LHSI/RHR trains:	10°C
Stabilised RCP [RCS] temperature:	About 65°C

SECTION 14.5.7 - TABLE 4 [Ref-1]

Initial Conditions at 4250 MWTH
20 cm² (∅ 50 mm) Cold Leg Break (State C, PCC-4)

Parameters	Limiting values	
Reactor coolant system		
Initial reactor power (% of full power)	0.91% (39.5 MW)	max
Initial average RCP [RCS] temperature	122.5°C	max (nom + 2.5)
Initial reactor coolant pressure	32.5 bar	max (nom + 2.5)
Number of LHSI/RHR trains in operation	2 (loops 1, 4)	
Number of reactor coolant pumps in operation	2 (loops 2, 3)	
Pressuriser water volume / level (%/m^3/m)[12]	39 / 28.5/4.29	max (nom + 5% R)
Steam generators		
Initial water temperature	122.5°C	max (nom + 2.5°C)
Initial steam pressure	2.15 bar	P_{sat} 122.5°C
Initial SG level	15.7 m	nom

[12] Corresponding to the SG nominal heat transfer surface with no plugging

SECTION 14.5.7 - TABLE 5 [Ref-1]

Sequence of Events
Typical results for 20 cm² (∅ 50 mm) Cold Leg Break (State C, PCC-4)

TIME (s)	EVENT
0	Break opening
0	Reactor coolant pumps trip
146	Pressuriser emptying
207	Safety injection signal on ΔP_{sat} < 5 bar
325	Minimum RCP [RCS] subcoolled margin
410	Leak flow matched by MHSI pump with 1 pump injecting
2500	End of calculation ($\Delta T_{sat} \approx 25°C$)

SECTION 14.5.7 - TABLE 6

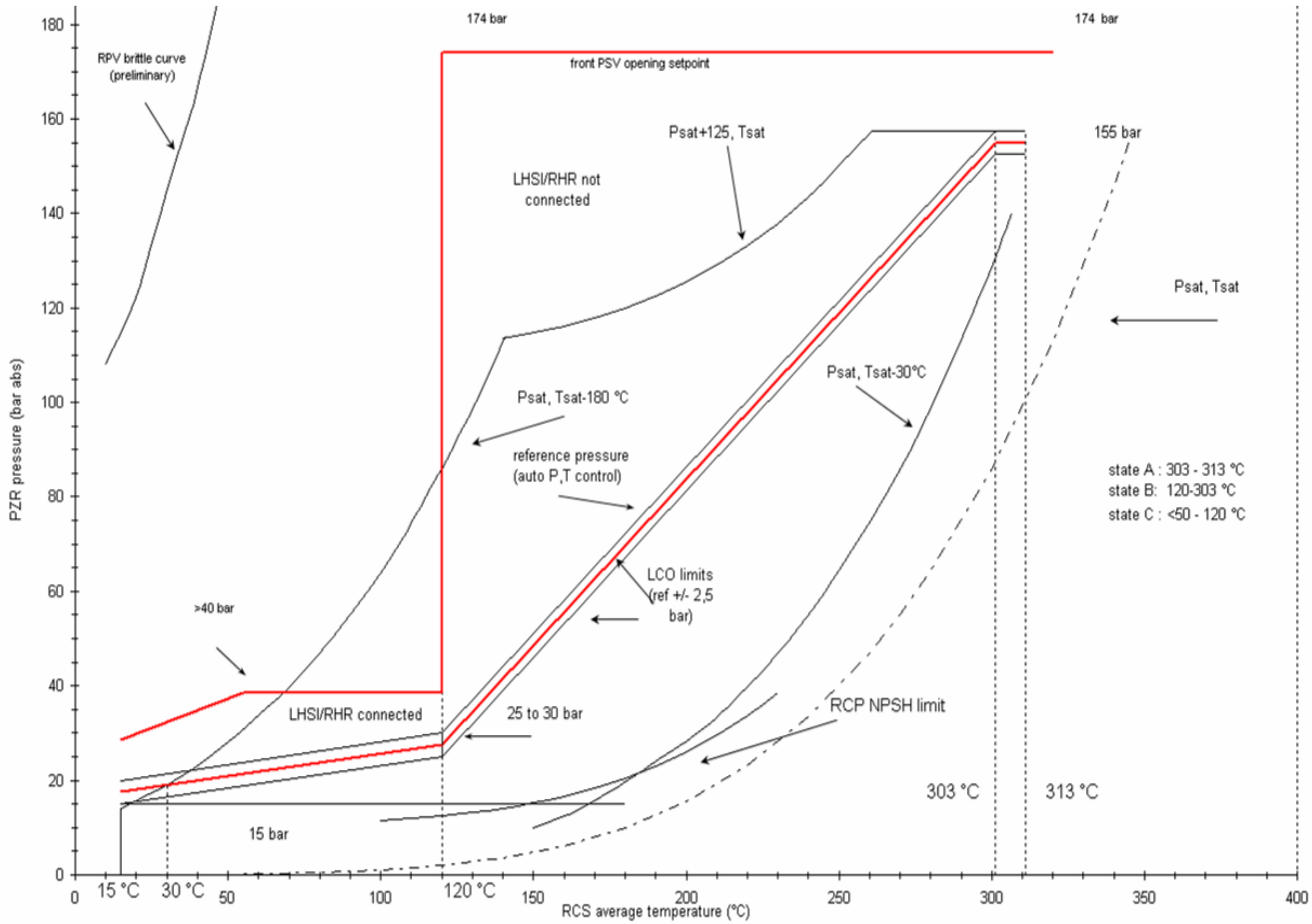
RCP [RCS] Plant Status in State C

	RCP [RCS] thermal-hydraulic STATUS			**EQUIPMENT OPERATING**		**EQUIPMENT on STAND-BY**		
						SAFETY INJECTION (RIS [SIS])	RCP [RCS] HEAT REMOVAL	
	RCP [RCS] Pres.	RCP [RCS] Temp.	RCP [RCS] water inventory	Reactor coolant pumps on	LHSI/RHR on	Automatic actuation on SI-signal	Manual actuation	Automatic actuation on SG-pressure
STATE C1	30 bar	120°C ↓ 100°C	Pressuriser level	2 reactor coolant pumps on	2 LHSI/RHR on	3 MHSI (1)	2 LHSI	2 SG
STATE C2	30 bar	100°C ↓ 55°C	Presssuriser level	2 reactor coolant pumps on (T> ≈70°C) 1 reactor coolant pump on (T> ≈55°C)	4 LHSI/RHR on	3 MHSI (1)		2 SG
STATE C3	30 bar ↓ 1bar	55°C	Pressuriser level ↓ 3/4 loop	0 reactor coolant pump on	3 LHSI/RHR on	3 MHSI (1)		2 SG

Note: (1) possible unavailability of 1 MHSI train

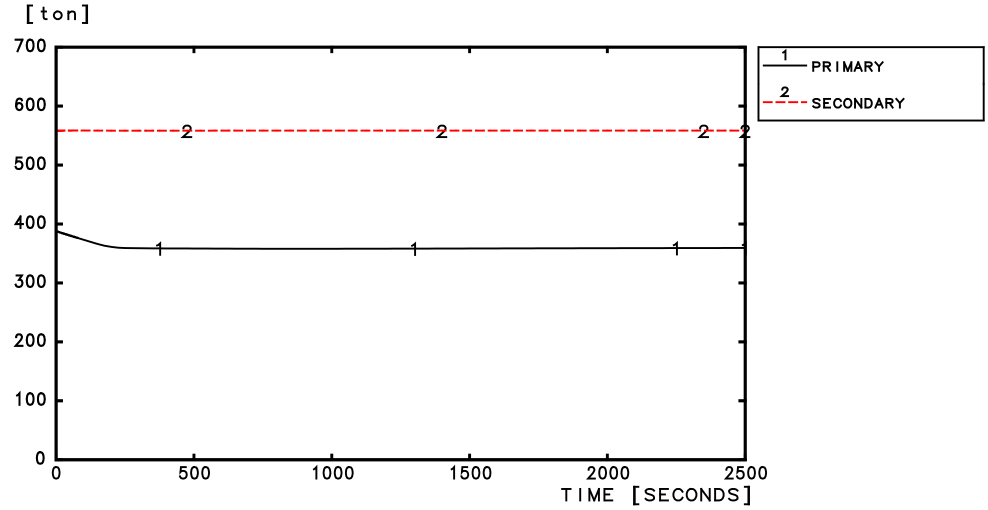
SECTION 14.5.7 - FIGURE 1 [Ref-1]

Typical (P, T) Reactor Operating Range

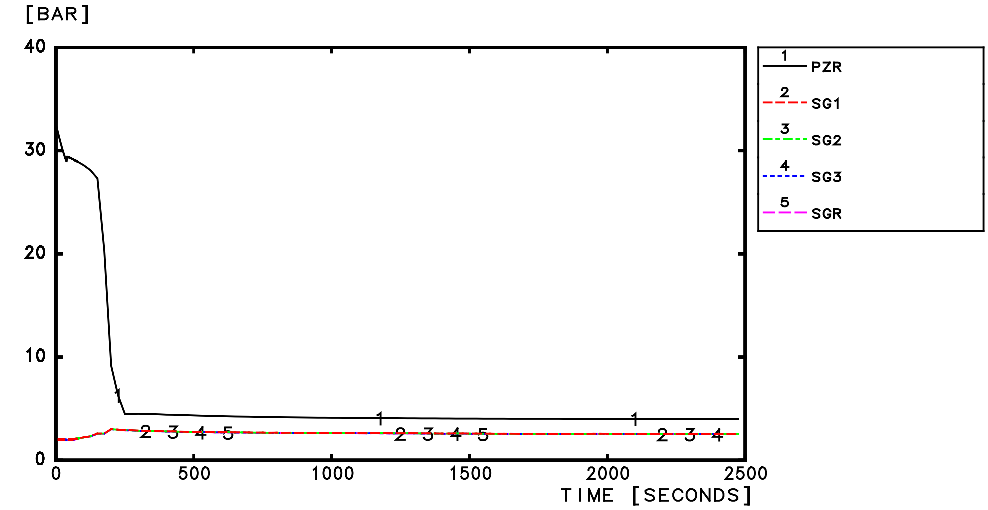


SECTION 14.5.7 - FIGURE 2 [Ref-1]

RCP [RCS] and Secondary Side Water Inventories
RCP [RCS] and Secondary Side Pressures
Typical Results for 20 cm2 (∅ 50 mm) Cold Leg Break (State C, PCC-4)



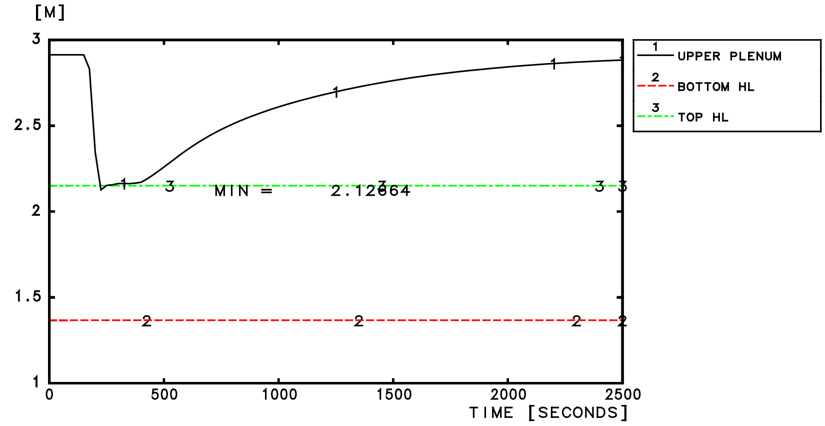
PRIMARY AND SECONDARY WATER MASS



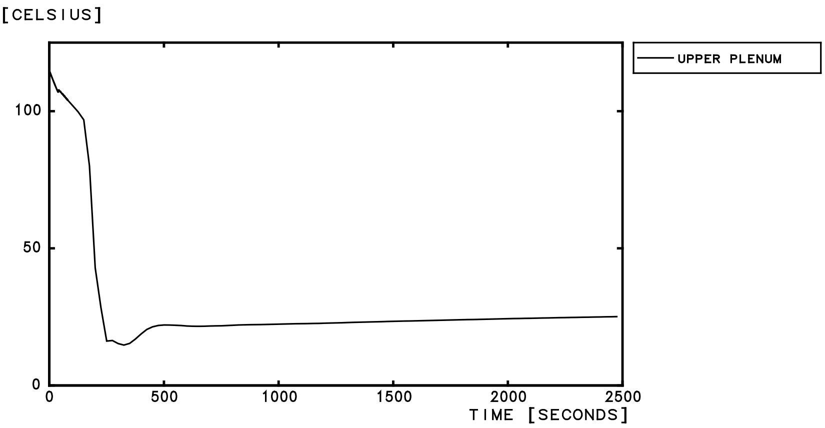
PRIMARY AND SECONDARY PRESSURES

SECTION 14.5.7 - FIGURE 3 [Ref-1]

RCP [RCS] Average Temperature
Total Break and RIS [SIS] Flow Rates
Typical Results for 20 cm² (∅ 50 mm) Cold Leg Break (State C, PCC-4)



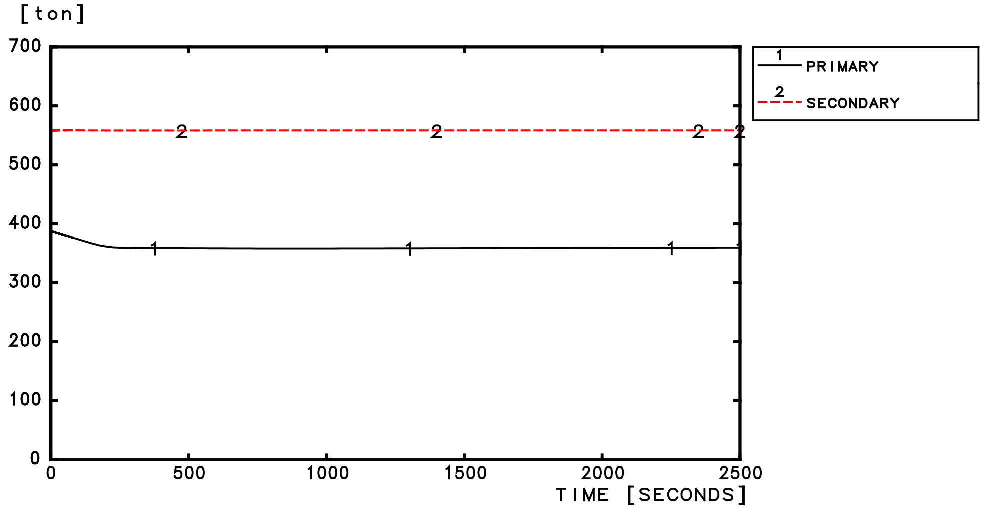
UPPER PLENUM COLLAPSED LEVEL



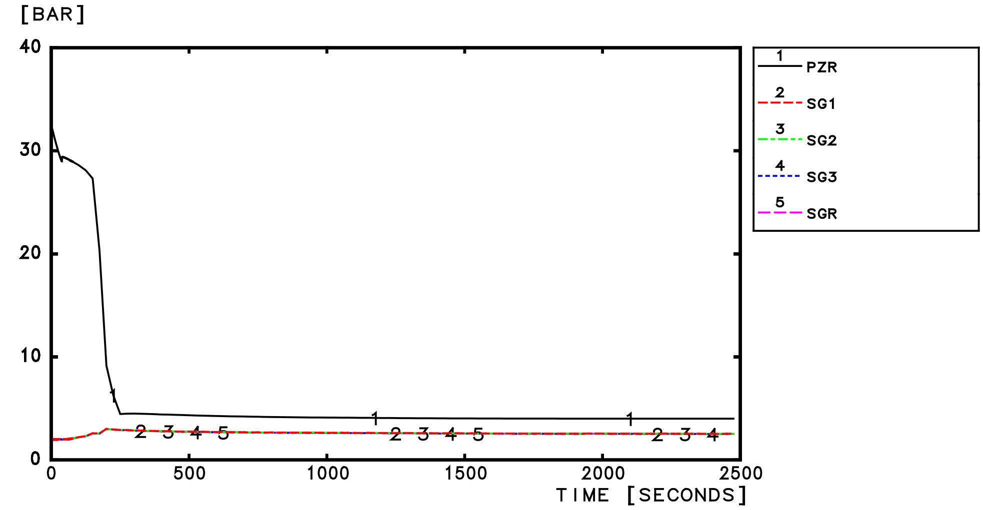
DELTA TSAT

SECTION 14.5.7 - FIGURE 4 [Ref-1]

RCP [RCS] and Secondary Side Water Inventories
RCP [RCS] and Secondary Side Pressures
Typical Results for 20 cm² (∅ 50 mm) Cold Leg Break (State C, PCC-4)



PRIMARY AND SECONDARY WATER MASS



PRIMARY AND SECONDARY PRESSURES

8. REACTOR COOLANT PUMP SEIZURE (LOCKED ROTOR)

The analysis is only performed in reactor state A. It also covers the PCC-4 event ‘Reactor Coolant Pump Shaft Break’, described in section 9 of this sub-chapter, which is virtually identical and bounded in terms of thermal-hydraulic behaviour.

8.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

The accident considered is an instantaneous seizure of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is assumed to be rapidly reduced to zero within 1 to 4 seconds. This leads to the initiation of a reactor trip following a “low-low loop coolant flow” signal in the affected loop. It is assumed that the flow reduction to zero in the affected loop occurs in 2 to 3 seconds. In the case of a locked rotor, a partial trip would also occur from low loop coolant flow in the affected loop. However, this is not considered and therefore the reactor trip is assumed to be triggered following the locked rotor.

Following the initiation of a reactor trip, heat stored in the fuel rods continues to be transferred to the primary coolant, although at a reduced rate as the core power falls. At the same time, heat transfer to the SG is reduced, due to the decrease in the primary flow and due to the increase in secondary temperature. The steam flow to the turbine is zero following reactor trip and turbine trip. The resultant expansion of the primary coolant causes a surge into the pressuriser and therefore a pressure increase throughout the reactor coolant system. The surge into the pressuriser compresses the steam volume, actuates the automatic spray system and may open the pressuriser safety valves.

In the analysis, the pressuriser spray is not taken into account as it has a beneficial effect on the maximum primary pressure and is not an F1A system.

8.1.1. Controlled State and Safe Shutdown State

It must be demonstrated that the controlled state can be reached using automatic F1A functions, and the safe shutdown state can be reached on F1A and F1B functions. During that process, the following safety and decoupling criteria must be met:

Safety Criteria

Radiological limits for PCC-3/PCC-4.

Decoupling Criteria

- The peak fuel cladding temperature must remain below 1482°C.
- No more than 10% of fuel rods should enter DNB.

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

- Reactor trip is initiated from a “low-low coolant flow in one loop” signal with a setpoint of 25% nominal flow rate.
- Four VDA [MSRT] for secondary side heat removal and pressure limitation actuated on reaching "high SG pressure” SG pressure > 95.5 bar.

- Four ASG [EFWS] trains for secondary side water supply actuated on reaching "low SG level" (SG level < 7.85 m).

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

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

8.2. METHODS AND ASSUMPTIONS

8.2.1. Method

The purpose of the analysis is to determine the maximum fuel clad temperature, the number of fuel rods entering DNB and the minimum DNBR.

The computer codes PANBOX/COBRA-3CP (Version 2.1RZ) described in Appendix 14A are used for the analysis.

Section 6 of Sub-chapter 14.3, assessing the LOOP event, describes the method used to calculate the DNBR LCO. In addition, the method used for the calculation of the transient DNBR (DNBR2 and DNBR3) is identical to the method used in section 6 of Sub-chapter 14.3.

The calculation of the number of fuel rods entering DNB considers the probability distribution associated with the DNBR calculation uncertainties.

It is judged that a dedicated analysis of the overall plant behaviour is not necessary.

8.2.2. Event and Qualification of the Panbox/Cobra-3cp Models

This event belongs to the Complete or Partial Loss of Reactor Forced Primary Coolant Flow category of transients.

The phenomena addressed here are similar to those of sections 6 and 8 of Sub-chapter 14.3. Therefore, the same scope of code qualification applies.

The specific phenomena for the present case, backflow in the affected loop, were verified against other cases by recalculation of a corresponding event at the KKG/BAG Nuclear Power Plant [Ref-1].

8.2.3. Initial and Boundary Conditions

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

For details of the parameters below, refer to the reference analysis presented in Appendix 14B.

The initial conditions are the most conservative for the DNBR limit and therefore the DNBR LCO is defined by the limiting PCC-2 event, LOOP described in section 6 of Sub-chapter 14.3.

The analysis is performed assuming a pump and core flow coast down time of 1 second.

8.2.3.1. Reactor Trip

RT on the low-low loop flow rate setpoint of < 25%, including measurement uncertainties, is claimed in the analysis. A conservative delay is assumed between the setpoint being reached and the start of RCCA drop.

The RCCA worth versus time is calculated based on the RCCA worth as a function of RCCA position. This is calculated by PANBOX using the actual 3D power distribution.

8.2.3.2. Moderator Temperature/Density Coefficient

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

The density reactivity feedback is calculated for the conservative top-peaked power shape with an axial offset of 18% and using the bounding Doppler coefficient.

8.2.3.3. Doppler Coefficient

The Doppler coefficient is at its maximum absolute value, including uncertainties. This minimises the negative reactivity feedback.

8.2.3.4. Evaluation of the Hot Spot Temperature (Transient)

The zirconium water reaction is not significant because the clad temperature remains below the admissible limit. The pellet/clad gap heat transfer coefficient assumed in the COBRA-code corresponds to the minimum value for α-gap of 9700 W/m^2K for a pessimistic power reduction. [Ref-1]

8.2.3.5. Choice of Single Failure and Preventive Maintenance

Single failure:

- No single failure is assumed, as has no effect. For example, a stuck rod following RT is only significant after the minimum DNBR occurs.

Preventive maintenance:

- No preventive maintenance is assumed prior to reaching the controlled state as it has no impact on the evaluated transient.

- From the controlled state to safe shutdown state, the assumptions for the reference case apply.

8.3. RESULTS AND CONCLUSIONS

8.3.1. From Initiating Event to Controlled State

The PCC transient is not recalculated in the PCSR. The capability to fulfil the safety and decoupling criteria derives from the results of the PCC transient analysis contained in section 15.2.8 of the BDR-99 as presented in Appendix 14B and from the comparison of the relevant characteristics between EPR 4500 MWth and EPR 4900 MWth modelled in the BDR-99.

8.3.1.1. BDR-99 Status

In the BDR-99 study discussed in section 2.8 of Appendix 14B the minimum DNBR including uncertainties is 0.8. This minimum value is obtained using very conservative decoupled assumptions especially for $F\Delta H$, which is increased to more than 1.9. This is beyond the actual core design limit of 1.65. As a result of this $F\Delta H$ increase, the local heat flux, over 500 W/cm is significantly higher than the maximum value of 470 W/cm defined by LOCA analysis.

The number of fuel rods entering DNB is 0.6% and the maximum heat-up of the fuel rod clad is 835°C. Consequently, the decoupling criteria concerning the number of fuel rods in DNB and the maximum clad temperature are both met with large margins.

8.3.1.2. Relative Differences between EPR 4500 MWth and EPR 4900 MWth

The main differences between the EPR 4500 MWth and the EPR 4900 MWth are the same as those outlined in section 6 of Sub-chapter 14.3 discussing the LOOP event.

8.3.1.3. Conclusions for EPR 4500 MWth

The core power for EPR 4500 MWth is lower, both globally: 4500 MW_{th} vs. 4900 MW_{th} and locally: 450 W/cm vs. 500 W/cm. Consequently, as the flow reduction transient is the same, the minimum DNBR calculated for BDR bounds the minimum DNBR to be expected for EPR 4500 MWth. Therefore the decoupling criteria of ‘less than 10% of fuel rods in DNB’ and ‘PCT < 1482°C’ are both met for EPR 4500 MWth.

8.3.2. From Controlled State to Safe Shutdown State

This transition is not explicitly analysed since it is bounded by the respective analyses of other events (reference cases). Section 14.5.8 - Table 1 below presents the reference cases and the associated safe shutdown criteria met in each case.

Impact of the safety classification change on the normal spray operations

The F1B safety classification of the pressuriser normal spray leads to the possibility of claiming this system in the transfer to the LHSI/RHR connecting conditions. The analysis of this modification of the classification is shown on the feedwater line break, section 3 of this sub-chapter for ASG [EFWS] tank sizing, and on the steam generator tube rupture, section 10 of this sub-chapter concerning radiological discharge to the atmosphere.

8.4. SYSTEM SIZING

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

SECTION 14.5.8 - TABLE 1

Safe Shutdown Criteria for Reference Cases

Criteria	Reference case	Remark/Reason
Sub-criticality	Section 13 of Sub-chapter 14.3 RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant	In case of a stuck rod the rod worth is sufficient
Max. activity release	Section 6 of this sub-chapter Large Break LOCA.	10% core damage is assumed for the LOCA based activity release calculation.
Heat removal (assured by RIS [SIS]/RRI [CCWS] /SEC [ESWS] or LHSI/RRI [CCWS] /SEC [ESWS]	Section 3 of this sub-chapter Feedwater system piping break	Only one train is available for cooldown.

9. REACTOR COOLANT PUMP SHAFT BREAK

9.1. INTRODUCTION

The thermal-hydraulic aspects and core protection are bounded by the analysis of ‘Reactor coolant pump seizure (locked rotor)’; presented in section 8 of this sub-chapter. Therefore, the results and conclusions of the locked rotor event fully apply to the RCP [RCS] pump shaft break.

9.2. SYSTEM SIZING

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

10 . STEAM GENERATOR TUBE RUPTURE (2 TUBES IN 1 SG)

A steam generator tube rupture involving failure of two tubes in one steam generator (4A-SGTR) is classified as a PCC-4 event.

10.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

10.1.1. General Concerns

The accident considered is due to a complete double ended severance of two steam generator tubes (4A-SGTR) located in the same steam generator.

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

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

Section 14.5.10 - Figure 16 shows a schematic of the Main Steam Line showing the main steam relief systems and valves relevant to the present study.

10.1.2. Typical Sequence of Events

The typical sequence of events in the 4A-SGTR, disregarding possible effects of an additional single failure, is described below.

10.1.2.1. Period from initiation to pressure equalisation between primary side and affected SG (short term)

a) From initiating event to the attainment of controlled state

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

The loss of primary coolant causes a decrease in the primary pressure and radioactive contamination of the secondary side. Reactor trip can occur on either “Pressuriser pressure < MIN2”, or “SG level > MAX1” in the affected SG. Which signal occurs first depends on the initial state and operating conditions of the plant.

At Power Operation

During power operation, the SGTR flow contributes to the production of steam. The level in the SGa does not significantly increase. Therefore the "Pressuriser pressure < MIN2" reactor trip signal is generated prior to the "SG level > MAX1" signal.

Zero Power Operation

At hot shutdown initial conditions, the heat transferred from the primary to the secondary side is insufficient to boil all the SGTR flow. Consequently, the level in the affected SG increases. Reactor trip occurs and the affected SG is isolated on the feed side following receipt of an "SG level > MAX1" signal. The ARE [MFWS] and ASG [EFWS], if already actuated, are both isolated from the affected SG.

For initial at-power operation, the reactor trip signal automatically trips the turbine, and the steam generator pressure rapidly increases. The GCT [MSB] is assumed to be unavailable as it is not F1A/F1B classified. (Note that the GCT [MSB] would be unavailable if a Loss of Offsite Power (LOOP) occurred following turbine trip.) When the VDA [MSRT] pressure setpoint is reached, the VDA [MSRT] isolation valve opens and contaminated steam is discharged to the atmosphere from the affected SG.

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

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

On generation of the SI signal, the MHSI pumps are actuated. However, they do not inject initially, as the RCP [RCS] pressure is above their shut-off head.

The controlled state is reached when MHSI injection, supplemented by RCV [CVCS] injection if available, are able to match the SGTR flow rate. However, since the leak is continuing at this point, the affected SG continues to fill with contaminated water and the activity release to atmosphere continues.

b) From the attainment of the controlled state to pressure equalisation between the primary and the affected SG

The affected SG is identified and isolated automatically or manually. The isolation of the affected SG is performed by raising the setpoint of the VDA [MSRT] to above the MHSI shut-off head but below the MSSV pressure setpoint, and closing the VIV [MSIV]. This prevents any further release of radioactivity to the atmosphere. The RCV [CVCS] charging lines are isolated automatically when the partial cooldown is complete and the level in the affected SG exceeds the MAX2 setpoint.

Following isolation of the affected SG by closure of the VDA [MSRT] and VIV [MSIV], the continuing flow from the break results in an increase in the pressure in the affected SG. Eventually the pressures on primary side and secondary side of the affected SG pressure equalise, and the break flow falls to close to zero. This corresponds to the end of the short term phase. A significant margin to overfilling is maintained; consequently, only steam is discharged to atmosphere during this phase of the transient.

The short term phase, until pressure equalisation, uses only automatic F1A/F1B signals and systems.

10.1.2.2. From pressure equalisation to the safe shutdown state (long term)

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

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

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

Boration and RCP [RCS] Cooldown

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

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

The RCP [RCS] cooldown is performed using the unaffected steam generator(s). The cooldown is performed with the MHSI operating, to avoid perturbing the pressure balance between the primary side and the affected SG.

