UK EPR	Title: PCSR – Appendix 14B – 4900MW Safety Analyses used in Chapter 14
	UKEPR-0002-148 Issue 05

Total number of pages: 482	Page No.: I / IV
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REVISION HISTORY

Issue	Description	Date
00	First issue for INSA review	14-01-2008
01	Integration of co-applicant and INSA review comments	29-04-2008
02	Remarking with RESTRICTED classification	16.01.09
03	PCSR June 2009 update including text clarifications	28-06-2009
04	Removal of RESTRICTED classification	10-06-2010
05	Consolidated Step 4 PCSR update: - Minor editorial changes - Appendix 14B.2.5.2 – Figure 2 (page 433) corrected in PDF version of document	26-03-2011

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TABLE OF CONTENTS

(The sections in this Appendix are numbered as in the original "Basic Design Report" issued in 1999)

0. ASSUMPTIONS USED IN SAFETY ANALYSES

0.1. SELECTION OF EVENTS

0.2. PLANT CHARACTERISTICS ASSUMED IN ACCIDENT ANALYSES

1. ANALYSIS RULES

1.1. ACCEPTANCE CRITERIA

1.3. INITIAL CONDITIONS

1.4. RULES FOR OPERATOR ACTIONS

1.5. MECHANICAL, ELECTRICAL AND I&C SYSTEMS MODELLED IN THE SAFETY ANALYSIS

1.6. APPLICATION OF THE SINGLE FAILURE CRITERION (SFC) IN THE SAFETY ANALYSIS

1.7. PREVENTATIVE MAINTENANCE

1.8. LOSS OF OFFSITE POWER (LOOP)

2. SAFETY ANALYSES

2.10. UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY (RCCA) BANK WITHDRAWAL "AT POWER" (STATE A)

2.14. EXCESSIVE INCREASE IN SECONDARY SYSTEM FLOW / INADVERTENT OPENING OF A SG RELIEF TRAIN OR A SG SAFETY VALVE

2.15. STEAM SYSTEM PIPING FAILURE

2.16.1 FEEDWATER SYSTEM PIPE BREAK (IN STATE A, PCC4)

2.17. STEAM GENERATOR TUBE RUPTURE (SINGLE TUBE IN STATE A, PCC-3)

2.18. STEAM GENERATOR TUBE RUPTURE (2 TUBES IN STATE A, PCC-4)

2.2. SMALL BREAK LOCA

2.3.1. INTERMEDIATE AND LARGE BREAK LOCA IN STATE A (PCC-4)

2.4. INADVERTENT CLOSURE OF ONE/ALL MAIN STEAM ISOLATION VALVE(S) (PCC-3)

2.5.1 LOSS OF NON-EMERGENCY AC POWER TO THE PLANT AUXILIARIES (PCC-2)

2.5.2 LOSS OF CONDENSER VACUUM (PCC-2)

2.6. DECREASE OF FORCED REACTOR COOLANT FLOW (PCC-3)

2.7. PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW (PCC-2)

2.8. REACTOR COOLANT PUMP SHAFT BREAK OR LOCKED ROTOR (PCC-4)

APPENDIX 14B - 4900 MW SAFETY ANALYSES USED IN CHAPTER 14

This appendix reproduces information from the EPR Basic Design Report 99 sections corresponding to accident analyses presented in the PCSR.

The section numbering used in this appendix maps to that used in BDR 99.

The content of this appendix is as follows:

- 0.1 Scope of events
- 0.2 Plant Characteristics assumed in the accident analyses
- 1 Analysis rules
- 2.2.1 Small Break LOCA in state A (PCC-3)
- 2.3.1 Intermediate and Large Break LOCA in state A (PCC-4)
- 2.4 Inadvertent closure of all/one VIV [MSIV] (PCC-3)
- 2.5.1 Loss of Non-Emergency AC Power to the Plant Auxiliaries (PCC-2)
- 2.5.2 Loss of condenser vacuum (PCC-2)
- 2.6 Forced decrease of reactor coolant flow (4 pumps)
- 2.7 Partial Loss of Forced reactor coolant flow (PCC-2)
- 2.8 Reactor Coolant Pump shaft Break or locked rotor (PCC-4)
- 2.10 Uncontrolled rod cluster control assembly (RCCA) bank withdrawal (state A) only for the transient "at power"
- 2.14 Excessive increase in secondary steam flow/Inadvertent opening of a SG relief train or of a SG safety valve
- 2.15 Steam system piping failure
- 2.16.1 Feedwater system pipe break (in state A, PCC-4)
- 2.17 Steam Generator Tube Rupture (1 tube in state A, PCC-3)
- 2.18 Steam Generator Tube Rupture (2 tubes in state A, PCC-4)

0. ASSUMPTIONS USED IN SAFETY ANALYSES

0.1. SELECTION OF EVENTS

0.1.1. General remarks

Events are selected according to their significance in challenging the main safety functions:

- reactivity and power control,
- heat removal from the fuel elements,
- containment of radioactivity.

Events are classified into four Plant Condition Categories (PCCs) and two Risk Reduction Categories (RRCs).

Appendix 14B deals primarily with the analysis of PCC events. The classification of PCC events is made according to their approximate expected occurrence frequency:

- PCC-1: transients occurring in normal operation,
- PCC-2: anticipated operational occurrences,
- PCC-3: infrequent accidents,
- PCC-4: limiting accidents.

PCCs are events caused by the failure of a single component or I&C function, or a single operator error (e.g. spurious starting of a Reactor Coolant Pump), or by a loss of off-site power.

Due to application of the break preclusion concept, double-ended guillotine break of a main coolant line (henceforth referred to as 2A-LOCA), is excluded by design. The term Large Break LOCA (LB(LOCA)) used in the safety analysis therefore refers to the rupture of the largest line connected to a main coolant line (MCL) and not to the double-ended guillotine break of an MCL, which was usually the case in the past.

The 2A-LOCA is neither a PCC nor a RRC-A event. However, a break mass flow rate corresponding to a 2A-LOCA is considered for the verification of the design of the emergency core cooling system and the containment, in order to provide safety margins in the design.

Break preclusion is also applied to the main steam lines (MSLs) between the SGs and the fixed point downstream the VIVs [MSIVs], and to the Feedwater Lines inside the containment. This means that a double ended (2A) break of these pipe sections is excluded. Consequently, the 2A-MSL break and the 2A-FWL break are not treated in this appendix in the framework of the PCC safety analysis rules.

Internal and external hazards are addressed in Chapter 13 of the PCSR.

0.1.2. Standard reactor states

Events addressed in the safety analysis are assumed to occur during normal plant operation. The initial conditions assumed in the safety analysis cover all possible reactor states from full power operation to cold shutdown. The following six standard reactor states are defined (see also Appendix 14B.0.1 - Table 1):

- **State A**: at power and hot and intermediate shutdown states. In the State A shutdown states all necessary automatic reactor protection functions are assumed to be available, as in the at-power state. In practice, certain protection functions might be deactivated at low powers, but sufficient automatic protection functions always remain available to ensure that acceptance criteria will be met if an event occurs.

- **State B**: intermediate shutdown above 120°C. Above this temperature the RIS/RRA [SIS/RHRS] is not connected to the RCP [RCS] in normal operation. It should be noted that the RIS/RRA [SIS/RHRS] can be connected to the RCP [RCS] at 180°C in emergencies, but this is not an initial state occurring in normal operation and therefore it is not considered in deterministic safety analysis. In state B, some automatic reactor protection functions available in the state A might be deactivated (e.g. automatic VIV [MSIV] isolation on low steam pressure).

- **State C**: intermediate and cold shutdown with RIS/RRA [SIS/RHRS] in operation. The RCP [RCS] is either closed or can be rapidly reclosed (e.g. when a vent line is open) so that the SGs can be used for decay heat removal if required. The RCP [RCS] is filled with water or at mid-loop operation level (e.g. for SG tube draining or RCP [RCS] sweeping).

- **State D**: cold shutdown with the RCP [RCS] open so that the SGs are unavailable for decay heat removal. The RCP [RCS] water level can be at mid-loop height.

- **State E**: cold shutdown with the reactor pool flooded for refuelling.

- **State F**: cold shutdown with the core totally unloaded. Work is performed on RCP [RCS] components in this state. This state has not to be analysed with regard to core protection.

0.1.3. List of PCC events

General remark: LOCAs which are defined as PCC events range from small RCP [RCS] leaks to guillotine failure of the largest lines connecting to the RCP [RCS]. The following terms are used:

- Small Break LOCA (SB(LOCA)): breaks with an equivalent diameter less than or equal to 50 mm,

- Intermediate Break LOCA (IB(LOCA)): breaks with an equivalent diameter greater than 50 mm but less than the diameter of an RIS/RRA [SIS/RHRS] injection line,

- Large Break LOCA (LBLOCA): full break of the largest lines connecting to the RCP [RCS]. These are the RIS/RRA [SIS/RHRS] injection lines on the cold leg side of the main coolant line and the pressuriser surge line on the hot leg side.

0.1.3.1. PCC-1 events: operational transients

- plant heat up and cooldown,
- step load changes,
- ramp load changes,
- load rejection up to and including the design full load rejection,
- loss of the main grid with auxiliary grid available,
- loss of the main feedwater system with the start-up and shutdown system available,
- partial reactor trip.

These transients are expected to occur frequently or even routinely in normal reactor operation. The events are not addressed as part of the safety analysis but are used to define load conditions for structural integrity analysis of the CPP [RCPB] and the SG.

0.1.3.2. PCC-2 events: anticipated operational occurrences (*)

- reactor trip (spurious),
- feedwater system malfunction causing a reduction in feedwater temperature,
- feedwater system malfunction causing an increase in feedwater flow,
- excessive increase in secondary steam flow,
- turbine trip,
- loss of condenser vacuum,
- short term loss of off-site power (≤ 2 hours), (states A, C and D),
- loss of normal feedwater flow (loss of all Main Feedwater Pumps (MFW(P)s) and of the start-up and shutdown pump),
- loss of 1 Reactor Coolant Pump without partial trip,
- uncontrolled rod cluster control assembly (RCCA) bank withdrawal (state A),
- RCCA misalignment up to rod drop, without limitation,
- start up of an inactive reactor coolant loop at an incorrect temperature,
- RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant, (states A to E),

(*) When the reactor state is not mentioned, the event is supposed to be analysed in power state.

- RCV [CVCS] malfunction causing increase or decrease in reactor coolant inventory,
- primary side pressure transients (spurious pressuriser spraying, spurious pressuriser heating),
- uncontrolled RCP [RCS] level drop during mid-loop operation (states C or D),
- loss of one cooling train of the SIS/RHRS in RHR mode (mid-loop operation in states C or D).

0.1.3.3. PCC-3 events: infrequent accidents (*)

- small steam or feedwater system piping failure,
- long term loss of off-site power (> 2 hours), (state A),
- inadvertent opening of a pressuriser pilot operated safety valve,
- inadvertent opening of a SG relief train or of a SG safety valve, (state A),
- SB(LOCA) (states A, B),
- Steam Generator tube rupture (one tube, state A),
- inadvertent closure of one/all main steam isolation valves,
- inadvertent loading and operation of a fuel assembly in an improper position,
- forced decrease of reactor coolant flow (four pumps),
- failures in Liquid or Gaseous Waste Systems,
- uncontrolled RCCA bank withdrawal (states B to D),
- uncontrolled single control rod withdrawal (state A),
- rupture of a line carrying primary coolant outside containment, e.g. sampling line.

0.1.3.4. PCC-4 events: limiting accidents (*)

- long term loss of off-site power (> 2 hours) (state C),
- steam system pipe break (states A, B),
- feedwater system pipe break (states A, B),
- inadvertent opening of a SG relief or safety valve (state B),
- spectrum of RCCA ejection accidents (states A, B),
- IB and LB(LOCA) (states A and B),

(*) When the reactor state is not mentioned, the event is analysed for the at-power state.