Final Depressurisation

At the end of the RCP [RCS] cooldown phase, the RCP [RCS] pressure is close to the MHSI shut-off head, as the MHSI is operating. This pressure is higher than the maximum operating pressure of the LHSI in RHR mode (30 bar). Before the final depressurisation, if the affected SG level is too high, the operator first opens the transfer line between the affected SG and its partner SG to limit the risk of water-hammer in the steam line of the affected SG. This also prevents overfilling of the affected SG, which could lead to liquid phase discharge to atmosphere and a consequential increase in the activity release. Prior to the fluid transfer from the affected SG, conditions in the partner SG are adjusted to allow it to accept the additional volume from the affected SG by:

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

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

The transfer of liquid to the partner SG results in a slight depressurisation of the affected SG, but the pressure in the affected SG remains above the RHR connection criteria of 30 bar. Once the level in the affected SG falls below MAX2 (NR), the VDA [MSRT] on the affected SG is opened. This allows depressurisation to below 30 bar.

10.1.2.3. Radiological consequences

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

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

The radiological consequences are assessed in Sub-chapter 14.6.

10.1.2.4. Precautions limiting event frequency and consequences

The probability and consequences of SGTR events are reduced through the following precautions:

- the SG tube material is highly ductile,
- the blowdown system is located at the bottom of the SG tube bundle and is designed to prevent solid deposits forming on the tube plate,
- the secondary water is conditioned chemically, thereby protecting the SG tubes from corrosion,
- the steam generators are designed to prevent any fragments of material released into the main feedwater system from directly impacting on the steam generator tubes,
- the SG support plates maintaining tube bundle are designed (type of material, geometry of borings) to prevent denting in the tubes and, in the case of a double-ended guillotine rupture, to prevent pipe whip which could cause propagation of damage to neighbouring tubes,
- activity of the secondary side water and steam is controlled by continuous monitoring to ensure compliance with defined limits.

10.2. SAFETY CRITERIA

The safety criteria to be complied with in this event are the radiological limits for PCC-4 events discussed in Sub-chapter 14.0.

The specific criteria to be met are the dose equivalent limits following activity release to the atmosphere.

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

- no core damage (fuel cladding integrity),

- MSSV opening to be avoided to prevent the possibility of an MSSV failing in the open position. A stuck open MSSV would not be isolable, so must be prevented in order to avoid the risk of the RCP [RCS] inventory being discharged to the environment via the ruptured SG tubes,

- requirement to return to RHR connection conditions of boration, depressurisation and heat removal with achievement of safe shutdown conditions using only F1A/F1B systems.

LOCA studies are performed to demonstrate avoidance of core damage, core coolability and prevention of increased activity concentrations on the primary side.

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

- prevention of SG overfilling, to avoid increased activity release by liquid discharge to the atmosphere,

- minimisation of the risk of SGTR leak backflow, to avoid problems due to water slugs with low boron concentration entering the primary circuit when the reactor coolant pumps are not operating (e.g. following a loss of offsite power).

10.3. METHODS AND ASSUMPTIONS

10.3.1. Methods of Analysis

The SGTR accident analysis is performed with the CATHARE code described in Appendix 14A using a realistic deterministic methodology [Ref-1].

The realistic deterministic methodology is characterised by two main features:

- the key code models are realistic though conservatively oriented, bounding the experimental results without excessive conservatism, and

- the initial and boundary conditions are conservatively selected.

The basic elements of the realistic deterministic methodology consist of:

- phenomenological analysis of the accident scenario, and the identification of the key phenomena,

- confirmation of adequacy of the code to calculate the accident scenario, based on physical understanding, experimental data base, code assessment examination, and supplemented when necessary by sensitivity studies,

- an evaluation of calculation uncertainty with emphasis on dominant parameters (through sensitivity studies as necessary), or verification based on a bounding conservative approach of key phenomena by the code, relying on the assessment matrix of the code,

- the introduction, where necessary, of conservative biases to reflect the uncertainties in key phenomena. These are introduced either in the code model, in the nodalisation scheme, or in the boundary conditions, and

- the use of conservative assumptions for initial and boundary conditions.

The dominant phenomena to be simulated in the SGTR transient are:

- the SGTR flow and the resultant SG filling behaviour,

- the RCP [RCS] draining (pressuriser) and depressurisation to reach equilibrium with the pressure in the affected SG,

- the asymmetric RCP [RCS] heat removal via the unaffected SGs in subcooled RCP [RCS] conditions,

- the RCP [RCS] cooling, and the RCP [RCS] depressurisation to LHSI in RHR mode connection conditions.

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

- the qualification of correlations and physical laws using separate effect tests (SET) or component tests, for example:

    - CATHARE SGTR model for accurate SGTR break flow prediction,

- validation of the axial SG model, with an N4 SG-type economiser using the MEGEVE small scale model tests [Ref-2],

- overall validation of the code by simulation of integral effect tests (IET), covering a wide range of representative PWR transients on small-scale facilities, for example:

    - BETHSY 4.3b ‘6 SGTR’ Test [Ref-3] in which CATHARE accurately predicted the mass discharged at the faulted SG relief valve, the faulted SG mass inventory, the RCP [RCS] mass inventory and distribution. It also accurately predicted the formation of the large steam void inside the faulted SG tubes, the slow RCP [RCS] depressurisation during the draining of the faulted SG, the restart of the loop circulation after the reactor coolant pump start in the affected loop, and the consequential rapid depressurisation due to the condensation and collapse of the faulted SG tubes steam void.

The transient analysis relies on the application of the conservative PCC analysis rules defined in Sub-chapter 14.0. The rules include adoption of the conservative deterministic assumptions for all relevant boundary conditions for the decoupling criteria being considered. The conservative assumptions adopted include:

- the characterisation of the initiating event,

- the assumed plant initial conditions which allow for control dead band limits, maximum measurement uncertainties, and

- the performance of the protection and mitigation actions which assume maximum uncertainty on each I&C measurement and signal delay, and on each system response time and capacity.

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

10.3.2. Main Assumptions

10.3.2.1. Accident Definition

The cases studied correspond to the double ended guillotine rupture of two tubes in the same steam generator, allowing unimpeded discharge from both ends of the severed tubes.

The tube rupture is located at the bottom of SG-tubes bundle, on the cold side. This location results in a maximum SGTR leak flow rate.

For the accident analyses, a loss of offsite power (LOOP) is included in the modelling of the accident, if this assumption is conservative.

10.3.2.2. Protection and Mitigation Actions

Following the SGTR event, automatic protection systems followed by operator actions, using F1-classified systems, act to trip the reactor, to remove residual heat, to match the primary to secondary leak flow rate, to limit contaminated SG liquid mass released to the atmosphere, and, finally, to take the reactor to a safe shutdown state.

The different automatic protection signals and alarms, which could occur during the in operation of mitigation systems following an SGTR event, are linked either to the reactor coolant depressurisation or to the level increase in the affected SG.

An F1A reactor trip is initiated on one of the following signals:

- “Pressuriser pressure < MIN2",
- “SG level > MAX1".

A high activity signal is received in the control room at the beginning of the transient due to the tube rupture. However, in contrast to the 1-tube rupture case, manual trip after the receipt of the high activity signal is not necessary because, in the 2-tube rupture case, one of the two automatic trips occurs within 50 minutes.

The other F1A automatic protection actions include the following:

- Turbine trip: the turbine trip is actuated on an RT check-back signal.
- Safety injection signal and Partial Cooldown: the RIS [SIS] is actuated on a "Pressuriser pressure < MIN3" signal. A partial cooldown using all SGs, including the affected one, is initiated if it has not already been actuated by another signal.

- Partial Cooldown can also be actuated on the "SG level > MAX2" signal[2] if partial cooldown has not already been actuated by another signal. The partial cooldown is performed by all SGs, including the affected SG.

- Isolation of the affected SG (feed side). The affected SG is isolated on a "SG level > MAX1" signal. This initiates isolation of the full load ARE [MFWS] (MAX1 narrow range) and the ASG [EFWS] line if the ASG [EFWS] was previously actuated (MAX1 wide range). This signal is SG related.

- Isolation of the affected SG (steam side). The affected SG is isolated on the "SG level > MAX2" at the end of partial cooldown signals. The isolation is performed by closure of the VIV [MSIV], increasing the VDA [MSRT] pressure setpoint to above the MHSI shut-off head but below the MSSV pressure setpoint.

- ASG [EFWS] actuation. The ASG [EFWS] train is actuated on "SG level < MIN2" in the corresponding SG. This signal is SG specific. The time delay between the setpoint being reached and effective ASG [EFWS] flow delivery is defined in accordance with the assumption of LOOP or no LOOP.

- VDA [MSRT] actuation. The VDA [MSRT] opens and performs the function of heat removal with pressure control, when the SG pressure reaches the VDA [MSRT] setpoint, "SG pressure > MAX1".

- VIV [MSIV] isolation. VIV [MSIV] closure is initiated on a "SG pressure < MIN1" or "SG pressure drop > MAX1" signal. In this case all the MS lines are isolated as the signal is not SG specific,

- VDA [MSRT] isolation. The VDA [MSRT] of the corresponding SG is isolated when the SG pressure reaches the VDA [MSRT] setpoint, "SG pressure < MIN3". This is performed by closure of the associated Main Steam Relief Isolation Valve (MSRIV). This signal is SG specific.

For long term mitigation of the event, operator actions are performed to transfer the plant to the safe shutdown state. The operator performs the boration and cooldown actions using the unaffected SG(s) prior to the final depressurisation of the RCP [RCS] and the affected SG to the connection conditions for LHSI in RHR mode.

10.3.2.3. Operator Actions

No operator action is considered before 30 minutes after the first indication of the event occurrence in the main control room (assumed to be reactor trip or high secondary activity depending on which occurs first). A high secondary activity signal would be received at the start of the transient: however, a conservative approach is adopted and operator action in the control room is not credited until 30 minutes after reactor trip. When operator action is needed local to the plant, the delay is assumed to be 1 hour.

The actions presented below are based on the version of the Emergency Operating Procedures (EOP) current at the time of the study.

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

The F1B operator actions associated with the affected SG isolation are:

- manual ASG [EFWS] isolation, if not done automatically (to control SG level, secondary side activity)
- manual ARE [MFWS] isolation, if not done automatically (to control SG level, secondary side activity)
- manual VIV [MSIV] isolation, if not done automatically (to control SG pressure)
- manual VDA [MSRT] setpoint increase, if not done automatically (to control SG pressure)
- manual actuation of cooldown, if not done automatically (to control SG level, secondary side activity)

The F1B operator actions related to the transfer to the safe shutdown state, which rely only on F1A/F1B indications and controls, are:

- ASG [EFWS] passive header (pump discharge) opening (F1B) (Only necessary when relevant preventive maintenance is being undertaken)
  - o manual re-alignment of ASG [EFWS] pump discharge to the unaffected SG, after an assumed 1 hour delay (to control ASG [EFWS] flow rate, SG level)
- RCP [RCS] cooldown by the unaffected SGs (F1B)
  - o manual VDA [MSRT] opening/closing, if there is no F1B automatic control of SG cooldown (control of RCP [RCS] temperature, SG pressure)
  - o manual ASG [EFWS] control, if there is no F1B automatic control of SG level
- RCP [RCS] boration (F1B)
  - o manual start of the RBS [EBS] pump (at the start of the RCP [RCS] cooldown)
  - o manual alignment to an unaffected loop (at the start of the RCP [RCS] cooldown)
  - o manual cut-off of RBS [EBS] pump or manual closing of the isolation valve (to control RBS [EBS] tank level)
- ASG [EFWS] passive header (pump suction) opening (F1B)
  - o manual realignment of the ASG [EFWS] tank to the ASG [EFWS] pump suction via the dedicated passive header (to control ASG [EFWS] tank level)
- RCP [RCS] and affected SG depressurisation (F1B)
  - o manual isolation of accumulators (to control RCP [RCS] pressure)
  - o manual MHSI pump shutdown, if previously actuated (to control RCP [RCS] pressure, MHSI flow rate)

- manual isolation of the partner SG
- manual opening of the dedicated transfer line for water transfer from the affected SG to the partner SG (SG pressure and wide range level) if necessary (level in affected SG > MAX2)
- manual opening of VDA [MSRT] (to control SG pressure)

- Connection of LHSI in RHR mode (F1B)
  - manual connection of LHSI (in RHR-mode) to the RCP [RCS] (to control RCP [RCS] pressure, RCP [RCS] temperature, RPVL, ΔT_{SAT})

10.4. DEFINITION OF CASES STUDIED

10.4.1. Short Term Phase

The short-term phase, as defined in the SGTR study, is the time period between the SGTR occurring and leak termination. This phase includes the achievement of the controlled state, when the SI flow matches the SGTR flow, and the termination of the activity release through the affected SG VDA [MSRT].

The purpose of the study of the short term phase is to determine the maximum amount of fluid that could be released to the atmosphere from the affected SG. This is done by analysing two separate cases:

- To quantify the maximum amount of activity released to the atmosphere by steam discharge, it is conservative to maximise the power to be removed from the RCP [RCS] since, prior to the isolation of affected SG, only steam is released. Therefore, the event is initiated at 102% of full power to maximise the decay heat, and no LOOP is assumed so that the reactor coolant pump power must be removed in addition to the decay heat.

- To demonstrate that SG overfilling does not occur, and thus no liquid is released to the atmosphere prior to leak termination, it is conservative to minimise the power to be removed from the RCP [RCS]. This minimises the steam release, and thus the boiling rate, and maximises the liquid volume in the SG. Therefore, the event is initiated from 2% of full power, and LOOP is postulated for this part of the study.

The two cases with these different assumptions are described below.

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

To produce the maximum steam release from the affected SG, it is necessary to maximise the heat to be removed by the SGs.

Thus, the most onerous case is:

- maximum initial power (102% FP)

- no LOOP

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

The other specific assumptions related to this case are described in section 10.5.1 of this sub-chapter.

Case 2: with LOOP

To produce the maximum liquid content in the affected SG the initial water inventory should be maximised. This occurs at a low power level. In addition, the steam released should be minimised, which occurs in conditions where the heat to be removed by the SGs is minimised.

Thus, the most onerous case is:

- low initial power (2% FP)
- LOOP assumed coincident with the turbine trip signal

The other specific assumptions related to this case are described in section 10.5.2 of this sub-chapter.

10.4.2. Long Term Phase

The long term phase is defined as the period between leak termination and the achievement of the safe shutdown state with connecting conditions for the LHSI in RHR mode. This phase includes the phases of boration and simultaneous cooldown of the RCP [RCS] using the unaffected SGs, and the final depressurisation of the RCP [RCS] and the affected SG.

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

- To confirm that LHSI in RHR mode connecting conditions and the safe shutdown state can be reached using only F1 systems.
  - The adequacy of ASG [EFWS] tank capacity is confirmed
  - The adequacy of the RBS [EBS] is confirmed
- To derive the maximum amount of activity released to the atmosphere. The release principally occurs during the final depressurisation phase of the RCP [RCS] and the affected SG.

Two cases with different assumptions are analysed to meet these objectives.

Case 3: without LOOP

This case, which does not assume LOOP, makes different assumptions to Case 1 with regard to single failure and preventive maintenance. The purpose is to show that the plant can be cooled down to connection conditions for LHSI in RHR mode using F1 systems only. The case confirms the adequacy of the ASG [EFWS] tank water contents for the cooldown by assuming maximum possible heat removal requirements.

For the long term phase, the final depressurisation is performed by transferring inventory from the affected SG to the unaffected partner SG and opening the VDA [MSRT] on the affected SG. Any activity release during this phase is evaluated.

Case 4: with LOOP

This case, which assumes LOOP, is an extension of Case 2. The purpose is to demonstrate the capacity of the F1 systems to borate the RCP [RCS], and depressurise the RCP [RCS] and the affected SG, while minimising reverse flow from the affected SG to the primary system when the reactor coolant pumps are not operating.

For the long term phase, the final depressurisation is performed by transferring inventory from the affected SG to the unaffected partner SG and, if necessary, opening the VDA [MSRT] on the affected SG. Any activity release during this phase is evaluated.

10.5. SHORT TERM STUDY

10.5.1. Case 1: without LOOP

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

The single failure is assumed on the Main Steam Relief Control Valve (MSRCV) of the affected SG. The valve is assumed to remain stuck in its initial, fully open, position. Therefore, instead of a controlled partial cooldown, the SF causes an uncontrolled pressure drop in the affected SG. The affected VDA [MSRT] is unable to close until the MSRIV setpoint is reached. Consequently, the activity released to the atmosphere is increased.

For this case, an assumption of no preventive maintenance is conservative.

The assumptions for SF and PM are summarised in Section 14.5.10 - Table 5.

10.5.1.2. Initial State

The conditions in the initial state, given in Section 14.5.10 - Table 1, are chosen to maximise the heat to be removed and the pressure difference between the RCP [RCS] and the affected SG.

10.5.1.3. Specific Assumptions

a) Neutronic data

Core power is assumed constant at 102% of full power prior to reactor trip. Following RT, the time-dependent A term with 1.645 σ uncertainties on the B+C term is used, as described in Sub-chapter 14.1.

b) Assumptions related to control systems (NC)

Turbine: Turbine control is assumed to maintain the turbine flow rate at 102% nominal flow rate until the turbine trip on reactor trip.

GCT [MSB]: Not considered.

ARE [MFWS]: SG level control is assumed to be in operation when the fault occurs and to continue working correctly until isolation on reactor trip. This is consistent with the analysis rules presented in Sub-chapter 14.0.

RCV [CVCS]: A maximum charging flow rate is assumed with two charging pumps in operation. This maximises the pressure difference between the RCP [RCS] and the affected SG. The letdown is isolated on a “Pressuriser level < MIN3” signal. RCV [CVCS] charging is isolated if the combination of an “SG Level > MAX2” signal and end of partial cooldown signals is generated.

Pressuriser heaters and spray: Maximum heater power is assumed corresponding to actuation of all the heaters. This maximises the pressure difference between the RCP [RCS] and the affected SG. In addition, spray flow is not considered. The heaters are shut off following a “Pressuriser level < MIN3.” signal

c) Assumptions related to F1 systems

Reactor trip (F1A): RT occurs following a “Pressuriser pressure < MIN2” signal at a setpoint of 135 - 1.5 bar. A maximum delay is assumed. These assumptions maximise the activity content of the affected SG when steam is released to atmosphere following VDA [MSRT] actuation.

VDA [MSRT] (F1A): a minimum VDA [MSRT] setpoint is assumed for the affected SG. A maximum VDA [MSRT] setpoint is assumed for the other SGs. This maximises the steam release from the affected SG.

On the affected SG, the VDA [MSRT] is actuated after RT at a pressure setpoint of 95.5 - 1.5 bar. The valve is assumed to remain stuck open (SF assumption) until isolation of the MSRIV on “SG pressure < MIN3” at a setpoint of 40 - 1.5 bar. Following the partial cooldown and the allowance for the operator delay time, the operator is assumed to increase the VDA [MSRT] setpoint to 100 bar. If the operator action does not take place, the setpoint would be increased automatically on a signal indicating that the partial cooldown is complete and SG level exceeds MAX2. Increasing the VDA [MSRT] setpoint does not prevent further activity release due to the assumed SF on the MSRCV, which is assumed to be stuck open. Activity release to the atmosphere via the affected SG is terminated by the isolation of the MSRIV (see Section 14.5.10 - Figure 16).

On the unaffected SGs, the VDA [MSRT] are actuated following RT at a pressure setpoint of 95.5 + 1.5 bar. They remain at this pressure level until the start of the partial cooldown, which is initiated following the SI signal. The VDA [MSRT] setpoints are decreased from 97 bar to 61.5 bar by the end of the partial cooldown.

VIV [MSIV] (F1A): the VIV [MSIV] of all SGs are closed following a low pressure signal on “SG pressure < MIN1”. The rapid depressurisation occurs via the fully open VDA [MSRT] on the affected SG. The uncertainty and delay on the low pressure signal are chosen to bring forward the time of VIV [MSIV] closure. This assumption maximises the activity release to atmosphere from the affected SG as the setpoints of the VDA [MSRT] are decoupled. The VIVs [MSIVs] may also be shut on a high rate of SG pressure drop signal (“SG $\Delta P/\Delta t$ > MAX1”).

MHSI (F1A): the SI signal is actuated following a “Pressuriser pressure < MIN3” signal at a setpoint of 115 + 1.5 bar. This signal initiates a partial cooldown of all four SGs, and MHSI injection. The maximum of the MHSI flow characteristic, which initiates injection below 97 bar, is used. This results in the maximum pressure difference between the RCP [RCS] and the affected SG at the end of the partial cooldown.

ASG [EFWS] (F1A): the ASG [EFWS] is actuated in the unaffected SGs on a “SG level < MIN2” signal with a setpoint of 40% WR – 2%. A minimum flow characteristic is assumed.

MSSV (F1A): A minimum setpoint of 105 - 1.5 bar is assumed for the MSSV. The calculation confirms the MSSV are not opened during the transient.

10.5.2. Case 2: with LOOP

10.5.2.1. Choice of Single Failure and Preventive Maintenance

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

The assumptions regarding single failure and the preventive maintenance have a negligible impact on the filling of the affected SG for overfilling study. Therefore, the choice of SF and PM is governed by the long term study described in section 10.6.2.1 of this sub-chapter:

- SF on one emergency diesel generator: loss of one safety division, associated with an unaffected SG
- PM on one emergency diesel generator: loss of one safety division associated with a second unaffected SG

The assumptions for SF and PM are summarised in Section 14.5.10 - Table 6.

10.5.2.2. Initial State

The conditions for the initial state, given in Section 14.5.10 - Table 2, maximise the initial water content in the affected SG. They also maximise the pressure difference between the primary and secondary side, and minimise the heat to be removed by the affected SG [Ref-1].

10.5.2.3. Specific Assumptions

a) Neutronic data

An initial core power of 2% full power is assumed. Following reactor trip, the core power decreases following the realistic decay heat curve described in Sub-chapter 14.1.

b) Assumptions related to control systems (NC)

Turbine: The turbine is assumed to be isolated at the beginning of the transient, as appropriate to model hot standby initial conditions.

GCT [MSB]: the GCT [MSB] is assumed available to remove the residual heat. This maximises the pressure difference between the RCP [RCS] and the affected SG until LOOP occurs. It is conservatively assumed that the GCT [MSB] is lost following the LOOP, which is assumed to coincide with the reactor trip.

ARE [MFWS]: the SG level control by the ARE [MFWS] is assumed to be in operation when the fault occurs. It continues working correctly until LOOP occurs, which is assumed coincident with reactor trip.

RCV [CVCS]: the RCV [CVCS] is assumed to operate correctly as for Case 1 described in section 10.5.1.3 of this sub-chapter.

The RCV [CVCS] charging is isolated following a combination of “SG Level > MAX2“ and end of partial cooldown signals.

Pressuriser heaters and sprays: Pressuriser pressure control is assumed to operate as described for Case 1 in section 10.5.1.3 of this sub-chapter.

c) Assumptions related to F1 systems

Reactor trip (F1A): A reactor trip signal occurs following a “SG level > MAX1” signal with a setpoint of 69 + 2% NR. Coincident LOOP is assumed to occur at this time.

VDA [MSRT] (F1A): The VDA [MSRT] setpoint is minimised on the affected SG and maximised on the other SGs, as was assumed for Case 1. This maximises the pressure difference between the primary and secondary sides.

The VDA [MSRT] on the affected SG is actuated following RT at a pressure setpoint of 95.5 - 1.5 bar. The affected SG remains at this pressure until the start of the partial cooldown. This is initiated following a “SG level > MAX2” signal at a setpoint of 85 + 2% NR, which occurs earlier than the SI signal. Following the PC signal, the VDA [MSRT] setpoint is decreased at a rate corresponding to a cooldown rate of 250°C/hour until the SG pressure is below 58.5 bar.

The VDA [MSRT] on the unaffected SGs are actuated after RT at a pressure setpoint of 95.5 +1.5 bar. The SGs remain at this pressure until the start of the partial cooldown. The VDA [MSRT] setpoints are decreased following the PC signal at a rate corresponding to a RCP [RCS] cooldown rate of 250°C/hour. This is continued until the SG pressure is below 61.5 bar.

The VDA [MSRT] setpoint of the affected SG is automatically increased to 99 bar on receipt of an “SG level > MAX2 + PC finished” signal.

VIV [MSIV] (F1A): the VIV [MSIV] of the affected SG is closed by the same signal that initiates the VDA [MSRT] setpoint increase. The VIV [MSIV] can also be assumed to be closed manually when operator actions are allowed.

MHSI (F1A): operation is the same as assumed for Case 1. The only exception is that only two MHSI pumps are assumed to be available to inject into the RCP [RCS]. This is due to the single failure and preventive maintenance assumptions.

ASG [EFWS] (F1A): following LOOP, the ASG [EFWS] is actuated manually when the SG level falls below the nominal level or automatically when the safety injection signal is actuated. Only the affected SG and one intact SG are assumed to be fed. ASG [EFWS] injection on the other SGs is discounted due to the SF and PM assumptions. Once actuated, the SG level in the unaffected SG is controlled at the reference setpoint. The level in the affected SG remains too high for actuation.

MSSV (F1A): same as Case 1.

10.5.3. Results

The analysis of the short term phases, assessing the radiological release criterion, is presented below.

10.5.3.1. Case 1:

The sequence of events for Case 1 is given in Section 14.5.10 - Table 3.

Key results for Case 1 are provided in Section 14.5.10 - Table 8.

Figures showing the main thermal hydraulic parameters are described below:

Description	Figure
Pressuriser and SG pressures	1
Mass flows	
SG level, NR	2
SG level, WR	
Affected SG, physical liquid level	3
Differential pressure across break	
Affected, Unaffected RCP [RCS] loop temperatures	4
Total steam masses	
SG steam outlet flow	5
Primary mass flow	

The results of Case 1 show that a maximum of 81 tons of contaminated steam is discharged to the atmosphere through the affected SG VDA [MSRT] during the short term phase. If flow from the non-contaminated steam is included, 203 tons are released via the affected SG VDA [MSRT]. Prior to turbine trip, 29 tons of primary side liquid are transferred to the affected SG. A total of 129 tons of primary side liquid are discharged to the affected SG and overfilling does not occur. There is no backflow from the affected SG to the primary side.

In addition, Case 1 shows that there is no demand on the MSSVs to open. This supports the assumption of single failure on the VDA [MSRT] instead of the MSSV.

10.5.3.2. Case 2:

The sequence of events for Case 2 is given in Section 14.5.10 - Table 9.

Key results for Case 2 are provided in Section 14.5.10 - Table 8.

The figures showing the main thermal hydraulic parameters are described below:

Description	Figure
Pressuriser and SG pressures	11
Mass flows	
SG level, NR	12
SG level, WR	
Affected SG, collapsed liquid level	13
Differential pressure across break	
Affected, Unaffected RCP [RCS] loop temperatures	14
Total steam masses	
SG steam outlet flow	15
Primary mass flow	

The results of Case 2 show that a maximum of 64 tons of contaminated steam are discharged to the atmosphere through the affected SG VDA [MSRT] during the short term phase. If flow from the non-contaminated steam is included, 211 tons are released via the affected SG VDA [MSRT]. Prior to turbine trip, 6 tons of primary side liquid are transferred to the affected SG. A total of 95 tons of primary side liquid are discharged to the affected SG and no overfilling occurs. There is no backflow from the affected SG to the primary side.

10.6. LONG TERM STUDY

10.6.1. Case 3 – without LOOP

10.6.1.1. Choice of Single Failure and Preventive Maintenance

A single failure is chosen to reduce the capacity of the F1 systems to borate and cool down the RCP [RCS] to connection conditions for the LHSI in RHR mode. This can be done with a single failure on either:

- one RBS [EBS] pump to reduce the RCP [RCS] boration, and consequently reduce the cooldown rate, or
- the VDA [MSRT] of one unaffected SG via failure of the MSRIV to open.. This reduces the heat removal capacity for RCP [RCS] cooldown at the end of the cooldown phase when at low SG pressure.

These two single failures are combined in the accident analysis, to limit the number of cases to be analysed. The assumption of a double failure is not considered unacceptably over-conservative, because the VDA [MSRT] SF has only a secondary impact on the demand on the ASG [EFWS] tank capacity.

Preventive maintenance has no impact on the availability of the RBS [EBS] or the VDA [MSRT]. Preventive maintenance results in the unavailability of one ASG [EFWS] train associated with one unaffected SG, which is conservative for the evaluation of the steam release.

The assumptions for SF and PM are summarised in Section 14.5.10 - Table 7.

10.6.1.2. Conditions at the Start of the Long Term Phase

Most of the assumptions made for the calculation of the long term of this study are identical to those applied for Case 1 in sections 10.5.1.2 and 10.5.1.3 of this sub-chapter for the 102% FP initial state. The main exceptions are associated with the SF and PM.

In Case 3, the SF is assumed to be on one RBS [EBS] pump and on one VDA [MSRT] (failure of MSRIV to open on an unaffected SG). This is compared to failure of VDA [MSRT] of the affected SG in Case 1, due to an MSRCV stuck fully open. This is conservative for the long term study.

PM is assumed to be on one ASG [EFWS] pump associated with an unaffected SG in Case 3. This is compared to PM of the ASG [EFWS] pump connected to the affected SG in Case 1. This is conservative for the long term study.

10.6.1.3. Specific Assumptions

The majority of the Case 1 system assumptions described in section 10.5.1.3 of this sub-chapter remain valid. The exceptions are as follows:

MHSI (F1A): 4 out of 4 trains are assumed to be available for use following the SI injection signal. The first three pumps are shut down by the operator once there is sufficient margin to saturation The fourth pump is kept operating until the temperature in the unaffected loops is at or below the RHR connection temperature of 180°C.