- SB(LOCA) (states C and D),
- Reactor Coolant Pump seizure (locked rotor),
- Reactor Coolant Pump shaft break,
- Steam Generator tube rupture (two tubes in one SG in state A),
- fuel handling accident,
- boron dilution due to a non isolable rupture of an heat exchanger tube (states A to E),
- leak outside containment on RIS/RRA [SIS/RHRS] in RHR mode (states C, D).

0.1.4. Events analysed in the BDR

All the PCC-2, PCC-3 and PCC-4 events listed in section 0.1.3 within this appendix are relevant to reactor safety.

Appendix 14B.0.1 - Table 2 gives an overview of the PCC events and the reactor states associated with each event (marked by crosses). The table indicates which PCC transients are analysed in the BDR and which are not. For cases analysed in BDR the corresponding BDR section is indicated. Those cases not analysed in the BDR are marked “not analysed”; a footnote gives the reason why the transient was not analysed.

APPENDIX 14B.0.1 – TABLE 1

<table>
<tr><th>Reactor state</th><th>LHSI/RHR connect. to RCP [RCS]</th><th>RCP [RCS] status</th><th>RCP [RCS] Coolant level</th><th>RCP [RCS] average temperature</th><th>RCP [RCS] pressure</th><th>Cont. hatch status</th><th>Remarks</th></tr>
<tr><td>A</td><td rowspan="2">no</td><td rowspan="4">closed (2)</td><td rowspan="2">full</td><td rowspan="2">311°C to 120°C</td><td>155 bar to (1)</td><td rowspan="2">closed (above ≅ 70°C in RCP [RCS])</td><td>All automatic protection functions are available as in power operation</td></tr>
<tr><td>B</td><td>(1) to30 bar</td><td>Some automatic protection functions with regard to LOCA or overcooling transients might be deactivated below this pressure (1)</td></tr>
<tr><td rowspan="2">C</td><td rowspan="4">yes</td><td>120°C to ≅ 55°C</td><td rowspan="2">30 bar to 1 bar</td><td rowspan="2">SIS/RHRS connected to RCP [RCS] at 120°C in normal operation but can be connected up to 180°C if needed

At least 2 SGs available for decay heat removal

Mid-loop operation for SG tube draining and nitrogen flushing</td></tr>
<tr><td rowspan="3">≅ 55°C</td><td>mid-

loop</td><td>open (below ≅ 70°C in RCP [RCS])</td></tr>
<tr><td>D</td><td>open</td><td>1 bar</td><td>closed (3)</td><td>RPV head handling: level near MCL top.
Work on Reactor Coolant Pump seals: level at mid-loop.
No maintenance on safety trains (but maintenance allowed on 1 train at low decay heats in the vessel, e.g. after refuelling).</td></tr>
<tr><td>E (4)</td><td>reactor cavity flooded</td><td>open</td><td>Core refuelling
Maintenance normally allowed on one train only (but allowed on 2 trains at low decay heat in RPV, e.g. with core partly unloaded)</td></tr>
</table>

(1) This pressure limit depends on the I&C design and is still to be confirmed (e.g. 120 bar with respect to overcooling and 60 bar for isolation of the accu.),
(2) RCP [RCS] can be rapidly re-closed when partly open (e.g. vent line),
(3) During mid-loop operation after core refuelling the containment can be open.
(4) State F is similar to state E; the only one difference is that the core is totally unloaded

APPENDIX 14B.0.1 – TABLE 2 : STANDARD REACTOR STATES FOR SAFETY ANALYSES

PCC-2 Events

Event	Appendix 14B Section	A	B	C	D
feedwater system malfunction causing a reduction in feedwater temperature	not analysed [1)]	X			
feedwater system malfunction causing an increase in feedwater flow	not analysed [2)]	X			
excessive increase in secondary steam flow	2.14	X			
turbine trip	not analysed [3)]	X			
loss of condenser vacuum	2.5.2	X			
short term loss of off-site power (≤ 2hours)	2.5.1	X		X	X
loss of normal feedwater flow (all ARE [MFWS] pumps and AAD [SSS])	not analysed [4)]	X			
loss of 1 Reactor Coolant Pump without partial trip	2.7	X			
uncontrolled rod cluster control assembly (RCCA) bank withdrawal	2.10	X			
RCCA misalignment up to rod drop, without limitation		X			
start up of an inactive reactor coolant loop at an incorrect temperature	not analysed [5)]	X			
RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant		X		X	X
RCV [CVCS] malfunction causing increase or decrease in reactor coolant inventory	not analysed [6)]	X			
primary side pressure transients (spurious pressuriser spraying/heating)	not analysed [7)]	X			
uncontrolled RCP [RCS] level drop during mid-loop operation				X	X
loss of one cooling train of RIS/RRA [SIS/RHRS] in RHR mode				X	X

PCC-3 Events

Event	Appendix 14B Section	A	B	C	D
small steam or feedwater system piping failure	not analysed [8)]	X			
long term loss of off-site power	2.5.1	X			
inadvertent opening of a pressuriser pilot operated safety valve	not analysed [9)]	X			
inadvertent opening of a SG relief train or of a SG safety valve	2.14	X			
SB(LOCA)	2.2	X	X		
steam generator tube rupture (one tube)	2.17	X			
inadvertent closure of one main steam isolation valve	2.4.2	X			
inadvertent closure of all main steam isolation valves	2.4.1	X			
inadvertent loading and operation of a fuel assembly in an improper position	not analysed [10)]	X			
forced decrease of reactor coolant flow (four pumps)	2.6	X			
failures in liquid or Gaseous Waste System	3.5	X			
uncontrolled RCCA bank withdrawal	2.9		X	X	X
uncontrolled single control rod withdrawal	2.11	X			
rupture of a line carrying primary coolant outside containment, e.g. sampling line	4.3	X			

PCC-4 Events

Event	Appendix 14B Section	A	B	C	D
long term loss of off-site power (> 2 hours)	2.5.1			X	
steam system pipe failure	2.15	X	X		
feedwater system pipe break	2.16	X	X		
inadvertent opening of a SG relief or safety valve	2.14		X		
spectrum of RCCA ejection accidents	2.12	X	X		
IB and LB(LOCA)	2.3	X	X		
SB(LOCA)	2.2			X	X
Reactor coolant pump seizure (locked rotor)	2.8	X			
reactor coolant pump shaft break	2.8	X			
steam generator tube rupture (two tubes in one SG in state A)	2.18	X			
fuel handling accidents	4.8	X			
boron dilution due to a non isolable rupture of an heat exchanger tube	2.13 [11)]			X	X
leak outside containment on RIS/RRA [SIS/RHRS] in RHR mode	2.19			X	X

1) Based on experience, a reduction in feedwater temperature leads to either a new steady state condition at an increased power level or to reactor trip from e.g. on "high reactor power" or "low DNBR" (depending on the burnup and detailed reactor trip criteria). This type of event is therefore bounded by other events involving a reactor power increase, such as RCCA bank withdrawal (see section 2.10 of this appendix).

2) An increase in feedwater flow is less important for reactor safety than a decrease in feedwater temperature, due to the fact that such an event can be terminated by closure of dedicated MFW isolation valves.

3) Turbine trip is bounded by the "loss of condenser vacuum" event (see section 2.5.2 of this appendix), since in the case of loss of condenser vacuum, the turbine bypass system is also lost.

4) Loss of all normal feedwater flow (ARE [MFWS] + AAD [SSS]) is of interest in two aspects: core protection and RCP [RCS] heat removal. Core cooling is not significantly impaired as in the limiting case reactor trip occurs on SG level < MIN1 (13.6 m) before the SG heat transfer capability is reduced. The variation of RCP [RCS] pressure (impacting on heat removal capability) is within limits experienced in more severe events.

5) Start-up of an inactive reactor coolant loop at an incorrect temperature is not analysed because operation at an inadmissible reactor power is prevented by design of the reactor trip function [x]. In general, mitigation of this event aims primarily at avoiding reactor trip [x].

6) An RCV [CVCS] malfunction resulting in increase or decrease of reactor coolant inventory does not represent a limiting case for safety analysis. Spurious increases would be detected by a PZR level increase which in severe cases would lead to reactor trip on "PZR level high". The spurious increase would be terminated by cut off of the RCV [CVCS] charging pumps. Spurious decreases would likewise be detected by a PZR level decrease, and would be terminated by isolation of the RCV [CVCS] letdown line. In a worst case scenario in which neither means of mitigation was available, both types of scenario could lead to a LOCA; LOCAs of equivalent break size are analysed in sections 2.2 and 2.3 of this appendix.

7) Spurious PZR spraying or heating do not represent limiting cases. Both events would be terminated by dedicated signals and means. Assuming a failure of these provisions, continuous spraying would result in reactor trip on e.g. "RCP [RCS] pressure low" or "DNBR low" so that core damage would be prevented; spray efficiency is lost anyway with filling of the PZR. In case of continuous PZR heating, the RCP [RCS] pressure increase would be limited by the PZR safety valves.

8) Minor piping failures in the feedwater or steam system are bounded by the analyses in sections 2.14B and 2.16 of this appendix.

9) Inadvertent opening of a PZR safety valve is bounded by analyses in section 2.3 of this appendix.

10) Inadvertent loading and operation of a fuel assembly in an improper position is prevented by various measures (administrative procedures, automatic identification of fuel assemblies and loading procedure, etc.) that 'practically eliminate' the potential for loading errors. These detailed operating procedures will be defined later and are outside the scope of GDA.

11) Boron dilution due to a non isolatable rupture of a heat exchanger tube is not possible in states A and B as the SIS/RHRS operation pressure is below RCP [RCS] operating pressure.

0.2. PLANT CHARACTERISTICS ASSUMED IN ACCIDENT ANALYSES

This sub-section defines the plant characteristics which are assumed in the accident analyses presented in this appendix. Some characteristics which are specific to a given accident analysis are specified within the sub-section describing the accident analysis.

The plant characteristics assumed in the accident analyses, relate to:

- plant geometrical data,
- plant initial conditions,
- reactivity coefficients,
- residual decay heat,
- I&C signals, related to RT and safety systems operation,
- safety systems characteristics.

Conservative values of these parameters (either minimum or maximum) are applied in PCC accident analyses, in accordance with the PCC analyses rules. Realistic parameter values used in the RRC-A accident analyses of Chapter 16 of the PCSR, are also presented for completeness.

0.2.1. Plant geometrical data

Appendix 14B.0.2 - Table 1 lists the main geometrical data related to the reactor coolant system and the steam generator secondary side.

Appendix 14B.0.2 - Figure 1 and 2 show the SG and pressuriser (PZR) geometries, indicating corresponding level measurements.

0.2.2. Plant initial conditions

In accident analyses, initial conditions are obtained by either adding or subtracting maximum steady state errors to or from nominal values (depending on which is pessimistic under PCC analyses rules).

The steady state errors include the measurement error, steady state fluctuations, and the control dead band if applicable.

The nominal values and the associated maximum errors are defined in Appendix 14B.0.2 - Table 2, for all the relevant parameters i.e.:

- core power,
- Pressuriser pressure,
- RCP [RCS] average temperature,
- Pressuriser level,

- SG level.

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimises adverse power distribution through the placement of control rods and adherence to operating instructions.

The most adverse power distribution which could occur during normal operation is considered as the starting point for transient. This corresponds to the surveillance limitation thresholds for maximum linear power density and minimum DNBR.

Appendix 14B.0.2 - Table 3 provides additional information on other pertinent parameters.

{ CCI Removed }

[b].

0.2.3. Reactivity coefficients

The transient response of the reactor system is dependent on reactivity feedback effects, in particular on the moderator temperature and Doppler power coefficients.

In the analysis of certain events, conservatism requires the use of large reactivity coefficients, whereas in the analysis of other events, conservatism requires the use of small reactivity coefficients. The values used are given in Appendix 14B.0.2 - Table 4. These values are applied in a core point kinetic model, and are not used in the analysis of some specific accidents such as ATWS where dedicated coefficient values are used.

RCP [RCS] boron concentration values assumed in the initial state (nominal operating conditions), and the required boron concentration at LHSI/RHR connection conditions are shown in Appendix 14B.0.2 - Table 5 for UO2 and MOX fuel cycles.