VDA [MSRT] (F1B):
Unaffected SG: the VDA [MSRT] setpoint remains constant at 60 + 1.5 bar before the beginning of the cooldown initiated by the operator. Their setpoints are decreased for the RCP [RCS] cooldown, at a rate corresponding to a RCP [RCS] temperature gradient of -25°C/hour. This is the rate with one RBS [EBS] pump operating, and it is maintained until the defined temperature is reached.

Affected SG: the VDA [MSRT] setpoint remains constant at 99 bar to ensure the VDA [MSRT] remains closed during the RCP [RCS] cooldown phase. Towards the end of the transient, the VDA [MSRT] is re-opened to complete the final depressurisation to the pressure required to enable RHR connection.

ASG [EFWS] (F1A): the ASG [EFWS] is manually controlled by the operator during the long term transient phase. The operator maintains the levels in the three unaffected SGs at the nominal value until preparation for fluid transfer from the affected SG, when the nominal control level of the partner SG is reduced. In the affected SG, the level remains too high to actuate the ASG [EFWS]. The total ASG [EFWS] tank capacity exceeds 1500 tons.

RBS [EBS] (F1B): the RBS [EBS] is manually actuated by the operator thirty minutes after RT. The injection is delivered to an unaffected loop by an appropriate alignment. The RBS [EBS] minimum flow rate is 2.8 kg/s. The total minimum useable volume of the two RBS [EBS] tanks is 72 m^3.

Transfer line (F1B): following the completion of the RCP [RCS] cooldown, the operator opens the dedicated transfer line connecting the affected SG to the unaffected partner SG. The partner SG has been prepared for the transfer by reducing its level and confirming its isolation. The pressure difference between the two SGs drives a water transfer from the affected SG thereby reducing its level and pressure. The pressure of the primary side follows that of the secondary side of the affected SG. The transfer line can remain open until the pressure in the two SGs has equalised.

10.6.2. Case 4 – with LOOP

10.6.2.1. Choice of Single Failure and Preventive Maintenance

The single failure and the preventive maintenance assumptions are chosen to reduce the capacity of the F1A/F1B systems to borate the RCP [RCS] and to cool down the RCP [RCS] and the affected SG to RHR connection conditions. The assumption that two emergency diesel generators are unavailable results in the unavailability of one RBS [EBS] pump supplied by an unavailable diesel generator. Consequently, the cooldown rate is limited to 25°C/hour and there is a greater challenge to the ASG [EFWS] and RBS [EBS] tank capacities.

Therefore the SF and PM assumptions are:

- SF on one emergency diesel generator causing the loss of the safety division assigned to an unaffected SG
- PM on one emergency diesel generator causing the loss of the safety division assigned to second unaffected SG

As discussed for Case 3, a SF of the VDA [MSRT] on an unaffected SG could reduce the heat removal capacity at the end of the cooldown phase. This second SF is therefore included in the accident analysis to limit the number of cases to be analysed.

The assumptions for SF and PM are summarised in Section 14.5.10 - Table 6.

10.6.2.2. Conditions at the start of the Long Term Phase

The long term phase starts at the end of the short term phase presented as Case 2.

10.6.2.3. Specific Assumptions

The assumptions on the systems described in section 10.5.2.3 of this sub-chapter remain valid.

The additional assumptions related to the long term phase are as follows:

MHSI and LHSI (F1A): operate as in Case 2 (section 10.5.2.3 of this sub-chapter).

VDA [MSRT] (F1B): operates as in Case 3 (section 10.6.1.3 of this sub-chapter).

ASG [EFWS] (F1A): operates as in Case 3 (section 10.6.1.3 of this sub-chapter).

The operator performs the re-alignment of the ASG [EFWS] train connected to the affected SG to deliver to an unaffected SG one hour after the RT.

RBS [EBS] (F1B): operates as in Case 3 (section 10.6.1.3 of this sub-chapter).

Transfer line (F1B): operates as in Case 3 (section 10.6.1.3 of this sub-chapter).

10.6.3. Results

The analysis for case 3 is as follows.

10.6.3.1. Case 3:

The sequence of events for Case 3 is given in Section 14.5.10 - Table 4.

Key results for the cases analysed for the PCSR are shown in Section 14.5.10 - Table 8.

The figures showing the main thermal hydraulic parameters are described below:

Description	Figure
Pressuriser and SG pressures	6
Mass flows	
SG level, NR	7
SG level, WR	
Affected SG, collapsed liquid level	8
Differential pressure across break	
Affected, Unaffected RCP [RCS] loop temperatures	9
Total steam masses	
SG Steam Outlet Flow	10
Primary mass flow	

The main results are as follows:

Case 3 involves the highest usage of ASG [EFWS] injection. 1307 tons of ASG [EFWS] feedwater are used prior to reaching the safe shutdown state. An additional 31 tons is added to this amount to account for the loss of SG mass from the continuous blowdown system in the time between the loss of ARE [MFW] flow and the startup of the ASG [EFWS]. This brings the total ASG [EFWS] consumption to 1338 tons. This is below the total ASG [EFWS] guaranteed capacity of 1500 tons. 57 tons of RBS [EBS] are used to reach the RCP [RCS] final temperature. The water used by the RBS [EBS] has been conservatively evaluated because the final RCP [RCS] cooldown temperature is below that required for RHR connection (160°C versus 180°C).

There is an additional release of contaminated steam (about 6 tons) through the VDA [MSRT] of the affected SG. This occurs because the transfer-line depressurisation is insufficient to reach the pressure required for RHR connection and the final depressurisation to below 30 bar must therefore be performed by opening the VDA [MSRT] on the affected SG. The total contaminated steam release of 64 tons is significantly less than that in Case 1. There is no potential for liquid release in this case as the VDA [MSRT] can only be opened when the level is below the MAX2 setpoint. 29 tons of primary side liquid are discharged to the affected SG prior to turbine trip. During the entire event, 164 tons of liquid was transferred from the primary side to the affected SG.

10.6.3.2. Case 4:

The sequence of events for Case 4 is given in Section 14.5.10 - Table 9.

Key results for the cases analysed for the PCSR are shown in Section 14.5.10 - Table 8.

The figures showing the main thermal hydraulic parameters are described below:

Description	Figure
Pressuriser and SG pressures	11
Mass flows	
SG level, NR	12
SG level, WR	
Affected SG, collapsed liquid level	13
Differential pressure across break	
Affected, Unaffected RCP [RCS] loop temperatures	14
Total steam masses	
SG steam outlet flow	15
Primary mass flow	

The main results are as follows:

Case 4, with LOOP, uses 834 tons of ASG [EFWS] to reach the safe shutdown state. The loss of SG mass from the continuous blowdown system is negligible. This is below the total ASG [EFWS] guaranteed capacity of 1500 tons. 61 tons of RBS [EBS] are used to reach the RCP [RCS] final temperature. The water used by the RBS [EBS] is conservatively evaluated because the final RCP [RCS] cooldown temperature is below that required for RHR connection (160°C versus 180°C).

There is an additional release of contaminated steam (34 tons) through the VDA [MSRT] on the affected SG that occurs after the SG is isolated. This occurs because the transfer-line depressurisation is insufficient to reach the pressure required for RHR connection. The contaminated steam release of 64 tons during the short term phase is significantly less than that of Case 1. There is no potential for liquid release in this case as the VDA [MSRT] can only be opened when the level is below the MAX2 setpoint. 6 tons of primary side liquid are discharged to the affected SG prior to turbine trip. During the entire event, 171 tons of liquid are transferred from the primary side to the affected SG.

The affected SG minimum free volume of 9.7 m^3 occurs before the transfer line is opened. The time to open the transfer line is conservatively extended since the cooldown is limited to 25°C/hr and the end of the cooldown occurs at 160°C instead of 180°C. This shows that there is no risk of overfilling the affected SG before the manual cooldown has ended and the transfer line is opened.

10.7. CONCLUSION

The present analysis of the 4A-SGTR accident shows that, even with the worst single failure and preventive maintenance assumptions:

- the controlled state is reached using only on F1A means:
  - RT for nuclear power shutdown,
  - ASG [EFWS] and VDA [MSRT], including partial cooldown, for RCP [RCS] heat removal,
  - MHSI for RCP [RCS] coolant inventory control.

- the safe shutdown state RHR connection conditions are reached using only F1A and F1B systems:
    - VIV [MSIV] closure, ARE/AAD/ASG [MFW/SSS/EFWS] isolation, VDA [MSRT] setpoint increase for isolation of the affected SG,
    - MHSI and LHSI for RCP [RCS] coolant inventory control,
    - ASG [EFWS] and VDA [MSRT] for RCP [RCS] cooling,
    - RBS [EBS] for boration,
    - Opening of the transfer line to the partner SG to enable depressurisation of the affected SG (and hence the RCP [RCS]).
    - LHSI/ RHR for long term heat removal.

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

- At full power, with reactor coolant pumps operating (no LOOP):
    - a maximum of 81 tons of steam is released before the affected SG is isolated. Increasing the duration of the RCP [RCS] cooldown adds 6 tons to the steam release during the RCP [RCS] cooldown and final depressurisation phase.
    - no liquid release occurs
- At lower power, with reactor coolant pumps off (LOOP assumed):
    - less than 64 tons of steam is released before the affected SG is isolated. A further 34 tons of steam is released during the depressurisation of the RCP [RCS] and affected-SG, immediately prior to RHR connection
    - no liquid release occurs

There is no demand on any MSSV to open and a margin is maintained to overfilling of the affected SG. The above assessment includes the effects from the worst single failure and preventive maintenance.

After the manual cooldown ends, the differential pressure across the break fluctuates as the ASG [EFWS] flow to the unaffected SGs switches on and off. This results in a small amount of backflow during the long term Case 3 (about 5 tons). There is also a small amount of backflow in Case 4, but the lack of forced circulation minimises the effect of the ASG [EFWS] cycling This transient results in less backflow (about 2 tons). This small amount should not present a concern from the potential boron dilution via this route and any associated criticality. The core remains covered throughout the transient, and adequate core cooling is maintained. Therefore, no clad heat-up occurs.

Therefore, all the decoupling criteria and targets are shown to be met.

10.8. SYSTEM SIZING

The maximum MHSI pump delivery pressure is sized for the SGTR transient to allow leak termination and prevent overfilling of the affected SG and opening of an MSSV.

SECTION 14.5.10 - TABLE 1

Plant Initial Conditions - Cases 1 and 3
(SGTR 2-tube Short and Long Term - Without LOOP)

Parameter	Value
◆ Initial reactor power (% of nominal power)	102%
◆ Initial average RCP [RCS] temperature	312.4 – 2.5 = 309.9°C
◆ Initial reactor coolant pressure	155 + 2.5 = 157.5 bar
◆ Reactor cooling flow rate	Thermal hydraulic
◆ Pressuriser level	56% + 5% = 61%
◆ Initial unaffected SG level	49% + 10.5% = 59.5% NR
◆ Main feedwater flow (% of nominal flow)	102%
◆ Initial ARE [MFWS] temperature	230°C

SECTION 14.5.10 - TABLE 2

Initial Conditions - Cases 2 and 4
(SGTR 2-tube Short and Long term - With LOOP)

Parameter	Value
Initial reactor power (% of nominal power)	2%
Initial average RCP [RCS] temperature	303.2 – 2.5 = 300.7°C
Initial reactor coolant pressure	155 + 2.5 = 157.5 bar
Reactor cooling flow rate	Thermal hydraulic
Pressuriser level	31% + 5% = 36%
Initial unaffected SG level	49% + 10.5% = 59.5% NR
Main feedwater flow (% of nominal flow)	2%
Initial ARE [MFWS] temperature	121°C

SECTION 14.5.10 - TABLE 3

Sequence of Events - Case 1
SGTR 2-tube: Short Term - without LOOP

Time (sec)	Event
0	SGTR occurrence, 2 tubes
466	Pressuriser level < MIN3
	Letdown isolation
	Heater isolation
549	Pressuriser pressure < MIN2
	Reactor trip
	Turbine trip
	High load ARE [MFWS] isolation in all loops
566	SGa MSRIV setpoint reached
	Beginning of short term radioactivity release
657	Pressuriser pressure < MIN3
	SI system signal
	Partial Cooldown at -250 deg/hour
713	SG level > MAX1
733	Low load ARE [MFWS] stopped in loop 3 (SGa)
798	Low load ARE [MFWS] isolation in remaining loops
1128	Partial cooldown complete
1354	SG pressure < MIN1
1360	All MSIVs shut
1621	SG pressure < MIN3
1627	SGa MSRIV closed
	End of short term radioactive release
2349	RT+30min: Operator actions could begin
3578	PC Complete + SGa > MAX2
	RCV [CVCS] charging isolated
3861	Leak terminated

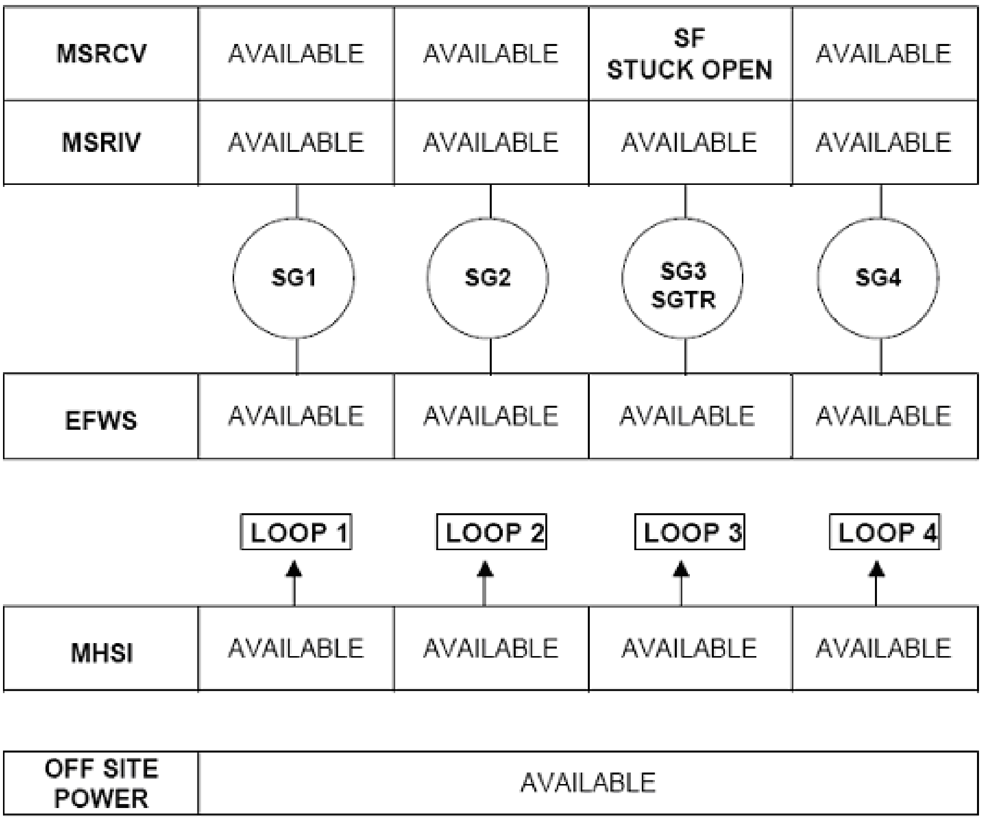
SECTION 14.5.10 - TABLE 4

Sequence of Events - Case 3
SGTR 2-tube Long Term - Without LOOP

Time (sec)	Event
0	SGTR occurrence, 2 tubes
466	Pressuriser level < MIN3
	Letdown isolation
	Heater isolation
549	Pressuriser pressure < MIN2
	Reactor trip
	Turbine trip
565	High load ARE [MFWS] isolation in all loops
713	SG level > MAX1
733	Low load ARE [MFWS] stopped in loop 3 (SGa)
785	Low load ARE [MFWS] isolation in remaining loops
947	Partial Cooldown at -250°C/hour on SG3 level > MAX2
1044	Pressuriser pressure < MIN3
	SI system signal
1420	Partial cooldown complete
2349	Operator Actions: RT + 30 minutes
	P12 permissive activated
	RBS [EBS] startup
	Manual cooldown at -25°C/hour
	Isolation of affected SG
	- SGa VDA [MSRT] setpoint increase
	- SGa VIV [MSIV] closure
	Stop 3 MHSI pumps
2366	ASG [EFWS] startup in SG1 and 2
2820	PC Complete + SGa > MAX2
	Isolation RCV [CVCS] charging
17277	TRIC < 180°C: Stop the last MHSI pump
22662	Beginning of post-cooldown actions
	Isolation of partner SG for transfer
	- Partner SG VDA [MSRT] setpoint increase
	- Partner SG VIV [MSIV] closure
	Stop RBS [EBS]
22668	Manual opening of the transfer line between SGa and partner SG
23654	Isolation of accumulators
30000	Begin depressurisation via SGa VDA [MSRT]
40374	RHR connection

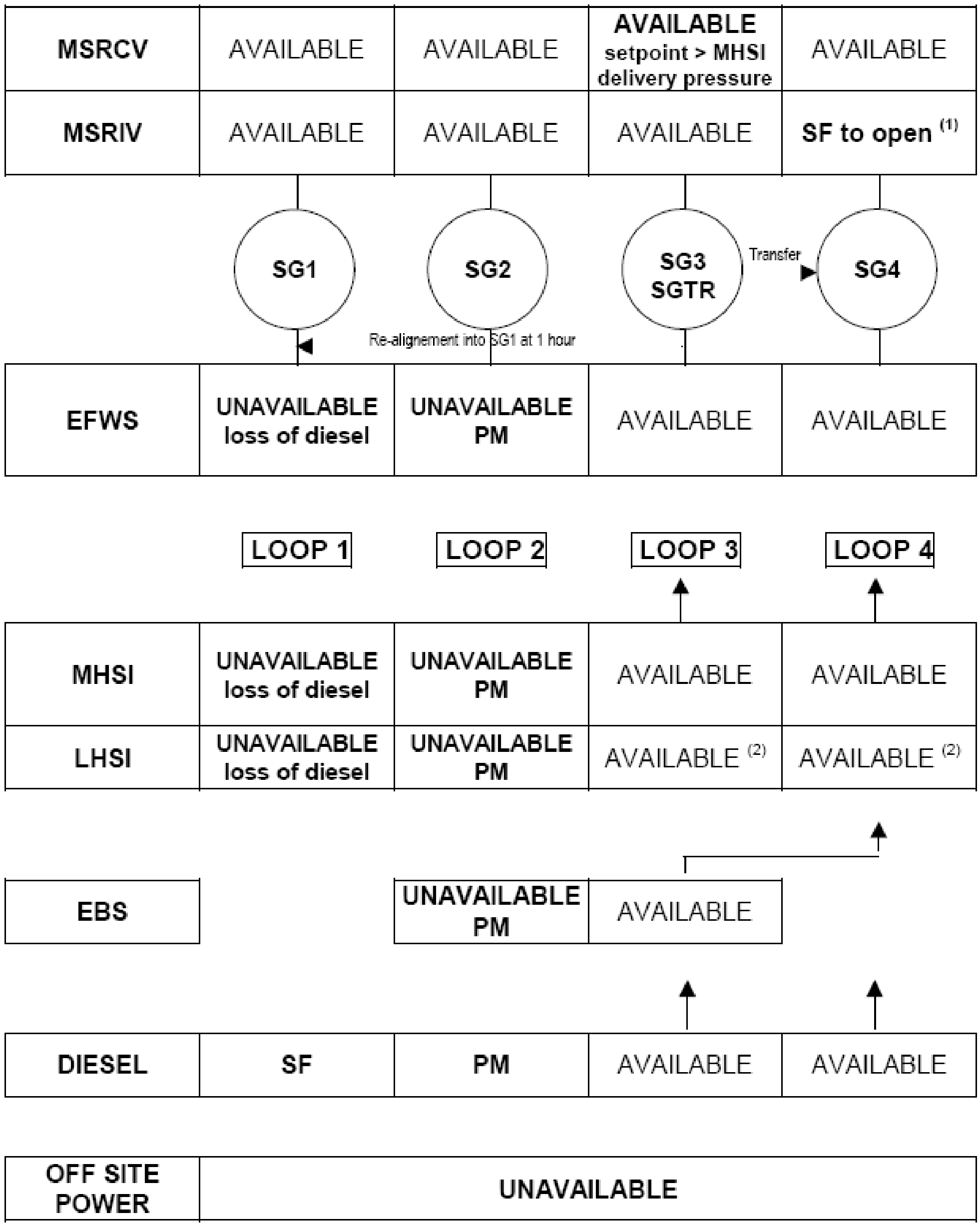
SECTION 14.5.10 - TABLE 5

Single Failure and Preventive Maintenance Assumptions- Case 1
SGTR 2-tube Short Term - Without LOOP



SECTION 14.5.10 - TABLE 6

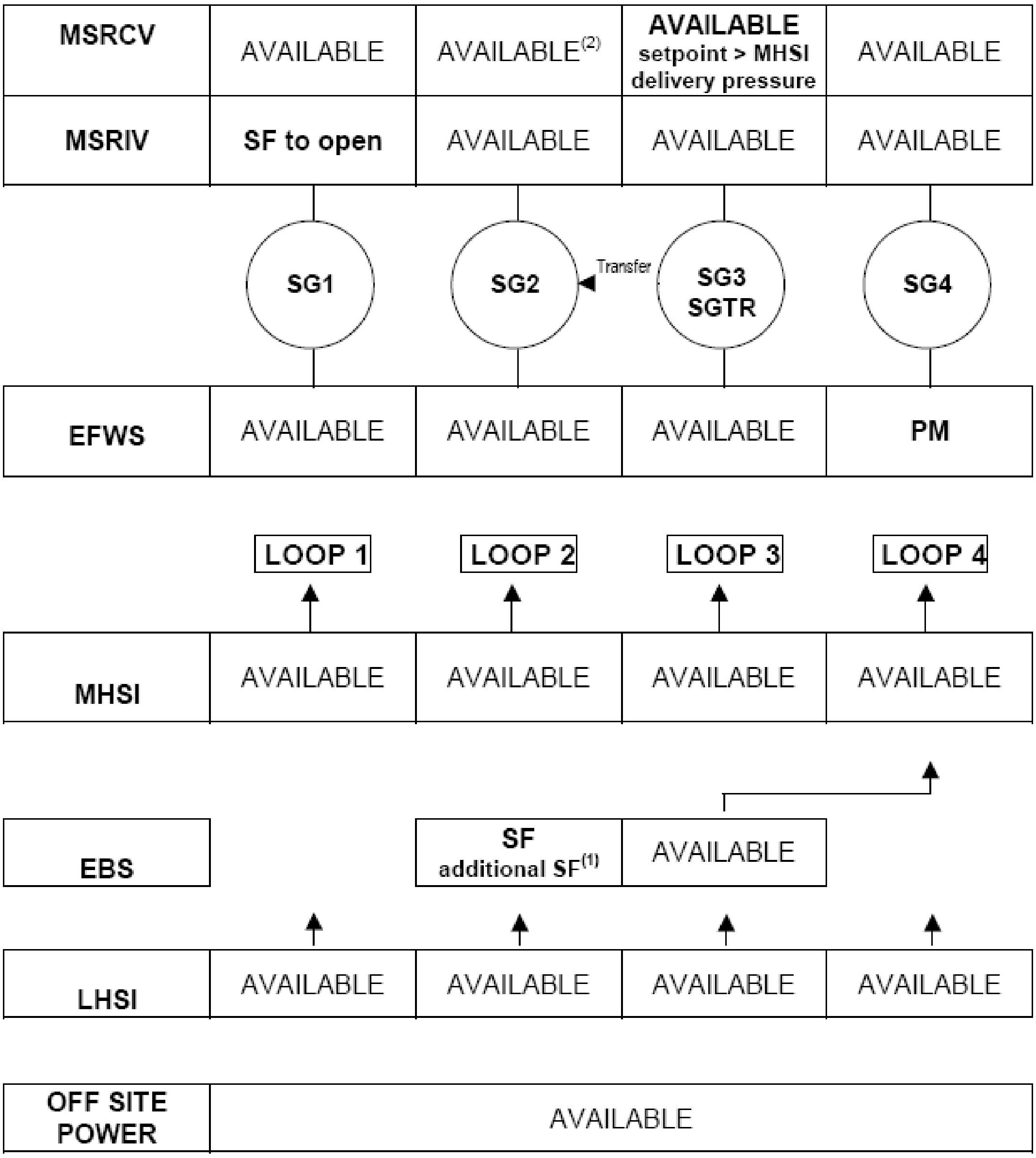
Single Failure and Preventive Maintenance Assumptions- Case 2 and 4
SGTR 2-tube Short Term and Long Term- With LOOP



(1) Additional SF in order to limit the number of cases presented
(2) The safe shutdown is achieved by 1 LHSI train in SI mode and the other in RHR mode

SECTION 14.5.10 - TABLE 7

Single Failure and Preventive Maintenance Assumptions- Case 3
SGTR 2-tube Long Term - Without LOOP



(1) Additional SF in order to limit the number of cases presented
(2) The setpoint is greater than the MHSI delivery pressure at the beginning of the transfer

SECTION 14.5.10 - TABLE 8

SGTR 2-tube Criteria-Relevant Results

Parameter	**Case 1**	**Case 3**	**Case 2**	**Case 4**
Contaminated steam release (tons)	81	64	64	98
Total SGa VDA [MSRT] steam released (tons)	203	202	211	245
Primary coolant liquid mass transferred to SGa (tons)	129	164	95	148
Primary coolant liquid mass transferred prior to turbine trip (tons)	29	29	6	6
Maximum SGa liquid mass (tons)	150	197	154	171
Minimum SGa free volume (m^3)	44.1	8.8	33.7	9.7
Time of RHR connection (seconds)	-	40,373	-	43,214
RBS [EBS] inventory used (tons - 72 tons available)	-	57	-	61
ASG [EFWS] inventory used (tons - 1500 tons available)	-	1338	-	834

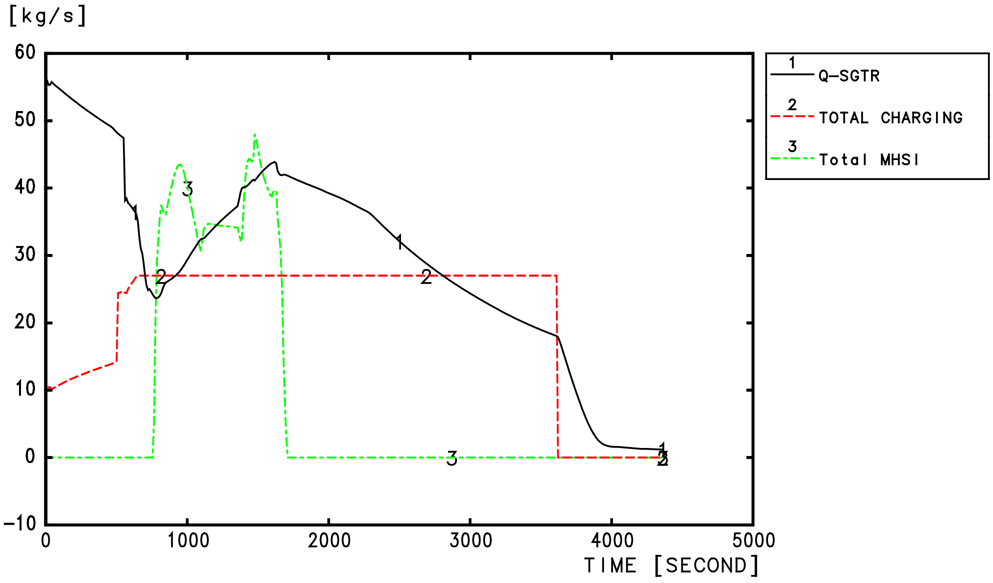
SECTION 14.5.10 - TABLE 9

Sequence of Events – Cases 2 and 4
SGTR 2-tube - With LOOP

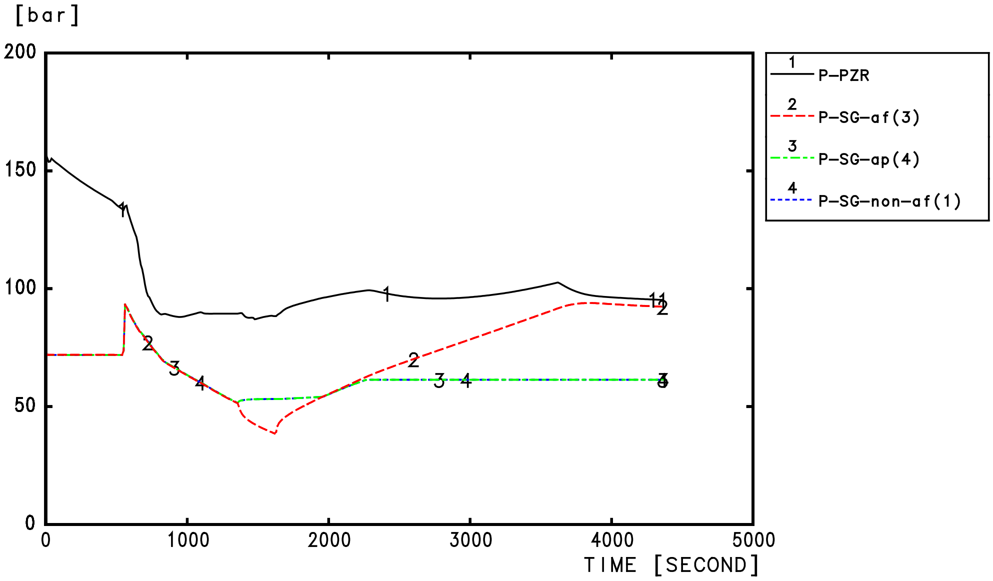
Time (sec)	Event
0	SGTR occurrence, 2 tubes
114	SG level > MAX1
116	Reactor trip
	Turbine trip (LOOP)
	RCPs stopped
527	Pressuriser level < MIN3
	Letdown isolation
	Heater isolation
884	SGa level > MAX2
	Partial Cooldown at -250°C/hour
1032	Pressuriser pressure < MIN3
	SI system signal
	EFW start up to SG 3 and 4 on LOOP + SI signal
~1100	SGTR leak flow compensated by SI and CVCS flow
	Controlled state reached
1356	Partial cooldown of 3 SGs complete
1914	Operator Actions: RT + 30 minutes
	P12 permissive activated
	Isolation of affected SG
	- SGa VDA [MSRT] setpoint increase
	- SGa VIV [MSIV] closure
2373	PC Complete + SGa > MAX2
2415	Isolation RCV [CVCS] charging
2940	**Leak termination** (End of Case 2)
3714	Local Actions: RT + 60 minutes
	- Re-align EFW3 to SG 1
	- Manual cooldown at -25°C/hour
	Stop 3 MHSI pumps
3737	RBS [EBS] startup (only one train available)
21635	TRIC < 180°C: Stop the last MHSI pump
25267	Beginning of post-cooldown actions
	Isolation of partner SG for transfer
	- Partner SG VDA [MSRT] setpoint increase
	- Partner SG VIV [MSIV] closure
	Stop RBS [EBS]
25272	Manual opening of the transfer line between SGa and partner SG
32974	Begin depressurisation via SGa VDA [MSRT]
33444	Isolation of accumulators
43214	RHR connection

SECTION 14.5.10 - FIGURE 1

Pressuriser and SG Pressures/Mass Flow
SGTR 2-tube Case 1: Short Term without LOOP



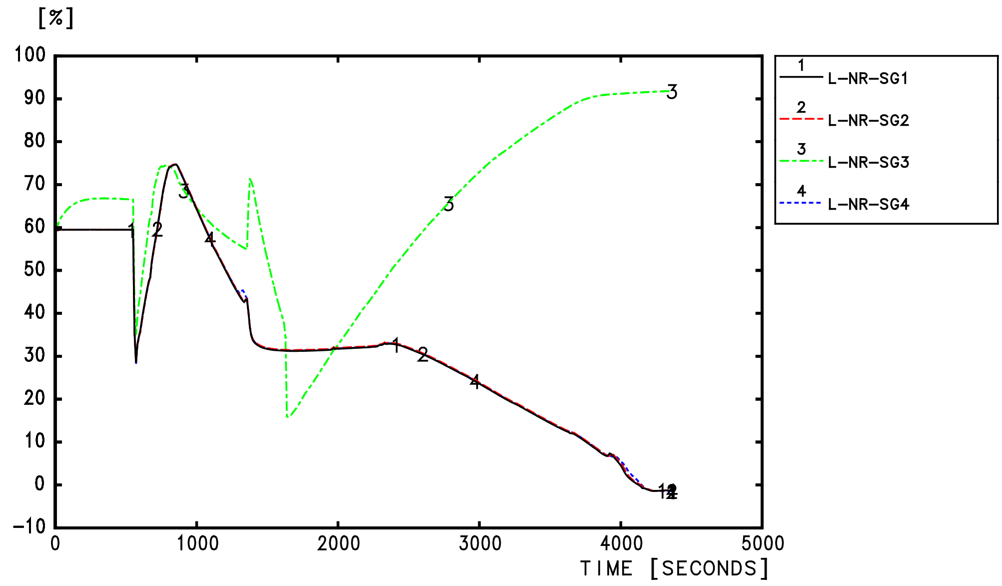
MASS FLOW RATE



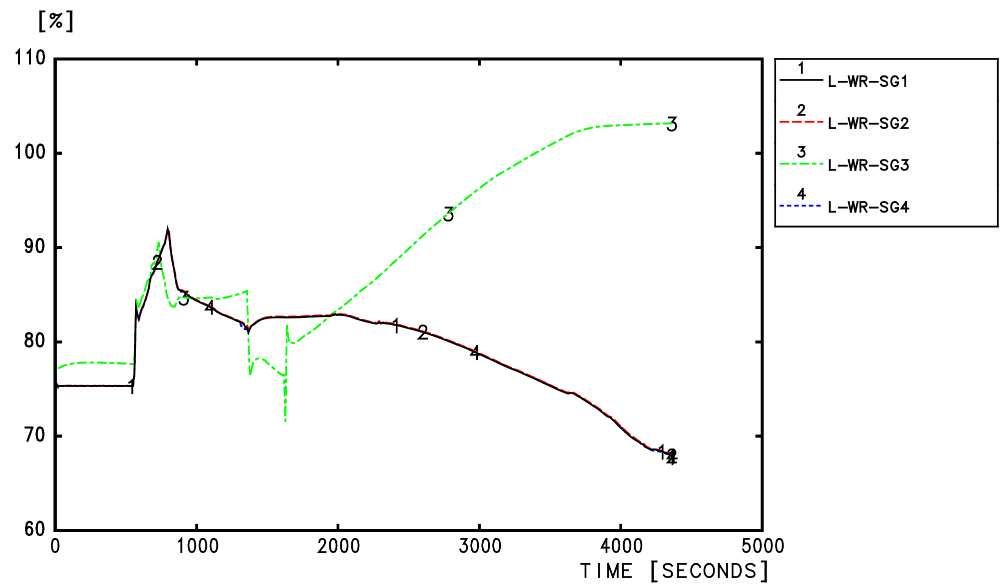
PRESSURE − PZR, RCS and SG

SECTION 14.5.10 - FIGURE 2

SG Levels: Narrow Range/Wide Range
SGTR 2-tube Case 1: Short Term without LOOP



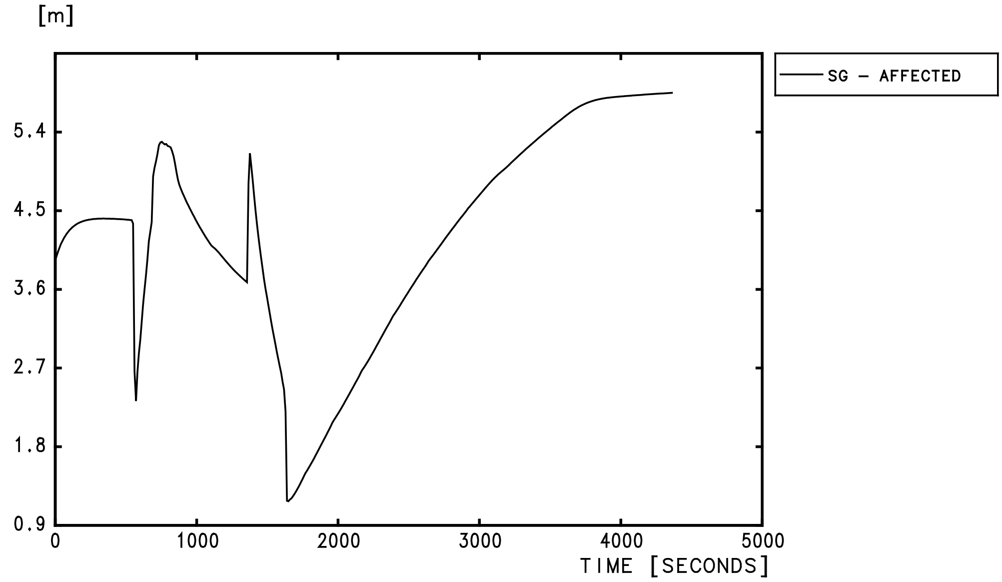
SG LEVEL – NR



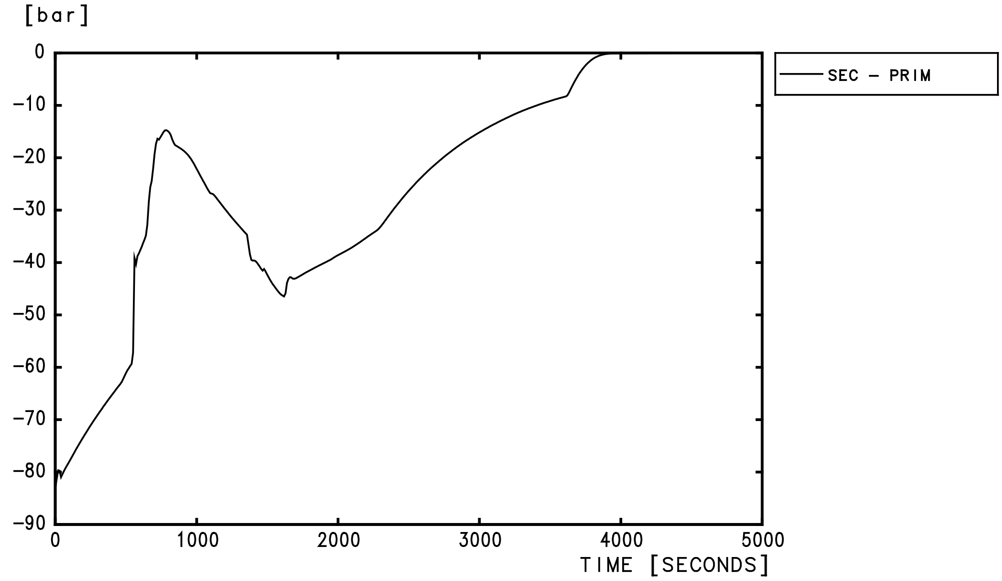
SG LEVEL – WR

SECTION 14.5.10 - FIGURE 3

SG Collapsed Liquid Level/Differential Pressure across the Break
SGTR 2-tube Case 1: Short Term without LOOP



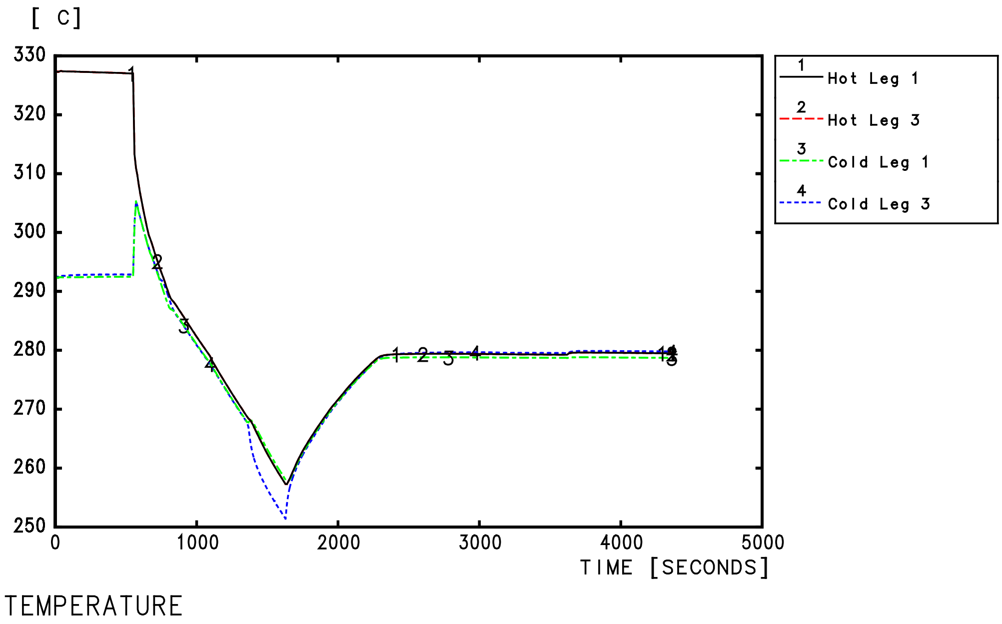
SG PHYSICAL LEVEL

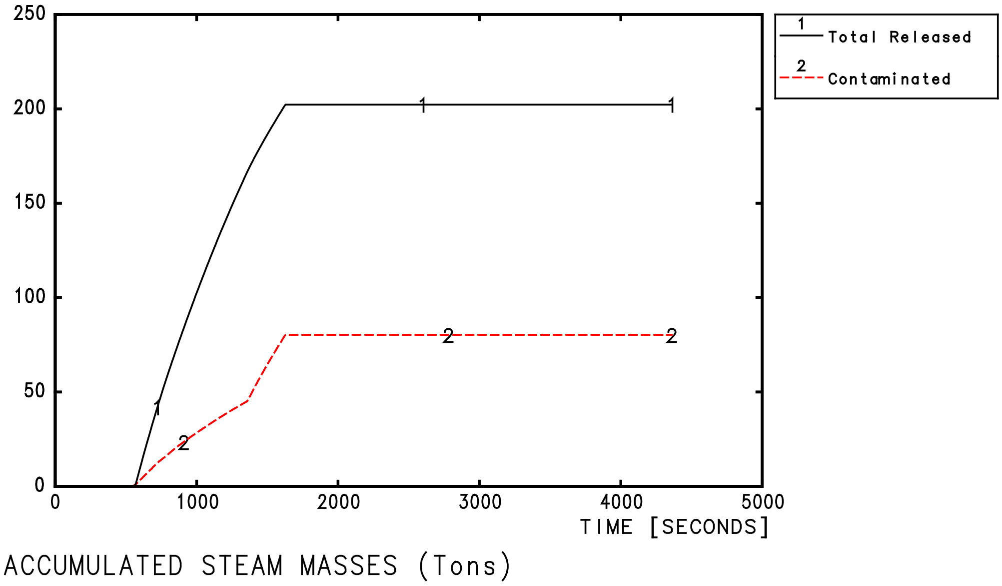


DIFFERENTIAL PRESSURE ACROSS THE BREAK

SECTION 14.5.10 - FIGURE 4

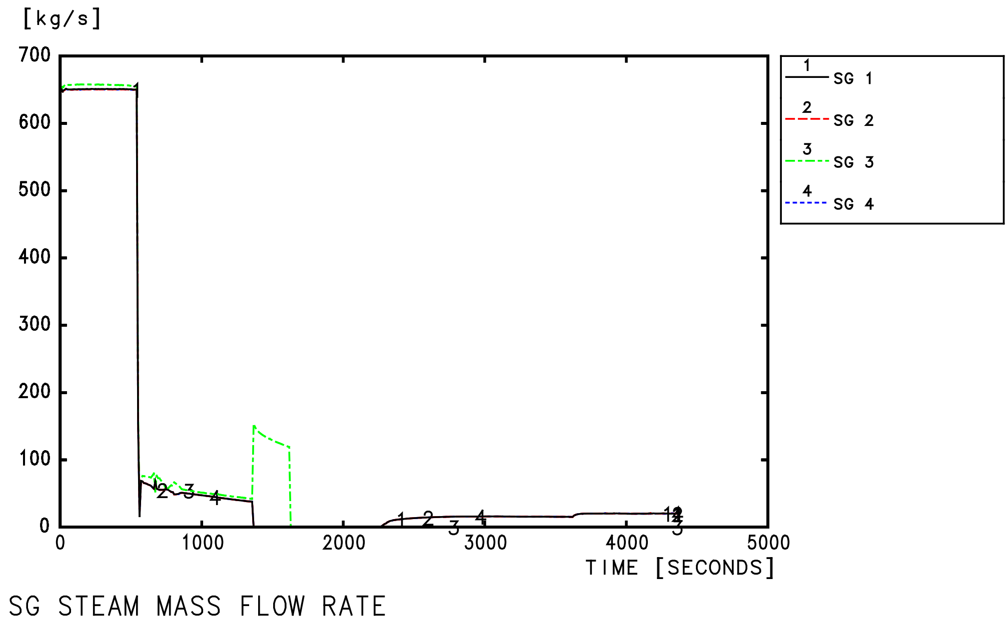
Temperatures/Total Steam Mass discharged through SGa VDA [MSRT]
SGTR 2-tube Case 1: Short Term without LOOP

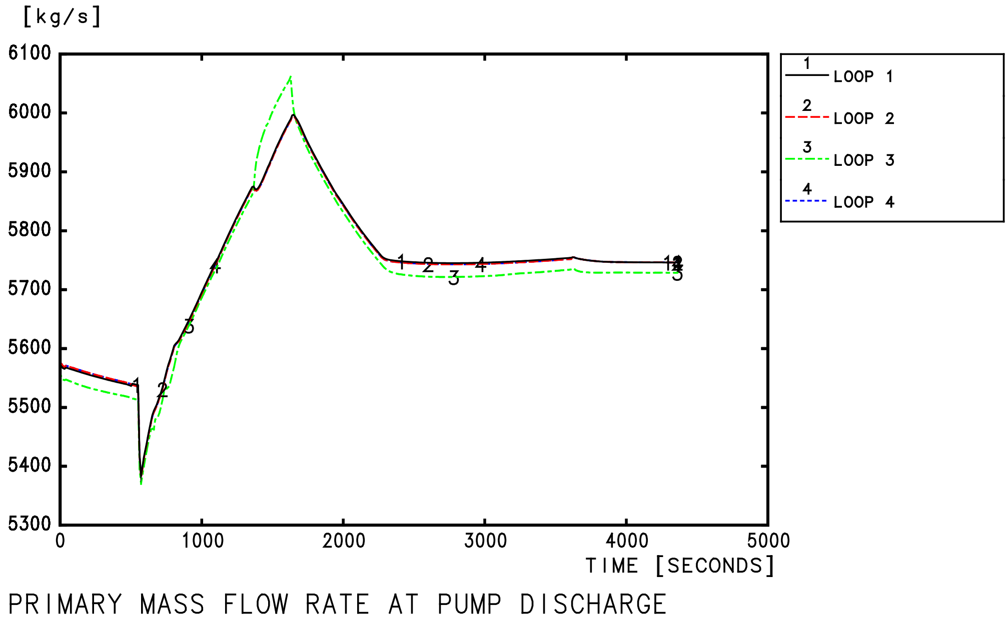




SECTION 14.5.10 - FIGURE 5

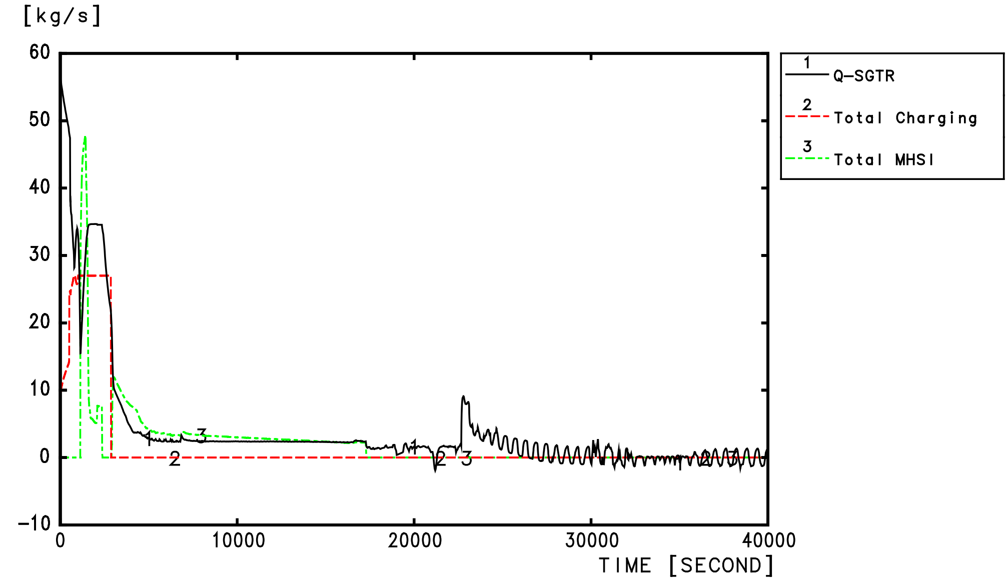
SG Steam Outlet Flow /Primary Mass Flow
SGTR 2-tube Case 1: Short Term without LOOP



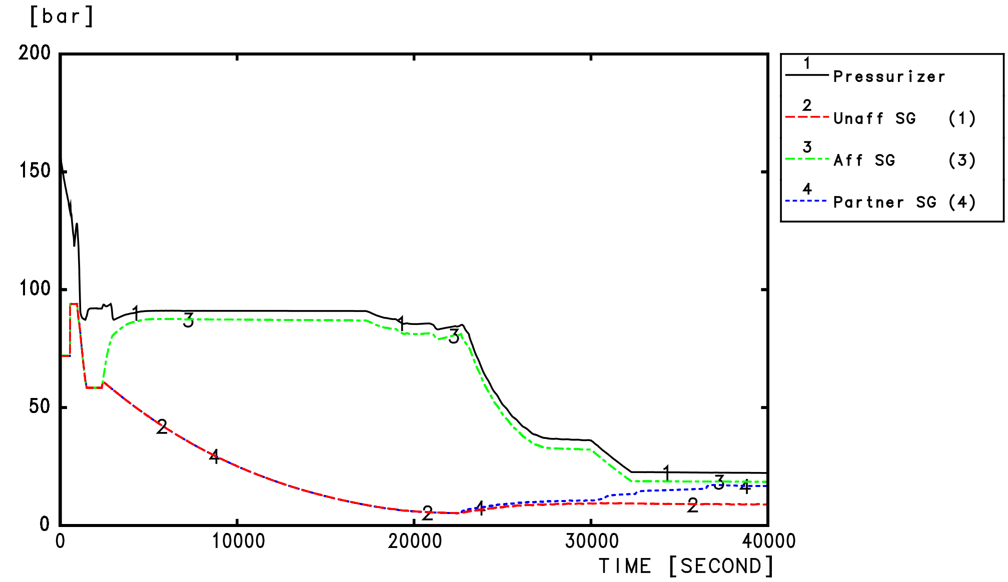


SECTION 14.5.10 - FIGURE 6

Pressuriser and SG Pressures/Mass Flows
SGTR 2-tube Case 3: Long Term without LOOP



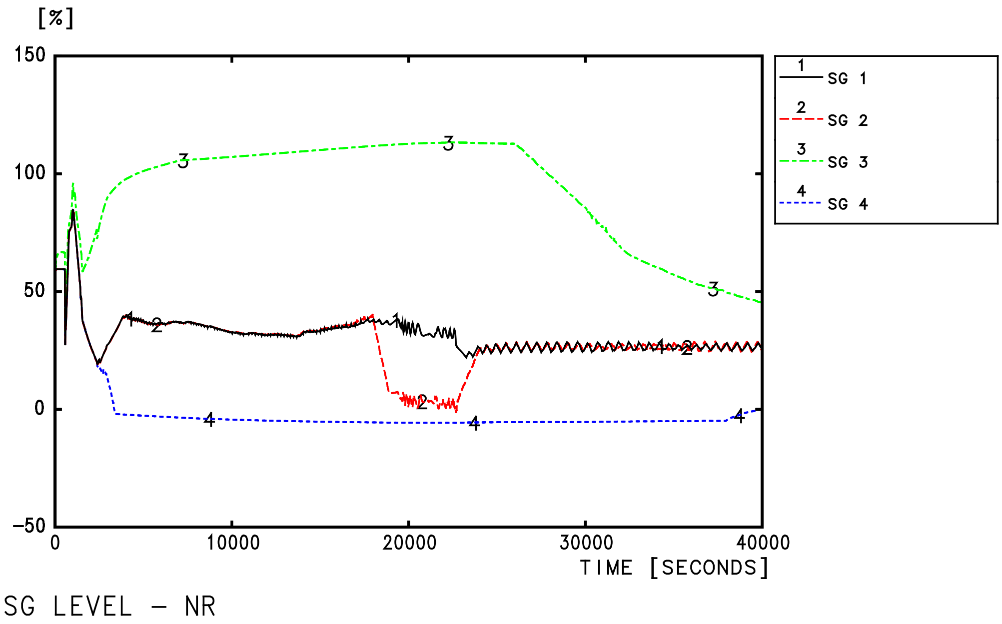
MASS FLOW RATE

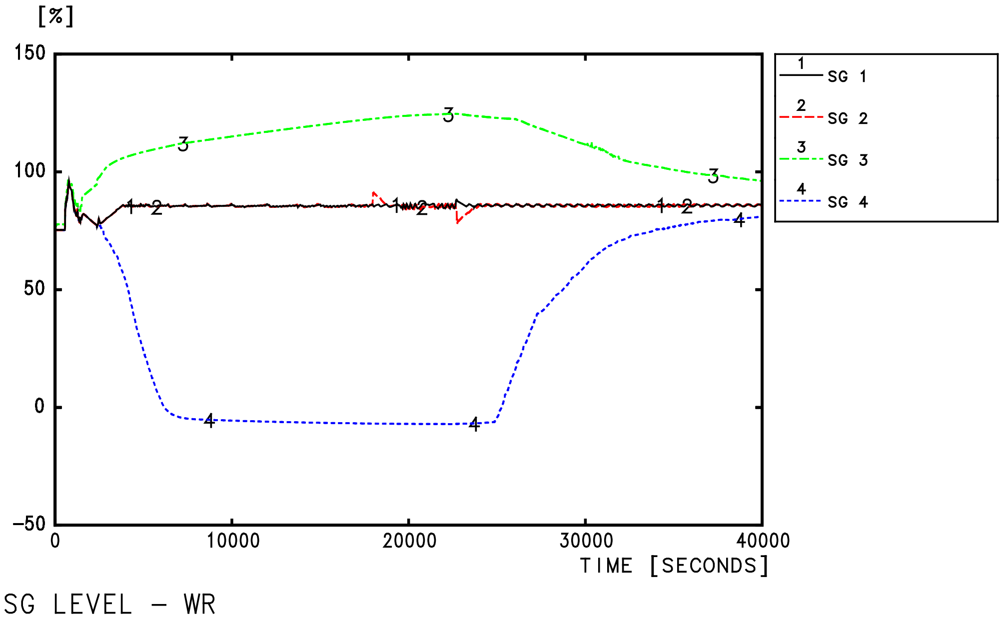


PRESSURE − PZR, RCS and SG

SECTION 14.5.10 - FIGURE 7

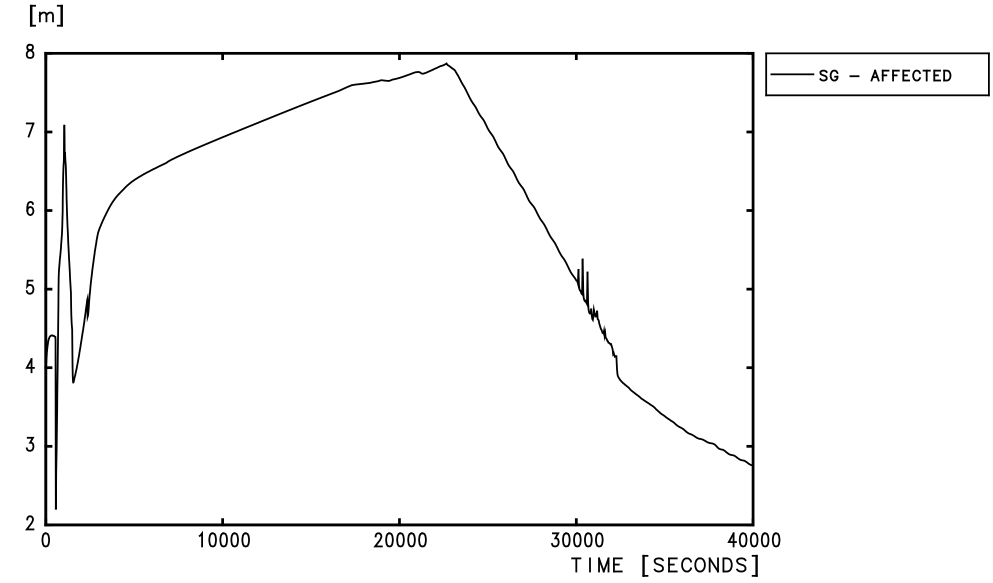
SG Levels: Narrow Range/Wide Range
SGTR 2-tube Case 3: Long Term without LOOP



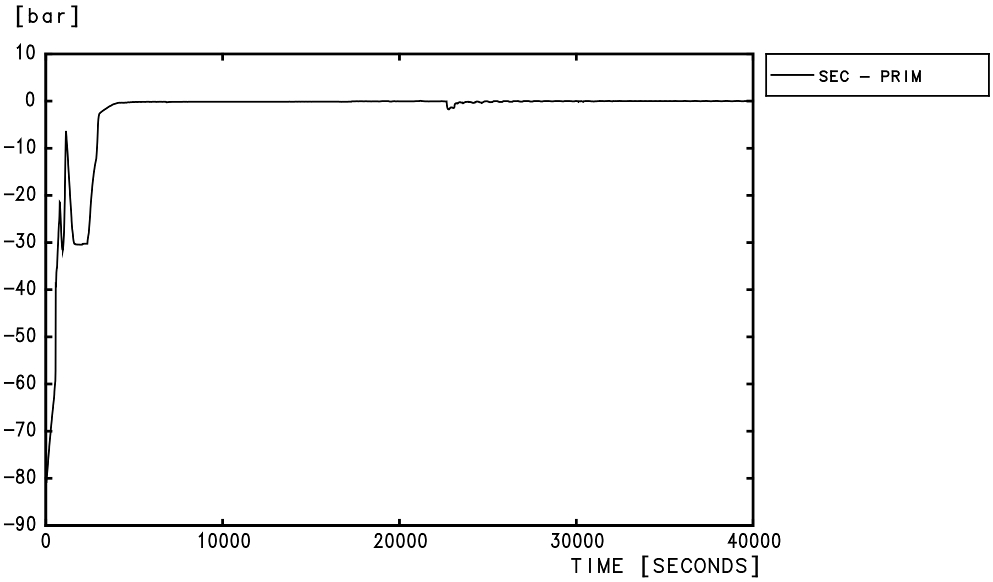


SECTION 14.5.10 - FIGURE 8

SG Collapsed Liquid Level/Differential Pressure across the Break
SGTR 2-tube Case 3: Long Term without LOOP