0.2.4. Residual fission and decay heat after RT

The residual heat in a subcritical core consists of:

- residual fission heat due to delayed neutrons (term A),
- decay heat from capture products U239 and Np239 (term B),

- decay heat from fission products and actinides except U239 and Np239 (term C).

The thermal power due to residual fissions (term A) after RT, depends on the number of neutrons generated by the different neutron sources in the core, including:

- decay of delayed neutron precursors,
- spontaneous fissions of actinides,
- (α, n) reactions.

The residual thermal power due to the decay of fission products and actinides (terms B+C) depend on the amount of these products present at RT. The main parameters having an influence on the nuclide composition in the core, which are linked to the fuel type and to the fuel management scheme, are as follows:

- the initial fuel enrichment,
- the number of batches in the core,
- the enrichment and the burnup of the different batches,
- the burnup history (irradiation sequences) of each batch (cycle length, specific irradiation power).

The code ORIGEN-S is used to calculate, the evolution of the nuclide composition inventory during fuel irradiation and the decay heat power of each chemical element in the core after RT. A total of 1700 nuclides are contained in the ORIGEN-S databases.

ORIGEN-S provides the B + C term of the residual decay heat. Different uncertainties are attached to the ORIGEN-S results as follows:

- for applications requiring realistic calculations (e.g. RRC-A accident analyses), no uncertainty is applied to the ORIGEN-S results (best-estimate (BE) decay heat curve used).
- for applications requiring conservative calculations (e.g. PCC accident analyses), the uncertainties applied are :
  - in case of a core containing UO2 assemblies exclusively (UO2 fuel management):
    - +15% for the short term (decay time less than 10 minutes),
    - +10% for the long term (decay time greater than 10 minutes).
  - in case of a core containing both UO2 and MOX assemblies (30% - MOX fuel management):
    - +15% whatever the decay time.

The above mentioned uncertainties correspond to a 95 % probability for a one sided distribution (1.645 σ) at a confidence level of 95%.

The resulting decay heat curve is referred to as the maximum decay heat (MAX).

Appendix 14B.0.2 - Table 6 depicts the two decay heat curves (BE and MAX) related to the B + C term: the curves are enveloping for all EPR UO2 and MOX fuel management schemes.

The A term depends on the characteristics of prompt and delayed neutrons and the variation of the core effective multiplication factor K during and after RT:

- kinetic parameters are defined for prompt neutrons and six groups of delayed neutrons,
- the variation of the multiplication factor as a function of time, K (t), is derived using the characteristics of the RT (rod reactivity worth as a function of time, based on the drop characteristic given in Appendix 14B.0.2 – Table 7) and the thermal-hydraulic characteristics of the core.

A specific calculation of the A term can be performed consistent with the accident analysis, on the basis of the shutdown rod characteristics given in Appendix 14B.0.2 - Table 8 and the reactivity coefficients given in Appendix 14B.0.2 - Table 4, using the actual core thermal-hydraulic behaviour.

This A term is simulated for most transients using the neutron kinetic model

In the particular cases of LOCA, FWLB, SGTR the A term is provided as an input quantity generated using a conservative RT-simulation decoupled from the accident analysis.

0.2.5. I&C signals

0.2.5.1. Primary and secondary (P/S) I&C signals

0.2.5.1.1. List of F1-signals

I&C signals considered in the accident analyses are those involved in either RT actuation or actuation of F1-classified systems (some non F1 classified signals may be also be considered if this is in accordance with the accident analyses rules).

The F1-signals related to the primary and secondary systems (P/S), which are used in PCC accident analyses are listed in Appendix 14B.0.2 - Table 9, along with their setpoints and associated uncertainties. F1-systems which are not actuated by the I&C, such as the pressuriser and MS safety valves, are also indicated.

The list of F1-signals does not address manual F1B actions; these are discussed, if applicable, in the section presenting the detailed accident analysis study.

When it is pessimistic to do so, a maximum time delay is considered for actuation of a signal and completion of the resulting action. Appendix 14B.0.2 - Table 11 and 12 define the time delays assumed in the PCC accident analyses.

0.2.5.1.2. Brief description of specific F1A P/S I&C functions

Explanations of specific I&C signals are provided below.

0.2.5.1.2.1. RIS [SIS] signal

For the BDR accident analyses in this appendix, the RIS [SIS] actuation signals considered are:

- state A : "pressuriser pressure < MIN3"
- state B, state C with Reactor Coolant Pump on : "hot leg ΔP_{sat} < MIN"
- state C with Reactor Coolant Pump off + state D : "hot leg RCP [RCS]-loop level < MIN"

The above signals rely on the following sensors:

- for "pressuriser pressure", 4 pressuriser pressure sensors with a 2 out of 4 logic,
- for "ΔP_{sat}", 4 hot leg temperature (1 per loop) and 4 hot leg pressure (1 per loop) sensors, with a 2outt of 4 logic (2 out of 4 loops),
- ΔP_{sat} is defined as ΔP_{sat} = actual P_{HL} - P_{sat} (actual T_{HL}),
- for "RCP [RCS]-loop level", 4 ΔP-level (1 per loop) measurements with sensing lines located respectively at the bottom and at the top of the hot leg, with a 2/4 logic (2/4 loops).

The RIS [SIS] components actuated are:

- in state A, state B before accumulator isolation
  - start all MHSI and LHSI pumps injecting into the RCP [RCS] cold legs,
  - confirm accumulator valve opening.
- in state B after accumulator isolation
  - start all MHSI and LHSI pumps injecting into the RCP [RCS] cold legs,
  - accumulator valves remain closed.
- in states C and D
  - start all MHSI pumps injecting into the RCP [RCS] cold legs,
  - LHSI pumps remain in RHR-mode operation or in standby,
  - accumulator valves remain closed,
  - note: at LHSI/RHR connection, the following changes are implemented in the PSV and MHSI systems,
  - the PSV setpoint is decreased to about 50 bar (I&C electrical signal),

- the large mini flow line of each MHSI pump is opened, in order to decrease the RCP [RCS] delivery pressure to about 40 bar.

0.2.5.1.2.2. Partial cooldown

On an RIS [SIS] actuation signal, a partial cooldown is automatically actuated. (As the MHSI delivery pressure is reduced in the EPR compared with earlier French NPP designs (in the range 80 to 90 bar), to avoid MSSV challenge in SGTR, an RCP [RCS] cooldown is required to ensure MHSI injection).

The cooldown is performed via the secondary side and consists in lowering the VDA [MSRT] setpoint of the four SGs from 93 to 60 bar, at a rate corresponding to -100°C/h. At the same time, the GCT [MSB] setpoint is reduced from 87.1 bar to 55 bar at the same rate. (Partial cooldown via the GCT [MSB] is discounted as it is not F1 classified).

Some I&C functions use the signal "partial cooldown finished": the end of the partial cooldown is detected when the pressure in the four SGs is lower than a target pressure (i.e. 60 bar).

0.2.5.1.2.3. Reactor Coolant Pump trip

In the EPR design, it is intended to provide an automatic Reactor Coolant Pump trip signal in case of LOCA.

Implementation of this signal aims at improving the LOCA accident mitigation, by avoiding a delayed Reactor Coolant Pump trip (the Reactor Coolant Pumps not being qualified to operate under LOCA conditions), with a potential severe penalty on the RCP [RCS] water inventory depletion, and consequently on the core heat-up.

The Reactor Coolant Pump trip signal considered is "ΔP over Reactor Coolant Pump < MIN1 and RIS [SIS] signal". The combination with the RIS [SIS] signal is aimed at avoiding spurious Reactor Coolant Pump trip. ΔP refers to the pressure difference between the Reactor Coolant Pump inlet (U-leg pressure) and the Reactor Coolant Pump outlet (cold leg pressure). With respect to LOCA mitigation from power state, one differential pressure sensor per Reactor Coolant Pump is sufficient, with a 2/4 logic involving the four Reactor Coolant Pumps (2/4 loops).

In the accident analyses, the most conservative hypothesis (either earliest or latest Reactor Coolant Pump trip) is selected, according to the decoupling criteria under consideration. "ΔP over Reactor Coolant Pump < MIN1 and RIS [SIS] signal" is credited when a late trip is pessimistic. “Reactor Coolant Pump trip at LOOP occurrence" (i.e. RT/TT) is assumed when an early trip is pessimistic.

0.2.5.1.2.4. SG pressure drop signals

The SG pressure drop signals are dedicated to mitigation of secondary side breaks.

Appendix 14B.0.2 - Figure 3 illustrates the principle of the signals: each function is in the form of an [SG pressure < MIN setpoint] where the MIN setpoint is variable and equal to the actual SG pressure minus 7 bar for the [SG pressure drop > MAX1] function and the actual SG pressure minus 17 bar for the [SG pressure drop > MAX2] function, with a limit on the rate of decrease of the setpoint of -2 bar/minute.

The maximum value of the setpoint is limited to 70 bar or 60 bar, in order to avoid VIV [MSIV] closure during SG pressure decreases following overpressure transients (e.g. reactor trip and VDA [MSRT] challenge on SG pressure > MAX1).

0.2.5.1.2.5. ARE [MFWS] isolation

Following any reactor trip signal, the ARE [MFWS] full load isolation and control valves are closed.

On the signals SG pressure drop > MAX1 or SG pressure < MIN1 in one SG, the ARE [MFWS] full load isolation and control valves are closed on all SGs: the purpose of this function is to limit the feed supply to the affected SG in case of a secondary side break. In addition, the four VIVs [MSIVs] are closed in order to isolate the unaffected SGs from the affected one.

As a first step, feedwater supplies to the SGs are not completely isolated (ARE [MFWS] low load line open in each SG), in order to take maximum benefit from the ARE [MFWS] (if pumps not tripped) or AAD [SSS], for the unaffected SG.

In the second step, on receipt of signals SG pressure drop > MAX2 or SG pressure < MIN2 in one SG, the ARE [MFWS] low load isolation and control valves are closed in this SG: the affected SG is detected via its further depressurisation after VIV [MSIV] isolation is performed in the first step, (the pressure in the unaffected SG increases). The objective of the SG related action is to completely terminate the normal feedwater supply to the faulted SG. Automatic isolation of ASG [EFWS] to the faulted SG is not currently planned. ASG [EFWS] isolation is instead performed by the operator (30 minute delay assumed in the accident analysis).

0.2.5.2. Core related I&C signals

The F1 signals related to the core protection (core related signals) for which credit is taken in the PCC accident analyses are listed in Appendix 14B.0.2 - Table 10. The corresponding time delays considered in the PCC accident analyses are listed in Appendix 14B.0.2 - Table 12.

A brief description of the protection channels which utilise these signals is given below, mainly with reference to the required inputs.

0.2.5.2.1. High linear power density

The maximum linear power density value is derived from signals from the 72 in-core Self Powered Neutron Detectors (SPNDs) which are distributed across the four divisions of the reactor Protection System (RPR [PS]).

This function initiates a Reactor Trip (and a Turbine Trip).

0.2.5.2.2. Low DNBR

The calculation of the minimum DNBR uses the following parameters:

- power density distribution in the hot channel,
- inlet temperature,
- pressure,

- core flow rate.

The hot channel power density distribution is derived from signals from the 72 in-core Self Powered Neutron Detectors (SPNDs) which are distributed across the four divisions of the reactor Protection System (RPR [PS]).

The inlet temperature is derived from the cold leg temperature sensors.

The pressure is derived from the primary pressure sensors.

The core flow rate is derived from the Reactor Coolant Pump speed sensors.

This function initiates a Reactor Trip (and a Turbine Trip).

0.2.5.2.3. Antidilution at power

The detection and mitigation of the spurious dilution at power is based on power dependent insertion limits (an alternate solution is to utilise an online calculation of the core reactivity balance).

This channel isolates the main sources of spurious dilution at power within the Chemical and Volume Control System (RCV [CVCS]) by closing two redundant in series valves downstream of the Volume Control Tank (VCT).

0.2.5.2.4. High neutron flux rate of change

The following relates to the high ex-core neutron flux rate of change channel, claimed in transient analysis of uncontrolled RCCA bank withdrawal at power (see section 2.10 of this appendix).

The signal used by this channel is the neutron flux rate of change (derivative module Tp / (1 + Tp)) obtained from the ex-core instrumentation:

- p is the Laplace transform and,
- T is a time constant.

The derivative function provides a step of unity times the input step, then decays to zero. As an example, the inverse Laplace transformation can be applied to this function with step and ramp inputs (there is no response to a static signal).

Due to the use of the derivative module, the setpoint value is given in terms of % of Nominal Power.

This function initiates a Reactor Trip (and a Turbine Trip).

0.2.5.2.5. High core power level

The core power level signal is derived from an enthalpy balance using:

- loop temperature measurements (cold and hot leg temperature sensors),
- the primary pressure,

- the core flow rate (derived from Reactor Coolant Pump speed sensors).

This function initiates a Reactor Trip (and a Turbine Trip).

0.2.5.2.6. Low reactor coolant flow rate (single loop)

Partial loss of forced reactor coolant flow and the reactor coolant pump shaft break or locked rotor are detected using the loop flow indication. At present, measuring principles applied are based on loop flow measurements from differential pressure sensors.

This function initiates a Reactor Trip (and a Turbine Trip).

0.2.5.2.7. Low Reactor Coolant Pump speed (loss of 4 Reactor Coolant Pumps)

For events affecting the electrical supply to all the Reactor Coolant Pumps, a specific protection channel is required with a high accuracy and a minimal response time.

Reactor Coolant Pump speed information is used to derive the core flow rate for such events.

0.2.5.2.8. High neutron flux (source range)

This function limits the consequences of reactivity increase events (especially homogeneous dilution) in shutdown conditions. The signal used is the neutron flux derived from the source range detectors. This function initiates a shutdown high neutron flux alarm.

For low coolant inventories (mid-loop operation) a permissive function based on loop level measurements allows the high neutron flux protection channel to automatically isolate the RCV [CVCS] (this is the same F1A actuation used for the low doubling time protection channel and the antidilution at power protection channel). Isolation occurs, if the signal (alarm) is still present, after a time delay which depends on the type of fuel management scheme and the maximum dilution flow rate assumed.

0.2.5.2.9. High neutron flux doubling time (source range)

This function limits the consequences of reactivity increase events (especially homogeneous dilution) in shutdown conditions. The signal used is the nuclear flux derived from the source range detectors.

During hot and normal cold shutdown states this function isolates the main source of spurious dilution by closing two redundant valves downstream the VCT.

0.2.6. Safety systems characteristics

In PCC accident analyses, all systems claimed to mitigate the consequences of an event must be F1-classified:

- F1A-classified systems are used to bring the plant to the controlled state,
- F1B-classified systems may be used (in addition to F1A systems) to bring the plant to the long term safe shutdown state.

The F1 mechanical-systems assumed for in the BDR accident analyses are:

- the core shutdown rods for reactor trip,
- the RCP [RCS] and SG isolation valves,
- the RCP [RCS] and SG fluid systems performing injection,
- the RCP [RCS] and SG fluid systems performing pressure relief,
- the RCV [CVCS] control tank isolation valves.

For the accident studies in this appendix, these systems are modelled using the conservative PCC analyses rules defined in section 1 of this appendix:

- assuming minimum guaranteed effectiveness,
- considering the most adverse additional single failure,
- considering the most adverse unavailability due to preventative maintenance.

Appendix 14B.0.2 - Table 13 to 22 provide the minimum or/and maximum characteristics of the F1 fluid-systems claimed in accident analyses, giving information pertinent to systems used in PCC accident analyses ,as well as systems relevant to overpressure protection analyses and RRC-A accident analyses.

The F1A systems involved in accident analyses (excluding support systems such as RRI [CCWS], SEC [ESWS] etc.) are:

- RT,
- RIS [SIS] (MHSI, LHSI, accumulator, IRWST),
- PSV,
- VDA [MSRT] (including partial cooldown),
- MSSV,
- VIV [MSIV],
- ASG [EFWS],
- Containment / RCP [RCS] isolation,
- ARE [MFWS] full load isolation,
- ARE [MFWS] low load isolation,
- SG blowdown isolation,
- Reactor Coolant Pump trip,

- Main diesel start-up,
- Dilution isolation,
- RBS [EBS]

The F1B systems involved in accident analyses (other than F1A systems previously listed) are:

- VIV [MSIV] bypass (opening function),
- ASG [EFWS] passive headers,
- ASG [EFWS] isolation,
- LHSI in hot leg SI-mode (after switchover to CL(I)/HL(I)),
- LHSI in RHR-mode (LHSI/RHR operation),
- Reactor Coolant Pump manual cut-off,
- MHSI manual cut-off,
- accumulators manual isolation,
- VDA [MSRT] manual operation,
- SG blowdown between 2 SG.

The characteristics of the shutdown rods are already provided in Appendix 14B.0.2 - Table 8.

In addition to the F1 systems, non-F1 systems may be taken into account in PCC accident analyses according to PCC analyses rules (either negative impact, or positive impact but not experiencing some discontinuity in their operation). Those systems may also be used in non-PCC accident analyses. The main relevant system characteristics assumed are listed in Appendix 14B.0.2 - Table 23.

Appendix 14B.0.2 - Table 24, summarises the key safety functions that need to be achieved to ensure a safe control of the plant, and the associated systems providing these functions.

Appendix 14B.0.2 – Figure 4 provides a simplified functional sketch of the main F1A fluid systems: RIS [SIS], ASG [EFWS], PSV, MSSV, VDA [MSRT], main steam isolation, main feed water isolation.

0.2.7. Computer codes used

A summary description of some of the principal computer codes used in transient analyses is given in Appendix 14A. Other specialised codes developed to simulate given accident conditions, such as analysis of reactor system pipe rupture, are described in the appropriate accident analysis sections.

The computer codes used for calculations are specified for each of the accident analyses in this appendix.

0.2.8. Approach used in accident analysis with respect to DNB prediction

Predictions of DNB in transient analysis depend on:

- the type of the protection function actuated,
- (low DNBR protection or specific protection),
- the LCO function which operates to limit the initial conditions,
- (low DNBR LCO or other),
- the method by which DNBR uncertainties are combined.

With regard to DNB prediction, three kinds of transients are considered:

1. Transients actuating low DNBR protection.
   The initial DNBR can be set to any value (no fixed initial DNBR), and the DNBR design limit is equal to 1.0. These transients are named 'type 1 transients'.

2. Transients actuating specific protection:

2.1 If the initial state is at power the initial MIN. DNBR is fixed to the 'DNBR limiting value' such that for bounding transient of this type, the minimum DNBR in the transient meets the DNBR design limit (equal to1.0).

These transients are named 'type 2 transients'.

Taking into account all the uncertainties arising in the DNBR calculation (U) due to the surveillance system, the 'DNBR limiting value' allows the site DNBR-LCO (Limiting Condition of Operation) value to be defined as:

DNBR-LCO = DNBR limiting value x U

2.2 If the initial state is at zero power, the minimum DNBR is calculated by applying uncertainties in a deterministic way. This type of transient is referred to as a 'type 3 transient'.

Appendix 14B.0.2 - Table 25 summarises the overall approach for the three kinds of transients.

The DNBR analyses use the following definitions:

- Safety criterion : Radiological limits
- DNBR decoupling criteria :
  - PCC-2 : No DNB (i.e. DNBR > DNBR design limit)
  - PCC-3, PCC-4 : Percentage of rods in DNB not higher than 10% of the total core
- $DNBR_{RT}$: Actuation threshold for the low DNBR protection function without considering uncertainties (see Sub-chapter 4.4)
- On site $DNBR_{RT}$: Actuation threshold for the low DNBR protection function allowing for uncertainties (see Sub-chapter 4.4)

- DNBR limiting value : Initial DNBR for type 2 transients (see Sub-chapter 4.4)
- $DNBR_{LCO}$: Actuation threshold for the low DNBR LCO function allowing for uncertainties (see Sub-chapter 4.4)
- DNBR design limit : The limit applied in each transient, above which the DNBR criterion is met.

  The design limit is the limiting value against which the DNBR calculated by the thermal-hydraulic design code is compared.

The transients presented in this appendix for which DNB is relevant, are classified as follows:

Type I transients:

- Uncontrolled RCCA bank withdrawal at power (PCC-2) (see section 2.10)
- Rod drop (PCC-2)
- Single RCCA withdrawal at power (PCC-3)
- Uncontrolled boron dilution [1] (PCC-2)
- Depressurisation of Main steam system at power [1] (PCC-2) (see section 2.14)

Type II transients:

- Uncontrolled RCCA bank withdrawal at power (very fast reactivity insertion rates) (PCC-2) (see section 2.10)
- Inadvertent closure of 4 VIV [MSIV] (PCC-3) (see section 2.4.1)
- Inadvertent closure of 1 VIV [MSIV] (PCC-3) (see section 2.4.2)
- Loss of non-emergency AC power to the plant auxiliaries (PCC-2) (see section 2.5.1)
- Forced decrease of reactor coolant flow (PCC-3) (see section 2.6)
- Partial loss of reactor coolant flow (PCC-2) (see section 2.7)
- Reactor coolant pump shaft break or locked rotor (PCC-4) (see section 2.8)
- Rod ejection (at power) [2] (PCC-4)

Type III transients:

- Uncontrolled RCCA withdrawal from subcritical condition (PCC-2)

[1] With regard to DNB behaviour in the transient, these events are covered by uncontrolled RCCA bank withdrawal at power

[2] To give a pessimistic assessment this transient is studied as a type III transient

- Rod ejection (from zero power) (PCC-4)
- Main steam line break (PCC-4) (see section 2.15)

APPENDIX 14B.0.2 – TABLE 1

Main Geometrical Data

CORE		
- number of fuel assemblies	241	(type 17x17)
- active height	4.20 m	
RCP [RCS] FLUID VOLUMES		
- RPV	150 m^3	
. RPV downcomer + lower plenum		53.5 m^3
. core		27.5 m^3
. RPV upper plenum + upper head		68.5 m^3
- Pressuriser	75m^3	
- RCP [RCS] loops	238 m^3	
. surge line		2.4 m^3
. hot/U/cold legs (incl. RCP [RCS])		14.9 m^3 x 4
. SG plenum and tubes		43.9 m^3 x 4
- TOTAL RCP [RCS]	463 m^3	
RCP [RCS]		
- surge line diameter (inside)	0.325 m	
- hot/U/cold leg diameter (inside)	0.780 m	
SG (per SG)		
- number of tubes	5980	
- heat transfer area	8171 m²	
- tube (inside diam./outside diam./thickness)	16.87/19.05/1.09 mm	
- secondary side fluid volume	247 m^3	
- steam flow limiter at SG outlet	0.13 m²	

APPENDIX 14B.0.2 – TABLE 2

Plant initial conditions and Maximum steady-state errors

Parameter	Nominal value at thermal-hydraulic flow rate		Maximum
	at 0% FP	at 100% FP	steady-state error
- core power	0% FP	100% FP	± 2% FP
- Pressuriser pressure	155 bar	155 bar	± 2.5 bar
- RCP [RCS] avg. temp.	301°C(1)	311.25°C(2)	± 2.5°C
- Pressuriser level	28% R	56% R	± 5% R
	(21 m^3)	(40 m^3)	(3.4 m^3)
- SG level	56% NR	56% NR	± 5% NR
	(16.2 m)	(16.2 m)	(0.35 m)
- SG water mass	116.3 t	87.6 t	

FP = Full Power
R = Range
NR = Narrow Range

(1) the corresponding SG saturation pressure is 87.1 bar abs
(2) the corresponding SG saturation pressure is 74.6 bar abs

APPENDIX 14B.0.2 – TABLE 3

Plant initial conditions and other pertinent parameters (at 100% FP)

Nominal NAAD [SSS] thermal power output	4930 MWth	
Nominal core thermal power	4900 MWth	
ARE [MFWS] flow temperature at SG inlet	230°C	
Main steam flow rate (per SG)	694 kg/s	
Steam moisture content at SG outlet (MAX)	0.25%	
Core average linear power(1)	178.6 W/cm	
	Thermalhydraulic flow rate	**Best-estimate flow rate**
RCP [RCS]-loop flow rate	26987 m3/h	28111 m^3/h
RCP [RCS] flow rate (4 loops)	22240 kg/s	23148 kg/s
Core flow rate bypass(2)	5.5%	3.4%
Core inlet temperature	292.5°C	292.8°C
Core outlet temperature	331.9°C	330.1°C
RPV outlet temperature	330.0°C	329.0°C
RPV dome temperature(3)	330.0°C	329.0°C

Note : Minor deviations from these values due to code specificities may occur in some accident analyses.