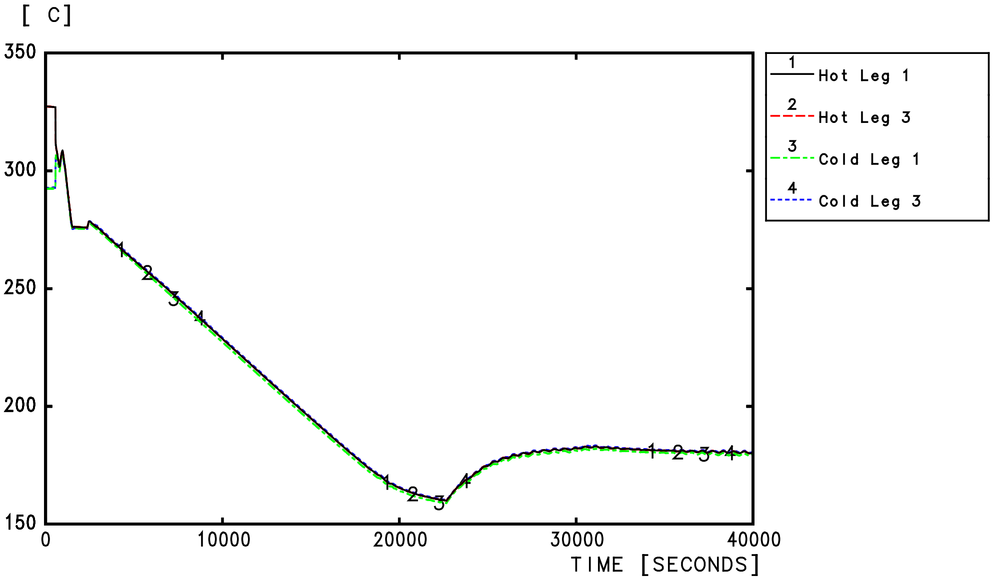
SG PHYSICAL LEVEL



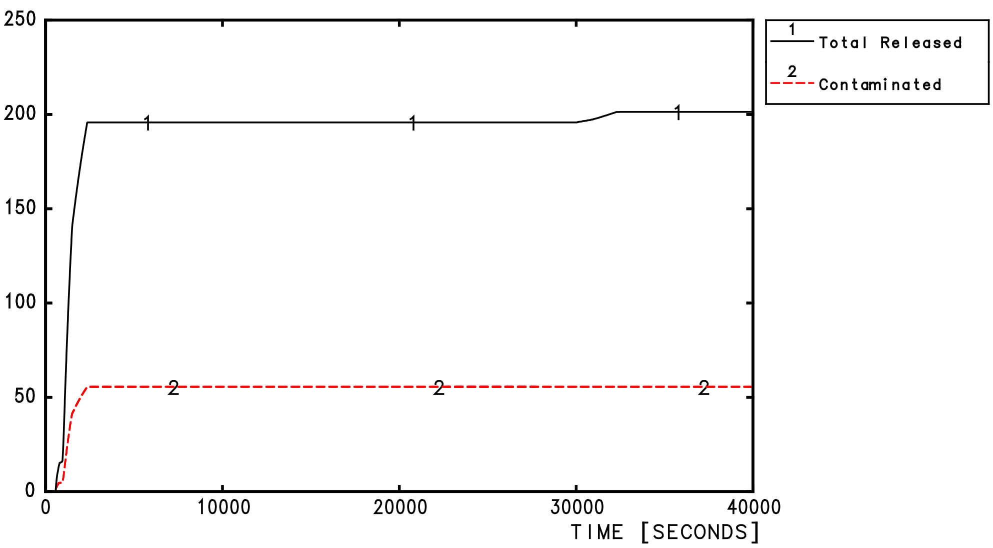
DIFFERENTIAL PRESSURE ACROSS THE BREAK

SECTION 14.5.10 - FIGURE 9

Temperatures/Total Steam Mass discharged through SGa VDA [MSRT]
SGTR 2-tube Case 3: Long Term without LOOP



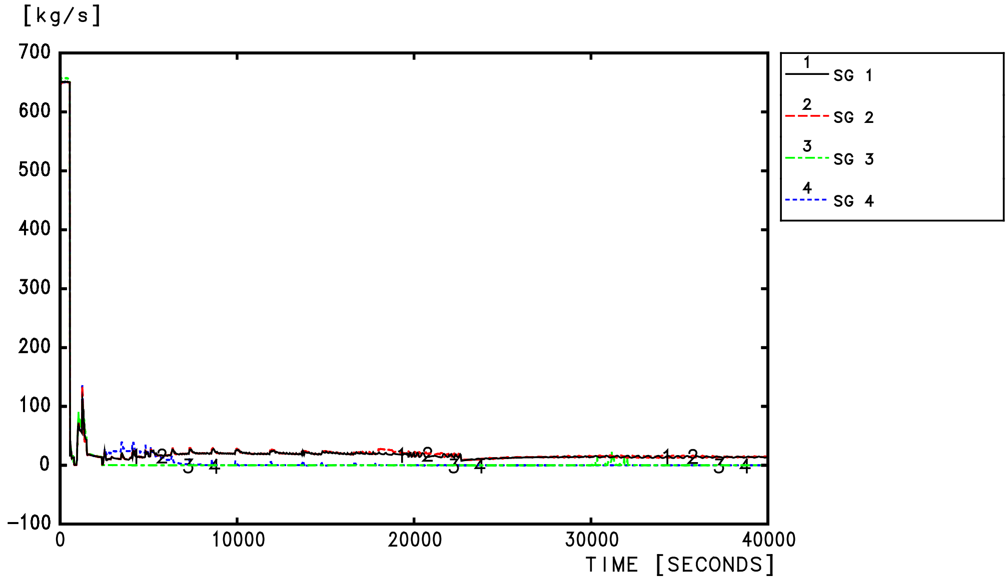
TEMPERATURE



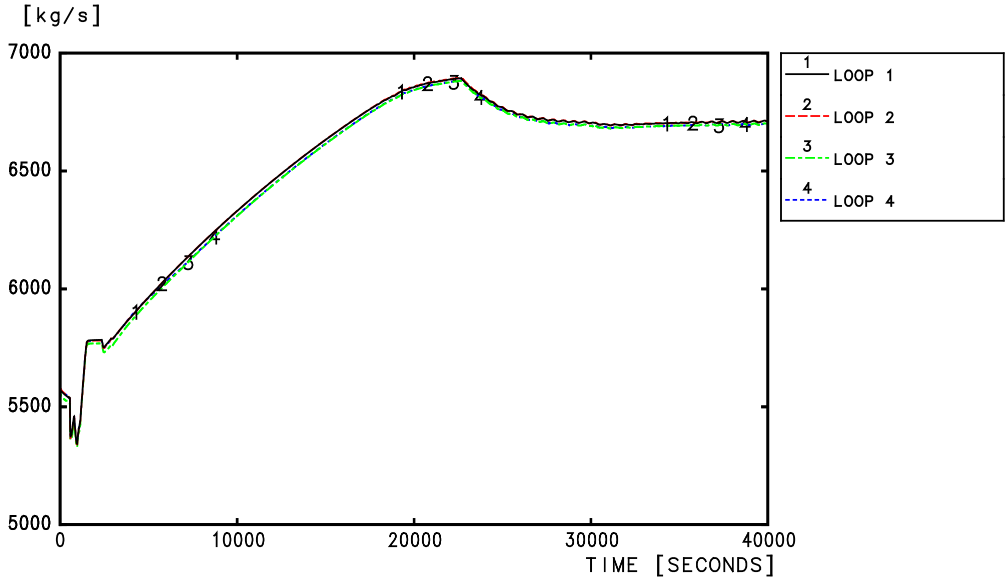
ACCUMULATED STEAM MASSES (Tons)

SECTION 14.5.10 - FIGURE 10

SG Steam Outlet Flow /Primary Mass Flow
SGTR 2-tube Case 3: Long Term without LOOP



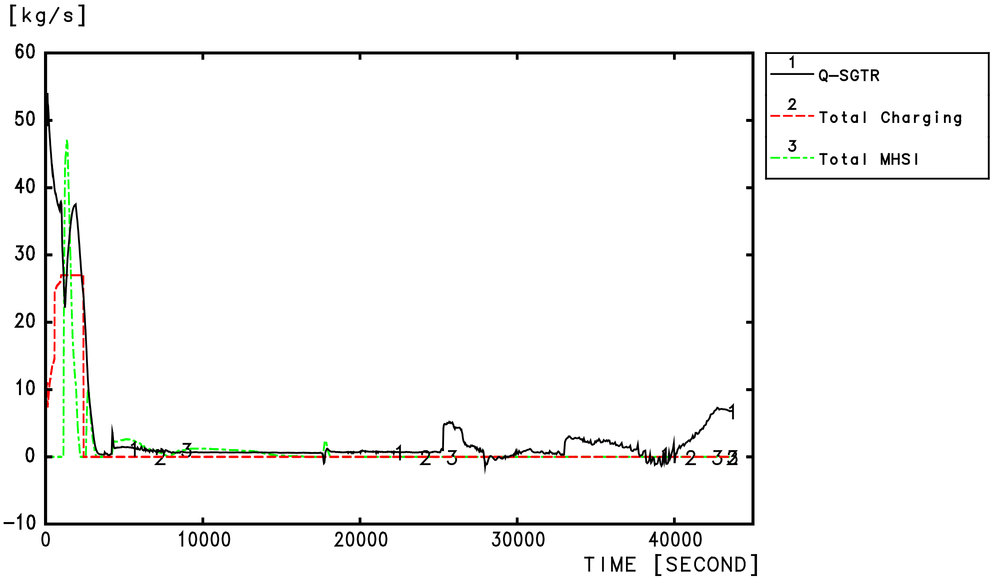
SG STEAM MASS FLOW RATE



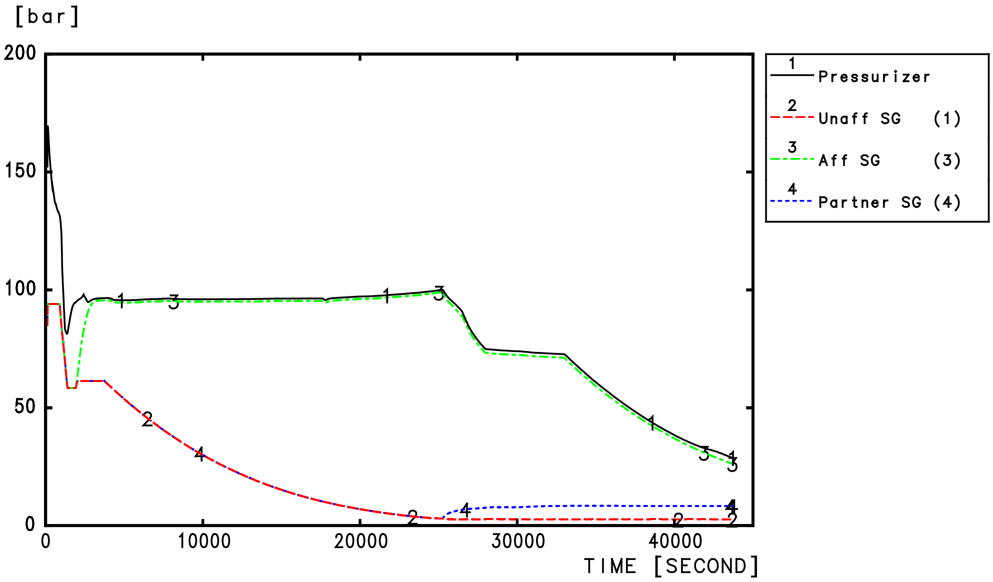
PRIMARY MASS FLOW RATE AT PUMP DISCHARGE

SECTION 14.5.10 - FIGURE 11

Pressuriser and SG Pressure/Mass Flows
SGTR 2-tube Case 2 and 4: Short and Long Term with LOOP



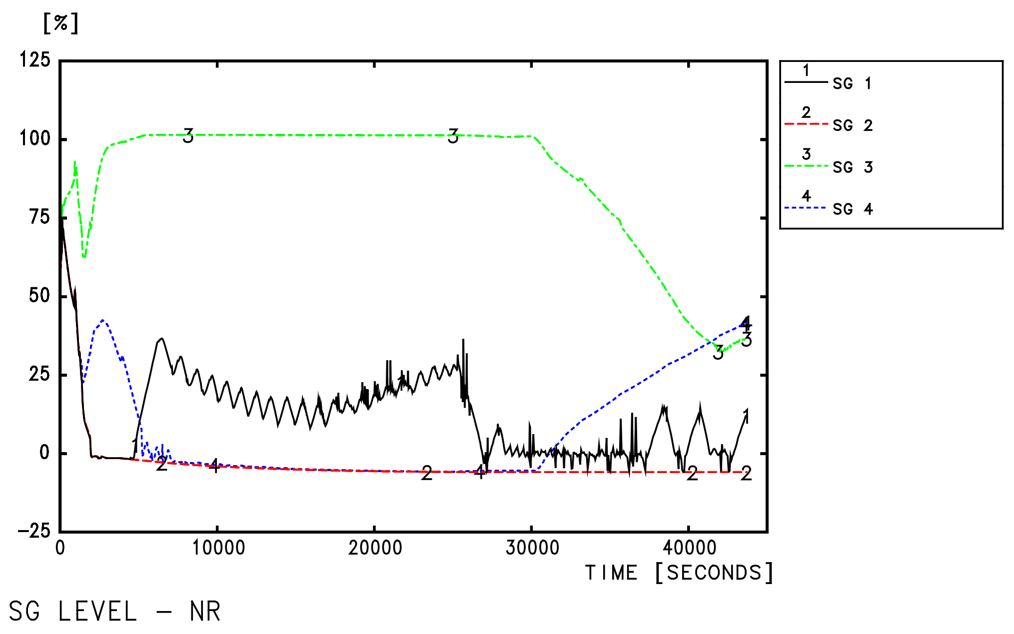
MASS FLOW RATE

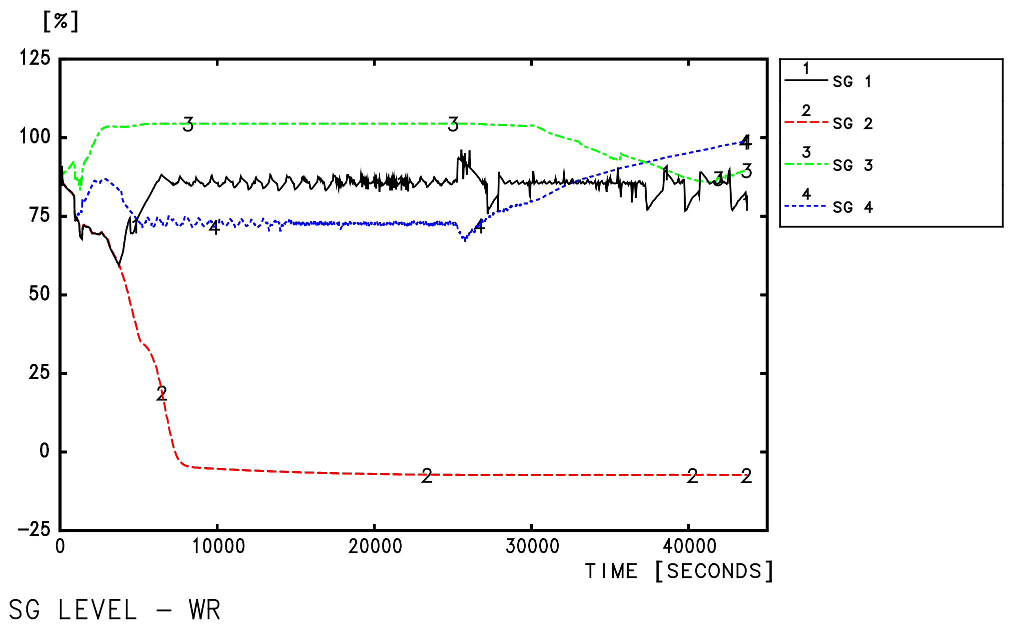


PRESSURE − PZR, RCS and SG

SECTION 14.5.10 - FIGURE 12

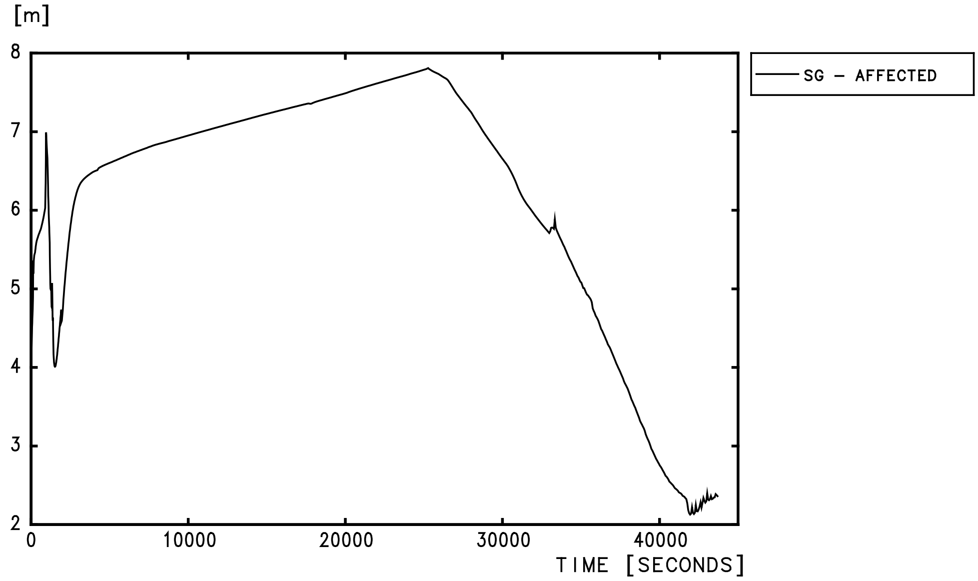
SG Levels: Narrow Range/Wide Range
SGTR 2-tube Case 2 and 4: Short and Long Term with LOOP



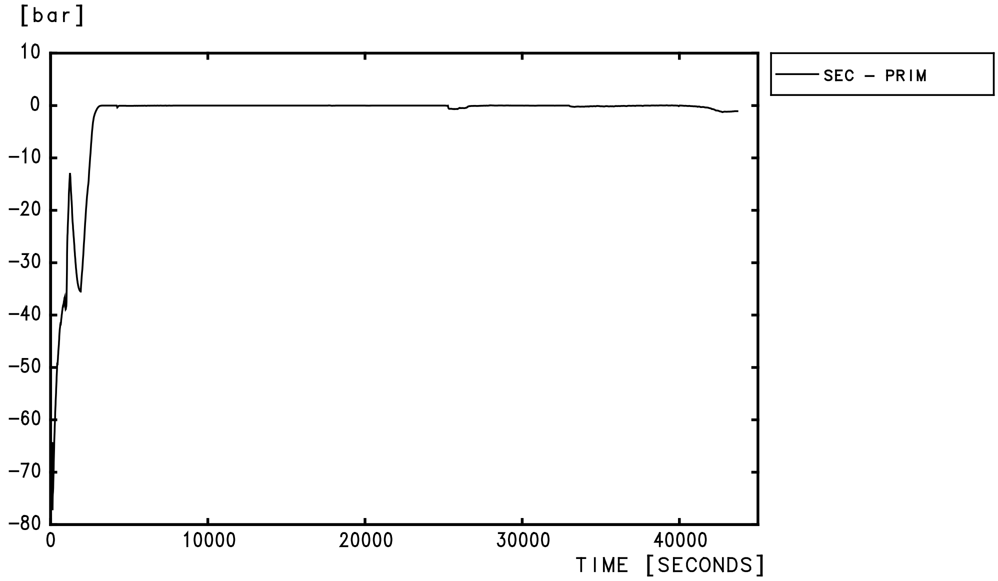


SECTION 14.5.10 - FIGURE 13

SG Collapsed Liquid Level/Differential Pressure across the Break
SGTR 2-tube Case 2 and 4: Short and Long Term with LOOP



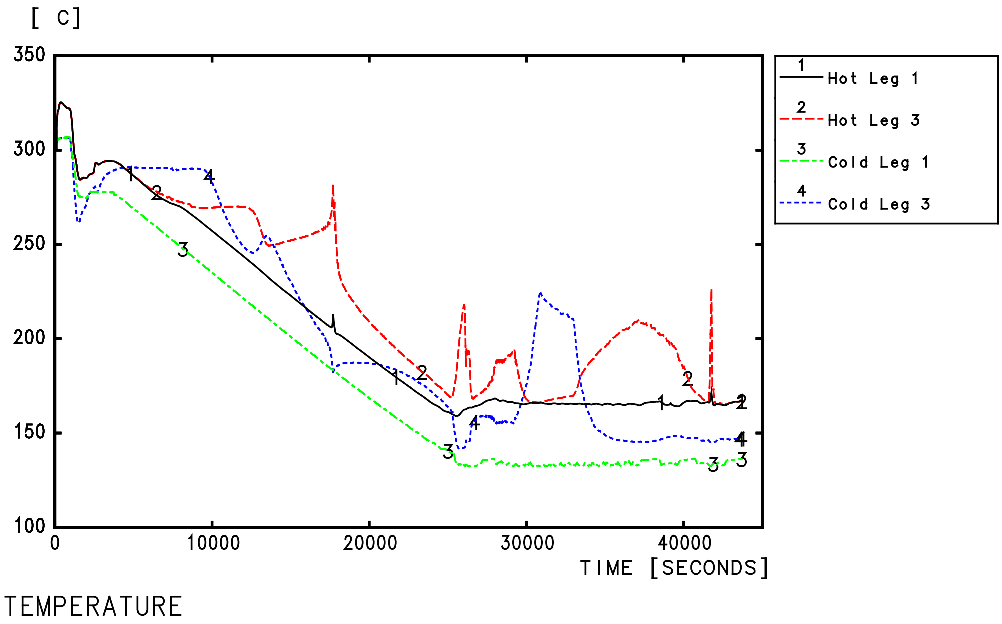
SG PHYSICAL LEVEL

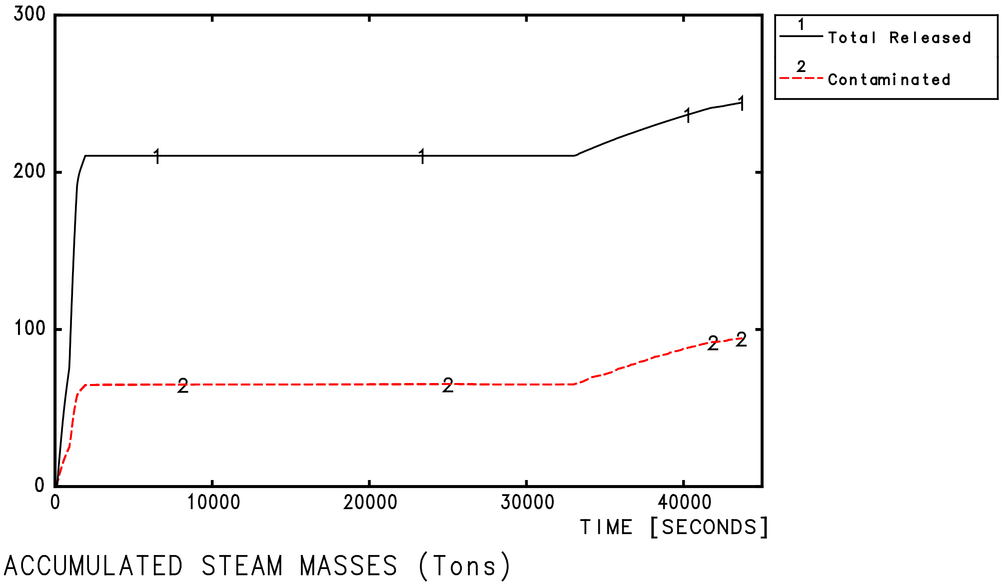


DIFFERENTIAL PRESSURE ACROSS THE BREAK

SECTION 14.5.10 - FIGURE 14

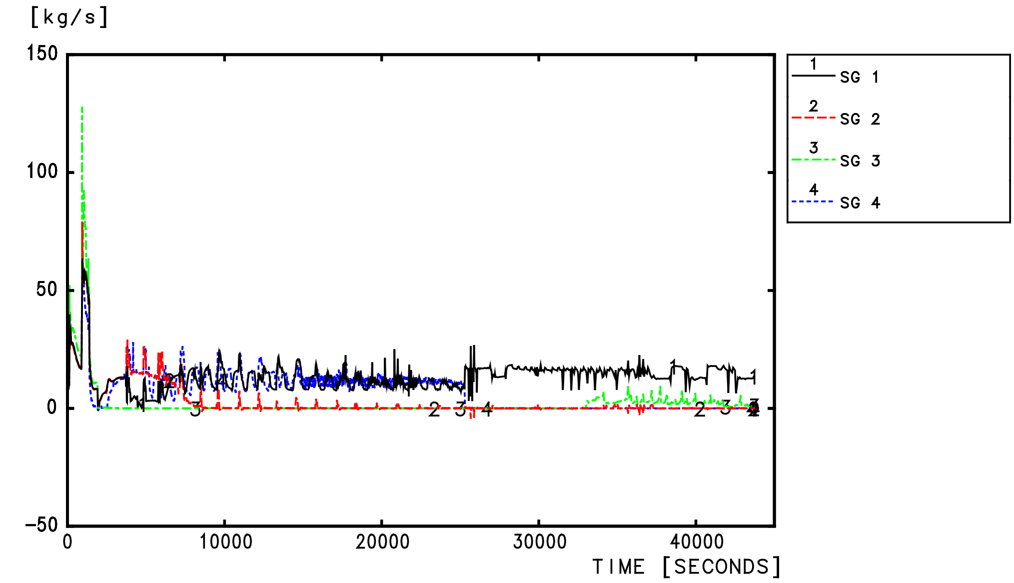
Temperatures/Total Steam Mass discharged through SGa VDA [MSRT]
SGTR 2-tube Case 2 and 4: Short and Long Term with LOOP



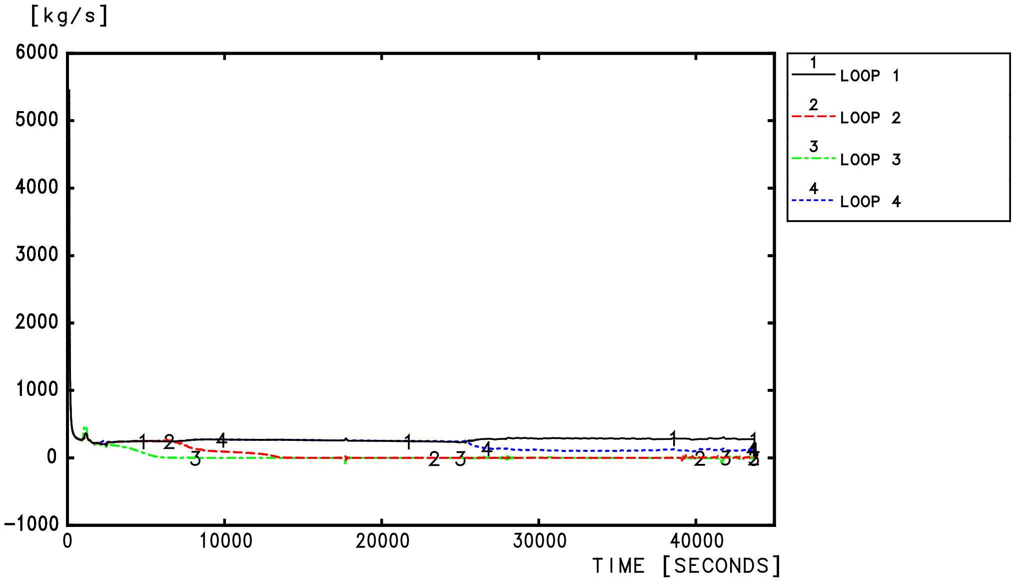


SECTION 14.5.10 - FIGURE 15

SG Steam Outlet Flow /Primary Mass Flow
SGTR 2-tube Case 2 and 4: Short and Long Term with LOOP



SG STEAM MASS FLOW RATE



PRIMARY MASS FLOW RATE AT PUMP DISCHARGE

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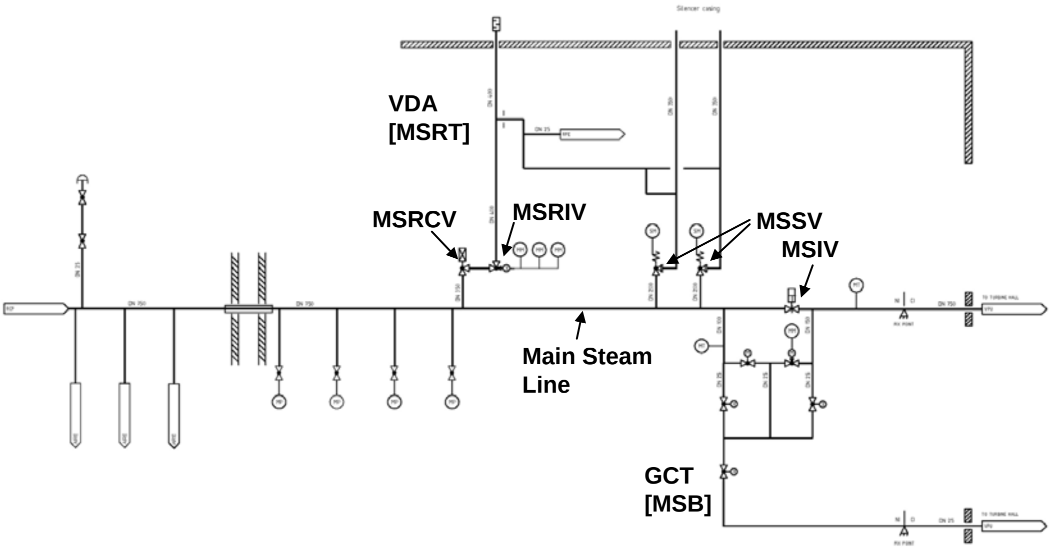
SECTION 14.5.10 - FIGURE 16

Schematic of Main Steam Line showing position of Main Steam Relief Routes and Valves



11. FUEL HANDLING ACCIDENT

11.1. POSTULATED ACCIDENT SEQUENCE AND ACTIVITY RELEASE TO THE ENVIRONMENT

It is assumed that a fuel assembly is damaged during handling, and that this accident takes place in the Fuel Building.