(1) related to power generated inside the fuel, corresponding to 97.4% of total power
(2) related to hot dome design
(3) RPV dome temperature is conservatively taken equal to the RPV outlet temperature (the highest possible one), while the realistic temperature is close to the average RCP [RCS]-loop temperature.

APPENDIX 14B.0.2 – TABLE 4

Conservative reactivity dataset used with point kinetics model (bounding data for all UO2 and MOX fuel management schemes)

MODERATOR DENSITY COEFFICIENT (1)

MIN (2)	MAX
0 ΔK/K per g/cm^3	0.49 ΔK/K per g/cm^3

DOPPLER TEMPERATURE COEFFICIENT

MIN	MAX
- 4.1 pcm/°C	**- 1.9 pcm/°C**

DOPPLER POWER COEFFICIENT

MIN	MAX
- 10.3 pcm / % FP at 100% FP	- 6.2 pcm / % FP
- 12.2 pcm / % FP at 75% FP	- 6.8 pcm / % FP
- 17.4 pcm / % FP at 50% FP	- 7.0 pcm / % FP
- 26.3 pcm / % FP at 25% FP	- 7.7 pcm / % FP
- 36.5 pcm / % FP at 5% FP	- 8.1 pcm / % FP

FP = Full Power

BORON DIFFERENTIAL WORTH

MIN	MAX
hot shutdown / cold shutdown	**hot shutdown / cold shutdown**
- 12.2 / - 17.1 pcm/ppm for 25 ppm	- 5.2 / - 8.5 pcm/ppm
- 10.6 / - 14.0 pcm/ppm for 1000 ppm	- 4.7 / - 7.2 pcm/ppm
- 9.2 / - 11.6 pcm/ppm for 2000 ppm	- 4.2 / - 6.0 pcm/ppm
- 8.1 / - 9.8 pcm/ppm for 3000 ppm	- 3.8 / - 5.1 pcm/ppm

Note : for ATWS analyses, a specific data set applies.

(1) More realistic values for a nominal coolant density are:
- BOL : 0.09 Δk/k per g/cm^3
- EOL : 0.32 Δk/k per g/cm^3

(2) Decoupling value

APPENDIX 14B.0.2 – TABLE 5

RCP [RCS] Boron Concentration (with uncertainties)

Plant operating conditions	EOL		BOL	
	UO2	MOX	UO2	MOX
Nominal operating conditions (including maximum Xenon build-up)	10 ppm	10 ppm	675 ppm	1660 ppm
RRA [RHRS] connection conditions (core subcritical at 150°C, (N-1) rods inserted)	530 ppm	620 ppm	910 ppm	1765 ppm

These data refer to natural boron, not accounting for B10 enrichment. They are chosen so that the difference in boron concentration between the nominal operation (full power) and the RHR-connection conditions (150°C) is maximised.

Accounting for the following B10 enrichments retained for $U0_2$ and MOX, the correspondence between 'enriched boron C%' and 'natural boron C%' is respectively:

	$U0_2$	**MOX**
B10 C% in enriched boron	28.5%	33.0%
Enriched boron C%	natural boron C% x 0.693	natural boron C% x 0.596

APPENDIX 14B.0.2 – TABLE 6

Residual Decay Heat (Term B + C)

Term B + C = decay of fission products and actinides

MAXIMUM DECAY HEAT (MAX) = refers to 'ORIGEN-S results + uncertainties'

Uncertainties for UO_2 fuel management: + 15% before 10 mn, + 10% after 10 mn.

Uncertainties for MOX fuel management: +15% all times.

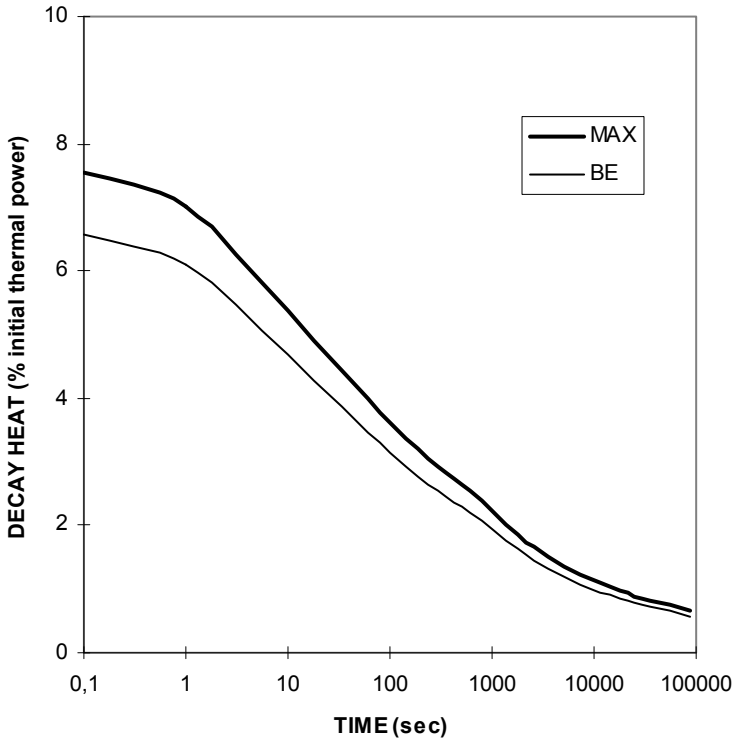
These uncertainties correspond to a 95% probability for a one sided distribution (1.645 σ), at a confidence level of 95%.

BEST ESTIMATE DECAY HEAT (B.E.) = refers to 'ORIGEN-S without uncertainties'

In both cases, the decay heat curve is enveloping for UO2 and MOX fuel management schemes.

MAX decay heat is used for PCC accident analyses.

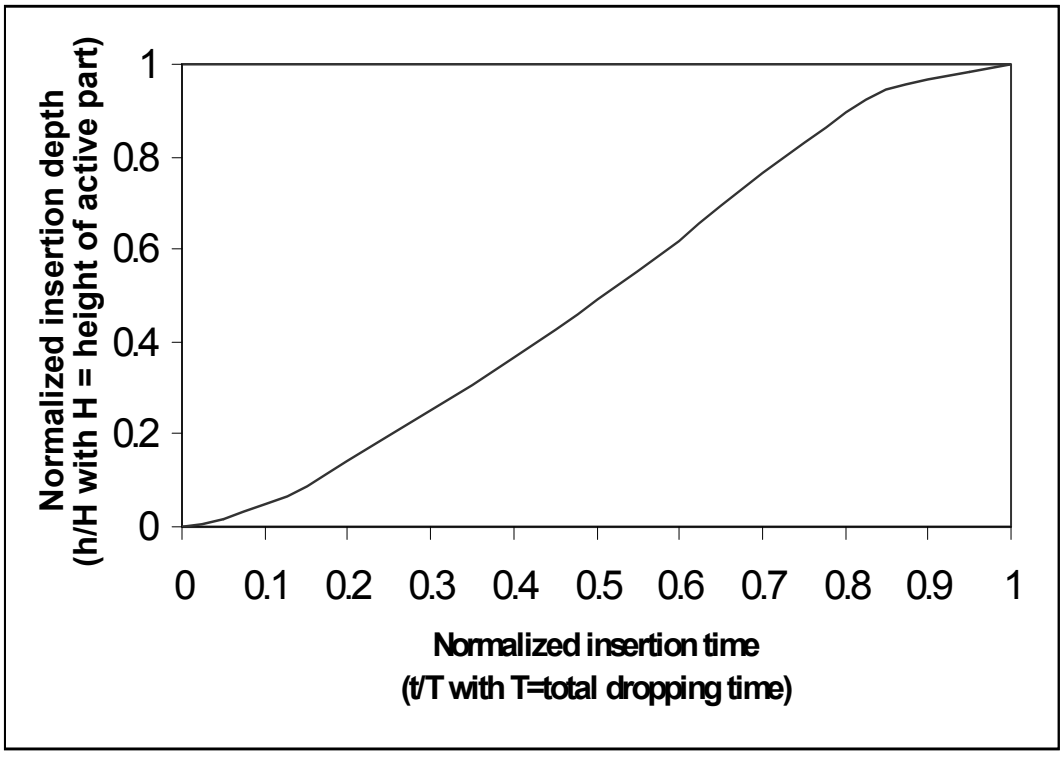
BE decay heat is used for RRC-A accident analyses.



APPENDIX 14B.0.2 – TABLE 7

RCCA drop characteristics (RT)

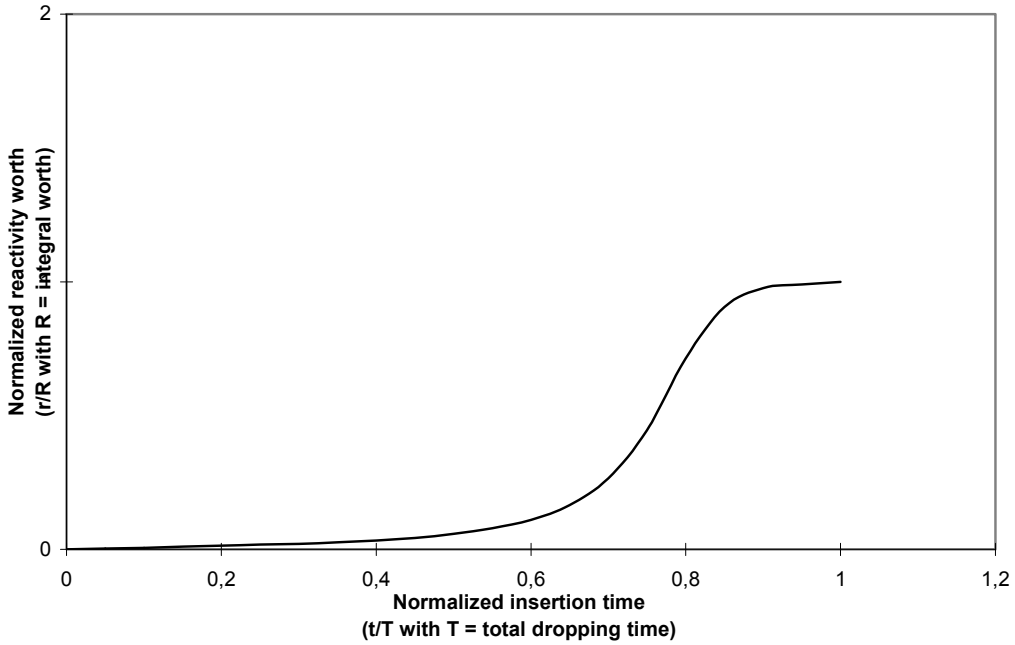
- Total drop time (MAX)	3.5 s without earthquake 5.0 s with earthquake
- Height of active core	4.20 m
- Insertion depth versus time (MIN)	see figure, below



APPENDIX 14B.0.2 – TABLE 8

- Total drop time (MAX)	3.5 s without earthquake 5.0 s with earthquake
- Integral reactivity worth (MIN) - with (N-1) rods	4000 pcm (1)
- Reactivity worth versus time (MIN)	see figure, below

(1) Enveloping value for UO_2 and MOX fuel management schemes



APPENDIX 14B.0.2 – TABLE 9

F1 Signals (P/S Related)

PRESSURISER PRESSURE SCALE

	ACTION	SETPOINT	UNCERTAINTY
(1)	3rd PSV opening (closing)	178 bar (168 bar)	± 1.5 bar
(1)	2nd PSV opening (closing)	178 bar (168 bar)	± 1.5 bar
(1)	1st PSV opening (closing)	174 bar (164 bar)	± 1.5 bar
MAX2	RT, TT	166.5 bar	± 1.5 bar
MIN2	RT, TT	135 bar	Normal conditions ± 1.5 bar Degraded conditions ± 3 bar
MIN3	RIS [SIS] (3) Partial cooldown (2) RCP [RCS] isolation	115 bar	Normal conditions ± 1.5 bar Degraded conditions ± 3 bar

(1) Included for completeness although not I&C related (hydraulic opening)
(2) Decrease of all VDA [MSRT] setpoints from 93 bar to 60 bar, at a rate of -100°C/h
(3) In state A (start of MHSI + LHSI)

PRESSURISER LEVEL SCALE

	ACTION	SETPOINT	UNCERTAINTY
MAX1	RT, TT	85% of MR (10.1 m from bottom of PZR)	Normal conditions ± 2% Degraded conditions ± 5%

MR = Measuring Range

RCP [RCS] LOOP LEVEL SCALE

	ACTION	SETPOINT	UNCERTAINTY
MIN	MHSI (*)	see section 2.2.3 and 4 within this appendix	see section 2.2.3 and 4 within this appendix

(*) in states C and D, with RCP [RCS] off (large mini flow line of MHSI already open)

SG PRESSURE SCALE

	ACTION	SETPOINT	UNCERTAINTY
(1)	MSSV opening in SGa	102.5 bar	± 1.5 bar
MAX1	RT, TT MSRIV opening in SGa	93 bar	± 1.5 bar
MIN1	RT, TT VIV [MSIV] closure, ARE [MFWS] full load isolation in all SG	50 bar	± 1.5 bar
MIN2	ARE [MFWS] low load isolation in SGa VDA [MSRT] isolation in SGa	40 bar	± 1.5 bar

(1) Indicated, however not being I&C related (spring-loaded valve)
SGa: Steam Generator affected

SG PRESSURE DROP SCALE

	ACTION	SETPOINT	UNCERTAINTY
MAX2	ARE [MFWS] low load isolation in SGa	- 2 bar/min (1) Variable limit, setpoint 17 bar below the actual pressure in steady state Maximum value 60 bar	± 1.5 bar (on setpoint)
MAX1	RT, TT VIV [MSIV] closure ARE [MFWS] full load isolation in all SG	- 2 bar/min (1) variable limit, setpoint 7 bar below the actual pressure in steady state Maximum value 70 bar	± 1.5 bar (on setpoint)

(1) See Appendix 14B.0.2 - Figure 3 and section 0.2.5.1.2
SGa : Steam Generator affected

SG LEVEL SCALE

	ACTION	SITE SETPOINT	UNCERTAINTY
MAX2	VIV [MSIV] closure in SGa VDA [MSRT] setpoint increase in SGa (if partial cooldown finished)	18.1 m (84% NR)	Normal conditions ± 2% of MR Degraded conditions ± 5% of MR
MAX2	Partial cooldown (1) in all SG	18.1 m (84% NR)	Normal conditions ± 2% of MR Degraded conditions ± 5% of MR
MAX1	ASG [EFWS] isolation in SGa (if ASG [EFWS] has started)	17.2 m (89% WR)	Normal conditions ± 2% of MR Degraded conditions ± 5% of MR
MAX1	RT, TT ARE [MFWS]/AAD [SSS] isolation in SGa (i.e. closure of 3 isolation valves)	17.2 m (71% NR)	Normal conditions ± 2% of MR Degraded conditions ± 5% of MR
MIN1	RT, TT	13.4 m (15% NR)	Normal conditions ± 2% of MR Degraded conditions ± 5% of MR
MIN2	ASG [EFWS] actuation in SGa	8m (40% of WR cold side)	Normal conditions ± 2% of MR Degraded conditions ± 5% of MR

(1) Decrease of all VDA [MSRT] setpoints from 93 bar to 60 bar at a rate of -100°C/h
WR: Wide Range, calibrated at hot standby on cold side
NR: Narrow Range, calibrated at 100% power
MR: Measuring Range
SGa: Affected Steam Generator

ΔP OVER RCP [RCS]

	ACTION	SETPOINT	UNCERTAINTY
MIN	RCP [RCS] trip	80% of nominal ΔP over RCP	Normal conditions ± 3% Degraded conditions ± 5%

HOT LEG SATURATION MARGIN ΔPSAT

	ACTION	SETPOINT	UNCERTAINTY
MIN	MHSI + LHSI (*)	See section 2.2.2 and 3 in this appendix	See section 2.2.2 and 3 in this appendix

(*) in state B, in state C with RCP [RCS] on (large mini flow line of MHSI already open, in state C)

APPENDIX 14B.0.2 – TABLE 10

F1 Signals (Core Related)

CORE PROTECTION

F1 SIGNAL	LIMIT	ACTION	SETPOINT (Note1)
High linear power density	MAX	RT, TT	590 W/cm
Low DNBR	MIN	RT, TT	1.0
Anti dilution at power	MIN	Stop/start RCV [CVCS] isolation	to be defined
High neutron flux rate of change	MAX	RT, TT	13% NP (to be confirmed)
High core power level	MAX	RT, TT	120% NP (to be confirmed)
Low reactor coolant flow rate (loop 1) (nuclear power level above a permissive threshold)	MIN	RT, TT	25% NF
Low reactor coolant flow rate (loop 2) (nuclear power level above a permissive threshold)	MIN	RT, TT	25% NF
Low reactor coolant flow rate (loop 3) (nuclear power level above a permissive threshold)	MIN	RT, TT	25% NF
Low reactor coolant flow rate (loop 4) (nuclear power level above a permissive threshold)	MIN	RT, TT	25% NF
Low Reactor Coolant Pump speed - Loss of Reactor Coolant Pumps	MIN	RT, TT	91% NS
High neutron flux (source)	MAX	Alarm	(Notes 2 & 3)
High neutron flux doubling time (source)	MIN	Stop/start RCV [CVCS] isolation	150 s

NP: Nominal Power
NF: Nominal Flow rate
NS: Nominal Speed

(1) Setpoints used in safety analyses
(2) Flux value equal to three times the current flux in shutdown conditions
(3) Automatic actions (stop/start RCV [CVCS] isolation) during cold shutdown for refuelling or maintenance.

APPENDIX 14B.0.2 – TABLE 11

Assumed I&C Signal Delays (PS Related)

Type of channel	$T_{I\&C}$ (MAX)	T_{RT} (MAX)	T_{SA} (MAX)
Pressure	0.9 s	1.2 s	0.9 s + T_D
Level	1.5 s	1.8 s	1.5 s + T_D
Temperature	4.5 s	4.8 s	4.5 s + T_D

$T_{I\&C}$ = response time of I&C channel, including sensor (up to the actuator)

T_{RT} = total delay to start RCCA dropping ($T_{I\&C}$ + 0.3 sec)
0.3 sec = RT breakers opening + gripper release

T_{SA} = total delay to achieve full completion of safeguard action (=$T_{I\&C}$ + T_D)
For definition of T_D see Appendix 14B.0.2 - Table 12

I&C Signal Delays (Core Related)

Type of channel	T (I&C) (MAX)	T (RT) (MAX)
High linear power density	0.6 s	0.9 s
Low DNBR (including SPND response time)	0.6 s	0.9 s
Anti dilution at power	0.5 s	(1)
High neutron flux rate of change (2)	0.3 s	0.6 s
High core power level (3)	0.5 s	0.8 s
Low reactor coolant flow rate (loop i)	0.9 s	1.2 s
Low Reactor Coolant Pump speed - Loss of Reactor Coolant Pumps	0.3 s	0.6 s
High neutron flux (source)	0.3 s	0.6 s
High neutron flux doubling time (source)	0.3 s	0.6 s

T (I&C) : response time of I&C channel, including sensor (up to the actuator)
T (RT) : total delay to start RCCA dropping (T (I&C) + 0.3 s)

(1) : The isolation of the RCV [CVCS] is done in 40 s including the T (I&C)
(2) : Time constant = 30 seconds
(3) : Not including sensor response time

APPENDIX 14B.0.2 – TABLE 12

Safeguard Action Delays

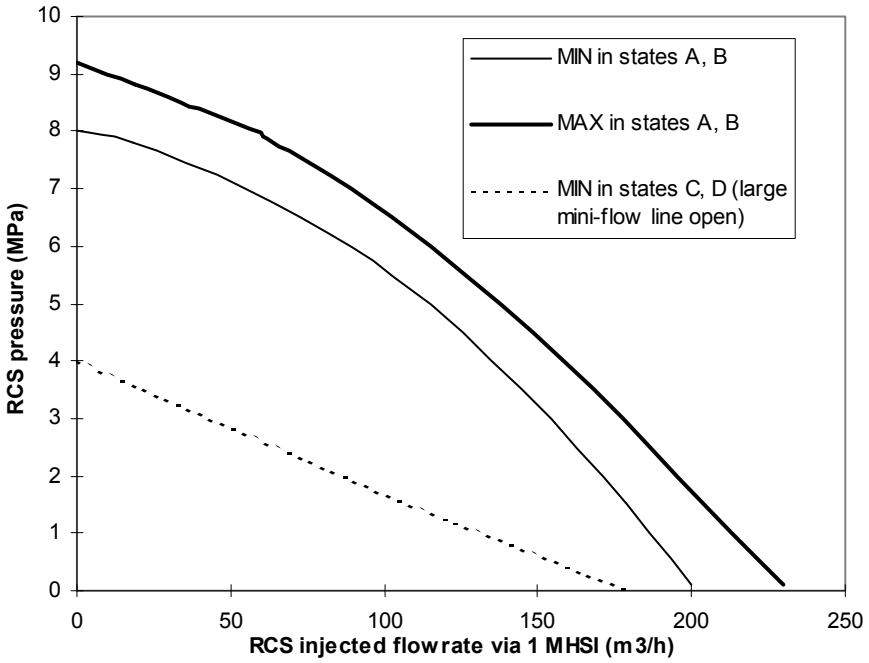
Safeguard action	Delay T_D (MAX)	Reason
RIS [SIS] actuation	15 s w/o LOOP	Pump start-up to full flow rate
(MHSI & LHSI)	40 s with LOOP	EDG reloading sequence
ASG [EFWS] actuation	15 s w/o LOOP 50 s with LOOP	Pump start-up to full flow rate EDG reloading sequence
ARE [MFWS] full load isolation	10 s	Valves closing delay
ARE [MFWS] low load isolation	10 s	Valves closing delay
ASG [EFWS] isolation	5 s	Valves closing delay
VIV [MSIV] closure	5 s	Valve closing delay
VDA [MSRT] opening	dead time 1.5 s opening time 0.5 s	MSRIV opening delay
VDA [MSRT] isolation	5 s	MSRIV closing delay
$Cont^t$/RCP [RCS] isolation	30 s	Valves closing delay
RCP [RCS] cut-off	0.15 s	Breaker opening delay
Turbine trip	0.3 s	Turbine valves closing delay

Note : except for VDA [MSRT] opening, all the safeguard actions are modelled as steps at time T_D (pessimistic approach).