The assumed accident sequence is as follows:

- The accident occurs 60 hours after reactor shutdown. This is the earliest time for the start of refuelling.
- Amount of damage – all the fuel rods located on an outer edge of the affected fuel assembly fail.

11.2. CONSEQUENCES

The event is only considered for the release of radioactivity. The radiological release calculations are provided in Sub-chapter 14.6.

11.3. SYSTEM SIZING

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

12. BORON DILUTION DUE TO A NON-ISOLATABLE RUPTURE OF A HEAT EXCHANGER TUBE

12.1. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

12.1.1. Definition, causes and description of the accident

The analysed accident is assumed to occur as a result of a Component Cooling Water System (RRI [CCWS]) reverse leak through the damaged tube of a heat exchanger (PCC-4).

The initial conditions are where the RRI [CCWS] pressure is above the operating pressure of the heat exchangers. In such situations the reactor coolant pumps have been stopped. The corresponding standard reactor states, described in Sub-chapter 14.0 are as follows:

- cold shutdown (state C),
- cold shutdown for refuelling or maintenance, including:
    - mid-loop operation with level at 3/4 loop and with the vessel closed (state C),
    - mid-loop operation with level at 3/4 loop and with the vessel open (state D),
    - RPV flange level operation with the vessel open (state D),
    - Reactor cavity flooded for refuelling (state E).

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

A typical sequence of events is given below, from the initiating event to the controlled state. A description of initiating events is given later, in section 12.1.2 of this sub-chapter.

From the initiating event to the controlled state

For all shutdown conditions the RCCAs are fully inserted. The uncontrolled boron dilution causes a reactivity insertion which could lead to unintentional criticality.

The progress of the transient could lead to transitory states, at power, for which the pressure, temperature and reactivity conditions would be uncontrolled.

The “High neutron flux (source range)” alarm (F1A) is actuated before criticality is reached. The operators are informed but the safety demonstration takes no credit for manual actions during a grace period of 30 minutes following the first significant information. This is consistent with the rules for operator actions discussed in Sub-chapter 14.0.

For the cases studied where the RRI [CCWS] pressure is above operating pressure, the initial boron concentrations are higher than those in normal cold shutdown and the RCP [RCS] level can be at mid-loop, 3/4 loop, or at RPV flange level.

The most conservative scenario occurs during mid-loop operation where the coolant inventory is low.

For dilution event which are not halted by RCV [CVCS] isolation the operator acts either after a 30 minute grace period from the Main Control Room (MCR) through isolation of RIS/RRA [SIS/RHRS] train (F1B) or, after a grace period of one hour, by a local action outside the MCR. An example of this type of event would be injection of RRI [CCWS] water into the LHSI in residual heat removal mode (LHSI/RHR) via a leakage of the RIS/RRA [SIS/RHRS] heat exchanger where the RCV [CVCS] tank and pump bypass line are operating. During the grace period, the core is kept sub-critical by boration via the RBS [EBS] (F1A) initiated from the MCR.

During the transient, the core remains sub-critical. When the dilution source is isolated, the controlled state, defined in Sub-chapter 14.0, is reached:

- Core sub-critical.
- Heat removal provided via an open loop cooling chain such as the SG and the ASG [EFWS].
- Core coolant inventory stable.

Transient analyses in state D bound those in state E.

12.1.2. Causes, initiator, events

The main heat exchangers considered for the ‘boron dilution due to a non isolatable rupture of a heat exchanger tube’ event (PCC-4) are:

- RCV [CVCS] high pressure heat exchangers.
- Nuclear sampling (RCP [RCS]) heat exchangers
- RIS/RRA [SIS/RHRS] heat exchangers
- RCP [RCS] pump thermal barriers

A double-ended rupture of one tube results in a maximum dilution flow of around 5 kg/s when the RRI [CCWS] pressure is above the operating pressure of these heat exchangers. This is calculated for the RCV [CVCS] high pressure coolers.

The maximum continuous dilution flow assumed in the analyses of this PCC-4 event is 5.5 kg/s (20 te/h).

12.1.3. Decoupling criteria

The boron dilution due to a heat exchanger leak is classified as a PCC-4 event.

The safety criteria are the radiological limits for PCC-4 events described in Sub-chapter 14.0.

In addition, the core must remain sub-critical during the transient.

12.1.4. Reactor protection system actions

The operator is initially informed of the dilution by the “source range high neutron flux (alarm)” protection channel which is F1A qualified.

This alarm is displayed in the reactor building and in the MCR when its setpoint is exceeded. The setpoint is a flux three times the current flux during shutdown conditions.

The core remains sub-critical for at least 30 minutes following the alarm actuation. The operator, acting in accordance with the emergency procedures, initiates boration and the core remains sub-critical. The source of dilution is then isolated either via a manual action from the MCR, at the earliest 30 minutes after the alarm actuation, or by a local manual action outside the MCR, at the earliest one hour after the alarm actuation.

Section 14.5.12 - Figure 1 provides a schematic representation of the high neutron flux (source range) protection channel.

12.2. METHODS AND ASSUMPTIONS

12.2.1. Choice of single failure and preventive maintenance

During the transient from the initiating event to the controlled state, the F1A functions or systems used for the safety demonstration conform with the single failure criterion.

For these events, the core remains sub-critical.

12.2.2. Method of analysis

A parametric study is performed using reactivity balance calculations to show that the initial boron concentration requirement is consistent with the minimum time assumed for operator action.

For these events, the core remains sub-critical.

12.2.3. Initial conditions

Continuous mixing in the reactor vessel is provided by one heat removal pump.

The analysis assumes a maximum dilution flow of 5.5 kg/s (20 te/h).

The transient with dilution via the rupture of a heat exchanger tube occurs with the RRI [CCWS] pressure above the pressure of the heat exchanger. In these situations the reactor coolant pumps are stopped. The initial boron concentration is thus the required boron concentration in cold shutdown for refuelling or maintenance. The initial boron concentration is the minimum In-containment Refuelling Water Storage Tank (IRWST) boron concentration. This is defined in Sub-chapter 14.1 to be 2405 ppm for UO_2 fuel management schemes and 2600 ppm for the MOX fuel management scheme. An uncertainty of -100 ppm is included in this study. This data refers to natural boron, and takes no account of B10 enrichment. The bounding UO_2 management scheme is the ‘UO_2 In Out 18 months’ scheme.

The volume of borated water required to be maintained in the reactor is

- At mid-loop level: 120 m³ (bounding case).
- At nominal level: 336 m³.

12.3. RESULTS AND CONCLUSIONS

The reactor protection relies on the “High neutron flux (source range)” alarm and associated operator actions.

The critical boron concentrations at BOL, cold zero power, 1 bar and 20°C, with all rods inserted are used. These are 1290 ppm for the "UO_2 IN-OUT 18 months" fuel management scheme, the bounding value, and 1402 ppm for the MOX fuel management scheme. An uncertainty of +100 ppm is included for this study. These concentrations have been calculated using the two-energy-group, three-dimensional nodal diffusion code SMART.

The calculations have been performed assuming the critical boron concentrations, the initial boron concentrations, the most onerous fuel management scheme for each type of fuel and a dilution flow of 5.5 kg/s (20 te/h). The results for the delay between the shutdown high neutron flux alarm signal and a subsequent return to criticality are as follows:

	Mid-loop level	Nominal level
UO_2	67 min	187 min
MOX	67 min	189 min

These delays are sufficient for an operator action after a 30 minute grace period to be successful.

The maximum dilution flow rates for a successful operator action at 30 minutes are also calculated.

In mid-loop conditions they are:

- 12.2 kg/s (44 te/h) for UO_2.
- 12.5 kg/s (45 te/h) for MOX.

12.4. TRANSITION TO THE SAFE SHUTDOWN ANALYSIS

The safe shutdown state is defined in Sub-chapter 14.0 to be:

- Core sub-critical even after the xenon depletion.

- Decay heat removed entirely by the closed-loop cooling chains LHSI/RRI/SEC [LHSI/CCWS/ESWS].

- Activity releases and barrier integrity within the limits of each PCC.

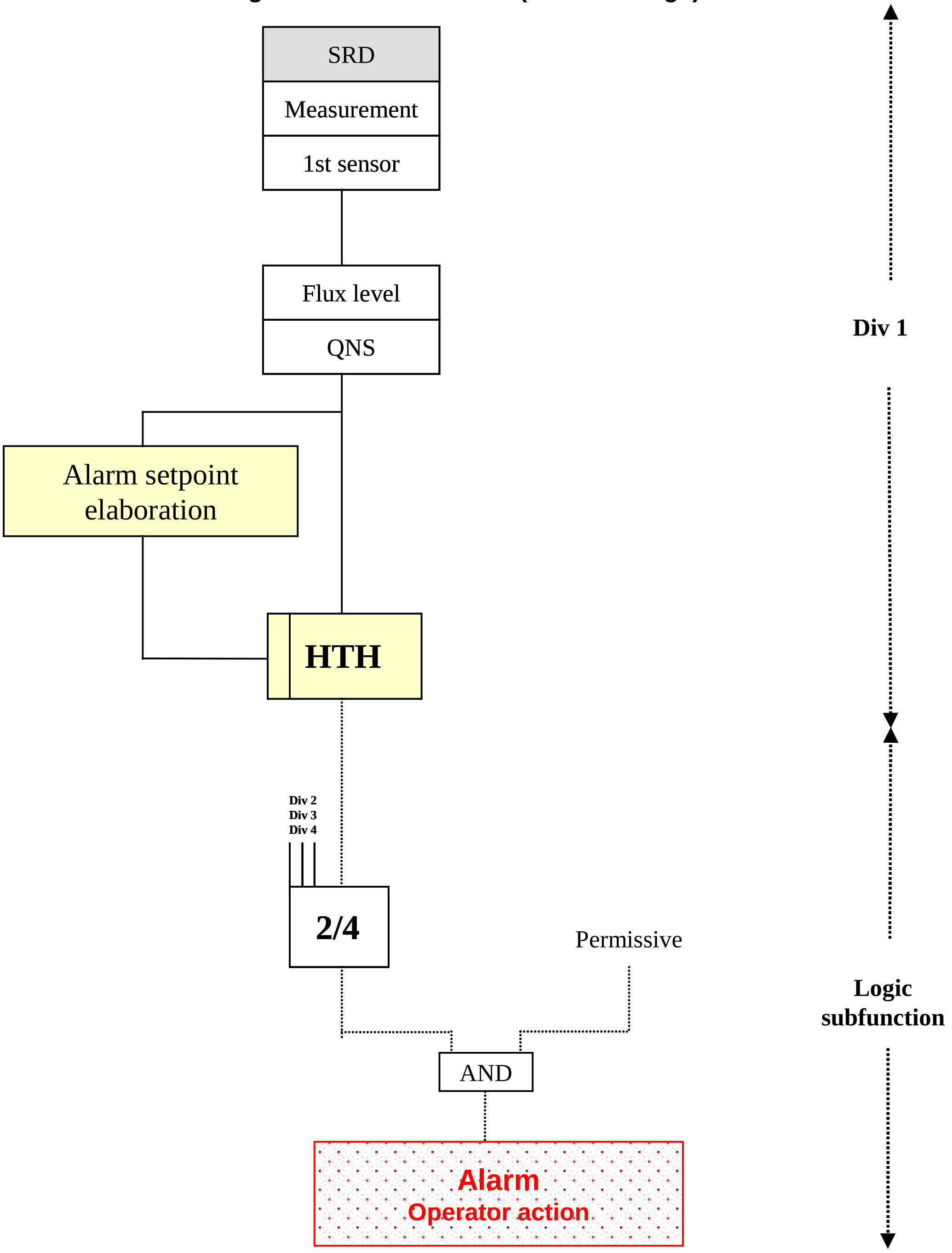
In addition, for this transient, the safe shutdown state is characterised by the recovery of the initial boron concentration. This transient is covered by the PCC-2 dilution event for the boration capability as discussed in section 13 of Sub-chapter 14.3.

12.5. SYSTEM SIZING

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

SECTION 14.5.12 - FIGURE 1

High Neutron Flux Alarm (Source Range) Channel



13. RUPTURE OF SYSTEMS CONTAINING RADIOACTIVITY IN THE NUCLEAR AUXILIARY BUILDING

This transient is only considered for the radiological consequence aspects. The relevant calculations are presented in Sub-chapter 14.6 ' Radiological Consequences of Design Basis Accidents '.

14. ISOLATABLE SAFETY INJECTION SYSTEM BREAK (DN ≤ 250 MM), IN RESIDUAL HEAT REMOVAL MODE (STATES C, D)

In states C and D, reactor heat removal is provided by the residual heat removal system RIS/RRA [SIS/RHRS]. If a break occurs in the system outside or inside the containment, it leads to a loss of primary coolant inventory and a discharge of radioactive primary fluid in the safeguards building or into the containment, respectively.

In states C and D, any break in the RIS/RRA [SIS/RHRS] system located beyond the reactor coolant pressure boundary (CPP [RCPB]) second isolation valve is considered as a PCC-4 event. The maximum break in terms of primary coolant loss is a guillotine break of the pipe connected to the hot leg of the primary coolant system which has a nominal diameter of DN 250.

PCC-4 breaks inside the containment in the reactor coolant system (RCP [RCS]) or in connected pipes up to 50 mm diameter (DN 50) which are non-isolable[1] and occur in a shutdown state, are dealt within section 7 of Sub-chapter 14.5.

14.1. BREAK OUTSIDE THE CONTAINMENT

14.1.1. Identification of Causes and Accident Description

14.1.1.1. General

The leaks considered range from operating leaks through mechanical pump seals and leaks in pipes, to a complete pipe fracture.

When compared to other LOCA scenarios considered for the RCP [RCS], RIS/RRA [SIS/RHRS] fractures outside the containment are subject to early detection.

Detection is based on the following F1A measurements described in Sub-chapter 6.3.

- In each safety system division there are four pressure measurement points sensitive enough to detect a rise in pressure on the order of 40 mbar. These are provided as part of the Reactor Protection System (RPR [PS]).

- In each safety system division there are four measurements of water level, which provide early detection of any leaks.

The RIS/RRA [SIS/RHRS] isolation signal is generated either by a high-water level signal or a high pressure signal in the safeguard building. This signal initiates the closure of the two RIS/RRA [SIS/RHRS] suction line isolation valves inside the containment. The high pressure signal is not generated for the smallest breaks, or when the RCP [RCS] temperature is below 100°C. In these circumstances, the safeguard building water-level signal is the only signal available to initiate isolation.

[1] On the primary circuit side of the CPP [RCPB] (primary pressure boundary) isolation valves.

Isolation of the RIS/RRA [SIS/RHRS] train in the division where the fracture is detected is automatic and is completed within two minutes for a nominal break diameter of 50 mm or above (≥ DN 50). This isolation action is classified F1A and meets the single failure criterion.

14.1.1.2. Typical Sequence of Events

The break causes a loss of primary coolant which, if not offset by the normal make-up, leads to the automatic start-up of safety injection on either a low ∆Psat or low loop level signal. This make-up is sufficient to prevent core uncovery. In addition, the release of fluid in the safeguard building leads to the automatic isolation of the hot leg suction of the RIS/RRA [SIS/RHRS] and a RIS/RRA [SIS/RHRS] pump trip in the affected division. Automatic isolation of the RIS/RRA [SIS/RHRS] injection flow to the cold leg is provided by closing of the check valves. Thus, following automatic isolation of the broken RIS/RRA [SIS/RHRS] train, the primary system is isolated from the break and the safety injection allows the primary water inventory to be replenished. Therefore, there is no further challenge to the primary system water inventory.

The loss of primary coolant water inventory can potentially lead to the automatic trip of the LHSI/RHR pump of the intact trains. This could occur in some cases on a very low loop level signal or on a signal indicating hot leg saturation. Two cases are considered.

- At least one LHSI/RHR pump is still operating after isolation of the break, i.e. no automatic tripping of the trains in the unaffected divisions. The operating LHSI/RHR train provides the long-term decay heat removal. Hence, the controlled state and the safe shutdown state are reached immediately following break isolation and without operator action.

- No LHSI/RHR pump is operating after the isolation of the break, i.e. automatic trip occurs of the trains in the unaffected divisions. In these circumstances it is necessary to restart[2] a RIS/RRA [SIS/RHRS] train for long-term decay heat removal or, if appropriate, other cooling means. Therefore, operator action is required to transfer the plant to the safe shutdown state.

Note: Following trip of the LHSI/RHR pumps in the intact trains in the latter case, and before putting long-term cooling into operation, the decay heat is removed partly by the break flow prior to break isolation, and partly by heating the safety injection flow. Subsequently:

- When the RCP [RCS] is closed, heat removal can be achieved using the atmospheric steam dump (VDA) [MSRT] of the two SGs on standby, fed by their ASG [EFWS] trains. The controlled state is reached automatically once the heat-up of the primary circuit is sufficient to promote heat transfer to the steam generator. This heat removal route will operate for as long as secondary water inventory is available.

- Alternatively, again with the primary system closed, heat can be removed by bleed off of primary coolant following boiling in the core. The controlled state is thus reached, and can be maintained in this way as long as the rise in containment pressure or temperature is not excessive.

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[2] Either, start an available RIS/RRA [SIS/RHRS] train that was on standby, or return to service an intact RIS/RRA [SIS/RHRS] train that was previously stopped.

14.1.2. Safety Criteria

The safety criteria for radiological limits for PCC-4 events must be satisfied. All the following criteria must be met as discussed in Sub-chapter 14.0:

- The core must not be uncovered to meet the LOCA criteria discussed in Sub-chapter 14.0.
- Long-term cooling must be assured. The core temperature must be maintained at an acceptable level and the decay heat must be removed.
- Activity release into the safeguard building and the containment, described in section 1 of Sub-chapter 6.2, must lead to radiological consequences within the PCC-4 limits defined in Sub-chapter 14.6.

It must be shown that the following two states can be attained whilst meeting the safety study rules given in Sub-chapter 14.0.

- Controlled state, using only F1A items.
- Safe shutdown state, using F1A and/or F1B systems. A single failure for the F1A or F1B systems is assumed. The use of F2 systems is inconsistent with the normal PCC rules for reaching a safe state. The relaxation in the rules is applied in studies of primary coolant loss in shutdown states where the RIS/RRA [SIS/RHRS], the F1 heat removal system, is initially in service. If the F2 means claimed is containment heat removal system (EVU [CHRS]), the safe state must be able to be maintained in the long term by other systems which are not dedicated to severe accidents. This is to limit the duration of EVU [CHRS] operation, as the EVU [CHRS] system must remain available to mitigate severe accidents. As with the EVU [CHRS], these other systems do not need to be F1.

14.1.3. Methods and Assumptions

This analysis is performed with the CATHARE code, described in Appendix 14A.

14.1.3.1. Principal Events and CATHARE Models

The event is an isolatable LOCA, described in section 6 of this sub-chapter, where the qualification of the CATHARE code for such breaks is also discussed.

14.1.3.2. Initial Conditions

The initial conditions considered cover the reactor states when the RIS/RRA [SIS/RHRS] is in service in standard C or D states. The main characteristics of these initial states are summarised in the table below [Ref-1]:

State	RCP [RCS]	Primary Coolant Pressure	Primary Temperature	Primary / Pool Reactor Level	RRA [RHRS] Trains in Service	RCP [RCS] (in service)	Available SG (on standby)
C1	Closed	24.5 bar - 32 bar	55°C - 120°C	Pressuriser at no load	2	2	2
C2	Closed	1 bar – 32 bar	55°C - 70°C	Pressuriser at no load	4	1 or 2	2
C3	Slightly open pressuris able	1 bar	15°C - 55°C	Pressuriser between no load and ¾ loop	3	0	2
D	Open non pres surisable	1 bar	15°C -55°C	Between ¾ loop and high water level in the reactor pool	3	0	0

14.1.3.3. Main Assumptions

14.1.3.3.1. Definition of the Accident

The initiating event is a leak or break occurring in the RIS/RRA [SIS/RHRS] pipe work, outside the containment. The maximum break size considered is a double-ended guillotine break of the main pipe. The maximum flow of primary coolant is produced by a break located in the Safeguard Auxiliary Building, in a suction line of DN 250, upstream of the containment isolation valve. The break is assumed to be located outside the containment.

14.1.3.3.2. Protection and Mitigation Actions

The relevant automatic actions during this accident are:

- Isolation of the broken RIS/RRA [SIS/RHRS] train,
- Safety injection,
- Tripping the RIS/RRA [SIS/RHRS] pumps,
- Tripping the reactor coolant pumps.

Isolation of the affected RIS/RRA [SIS/RHRS] train

The consequences of the break in the safeguard building are:

- Pressure increases in the relevant rooms if the RIS/RRA [SIS/RHRS] temperature is above 100°C (state C1)
- Water appears in the safeguard building.

Automatic isolation of the broken RIS/RRA [SIS/RHRS] train occurs either on a high pressure signal or on high water level signal in the safeguard building.

The characteristics of these signals are:

- High pressure in the safeguard building (2/4)
  - Sensors: four pressure sensors per building
  - Classification: F1A
  - Qualification: qualified to ambient conditions following a guillotine break on the RIS/RRA [SIS/RHRS] in the safeguard building
  - Setpoint: 40 mbar (preliminary value)
  - I&C delay time: < 2 seconds
- High water level in the safeguard building (2/4)
  - Sensors: four pressure sensors per building
  - Classification: F1A
  - Qualification: qualified to ambient conditions following a guillotine break on the RIS/RRA [SIS/RHRS] in the safeguard building
  - Setpoint: The channel must allow detection of a break size of 50 mm in less than 90 seconds. This is a preliminary value for the requirement.
  - I&C delay time: < 2 seconds.

The isolation signal initiates the closing (F1A) of the two primary circuit isolation valves inside the containment located on the hot suction leg of the affected RIS/RRA [SIS/RHRS] train. It also trips the RIS/RRA [SIS/RHRS] pump. F1 classification of this action is not required. Full closure of the valves is conservatively assumed to occur 30 seconds after the isolation signal.

Safety injection

In states C and D, the safety injection signal starts the four MHSI pumps, with the high-flow recirculation lines open under P14[3] permission. The starting signal depends on the initial conditions:

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[3] P14 : permission to go from State B to State C

- Above P15[4]: low ΔPsat
- Below P15: low loop level

The logic of these signals is described in Sub-chapter 14.1.

Tripping the RIS/RRA [SIS/RHRS] pumps of the intact trains

The RIS/RRA [SIS/RHRS] pumps must be shut down quickly enough to prevent damage from cavitation following the voiding of the hot legs, or from vortex formation following the loss of water level in the loops. The type of signal depends on the initial operation of the reactor coolant pumps and therefore on the state of the reactor.

The characteristics of the signal are:

- States C1 and C2, reactor coolant pumps in operation: ΔPsat signal using pressure and temperature measurements in hot leg, with the following characteristics:
  - Sensors: one pressure sensor and one temperature sensor in each hot leg
  - $\Delta Psat = Psat(T_{HL}) - P_{HL}$
  - Classification: F1A
  - Qualification: qualified to degraded conditions in the containment
  - Setpoint: to be defined.
- States C3 and D, reactor coolant pumps not in operation: signal generated from the loop level measurement (2/4), with the following characteristics:
  - Sensors: one loop level sensor in each hot leg
  - Classification: F1A
  - Qualification: qualified to LOCA ambient conditions in the containment
  - Setpoint: low-low level (0.33 m, preliminary value)
  - I&C delay time: < 1.5 seconds
  - Accuracy: +/- 2%

As the single failure has been modelled on the RIS/RRA [SIS/RHRS] trains overall, it is unnecessary to assume failure of the RIS/RRA [SIS/RHRS] pump circuit breakers. The consequences of the failure of a pump to stop, and hence making it unavailable due to damage, are acceptable. Two of the RIS/RRA [SIS/RHRS] pumps have been correctly stopped in these conditions, and are therefore available to be restarted. A single pump is sufficient to provide the required decay heat removal.

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[4] P15 : permission to change the SI signal from ΔPsat signal to low loop level after four reactor coolant pumps stop

Tripping the reactor coolant pumps (states C1 and C2)

For initial conditions where the reactor coolant pumps are running, they are automatically tripped via a low pump ∆P signal combined with a safety injection signal.

14.1.3.3.3. Operator Actions

The safe shutdown state is immediately reached following the automatic isolation of the break if the RIS/RRA [SIS/RHRS] pumps in the intact trains are not automatically tripped. Consequently no operator action is required to reach the safe state.

If all the RIS/RRA [SIS/RHRS] pumps of the intact trains have been automatically tripped, the safe state can be reached by one of two methods. It either requires the starting of a standby RIS/RRA [SIS/RHRS] train by bringing back into service at least one tripped RIS/RRA [SIS/RHRS] train, or by bringing other cooling systems into service.

14.1.3.3.4. Residual Heat Removal Systems

The following table shows the systems available for decay heat removal following the isolation of the break for different initial conditions, and the associated residual heat removal capacities. [Ref-1]

System	Associated Initial States	Residual Heat Removal Capacity (including reactor coolant pump heat at start of state C1)
Two SG on standby (ASG [EFWS] and VDA [MSRT] (steam dump) class F1A)	C1, C2, C3 (primary system closed)	One SG removes the pump plus residual heat Two ASG [EFWS] tanks provide sufficient capacity for about 24 hrs without replenishment
RIS/RRA [SIS/RHRS] train in RHR mode (class F1A) or RIS/RRA [SIS/RHRS] train in SI mode (classed F1A) + Pressuriser relief valves open (classed F1A) if RCP [RCS] closed	All states C and D	One RIS/RRA [SIS/RHRS] train removes the pump plus residual heat with a hot leg temperature ~120°C Two RIS/RRA [SIS/RHRS] trains remove the pump plus residual heat with a hot leg temperature ~80°C One LHSI train removes the pump plus residual heat with an IRWST temperature ~120°C and a RCP [RCS] pressure ~containment pressure
EVU [CHRS] train (classed F2) + MHSI and pressuriser depressurisation system (classed F1A)	All states C and D	The two EVU [CHRS] trains remove the pump plus residual heat with an IRWST temperature ~120°C

14.1.4. Definition of the cases studied

All the initial conditions (states C and D) and the full break range up to the DN 250 double guillotine break must be covered by the analysis. Currently, no study is available whose assumptions strictly meet the requirements of the PCC-4 accident studies.

The results concerning the RCP [RCS] blow-down until break isolation are extrapolated from available studies [Ref-1]. The aim of these studies was to determine, for each initial state, the minimum break sizes for which break isolation occurs before the RIS/RRA [SIS/RHRS] automatic trip signal is generated. This analysis was performed to justify the acceptability of this accident for a specific study. Therefore, the study is not directly applicable for a PCC-4 analysis. The main difference in assumptions between those studies and the PCC-4 study is the injection to the RCP [RCS] with two MHSI trains instead of the one which must be assumed for the PCC study.

The analysis of later phases after the break isolation is mainly based on energy balances.

14.1.5. Description of the cases analysed, from the initiating event to the controlled state

14.1.5.1. Choice of Single Failure and Preventive Maintenance

The worst single failure for core uncovery is that of an MHSI train due to either an electrical board, pump, or valve failure.

In addition, it is conservatively assumed that a MHSI train is unavailable when the plant is functioning normally in states C and D.

14.1.5.2. Specific Assumptions

The assumptions are similar to those for the ‘Small break LOCA (< DN50, states C and D)’ described in section 7 of this sub-chapter, and also similar to those used for studies of the 2A-RHR break [Ref-1] (see section 4 of Sub-chapter 14.1), except for the number of MHSI trains injecting into the RCP [RCS].

14.1.5.2.1. Decay Heat

The maximum residual powers considered are: [Ref-1]

- 36 MW in State C corresponding to 10 hours after reactor trip
- 27.3 MW in State D corresponding to 30 hours after reactor trip

14.1.5.2.2. Assumptions Related to Non-F1 Systems

The minimum number of RIS/RRA [SIS/RHRS] trains initially in operation is conservatively assumed for each state:

- The affected train is assumed totally lost at the start of the event
- The RIS/RRA [SIS/RHRS] performance is held constant for the intact trains.

Minimum characteristics are assumed for the cooling system to minimise the heat removal.

14.1.5.2.3. Assumptions Related to F1 Systems

Only one MHSI pump is claimed due to the following assumptions:

- The single failure
- Unavailability of one train in normal operation for States C and D
- No credit for the MHSI pump injecting from the affected train.

The MHSI characteristics are as follows [Ref-1]:

- Maximum injection time: 25 seconds (preliminary value)

- Maximum injection characteristic with large mini-flow line open (injection at 40 bar, see Sub-chapter 14.1)

- Maximum temperature of the IRWST: 50°C

14.1.5.3. Results

14.1.5.3.1. Break Diameter < DN 50

Small RIS/RRA [SIS/RHRS] breaks are clearly covered by the analyses of reactor coolant system breaks smaller than or equal to DN 50 discussed in section 7 of this sub-chapter for reaching the controlled state and changes in the primary coolant inventory. This conclusion is justified as follows:

- The RIS/RRA [SIS/RHRS] break is isolated automatically, which allows the MHSI to replenish the primary coolant inventory.

- The break flow is less for a RIS/RRA [SIS/RHRS] break than in the same size primary break due to the pressure drop between the primary loop and the break.

For breaks up to DN 50, the hot leg remains sub-cooled. Thus, the RIS/RRA [SIS/RHRS] pumps in the intact trains continue to operate. The controlled state and the safe state are reached as soon as the break is isolated.

14.1.5.3.2. Break Diameter > DN 50

First phase: From initiating event to isolation of the break

a) State C1

Studies of the 2A-RHR break [Ref-1] show that the DN 250 break is isolated on a “high pressure in safeguard building” signal before the safety injection is started. Consequently, the results are independent of the number of MHSI trains assumed to operate. The results from these studies show that the hot legs remain full of water. The margin to uncovery of the core is therefore large.

The conditions in the hot legs can be at or near saturation following draining of the pressuriser, and before initiation of the MHSI.

b) States C2 and C3: Pressuriser level at zero (i.e. drained)

The results of studies of the 2A breaks in the RIS/RRA [SIS/RHRS] [Ref-1] are not directly applicable to these conditions. The calculations assume two MHSI pumps operate to provide the required injection. Nevertheless, they show that only a relatively small reduction in primary coolant inventory occurs in the case of a DN 250 break. This is a consequence of the rapid automatic isolation of the break. It is possible to conservatively estimate the water loss assuming no MHSI injection occurs. This can then be compared to the initial mass to ensure that the core uncovery is avoided and the hot legs remain full.

- The break is isolated after 120 seconds, as in the studies of the 2A break in the RIS/RRA [SIS/RHRS] [Ref-1]. This value is very conservative for DN 250, as it is based on the DN 50 break size.

- Initial water mass in the RCP [RCS] is 384 te. This is evaluated at 120°C and is a bounding value for state C. It includes 20 te in the pressuriser and 30 te in each SG tube bundle.
- Mass of water discharged through the break in 120 seconds is 60 te at 50°C, the bounding temperature for state C, as calculated in the studies of the 2A break in the RIS/RRA [SIS/RHRS] [Ref-1].

Consequently, the water mass lost through the break is small compared with the initial inventory of the RCP [RCS]. Core uncovery is avoided, and the hot legs remain full. Only the upper parts of the RCP [RCS] drain via the break.

The conditions in the hot legs can be near or equal to saturation following draining the pressuriser, and before initiation of the MHSI.

c) State C3 (3/4 loop)

In this state, the initial primary coolant inventory is significantly less than in the states discussed above, and the thermal-hydraulic behaviour during the draining is different: The primary pressure is initially 1 bar. Therefore, the draining is solely due to gravity. Consequently, the DN 250 break cannot drain the reactor vessel below the bottom of the hot legs.

To ensure that the core stays covered, it is sufficient to show that there is no steam production that could raise the pressure or lower the water inventory before the break is isolated.

To evaluate this, the water volume of the reactor vessel alone is considered. The thermal inertia of the water volume contained in the core and the upper plenum, up to the bottom of the hot legs, is enough to absorb the decay heat without boiling until isolation of the break. This assumes the isolation is completed within 120 seconds. This is a conservative value based on the delay in detection of a DN 50 break.

Consequently, there is no risk of bulk boiling in the reactor vessel before the break is isolated, and the reduction in primary water inventory is insufficient to uncover the core.

d) State D with low water level in the reactor cavity

In this state, three RIS/RRA [SIS/RHRS] trains are initially in service. The remaining train is either isolated from the RCP [RCS] on standby and available, or being used in LHSI mode to refill the reactor cavity. In the latter case, the LHSI pump takes suction from the IRWST and cannot be affected by an automatic RIS/RRA [SIS/RHRS] pump trip signal in the intact trains.

The break is assumed to occur in an operating train in residual heat removal mode. A break in the train in LHSI mode would be immediately isolated by the closure of the primary coolant check valves and would have no effect on continued residual heat removal. This would continue to be provided by the three operating RIS/RRA [SIS/RHRS] trains.

In this state, the worst case occurs for an initial primary water level at 3/4 loop.

As in state C at 3/4 loop, the thermal capacity of the water contained in the core and upper plenum up to the bottom of the hot legs is sufficient to absorb the decay heat without boiling until isolation of the DN 250 break. The break isolation is assumed to be complete by 120 seconds. This delay is conservatively based on detection of a DN 50 break. In addition, the RIS/RRA [SIS/RHRS] train in LHSI mode contributes to limiting the drop in primary water level if it is initially in service.

Consequently, there is no risk of bulk boiling in the reactor vessel before the break is isolated, and the loss of primary coolant inventory is insufficient to uncover the core.

e) State D with high water level in the reactor cavity

In this state, three RIS/RRA [SIS/RHRS] trains are initially in service. The remaining train is either isolated from the RCP [RCS] on standby and available, or is being used in LHSI mode to refill the reactor cavity.

As discussed above, the DN 250 break is assumed to occur on a train operating in residual heat removal mode.

The water inventory in the reactor cavity is sufficient for the break to be isolated without significantly degrading the primary coolant inventory and without initiating an automatic trip signal to the RIS/RRA [SIS/RHRS] pumps of the intact trains.

<u>Second phase: from break isolation to reaching a controlled state</u>

a) States C1 to C3

After isolating the break, the primary coolant system and the intact RIS/RRA [SIS/RHRS] trains are pressurised to the shut-off head of the MHSI with the high-flow recirculation line open, about 40 bar.

It is conservatively assumed that the automatic trip signal for the pumps in the intact RIS/RRA [SIS/RHRS] trains has been initiated, either for conditions near hot leg saturation (states C1 and C2) or at very low loop level (state C3). This is particularly conservative considering the delay in isolating the break.

The decay heat removal is by the VDA [MSRT] on the two available SG, with the ASG [EFWS] supplying feedwater. The secondary pressure stabilises at about 5 bar, which corresponds to a primary temperature of around 150°C.

This controlled state at 40 bar / 150°C is safe with the break flow halted, the primary coolant inventory restored and redundant heat removal available. This state can be maintained for as long as water is available in the ASG [EFWS] tanks. This is about 24 hours assuming only two ASG [EFWS] tanks are available, or more than 48 hours with four ASG [EFWS] tanks. These times claim no credit for possible replenishment of the tanks. This state is maintained using only F1 systems.

b) State D with low water level in the reactor cavity

After isolating the break, the primary water inventory is restored by the MHSI and potentially also by the LHSI train initially in service.

The decay heat is removed, either by the RIS/RRA [SIS/RHRS] if it is still in service or by the MHSI and potentially by an LHSI train initially in service with the reactor vessel overflowing into the sumps, and heat being rejected to the IRWST.

c) State D with high water level in the reactor cavity

Isolating the break allows the primary system water inventory to be adequately maintained.

The heat is removed by at least one RIS/RRA [SIS/RHRS] train still in service after the break has been isolated.

14.1.5.4. Conclusion

The break leads to a loss of water inventory from the primary system. This is rapidly halted by automatic isolation of the break following the deterioration of the conditions in the safeguard building concerned. The break can potentially lead to draining of the primary loops. However, a single MHSI pump is sufficient to ensure the core remains covered. Consequently all the LOCA criteria relating to core integrity are met.

The reactor then stabilises in the following controlled states:

- State C:
  - The water inventory in the primary system is restored by the MHSI.
  - The decay heat is removed, either by the RIS/RRA [SIS/RHRS] if it is still in service, or by the steam generators.
- State D (low cavity level):
  - The water inventory in the primary system is restored by the MHSI and potentially by an LHSI train initially in service.
  - The decay heat is removed, either by the RIS/RRA [SIS/RHRS] if it is still in service, or by heating the IRWST. Heat is transferred to the IRWST via the reactor vessel overflowing due to SI delivery from the MHSI train and potentially by an LHSI train initially in service.
- State D (high cavity level):
  - The water inventory is maintained by the automatic isolation of the break.
  - The decay heat is removed by the RIS/RRA [SIS/RHRS].

14.1.6. Description of the Cases Analysed, from the Controlled State to the Safe State

If at least one RIS/RRA [SIS/RHRS] intact division train is still in service after the isolation of the break, i.e. no automatic trip of the RIS/RRA [SIS/RHRS] pumps occurs, the safe state is reached. In these conditions, the RIS/RRI/SEC [SIS/CCWS/ESWS] cooling chain removes the decay heat in the long-term.

If all the intact RIS/RRA [SIS/RHRS] trains are automatically tripped, manual start-up of the pumps initially in standby and/or the pumps stopped automatically on adverse conditions, allows continued operation. A single failure must therefore be assumed for assessing the return of the available trains to service.

14.1.6.1. Break Diameter < DN 50

For breaks of this size, the intact RIS/RRA [SIS/RHRS] trains remain in service after the break has been isolated and the safe state is reached immediately.

14.1.6.2. Break Diameter > DN 50

14.1.6.2.1. State C

In the analysis below, the conservative scenario is considered of an automatic RIS/RRA [SIS/RHRS] pump trip in the trains initially in service. However, it is important to emphasise that for the smallest breaks in this range; the intact RIS/RRA [SIS/RHRS] trains would be expected remain in service. Thus the safe state would be reached immediately.

Controlled state:

- The water inventory in the primary system is restored by the MHSI.
- The residual heat is removed by the steam generators.

After reaching a controlled state, the operator can perform the following actions:

- Fully open the steam dump valve (VDA [MSRT]) to reduce the primary temperature to about 130°C. This is an F1 action.
- Manually stop the MHSI pumps. This is an F1 action.
- Depressurise the primary system to between 5 and 10 bar by opening the pressuriser relief valves. This is an F1 action.

This new state at 5-10 bar and ~130°C can be reached within one hour after automatic isolation of the break, including the 30 minute operator delay. This state can be maintained for at least 24 hours using only F1 systems.

State C1

In state C1, two initially isolated RIS/RRA [SIS/RHRS] trains are on standby and available for start in an accident situation. Once the water inventory of the primary system has been restored by the MHSI, these trains can be started in residual heat removal mode. This allows long-term decay heat removal to be provided, while also meeting the single failure criterion as only one train is required. This action must be performed within about 24 hours of the break occurring. The safe state is thus reached using only F1 systems.

State C2

In state C2, the four RIS/RRA [SIS/RHRS] trains are initially in service. For the worst initial conditions of a primary temperature of 100°C and reactor coolant pump trip assumptions, it is possible that the conditions reached in the hot leg are near or equal to saturation. This would lead to initiation of an automatic trip of the RIS/RRA [SIS/RHRS] pumps in the intact trains on a ΔPsat signal.

However, the break is isolated quickly enough to keep the hot legs water-solid and maintain the thermal-hydraulic conditions in the RIS/RRA [SIS/RHRS] pipe. This allows the subsequent restart of the RIS/RRA [SIS/RHRS] pumps. It should be noted that the RIS/RRA [SIS/RHRS] pumps are located about 15 m below the level of the hot legs. Therefore temporary saturation in the hot leg will not result in pump damage.

In addition, the RIS/RRA [SIS/RHRS] pumps can be restarted once the MHSI has re-established appropriate conditions in the primary system. Bringing these three trains into service in residual heat removal mode provides long-term decay heat removal, while also meeting the single failure criterion as only one train is required. This action is required within 24 hours of the break.

The safe state is thus reached using only F1 systems.

State C3

In state C3, three RIS/RRA [SIS/RHRS] trains are initially in service, the remaining train being isolated from the RCP [RCS] on standby and available for operation. It is conservatively assumed that the three pumps in service are lost through damaged from adverse suction conditions. This is despite the initiation of the automatic pump trip signal on very low loop level. Starting the train initially on standby provides the long-term decay heat removal. However, applying the single failure criterion while starting this train requires the following actions in order to reach a safe state if the train is assumed to fail:

- Manually initiate injection via the MHSI and relief via the pressuriser relief valves to remove RCP [RCS] decay heat.
- Manually bring into service the two EVU [CHRS] trains to remove the heat in the containment and maintain the temperature of the IRWST water below 120°C. This is required to provide adequate operating conditions for the MHSI pumps.

A combination of F1 and F2 systems are therefore required to reach a safe state following a break outside the containment in state C3.

To assure a safe state in the long term, it is necessary to subsequently restart at least one RIS/RRA [SIS/RHRS] train by actions or methods that are not F1 classified.

14.1.6.2.2. State D

State D with low water level in the reactor cavity (3/4 loop)

Controlled state:

- The water inventory in the primary system is restored by the MHSI and potentially also by an LHSI train initially in service.
- The decay heat is removed, either by the RIS/RRA [SIS/RHRS], if it is still in service, or by heating the IRWST. Heat is transferred to the IRWST via the reactor vessel overflowing due to SI delivery from the MHSI, and potentially by an LHSI train initially in service.

As in the state C3 case above, it is conservatively assumed that the three RIS/RRA [SIS/RHRS] pumps in service are unavailable despite the initiation of the automatic pump trip signal on very low loop level. Two cases are considered:

a) An RIS/RRA [SIS/RHRS] train is initially in service in LHSI mode.

The LHSI train still in service can be used for long-term RCP [RCS] decay heat removal. The controlled state is reached.

b) An RIS/RRA [SIS/RHRS] train is initially isolated.

The decay heat is transferred to the containment, which requires the use, in due course, of a system to cool the IRWST. This is initially performed by the LHSI.

The operator therefore has several hours to carry out these actions. The transfer of decay heat to the containment can be seen in terms of two main phases:

- Overall heating of the IRWST from 50°C to100°C without steam production
- Steam production and a rise in containment pressure.

The time to warm IRWST water to 100°C is about 4 hours. This is based on simplified and conservative assumptions that there are 1830 te of water in the IRWST and 550 te of steel in the lower part of the vessel [Ref-1].

The decay heat can be removed by an initial steam flow of about 12 kg/s. A calculation of the change in containment pressure and temperature shows that the pressure rise is close to 0.225 bar/hour and that the IRWST temperature remains below 120°C for 24 hours. [Ref-1]

Bringing the initially isolated train into service in LHSI mode rapidly halts the steam flow and reduces the containment pressure. The IRWST is cooled by the RRI [CCWS]. However, when the single failure criterion is applied, bringing this train into service requires bringing into service two EVU [CHRS] trains to cool the IRWST. This action is necessary to prevent the IRWST temperature exceeding 120°C. This ensures that conditions remain suitable for normal MHSI pump operation.

Hence, attaining a safe state following a break outside the containment in state D with low water level in the reactor cavity can require a combination of F1 and F2 protection while the LHSI train used for replenishing the pool is not in service.

To assure a safe state in the long term, it is necessary to restart at least one RIS/RRA [SIS/RHRS] train by actions or means that are not required to be F1 classified.

State D with high water level in the reactor cavity

Controlled state:

- The primary water inventory is maintained by isolating the break
- Decay heat is removed by the two RIS/RRA [SIS/RHRS] trains still in service after the break has been isolated.

In this state, the water inventory in the reactor cavity is sufficient for the break to be isolated without degrading the primary coolant inventory. Consequently, the automatic RIS/RRA [SIS/RHRS] pump trip signal in the intact trains is not initiated.

The two trains used in residual heat removal mode that are still available after the break has been isolated allow long-term decay heat removal. The safe state is thus reached immediately, using only F1 protection.

14.1.6.3. Conclusion

Following the intervention of the protection systems, the reactor reaches a controlled state. If at least one RIS/RRA [SIS/RHRS] pump is still in service, i.e. there is no automatic trip signal in the intact RIS/RRA [SIS/RHRS] trains, this controlled state is also a safe state. In these conditions, long term decay heat removal is provided. This is the case for:

- Small breaks, in particular < DN 50, for all initial conditions
- Breaks up to a double ended guillotine break of an RIS/RRA [SIS/RHRS] line for state D with a high water level in the reactor cavity.

Conversely, for larger breaks in states C, and in state D with low water level in the reactor cavity, an automatic RIS/RRA [SIS/RHRS] pump trip signal in the intact trains could be initiated. If such a signal is generated, the controlled state is reached:

- In state C1, by starting the intact RIS/RRA [SIS/RHRS] train, initially on standby
- In state C2, by starting the intact RIS/RRA [SIS/RHRS] initially running but stopped automatically
- In state C3 and in state D with low water level in the reactor cavity, by bringing into service the F2 protection (EVU [CHRS]) in addition to F1. The controlled state previously reached allows a significant delay in implementing these actions.

When a safe state is reached using F2 protection (EVU [CHRS]), an RIS/RRA [SIS/RHRS] train must be brought into service to maintain a long-term safe state. The exact methodology for implementing these actions is being reviewed but F1 classification is not required.

The means available to reach a safe state from each initial reactor state, with the conservative assumption that the automatic signal to trip the intact RIS/RRA [SIS/RHRS] trains has been initiated, are summarised in Section 14.5.14 - Figure 1. [Ref-1]

14.2. ISOLATABLE BREAK INSIDE THE CONTAINMENT

The following analysis deals with the particular case of isolatable breaks in the RIS/RRA [SIS/RHRS] inside the containment. In practice, this study bounds all isolatable breaks inside the containment in systems connected to the RCP [RCS], noting that:

- The RIS/RRA [SIS/RHRS] pipe work includes the larger lines connected to the RCP [RCS].
- A RIS/RRA [SIS/RHRS] break directly affects the system that provides decay heat removal, leading to a greater heat-up of the primary system.

The main differences for RIS/RRA [SIS/RHRS] breaks inside the containment compared to equivalent breaks outside the containment are:

- Transfer of part or all of the decay heat directly into the containment, as well as part of the initial mass and energy inventory of the RCP [RCS]

- Absence of an automatic isolation signal for the affected train, and thus the automatic isolation of the break. Isolation takes place following operator action, assumed to occur 30 minutes after the first significant alarm. This is assumed to be the safety injection signal.

The criteria for safety, initial states, methods and assumptions of the study, sequence of main events, and systems necessary to reach controlled and safe states are identical to those described in section 14.1 of this sub-chapter. The key difference in the development of the accident transient is the additional discharge from the RCP [RCS] due to the later isolation of the break.

The increased discharge from the RCP [RCS]:

- Has no impact on core cooling as the core remains covered due to the injection from at least one MHSI pump. This has sufficient capacity for decay heat removal

- Can cause difficulties in restarting the automatically tripped LHSI pumps. This will depend on how the various protection setpoints are calibrated.

As with breaks outside the containment, the safe state relies primarily on starting the RIS/RRA [SIS/RHRS] trains which are on standby when the fault occurs. They have not therefore been affected by the temporary reduction in the primary coolant inventory. Difficulties can arise in restarting the automatically tripped RIS/RRA [SIS/RHRS] trains in the C2 state where all the category F1 energy removal systems, the four RIS/RRA [SIS/RHRS] trains, are initially in service. The requirement to restart at least one RIS/RRA [SIS/RHRS] train dictates the requirements for the automatic RIS/RRA [SIS/RHRS] pump trip signal in state C2 on a very low ∆Psat signal:

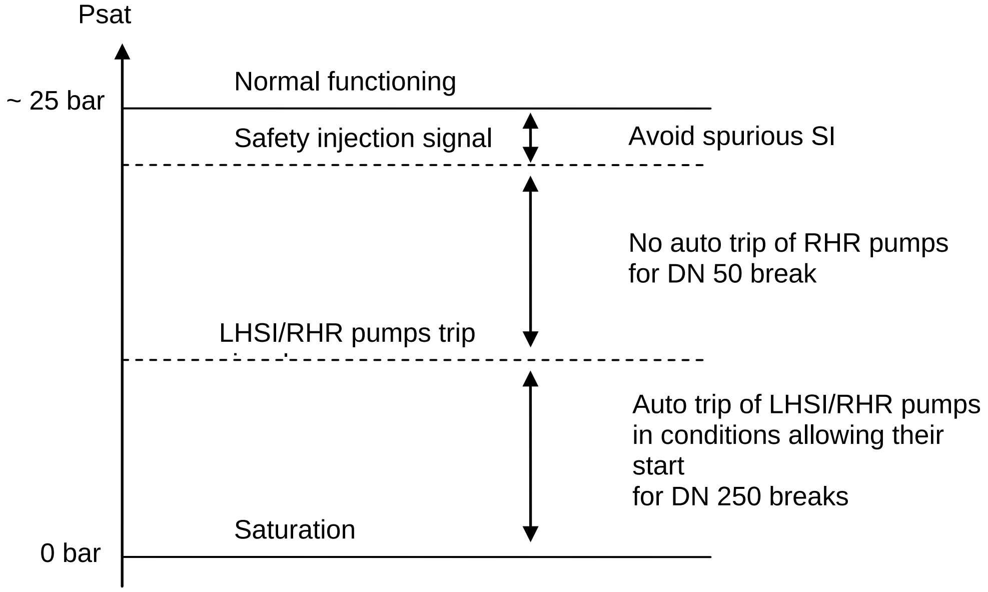
- For breaks outside the containment, the requirement is limited to stopping the pumps before they sustain damage due to cavitation. In state C2, the very rapid isolation of the break limits hot leg draining and ensures the pumps are capable of being restarted.

- For breaks inside the containment, the break is isolated later. Consequently, the setting of the ”very low ∆Psat” signal must ensure that adequate thermodynamic conditions are maintained within the RIS [SIS] system. These are required to ensure the LHSI pumps are capable of being restarted without local action on non-classified systems.

This change in requirements involves raising the automatic trip setpoint of the RIS/RRA [SIS/RHRS] pumps on “very low ∆Psat” to above the setpoint required if only RIS/RRA [SIS/RHRS] breaks outside the containment were considered. Raising the automatic “very low ∆Psat” trip setpoint for the RIS/RRA [SIS/RHRS] pumps requires a comparable increase in the IS [SI] signal on ”low ∆Psat”. The gap between the two setpoints allows the MHSI to operate and prevent an automatic RIS/RRA [SIS/RHRS] pump trip for breaks < DN 50. Avoiding tripping a heat removal function normally in service for the most frequent breaks, when it is unnecessary, is a safety benefit.

The resulting margin between the SI threshold and the normal operating range must be sufficient to reduce the possibility of spuriously initiating the MHSI pumps to an acceptable level.

The diagram below summarises the respective setpoint requirements: [Ref-1]



The setpoints for the safety injection and automatic RIS/RRA [SIS/RHRS] trip signals required for RIS/RRA [SIS/RHRS] breaks in the containment are not yet available.

14.3. SYSTEM SIZING

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

SECTION 14.5.14 - TABLE 1 [Ref-1]

RIS/RRA [SIS/RHRS] Breaks outside the Containment – Systems Available for Long-Term Heat Removal (Attainment of Safe State)

Initial State of the Reactor	**C1**	**C2**	**C3**	**D (low water level in the reactor pool)**		**D (high water level in the reactor pool)**
Initial state of the RIS [SIS]/RRA [RHR] trains	- 2 RIS/RRA [SIS/RHRS] trains in service -2 RIS/RRA [SIS/RHRS] trains on standby	- 4 RIS/RRA [SIS/RHRS] trains in service	- 3 RIS/RRA [SIS/RHRS] trains in service - 1 RIS/RRA [SIS/RHRS] train on standby	- 3 RIS/RRA [SIS/RHRS] trains in service - 1 LHSI train on standby	- 3 RIS/RRA [SIS/RHRS] trains in service - 1 LHSI train in service	- 3 RIS/RRA [SIS/RHRS] trains in service - 1 LHSI train in service/ on standby
Means available for long-term decay heat removal before applying the single failure criterion	-2 RIS/RRA [SIS/RHRS] trains initially on standby	- 3 RIS/RRA [SIS/RHRS] trains stopped	- 1 RIS/RRA [SIS/RHRS] train initially on standby - 2 EVU [CHRS] trains and pressure relief (MHSI+pressuriser relief valves)	- 1 LHSI train initially on standby - 2 EVU [CHRS] trains and pressure relief (MHSI+RCP [RCS] open)	- 1 LHSI train initially in service	- 2 RIS/RRA [SIS/RHRS] trains initially in service - 1 LHSI train initially in service/ on standby
Ways of bringing these systems into service	- Manual start (24hr delay)	- Manual restart (24hr delay)	- Manual start (24hr delay)	- Manual start - Manual start	- Already in service	- Already in service - Already in service / manual start
Means available for long-term decay heat removal after applying the single failure criterion	- 1 RIS/RRA [SIS/RHRS] train	- 2 RIS/RRA [SIS/RHRS] trains	- 2 EVU [CHRS] trains and pressure relief (MHSI+ pressuriser relief valves)	- 2 EVU [CHRS] trains and pressure relief (MHSI+ RCP [RCS] open)	- 1 LHSI train initially in service	-2 RIS/RRA [SIS/RHRS] trains initially in service
Functional safety classification of the systems used to reach a safe state	F1	F1	F1+F2	F1+F2	F1	F1

15. NON ISOLATABLE SMALL BREAK (< 50 MM) OR ISOLATABLE RIS [SIS] BREAK (< 250 MM) IN RHR MODE: SPENT FUEL POOL DRAINAGE ASPECTS (STATE E)

15.1. INTRODUCTION

A piping failure may occur on:

- An RIS [SIS] line (DN 250) in Residual Heat Removal (RHR) mode.
- A line (DN < 50 mm) connected to the primary cooling system.

The transfer tube is open in state E, and consequently such piping failures may affect the fuel pool cooling function.

An overview of Reactor Building and Fuel Building pools is given in Section 9.1.3 - Figure 1.

The methodology used for the analyses given in this chapter is described in Sub-chapter 14.3.

The upper end of a fuel assembly being handled is at a level of +16.20 m and that of an assembly stored in the spent fuel pool rack is at a level of +10.30 m. [Ref-1][1] [Ref-2]

15.2. PRE-ACCIDENT CONDITIONS (NORMAL OPERATION – PCC-1)

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 corresponds to a decay heat of 20.81 MW and occurs approximately 111 hours after shutdown [Ref-1].

The initial fuel pool water temperature is assumed to be 50°C. This value covers all potential operating states.

15.3. GRACE PERIOD

For PCC events involving fuel pool drainage, the grace period is calculated assuming a reduced fuel pool water volume of 1348 m^3. This corresponds to a water level of +17.90 m, reached after the RIS [SIS]/RHR suction valves have been automatically isolated or the MHSI pumps have been started.

In this case, the fuel pool water temperature will reach 97°C in 3.4 hours after a loss of the cooling function, assuming a decay heat of 20.81 MW [Ref-1].

[1] This analysis is a Flamanville 3 (FA3) study that takes into account some specific features of the FA3 design but the results are bounding for the UK design. In particular, the analysis considers that the third PTR [FPCS] train is lower than for the UK design. Therefore, the calculations are conservative.

The water volume considered in the grace period calculations is conservatively assumed to consist only of the spent fuel pool and does not include any make-up water. In addition, the analysis conservatively ignores any positive benefit from the reduced temperature of the make-up water.

15.4. BOUNDARY CONDITIONS

As for all other PCC events, the transient is analysed using conservative assumptions. Therefore, the assumed fuel pool heat load includes a safety margin.

Single failure and preventive maintenance are considered in combination with the initiating event.

As required by the general study rules, a loss of offsite power (LOOP) following an earthquake is considered in the analyses, but without additional failures as discussed in Sub-chapter 14.0.

In state E, maintenance is not allowed on the PTR [FPCS] cooling trains but limited maintenance may be performed on supporting systems. For example, maintenance may be performed on one electrical division in state E. Adequate cross-connections are installed prior to maintenance on a supporting system to ensure the system availability.

15.5. DECOUPLING CRITERIA

For PCC events involving fuel pool drainage, the decoupling criterion for the PTR [FPCS] design is to avoid pool boiling throughout the transient. Therefore, the studies aim to show that the fuel pool water temperature remains below a maximum temperature of 97°C.

In the long term, following restoration of a main PTR [FPCS] train, the water temperature in the fuel pool must not exceed 80°C as discussed in Sub-chapter 14.0.

15.6. TRANSIENTS

All the systems and equipment studied in the analysis below are robust to single failure and can be supplied by alternative means in the event of loss of their main electrical supplies. The application of the single failure criterion does not impact on systems already in operation.

A LOOP is assumed to occur coincident with the start of the drainage initiating event. Although power is restored to the main PTR [FPCS] trains automatically upon start-up of the emergency diesel generators, it is conservatively assumed that the PTR [FPCS] trains will have to be restarted manually, leaving the pool without cooling for one hour. Fuel pool drainage leads to an automatic shutdown of the PTR [FPCS] pumps at a water level of +18.00 m. Therefore, in the analysis, the one hour loss of cooling due to LOOP is added to the duration of loss of cooling due to low water level. An initial level of 19.00m is therefore conservatively assumed, to maximise the drainage duration and the duration of loss of cooling linked to LOOP. In addition, the analysis conservatively ignores the beneficial cooling effect once the PTR [FPCS] trains have restarted.

15.6.1. Isolatable break on a RIS [SIS]/RHR line (< 250 mm)

15.6.1.1. Introduction

This sub-section describes the analysis of a break on a RIS [SIS] line located inside the Reactor Building whilst the reactor is operating in RHR mode (state E) (see note 2 below).