APPENDIX 14B.0.2 – TABLE 13

RIS [SIS] Characteristics (MHSI)

- Number of MHSI trains	4
- Location	Separated divisions
- Suction from	IRWST
- Injection into	RCP [RCS] cold leg
- MHSI injection temperature (MIN/MAX)	10°C/50°C (if no IRWST heating)
- MHSI injection flow rate (MIN/MAX)	See figures, below



APPENDIX 14B.0.2 – TABLE 14

RIS [SIS] Characteristics (LHSI)

- Number of LHSI trains	4
- Location	Separated divisions
- Suction from	IRWST
- Injection into	RCP [RCS] cold leg RCP [RCS] hot leg (in LOCA long term)
- Heat exchanger (MIN)	1.3 MW/°C
- LHSI injection temperature (MIN/MAX)	10°C/50°C (if no IRWST heating)
- LHSI injection flow rate (MIN)	See figure, below
- LHSI flow rate towards IRWST (MIN)	180 m^3/h at RCP [RCS] $\geq$ 20 bar

APPENDIX 14B.0.2 – TABLE 15

RIS [SIS] Characteristics (accumulator)

- Number of accumulators (ACCU)	4
- Location	Separated trains
- Injection into	RCP [RCS] cold leg
- ACCU injection temperature (MIN/MAX)	10°C/50°C
- ACCU initial pressure (MIN/MAX)	45/50 bar
- ACCU initial water volume (MIN/MAX)	30/35 m^3
- ACCU total volume	47 m^3
- ACCU line pressure loss (MIN/MAX)	1700/2500 m^{-4}
- ACCU boron concentration (MIN/MAX)	see IRWST boron concentration on Appendix 0.2 - Table 16

APPENDIX 14B.0.2 – TABLE 16

RIS [SIS] Characteristics (IRWST)

- IRWST initial water volume (MIN/MAX) 1850 m^3 / 1940 m^3
The maximum volume that could be trapped in the reactor building (excluding reactor pit) in course of LOCA is 400 m^3.

- IRWST initial water temperature (MIN/MAX) 10°C/50°C

- IRWST initial boron concentration (MIN) :

	UO2	MOX
Natural boron	MIN: 2300 ppm MAX: 2600 ppm	MIN: 2700 ppm MAX: 3000 ppm
Enriched boron (1)	MIN: 1600 ppm MAX: 1800 ppm	MIN: 1600 ppm MAX: 1800 ppm

(1) The conversion of the boron concentration from natural boron to enriched boron is given by:
- C (enriched boron) = 0.693 x C (natural boron) for UO2
- C (enriched boron) = 0.596 x C (natural boron) for MOX

APPENDIX 14B.0.2 – TABLE 17

ASG [EFWS] Characteristics

- Number of ASG [EFWS] trains	4 (1 per SG)
- Location	Separated divisions, with passive cross-headers
- Suction from	Dedicated ASG [EFWS]-tank, with passive cross-headers
- Injection into	Dedicated SG
- ASG [EFWS] injection temperature (MIN/MAX)	10°C/50°C
- ASG [EFWS] tank effective water volume (MIN)	375 m^3 per tank
- ASG [EFWS] injection flow rate (MIN/MAX)	See table, below

SG pressure	ASG [EFWS] injection flow rate per train (MIN)	(MAX)
104 bar (MSSV setpoint, with uncertainties)	30 m^3/h	Not relevant
94.5 bar (MSRV setpoint, with uncertainties)	93.5 m^3/h	130 m^3/h (1)
1 bar	93.5 m^3/h	200 m^3/h (1)

(1) If the active limitation is OFF
If the active limitation is ON, the maximum ASG [EFWS] flow rate is 120 m^3/h per train whatever the SG pressure

APPENDIX 14B.0.2 – TABLE 18

VIV [MSIV] Bypass

- Number of VIV [MSIV] bypass trains	4 (1 per SG)
- Location	In parallel to the VIV [MSIV] (MSS warm-up line)
- suction from	upstream VIV [MSIV]
- relief into	downstream VIV [MSIV]
- equipment per train	2 isolation valves in series
- VIV [MSIV] bypass relief flow rate (MIN)	4 Kg/s saturated steam at SG = 5 bar (upstream VIV [MSIV] -BP) and MSH = 1 bar (downstream VIV [MSIV] -BP)

APPENDIX 14B.0.2 – TABLE 19

VDA [MSRT] Characteristics

- Number of VDA [MSRT], per SG	1
- Location	MS line, outside containment
- Suction from	MS line, upstream VIV [MSIV]
- Relief into	Environment
- Equipment, per train	1 isolation valve (MSRIV), initially closed, and 1 control valve (MSRCV), initially open, in series
- VDA [MSRT] relief flow rate (MIN/MAX) per train	50% / 55% nominal steam flow (1250 / 1375 t/h saturated steam) under 97 bar
- VDA [MSRT] opening dynamics (MAX) refers to MSRIV, closed at standby MSRCV is fully open at standby	1.5 sec dead time 0.5 sec opening time
- Partial cooldown characteristics	From 93 bar to 60 bar, with a rate corresponding to -100°C/h

APPENDIX 14B.0.2 – TABLE 20

MSSV Characteristics

- Number of MSSV trains, per SG	2
- Location	MS line, outside containment
- Suction from	MS line, upstream VIV [MSIV]
- Relief into	Environment
- Equipment, per train	1 spring-loaded safety valve
- MSSV relief flow rate (MIN) per train	25% nominal steam flow (625 t/h saturated steam) under 97 bar
- MSSV opening dynamics (MAX)	3% accumulation

APPENDIX 14B.0.2 – TABLE 21

RBS [EBS] Characteristics

- Number of RBS [EBS] trains	2
- Location	Separated trains
- Injection into	RCP [RCS] cold leg (between RPV and RCP [RCS]) each RBS [EBS] train injects into 2 cold legs, with 1 isolation valve per cold leg injection line
- RBS [EBS] tank water content (MIN) per train	27 m^3
- RBS [EBS] injection flow rate (MIN/MAX) per train	2.8 / 3.2 Kg/s up to 180 bar RCP [RCS] pressure
- Injected boron concentration (1)	Enriched boron : 7000 ppm Natural boron UO2 : 10100 ppm Natural boron MOX : 11750 ppm

(1) The conversion of the boron concentration from natural boron to enriched boron is given by:
- C (enriched boron) = 0.693 x C (natural boron) for UO2
- C (enriched boron) = 0.596 x C (natural boron) for MOX

APPENDIX 14B.0.2 – TABLE 22

PSV Characteristics

- Number of PSV trains	3
- Location	Top of pressuriser
- Suction from	Pressuriser steam phase
- Relief into	IRWST
- Equipment, per train	2 safety valves in series
- PSV relief flow rate (MIN) per train	300 t/h sat. steam under 176 bar 450 t/h sat. liquid under 176 bar
- PSV opening / closing dynamics (MAX)	0.5 s dead time 1.5 s opening time 1.5 s closing time

APPENDIX 14B.0.2 – TABLE 23

Non-F1 Systems Characteristics assumed in PCC analysis

Partial trip (PT)	
- Total drop time	3.5 s
- Integral reactivity worth	depends on the I&C limitation functions actuating the RCCAs.
- Reactivity worth versus time	same as RT
Pressuriser normal spray (3 stages)	
- Setpoint	156/158/160 bar
- Capacity per stage (MIN)	2 x 10 / 25 / 25 Kg/s
- Capacity per line (MIN/MAX)	25 / 35 kg/s
- Opening time (MAX)	10 / 2 / 2 sec
Pressuriser heaters	
- Total heating power	2500 kW
- Emergency power supplied heating	300 kW
MSB	
- Steam relief capacity (MIN)	50% nominal steam flow (5000 t/h saturated steam) under 71 bar
ARE [MFWS]	
- Flow rate of low-load line after RT (MIN/MAX)	0% / 20% nominal ARE [MFWS] flow
- Closing time of high-load line after RT (MIN/MAX)	0 s / 10 s (step)
AAD [SSS]	
- Injection rate (only one pump) (MIN/MAX)	350 / 400 m³/h
RCV [CVCS] (1)	
- Net RCP [RCS] injection flow rate, 1st step	
Pressuriser level > reference + dead band	- 10 Kg/s
Pressuriser level < reference + dead band	+ 10 Kg/s
- Net RCP [RCS] injection flow rate, 2nd step	
Pressuriser level > 100% R (~ 70 m3)	- 20 Kg/s (injection isolation)
Pressuriser level < 12% R (~ 10 m3)	+ 20 Kg/s (2 injection pumps)

(1) The simulation of the RCP [RCS]-inventory control as presented is simplified, but is considered to be bounding

.

APPENDIX 14B.0.2 – TABLE 24

Safety Functions and Associated Systems

	C/R Control of fuel integrity at power	C/R Control of core reactivity at shutdown	C/I Control of RCS water inventory	C/T Control of RCS water temperature	C/P Control of RCS pressure	C/C Control of Containment	
PCC	**RCCA**	**dedicated RT** **dedicated isolation** Short term : **RCCA** Long term : **RBS [EBS]**	**dedicated RT** **dedicated isolation** **RCP [RCS] trip** SB-LOCA : **MHSI + MSRT** LB/IB LOCA : **RIS [SIS]**	**dedicated RT** **dedicated isolation** Hot shutdown : **ASG [EFWS] + VDA [MSRT]** Cold shutdown : **LHSI in RHR-mode**	**dedicated RT** **dedicated isolation** overpressure protection : **RT + PSV + MSSV** Depressurization to cold shutdown : **PSV**	**dedicated RT** **dedicated isolation** Containment heat removal : **LHSI** SGTR bypass : **Dedicated design of MHSI, MSRT, MSSV VIV [MSIV]-bypass, APG [SGBS]**	F1 SYSTEMS
RRC	Loss of RCCA : **Core reactivity** **feedback, RCV [CVCS], RBS [EBS]**	Loss of EBS : **RCV [CVCS], RIS [SIS]**	SB-LOCA w/o MHSI : **ACCU + LHSI + MSRT** SB-LOCA w/o MSRT : **MHSI + MSB**	Loss of EFWS : **SSS** Loss of EFWS + SSS + MFW : **RIS [SIS] + RCV [CVCS] + PSV** Loss of MSRT : **MSB** Loss of LHSI in RHR-mode **ASG [EFWS] + MSRT** **RIS [SIS] + PSV**	Loss of RT: **PSV + MSSV + MSRT** Loss of PSV : **PZR spray, PT** Loss of PSV for depr: **PZR / RCV [CVCS] spray** **RCV [CVCS] letdown**	Loss of LHSI : **EVU [CHRS]** SGTR bypass : **MSB**	BACK UP (1)

(1) Only relevant for the most probable events

APPENDIX 14B.0.2 – TABLE 25

Principal Approaches used in Accident Analyses involving DNBR Prediction

Category of the transients	Aim of the transients analyses	Consideration of Uncertainties		DNBR decoupling criterion	DNBR protection setpoint (used for transient analyses, including uncertainties)	Initial DNBR value (used for transient analyses, including uncertainties)	site $DNBR_{RT}$ setpoints + uncertainties (U)	site $DNBR_{LCO}$ setpoints + uncertainties (U)	DNBR design limit (used for transient analyses)
Transients actuating the low DNBR protection function **TYPE 1**	Demonstration of the effectiveness of the low DNBR protection function for the largest initial domain : * setting of time constant * definition of protection function operating limits, e.g. - MAX. reactivity insertion rate - MIN. initial DNBR * accuracy	statistical (all of them grouped together, including the DNBR prediction uncertainty)	All included within the $DNBR_{RT}$ threshold	PCC-2 DNBR > DNBR design limit PCC-3 number of rods which experience DNBR < 10 %	1.0	Not fixed (any value compatible with the DNBR criterion meeting)	1.0; U	-	1.0
Transients actuating a specific protection function DNB_{LCO} I&C function used at initiation of transient **TYPE 2**	Demonstration that DNBR criterion is met when the transient starts from the $DNBR_{LCO}$		All included within the $DNBR_{LCO}$ threshold		Not considered	DNBR limiting value	-	DNBR limiting value; U	1.0
$DNBR_{LCO}$ I&C function used at initiation of transient (power level=0) **TYPE 3**	Demonstration that the DNBR criterion is met when the transient starts from the worst initial condition within the operating domain bounded by LCOs.	deterministic	*partially included via initial conditions (e.g. on pressure, temperature, …) *partially included via DNBR design limit (e.g. DNBR prediction – effective of rod bow): This contribution depends on the initia pressure		Not considered	Not significant	-	-	High pressures 1. *Up=1.18 Low pressures 1.* Up=1.20