Different drainage flow values were analysed from a set of values ranging between 50 and 800 m^3/h. The most limiting value of the drainage flow in terms of the duration of loss of cooling is 50 m^3/h and this bounding case is described below.

Note 1: breaks located upstream of the two isolation valves are not taken into account (see Chapter 9)

Note 2: for a piping failure located outside the reactor building, RIS [SIS]/RHR lines are assumed to be automatically isolated by the actuation of a high level signal in the relevant safeguard building sumps. The discharge is therefore limited to approximately 60 m^3. This corresponds to a level drop of 23 cm in the fuel pool with the transfer tube open, which does not lead to a loss of the fuel pool cooling function.

Note 3: The limiting flow rate is not the maximum flow rate: breaks on a RIS [SIS]/RHR train with a drainage flow rate greater than 800 m^3/h are not studied.

15.6.1.2. Transient analysis

Following detection of the leakage, one of the required actions is the closure of the transfer tube valve to isolate the fuel building pool from the reactor building pool. The subsequent transient progression differs depending on whether the transfer tube is successfully closed. The bounding situation is failure to close the transfer tube, as this maximises the duration of loss of cooling.

The resulting fuel pool drainage leads to an automatic shutdown of the PTR [FPCS] pumps at a water level of +18.00 m and therefore to a loss of the fuel pool cooling function [Ref-1] [Ref-2][2].

Between water levels of +19.00 m and +18.00 m, loss of cooling due to LOOP is considered: the duration of loss of cooling is one hour in this case, corresponding to the maximum time for manually restarting a PTR [FPCS] train.

The drainage through the RIS [SIS]/RHR suction line is automatically isolated by the closure of two redundant motorised valves. This is initiated following detection of a low water level in the reactor building transfer compartment at a setpoint of +17.90 m. Check valves prevent drainage through the RIS [SIS]/RHR discharge lines.

Therefore, the controlled state with the leakage stopped is reached automatically.

Water make-up is then performed using the Classified Fire Fighting Water Supply System (JAC/JPI [NIFPS]) at a flow rate of approximately 150 m^3/h. This raises the fuel pool water level to +18.05 m, which is sufficient to allow starting of a main PTR [FPCS] train.

The water make-up volume required to raise the fuel pool water level from +17.90 m to +18.05 m is approximately 41 m^3 in state E.

[2] This analysis is a Flamanville 3 (FA3) study that takes into account some specific features of the FA3 design but the results are bounding for the UK design. In particular, the analysis considers that the third PTR [FPCS] train is lower than for the UK design. Therefore, the calculations are conservative.

The most limiting drainage situation leads to a loss of cooling for a total of 1.85 hours, which corresponds to:

- The loss of cooling due to LOOP = 1 hour.
- The drainage time from +18.00 m to +17.90 m = 0.55 hours.
- The JPI [NIFPS] make-up time = 0.3 hours.

The average fuel pool water temperature does not exceed 75°C during the transient. The calculation assumes a heat load of 20.81 MW, which includes a safety margin. In addition, the calculation of water temperature conservatively considers only the volume of the spent fuel pool at a water level of +17.90 m and ignores the additional make-up water. The safe shutdown state is therefore reached and the long term fuel pool water temperature is stabilised at 65°C with one main PTR [FPCS] cooling train in operation, assuming an RRI [CCWS] design temperature of 38°C.

15.6.2. Non isolatable break (< 50 mm) on a line connected to the primary cooling system

15.6.2.1. Introduction

This sub-section describes the analysis of a non-isolatable break on a line connected to the primary cooling system in state E.

Leakage flow rates between 30 m^3/hour and 120 m^3/hour are considered in this study. The most limiting value of the drainage flow in terms of the duration of loss of cooling is 120 m^3/h and this bounding case is described below [Ref-1][3].

Since the break is, by definition, not isolatable, the resulting inventory loss cannot be manually or automatically stopped. It is therefore necessary to compensate for the leakage with a permanent source of water make-up. The make-up must first stabilise the water level in the pools to reach the controlled state, and then restore the water level in the spent fuel pool to a level sufficient to allow starting of a main PTR [FPCS] train so as to reach the safe shutdown state.

During the transient, once the capacity of the floor sumps (approximately 200 m^3) is exceeded, the water lost is collected in the In-Containment Refuelling Water Storage Tank (IRWST). This is a consequence of the reactor building design [Ref-2].

Water make-up is performed using the MHSI pumps in recirculation mode between the IRWST and the primary cooling system. Make-up can be performed once the reactor compartment overflow line and the instrumentation lance compartment draining line, toward the IRWST, have been opened.

MHSI injection flow rates between 146 m^3/hour and 210 m^3/hour are assumed in the study.

To prevent any loss of water outside the reactor building, floor drain exhaust lines 1 and 2 are automatically isolated when the spent fuel pool water level drops to +18.90 m in state E [Ref-3].

[3] This analysis is a Flamanville 3 (FA3) study that takes into account some specific features of the FA3 design but the results are bounding for the UK design. In particular, the analysis considers that the third PTR [FPCS] train is lower than for the UK design. Therefore, the calculations are conservative.

15.6.2.2. Transient analysis

Following detection of the drainage, one of the required actions is the closure of the transfer tube valve to isolate the fuel building pool from the reactor building pool. The bounding situation is failure to close the transfer tube, as it maximises the duration of the loss of cooling.

Between water levels of +19.00 m and +18.00 m, loss of cooling due to LOOP is considered. The duration of loss of cooling is one hour in this case, corresponding to the maximum time assumed for manually restarting a PTR [FPCS] train.

The first significant alarm is actuated when the pool water level drops to 18.90 m [Ref-1]. More than 30 minutes later, a low water level of +18.40 m is detected in the spent fuel pool and the instrumentation lance compartment draining line toward the IRWST is opened from the main control room.

Continuing fuel pool drainage leads to an automatic shutdown of the PTR [FPCS] pumps at a water level of +18.00 m, and therefore to a loss of the fuel pool cooling function.

At 75 minutes after initiating drainage of the instrumentation lance compartment (the time period considered for complete drainage of the lance compartment), the overflow line is opened and a MHSI pump is started from the main control room. The water level in the reactor building pool is thus restored and stabilised, and the controlled state is therefore reached. At this point, the level in the spent fuel pool is +17.85 m.

Water make-up is also undertaken using the Classified Fire Fighting Water Supply System (JAC/JPI [NIFPS]) at a flow rate of approximately 150 m^3/h.

The safe shutdown state is finally reached when the spent fuel pool water level has risen to + 18.05 m, which is sufficient to allow starting of a main PTR [FPCS] train.

The water make-up volume required to raise the fuel pool water level from +17.85 m to +18.05m to allow start-up of a main PTR [FPCS] cooling train, is approximately 56 m^3 in state E.

The minimum water level in the IRWST is 0.5 m and its value at equilibrium (MHSI injection, break flow rate + drainage via overflow) is 0.55m.

The most limiting drainage situation (drainage rate of 120 m^3/h) leads to a loss of cooling of 1.7 hours duration, which corresponds to:

- A loss of cooling due to LOOP = 1 hour
- A drainage time from +18.00 m to +17.85 m = 0.35 hours
- A MHSI make-up time to level of +18.05 m = 0.32 hours
  (MHSI flow rate = 146 m^3/h + JAC/JPI [NIFPS] flow rate = 150 m^3/h).

The water temperature in the fuel pool does not exceed 73°C during the transient. This assumes a conservative heat load of 20.81 MW, which includes a safety margin. In addition, it is conservatively assumed that only the volume of water in the spent fuel pool up to a level of +17.85 m contributes to the available heat sink. The safe shutdown state is therefore reached and the long-term fuel pool water temperature is stabilised at 65°C, with one main PTR [FPCS] cooling train in operation, assuming a RRI [CCWS] design temperature of 38°C.

15.7. CONCLUSIONS

In all cases, the water temperature in the fuel pool remains below 97°C throughout the transient, and below 80°C in the long term after a PTR [FPCS] train has been restored. Therefore, the decoupling criteria for PCC events involving fuel pool drainage are fulfilled for the transient "non-isolatable small break (< 50 mm) or isolatable RIS [SIS] break (< 250 mm) in RHR mode, spent fuel pool drainage aspects (state E)".

The main assumptions and results of this study are summarised in Section 14.5.15 - Table 1 below.

15.8. EFFECT OF REACTOR EVENT ON THE FUEL POOL TEMPERATURE

In addition to considering the PTR [FPCS], analysis is performed to demonstrate that PCC-3 or PCC-4 events affecting the reactor will not have a major effect on the fuel pool temperature.

This is demonstrated by calculating the pool temperature in steady state conditions, making the following assumptions:

- normal operation of the PTR [FPCS] cooling trains,
- different PTR [FPCS] configurations: beginning of cycle, end of cycle and end of refuelling, with either one or two main trains in service,
- maximum decay heat (MOX fuel management) with and without safety margin,
- PTR [FPCS] pool water volume for normal operation of 1463 m^3,
- RRI [CCWS] temperature corresponding to a decoupling value 45°C, which is representative of the maximum temperature that may arise in the RRI [CCWS] during a PCC-3 or PCC-4 transient affecting the reactor core.

The maximum fuel pool temperature calculated is 59°C, based on a decay heat of 5.85 MW (maximum including safety margin) and only one main PTR [FPCS] train in operation. This result demonstrates that PCC-3 and PCC-4 reactor events do not have a significant effect on the fuel pool temperature.

SECTION 14.5.15 - TABLE 1

NON ISOLATABLE SMALL BREAK (< 50 MM) OR ISOLATABLE RIS [SIS] BREAK (< 250 MM) IN RHR MODE, SPENT FUEL POOL DRAINAGE ASPECTS (STATE E)

Main assumptions and results (4500 MWth)

	End of refuelling	
	Isolatable RIS [SIS]/RHR break (< 250 mm)	Non isolatable small break (< 50 mm)
Decay heat (MW) (with safety margin)	20.81 [Ref-1]	
$T_{SEC\ [ESWS]}$ / $T_{RRI\ [CCWS]}$ / $T_{SRU\ [UCWS]}$ (°C)	30 / 38 / 30	
$T_{fuel\ pool}$ (initial) (°C)	50	
Fuel pool water volume (m^3)	1348	1342
Grace period without any cooling (h)		
To reach 97°C	3.4	
Transient duration (without any cooling) (hours)	1.9	1.7
Maximum $T_{fuel\ pool}$ without any cooling (°C)	75	73
$T_{fuel\ pool}$ in the long term (after restoration of the cooling function) (°C)	65	65

SUB-CHAPTER 14.5 – 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.

1. LONG-TERM LOSS OF OFFSITE POWER IN STATE C (>2 HOURS)

1.2. IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

[Ref-1] System Design Manual - Reactor Coolant System (RCP [RCS]) - Part 2: System Operation. NESS-F DC 538 Revision A. AREVA. May 2009. (E)

1.2.1. Relevant Cases and event mitigation

[Ref-1] EPR FA3 Preliminary Safety Analysis Report, Section 15.2.4a. “Loss of external power supplies (> 2 hours) (State C)”. Edition 2006. EDF. (E)

2. MAIN STEAM LINE BREAK

2.1. MAIN STEAM LINE BREAK (STATE A)

2.1.3. Methods and assumptions

2.1.3.2. Specific assumptions

2.1.3.2.1 Neutronic data and decay heat

2.1.3.2.1.1 At hot shutdown

[Ref-1] J-B Decaudin. Main Steam Line Break. PEPR-F DC 3 Revision A. AREVA. May 2010. (E)

2.1.3.2.5. Assumptions related to F1 systems

2.1.3.2.5.1. VIV [MSIV] (F1A)

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

2.1.3.2.5.2. ARE [MFW] high-load line isolation (F1A)

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

2.1.3.2.5.3. ARE [MFW] low-load line isolation (F1A)

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

2.1.3.2.5.4. MHSI (F1A)

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

2.1.3.2.5.5. ASG [EFWS] (F1A)

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

2.1.3.2.6. Other assumptions

2.1.3.2.6.2. Break flow (or stuck-open valve flow) correlation:

[Ref-1] D Venturi. 79/19TE Steam generator secondary side pressure vessel details right and left SGs. NEEG-F DC 1210 Revision F. AREVA. April 2008. (E)

[Ref-2] M Petiard – H Lafitte. EPR Preliminary Safety Analysis Report, Section 15.2.4b.1-Steam system piping break (State A) Edition 2003. PSRR DC 13 Revision B. AREVA. November 2003. (E)

2.1 3.2.6.3. Other assumptions maximising core cooling:

[Ref-1] N. Goreaud. EPR – Lower Plenum Hydraulics Design of the Flow Distribution Device. EPRR DC 1651 Revision D. AREVA. December 2001. (E)

3. FEEDWATER SYSTEM PIPE BREAK

3.1. FEEDWATER SYSTEM PIPE BREAK IN STATE A

3.1.3. Methods and Assumptions

3.1.3.1. Method of analysis

[Ref-1] A Barbier. CATHARE 2 – Code synthetic qualification assessment. EPD DC 490 Revision C. AREVA. September 2007. (E)

3.1.4. Definition of Cases Studied

[Ref-1] L Foucart. EPR Preliminary Safety Analysis Report, Section 15.2.4c.2 Feedwater System Piping Failure (State B) Edition 2003. EPRRDC 1706. AREVA. December 2003. (E)

3.1.5. Description of Studied Cases 1 and 2 (from the initiating event to the controlled state)

3.1.5.2. Initial state

[Ref-1] R Gagner. EPR™ operation at 4250MWth. EPRR DC 1701. Revision B. AREVA. February 2004. (E)

[Ref-2] J Feingold. EPR Preliminary Safety Analysis Report, Sub-chapter 15.1 “Plant characteristics assumed in the accident analyses”. Edition 2003. EPRR DC 1693 Revision C. AREVA. December 2003. (E)

3.1.5.3. Specific assumptions

c) Assumptions related to F1 systems

[Ref-1] R Gagner. EPR™ operation at 4250MWth. EPRR DC 1701. Revision B. AREVA. February 2004. (E)

[Ref-2] J Feingold. EPR Preliminary Safety Analysis Report, Sub-chapter 15.1 “Plant characteristics assumed in the accident analyses”. Edition 2003. EPRR DC 1693 Revision C. AREVA. December 2003. (E)

3.1.5.4. Results

3.1.5.4.1. Case 1: EPR$_{4250}$ accident analysis

[Ref-1] L Foucart. EPR Preliminary Safety Analysis Report, Section 15.2.4c.2 Feedwater System Piping Failure (State B) Edition 2003. EPRRDC 1706. AREVA. December 2003. (E)

3.2. FEEDWATER LINE BREAK IN STATE B

3.2.5. Description of cases studied (from the initiating event to the controlled state)

b) Initial state

[Ref-1] S Laurent. EPR – FA3 NSSS Operating Parameters. NFPSC DC 1042 Revision C. AREVA. December 2006. (E)

c) Result

[Ref-2] R Gagner. EPR™ operation at 4250MWth. EPRR DC 1701. Revision B. AREVA. February 2004. (E)

[Ref-3] J Feingold. EPR Preliminary Safety Analysis Report, Sub-chapter 15.1 “Plant characteristics assumed in the accident analyses”. Edition 2003. EPRR DC 1693 Revision C. AREVA. December 2003. (E)

SECTION 14.5.3 - TABLE 5

[Ref-1] R Gagner. EPR™ operation at 4250MWth. EPRR DC 1701. Revision B. AREVA. February 2004. (E)

[Ref-2] J Feingold. EPR Preliminary Safety Analysis Report, Sub-chapter 15.1 “Plant characteristics assumed in the accident analyses”. Edition 2003. EPRR DC 1693 Revision C. AREVA. December 2003. (E)

SECTION 14.5.3 - TABLE 6

[Ref-1] L Foucart. EPR Preliminary Safety Analysis Report, Section 15.2.4c.2 Feedwater System Piping Failure (State B) Edition 2003. EPRR DC 1706. AREVA. December 2003. (E)

SECTION 14.5.3 - FIGURES 3 TO 8

[Ref-1] L Foucart. EPR Preliminary Safety Analysis Report, Section 15.2.4c.2 Feedwater System Piping Failure (State B) Edition 2003. EPRR DC 1706. AREVA. December 2003. (E)

6. INTERMEDIATE AND LARGE BREAK LOCA (UP TO SURGE LINE BREAKS – STATES A AND B)

6.1. INTERMEDIATE AND LARGE BREAK LOCA IN STATE A (PCC-4)

6.1.3. Methods and Assumptions

6.1.3.1 Codes and methods

[Ref-1] A. Barbier. CATHARE 2 – Code synthesis qualification assessment. EPD DC 490 Revision C. AREVA. September 2007. (E)

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

6.1.5.3 Specific assumptions

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

[Ref-2] S Laurent. Residual Decay Heat Curves for System Design and Accident Analysis (Update 4500 MWth). NFPSC DC 283 Revision C. AREVA. November 2005. (E)

6.1.6. Description of studied cases from controlled state to safe shutdown state

6.1.6.1. Acceptance criteria

[Ref-1] C Hove. Core Reactivity Control (update 4500 MWth). NFPSC DC 284 Revision B. AREVA. January 2006. (E)

[Ref-2] P Cohen. Water Coolant Technology of Power Reactors. 1980. (E)

6.2. INTERMEDIATE AND LARGE BREAK LOCA IN SHUTDOWN STATE, RIS/RRA [SIS/RHRS] NOT CONNECTED (PCC-4)

6.2.1. Accident definition

[Ref-1] EPR FA3 – Primary Loops – Assembly. AREVA-NP drawing. NFPMR DB 1207 Revision G. AREVA. March 2009 (E)

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

6.2.5.2. Initial state

[Ref-1] S Laurent. Residual Decay Heat Curves for System Design and Accident Analysis (Update 4500 MWth). NFPSC DC 283 Revision C. AREVA. November 2005. (E)

6.2.5.3. Specific assumptions

b) Assumptions related to F1 systems

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

6.2.7. Results

State B2: accumulators unavailable (isolated)

EPR at 4500 MWth

[Ref-1] S Laurent. Residual Decay Heat Curves for System Design and Accident Analysis (Update 4500 MWth). NFPSC DC 283 Revision C. AREVA. November 2005. (E)

SECTION 14.5.6 - TABLES 9 TO 10

[Ref-1] B Cournarie. Rapport de sûreté Chapitre 15.2.4f2 APRP brèches intermédiaires et grosses brèches (jusqu'à la rupture de la ligne d'expansion dans l'état B) (Update 4500 MWth).
[Safety report of Chapter 15.2.4 f2 LOCA large and intermediate breaks (until the rupture of expansion lines in state B]
Edition 2005. AREVA. March 2005.

SECTION 14.5.6 - FIGURES 76 TO 87

[Ref-1] B Cournarie. Rapport de sûreté Chapitre 15.2.4f2 APRP brèches intermédiaires et grosses brèches (jusqu'à la rupture de la ligne d'expansion dans l'état B) (Update 4500 MWth).
[Safety report of Chapter 15.2.4 f2 LOCA large and intermediate breaks (until the rupture of expansion lines in state B]
Edition 2005. AREVA. March 2005.

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

7.1. SMALL BREAK LOCA IN STATE C (PCC-4)

7.1.5. Description of cases studied

7.1.5.2. Initial State

7.1.5.2.2. Assumptions related to F1 systems

LHSI/RHR trains (F1B):

[Ref-1] R Gagner. EPR Operation at 4250MWth. EPRR DC 1701 Revision B. AREVA. February 2004. (E)

7.1.5.3. Results

[Ref-1] EPR FA3 Preliminary Safety Analysis Report, Section 15.2.4g.
Small break LOCA (≤ DN 50) (states C, D). Edition 2003. AREVA. 2003. (E)

Applicability of EPR_{4250} analysis results to EPR_{4500}

[Ref-2] S Laurent. Residual Decay Heat Curves for System Design and Accident Analysis (Update 4500 MWth). NFPSC DC 283 Revision C. AREVA. November 2005. (E)

7.2. SMALL BREAK LOCA (DN50 IN STATE D)

7.2.5. Initial and Boundary Conditions

7.2.5.3. Specific assumptions

7.2.5.3.1 Decay heat

[Ref-1] S Laurent. Residual Decay Heat Curves for System Design and Accident Analysis (Update 4500 MWth). NFPSC DC 283 Revision C. AREVA. November 2005. (E)

7.2.5.3.2. Assumptions related to MHSI and LHSI/RHR (F1A)

[Ref-1] EPR FA3 Preliminary Safety Analysis Report, Section 15.2.4g.
Small LOCA (≤ DN 50) (states C, D). Edition 2003. AREVA. 2003. (E)

7.2.6. Results and Conclusions

EPR at 4500 MWth

[Ref-1] S Laurent. Residual Decay Heat Curves for System Design and Accident Analysis (Update 4500 MWth). NFPSC DC 283 Revision C. AREVA. November 2005. (E)

SECTION 14.5.7 - TABLE 3

[Ref-1] S Laurent. Residual Decay Heat Curves for System Design and Accident Analysis (Update 4500 MWth). NFPSC DC 283 Revision C. AREVA. November 2005. (E)

SECTION 14.5.7 - TABLES 4 TO 5

[Ref-1] EPR FA3 Preliminary Safety Analysis Report, Section 15.2.4g.
Small break LOCA (≤ DN 50) (states C, D). Edition 2003. AREVA. 2003. (E)

SECTION 14.5.7 - FIGURES 1 TO 4

[Ref-1] EPR FA3 Preliminary Safety Analysis Report, Section 15.2.4g.
Small break LOCA (≤ DN 50) (states C, D). Edition 2003. AREVA. 2003. (E)

8. REACTOR COOLANT PUMP SEIZURE (LOCKED ROTOR)

8.2. METHODS AND ASSUMPTIONS

8.2.2 Event and Qualification of the Panbox/Cobra-3cp Models

[Ref-1] R Muller. Analysis of Grafenrheinfeld Pump Shaft Break Event with PANBOX 2 including COBRA 3-CP. Work Report A1C-1 305860-0. AREVA. April 1999. (E)

8.2.3. Initial and Boundary Conditions

8.2.3.4. Evaluation of the Hot Spot Temperature (Transient)

[Ref-1] Dr Goll. Mikrostrukturelle Bestimmung der Brennstoffzentraltemperatur im Normalbetrieb.
[Determination of the fuel temperature in normal operation].
Work Report A1C-1 307321. AREVA. January 2000.

10. STEAM GENERATOR TUBE RUPTURE (2 TUBES IN 1 SG)

10.3. METHODS AND ASSUMPTIONS

10.3.1. Methods of Analysis

[Ref-1] A Barbier. CATHARE2 – Code Synthetic qualification assessment.
EPD DC 490 Revision C. AREVA. September 2007. (E)

[Ref-2] Rapport d'évaluation du code CATHARE 2 (version 1.3, revision 5)
[Calculation code CATHARE 2 (version 1.3 revision 5) assessment report]
STR/LML/EM/94-217.

[Ref-3] BETHSY – Test 4.3b – Multiple SGTR – test report. SETH/LES/90-101. (E)

10.5. SHORT TERM STUDY

10.5.2. Case 2: With LOOP

10.5.2.2. Initial State

[Ref-1] EPR FA3 – Preliminary safety analysis report. Section 15.2.4k. "Steam Generator Tube Rupture 2 Tubes in 1 SG (State A)". Edition 2003. AREVA. 2003. (E)

14. ISOLATABLE SAFETY INJECTION SYSTEM BREAK (DN ≤ 250 MM), IN RESIDUAL HEAT REMOVAL MODE (STATES C, D)

14.1. BREAK OUTSIDE THE CONTAINMENT

14.1.3. Methods and Assumptions

14.1.3.2. Initial Conditions

[Ref-1] System Design Manual - Reactor Coolant System (RCP [RCS]) - Part 2: System Operation. NESS-F DC 538 Revision A. AREVA. May 2009. (E)

14.1.3.3. Main Assumptions

14.1.3.3.4. Residual Heat Removal Systems

[Ref-1] EPR FA3 Preliminary Safety Analysis Report, Section 15.2.4q. Isolatable break in the RIS system in RRA mode (to DN 250) outside or inside the enclosure (States C, D). Edition 2006. AREVA. 2006. (E)

14.1.4. Definition of the cases studied

[Ref-1] L Foucart. Justification of the 2A-RHR break as a "specific study" – Functional aspect. NFPSR DC 1080 Revision A. AREVA. April 2005. (E)

14.1.5. Description of the cases analysed, from the initiating event to the controlled state

14.1.5.2. Specific Assumptions

[Ref-1] L Foucart. Justification of the 2A-RHR break as a "specific study" – Functional aspect. NFPSR DC 1080 Revision A. AREVA. April 2005. (E)

14.1.5.2.1 Decay Heat

[Ref-1] S Laurent. Residual Decay Heat Curves for System Design and Accident Analysis (Update 4500 MWth). NFPSC DC 283 Revision C. AREVA. November 2005. (E)

14.1.5.2.3. Assumptions Related to F1 Systems

[Ref-1] EPR FA3 Preliminary Safety Analysis Report, Section 15.2.4q. Isolatable break in the RIS system in RRA mode (to DN 250) outside or inside the enclosure (States C, D). Edition 2006. AREVA. 2006. (E)

14.1.5.3. Results

14.1.5.3.2. Break Diameter > DN 50

First phase: From initiating event to isolation of the break

[Ref-1] L Foucart. Justification of the 2A-RHR break as a "specific study" – Functional aspect. NFPSR DC 1080 Revision A. AREVA. April 2005. (E)

14.1.6. Description of the Cases Analysed, from the Controlled State to the Safe State

14.1.6.2. Break Diameter > DN 50

14.1.6.2.2 State D

State D with low water level in the reactor cavity (3/4 loop)

Controlled state:

b) An RIS/RRA [SIS/RHRS] train is initially isolated.

[Ref-1] R Gagner. EPR sizing 4500 MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

14.1.6.3. Conclusion

[Ref-1] EPR FA3 Preliminary Safety Analysis Report, Section 15.2.4q. Isolatable break in the RIS system in RRA mode (to DN 250) outside or inside the enclosure (States C, D). Edition 2006. AREVA. 2006. (E)

14.2. ISOLATABLE BREAK INSIDE THE CONTAINMENT

[Ref-1] EPR FA3 Preliminary Safety Analysis Report, Section 15.2.4q. Isolatable break in the RIS system in RRA mode (to DN 250) outside or inside the enclosure (States C, D). Edition 2006. AREVA. 2006. (E)

SECTION 14.5.14 - TABLE 1

[Ref-1] EPR FA3 Preliminary Safety Analysis Report, Section 15.2.4q. Isolatable break in the RIS system in RRA mode (to DN 250) outside or inside the enclosure (States C, D). Edition 2006. AREVA. 2006. (E)

15. NON ISOLATABLE SMALL BREAK (< 50 MM) OR ISOLATABLE RIS [SIS] BREAK (< 250 MM) IN RHR MODE, SPENT FUEL POOL DRAINAGE ASPECTS (STATE E)

15.1. INTRODUCTION

[Ref-1] Functional study on the treatment of PCCs and RRC-As involving spent fuel pool cooling loss and draining. ECEF080499 Revision B1. EDF. November 2012. (E)

[Ref-2] System Design Manual - Fuel Pool Cooling System (PTR [FPPS/FPCS]) - Part 2: System Operation, SFL–EF MF 2006.712 Revision G1. Sofinel. August 2009. (E)

15.2. PRE-ACCIDENT CONDITIONS (NORMAL OPERATION – PCC-1)

[Ref-1] Residual Decay Heat Curves for Major Components Design Purposes Heat Load inside the Fuel Pool. NEPC-F DC 164 Revision B. AREVA NP. November 2008. (E)

15.3. GRACE PERIOD

[Ref-1] Residual Decay Heat Curves for Major Components Design Purposes Heat Load inside the Fuel Pool. NEPC-F DC 164 Revision B. AREVA NP. November 2008. (E)

15.6. TRANSIENTS

15.6.1. Isolatable break on a RIS [SIS]/RHR line (< 250 mm)

15.6.1.2. Transient analysis

[Ref-1] System Design Manual - Safety Injection and Residual Heat Removal System - Part 4: Flow diagrams. NESS-F DC 546 Revision A. AREVA. June 2009. (E)

[Ref-2] Functional study on the treatment of PCCs and RRC-As involving spent fuel pool cooling loss and draining. ECEF080499 Revision B1. EDF. November 2012. (E)

15.6.2. Non isolatable break (< 50 mm) on a line connected to the primary cooling system

15.6.2.1. Introduction

[Ref-1] Functional study on the treatment of PCCs and RRC-As involving spent fuel pool cooling loss and draining. ECEF080499 Revision B1. EDF. November 2012. (E)

[Ref-2] System Design Manual - Fuel Pool Cooling System (PTR [FPPS/FPCS]) - Part 2: System Operation, SFL–EF MF 2006.712 Revision G1. Sofinel. August 2009. (E)

[Ref-3] System Design Manual Fuel Pool Purification and Cooling Systems (PTR [FPPS/FPCS]) Part 5 – Instrumentation and Control. SFL EF MF 2006.751 Revision F1. Sofinel. September 2009. (E)

15.6.2.2. Transient analysis

[Ref-1] System Design Manual - Fuel Pool Cooling System (PTR [FPPS/FPCS]) - Part 2: System Operation, SFL–EF MF 2006.712 Revision G1. Sofinel. August 2009. (E)

SECTION 14.5.15 - TABLE 1

[Ref-1] Residual Decay Heat Curves for Major Components Design Purposes Heat Load inside the Fuel Pool. NEPC-F DC 164 Revision B. AREVA NP. November 2008. (E)