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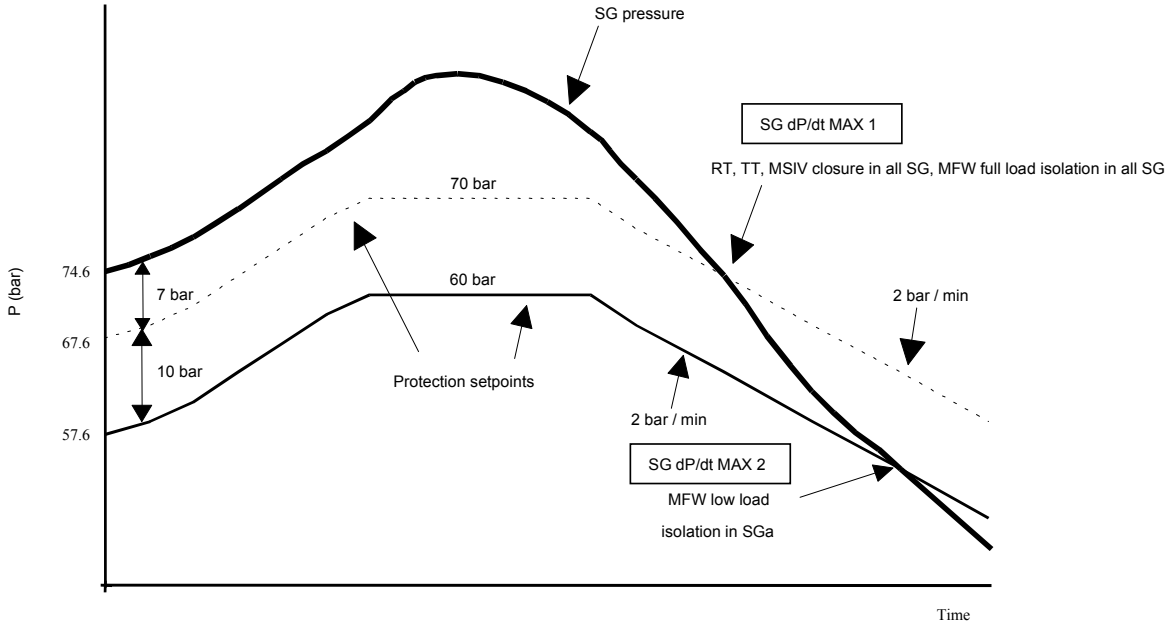
APPENDIX 14B.0.2. – FIGURE 3

SG Pressure Drop Protection Principles



SG dP/dt MAX1 = Actual SG pressure –7 bar (decrease limited to 2 bar/min and maximum absolute value limited to 70 bar)

SG dP/dt MAX2 = Actual SG pressure –17 bars (decrease limited at 2 bars/min and maximum absolute value limited at 60 bar

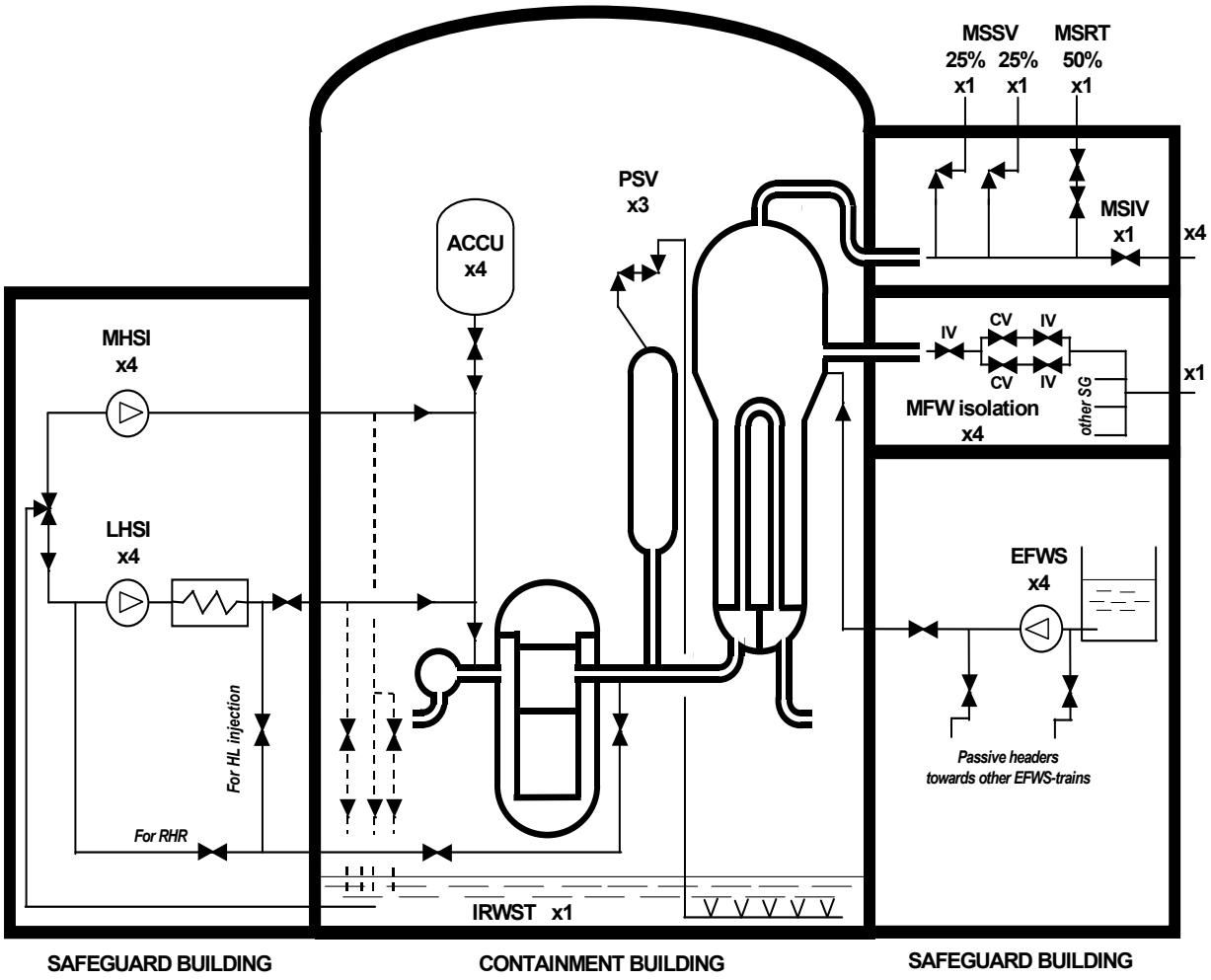
Additionally, other signals are involved in protection against low SG pressure as follows :

- SG pressure MIN1 (50 bar) ➔ RT (TT, ARE [MFWS] full load isolation)
  VIV [MSIV] closure and ARE [MFWS] full load isolation in all SG
- SG pressure MIN2 (40 bar) ➔ ARE [MFWS] low load isolation in SGa
  VDA [MSRT] isolation in SGa

[SGa: Affected Steam Generator]

APPENDIX 14B.0.2 – FIGURE 4

MAIN F1-A FLUID SYSTEMS (simplified functional sketch)



1. ANALYSIS RULES

The safety analysis rules provide a conservative methodology to confirm that safety systems are suitably designed. The degree of conservatism in these analysis rules is considered sufficient to provide appropriate margins in design.

Safety analyses (thermal-hydraulic and neutronic transient calculations) and radiological calculations demonstrate the suitability of a design. A global probabilistic safety assessment is finally carried out to demonstrate compliance with EPR general safety objectives and specific UK EPR risk targets.

1.1. ACCEPTANCE CRITERIA

The acceptance criteria are split into safety criteria and decoupling criteria.

1.1.1. Safety criteria

Safety criteria are defined in terms of radiological limits. These criteria must be met in the safety analysis. The most stringent criteria apply to the most probable events, i.e. PCC-2 events.

Radiological limits for PCC-2 events are those applying in normal operation. For the public the radiological limit corresponds to a deterministic dose of 0.3 mSv/year.

There is no difference between PCC-3 and PCC-4 events in terms of radiological limits. The limits for these events are 50 mSv effective dose and 150 mSv organ dose.

1.1.2. Decoupling criteria

In addition to safety criteria, it is convenient for practical purposes to introduce certain decoupling criteria, which can be applied in the thermal-hydraulic calculations. In this way, thermal-hydraulic calculations and radiological calculations can be decoupled and carried out separately.

Decoupling criteria are defined in such a way that meeting them guarantees that the safety criteria, i.e. the radiological limits, will also be met.

The decoupling criteria must be met with application of all the conservative rules applicable to the safety analysis.

The following decoupling criteria are used in the safety analysis:

a) There shall be no fuel cladding failure in any PCC-2 event and in PCC-3 or PCC-4 events involving a failure of the secondary side pressure boundary (e.g. main steam line break). The de-coupling criterion is: DNBR > 1.0.

b) The fraction of fuel rods experiencing DNB in PCC-3/PCC-4 must remain below 10%.

c) Decoupling criteria for LOCA:

- the peak cladding temperature must remain lower than 1204°C,
- the maximum cladding oxidation must remain lower than 17% of the cladding thickness,
- the maximum hydrogen generation must remain lower than 1% of the amount that would be generated if all the active part of the cladding had reacted,
- the core geometry shall remain coolable, i.e. calculated changes in core geometry shall be such that the core remains amenable to cooling,
- long term core cooling shall be ensured, i.e. the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed.

Note: The radiological calculations for LOCA are performed with the conservative (decoupling) assumption that 10% of fuel rod cladding fails.

d) The peak cladding temperature must remain lower than 1482°C for the fast transients, which do not involve fuel cladding oxidation.

e) The maximum linear power density must remain lower than 590 W/cm in PCC-2 events.

f) Fuel melting at the hot spot must not exceed 10% by volume for PCC-3 and PCC-4 events, i.e. considering a cross section of the hottest fuel rod at the elevation of the power peak, less than 10% of this area is allowed to reach the melting temperature.

g) The core must be subcritical at safe shutdown.

Use of decoupling criteria makes safety assessment easier. Nevertheless, if decoupling criterion a) was not met, fuel cladding degradation after DNB could be assessed.

For accidents occurring in cold shutdown, the initial state of various barriers may be different from that in power operation. For instance the containment or the RCP [RCS] could be open. Then the decoupling criteria related to barrier integrity would be adapted accordingly.

1.1.3. Safe state

The safety analysis is continued to a safe state. Two states have been defined: the controlled state and the safe shutdown state (henceforth referred to as safe shutdown).

For each postulated initiating event (PIE) it must be demonstrated that the controlled state can be reached. The analysis of the transition from controlled state to safe shutdown may be performed once for a set of similar PIEs.

1.1.3.1. Controlled state

The controlled state is a state in which the fast transient is finished and the plant is stabilised with:

- reactivity under control,

- core power being removed (an open cooling chain such as the SGs and the ASG [EFWS] may be used),
- core coolant inventory stable.

'Reactivity under control' means no further power increase is occurring. 'Core power removed' means that core power does not exceed heat removal capacity of F1A systems (i.e. ASG [EFWS] or RIS/RRA [SIS/RHRS]).

For the majority of accident scenarios subcriticality is achieved by reactor trip. Typical examples are:

a) most PCC-2 events, for which the controlled state is the hot shutdown and

b) small break LOCAs, for which the controlled state will be:

- RIS [SIS] flow rate compensates the break flow,
- heat removed from RCP [RCS] by SG (SG relief valves and ASG [EFWS]),
- heat removed from IRWST by the cooling chain LHSI/RRI [CCWS]/SEC [ESWS].

In a few cases subcriticality is not achieved, e.g.:

Overcooling transients on the secondary side: During a large steam line break, recriticality occurs, but the reactivity variation and thermal power excursion are limited and stabilised within a short period of time. After the emptying of the affected SG, the core power level stabilises at the level corresponding to the heat removal capacity of one ASG [EFWS] train (i.e. the ASG [EFWS] train which provides water to the affected SG). The controlled state is thus reached. Subsequently, after manual isolation of the ASG [EFWS] of the affected SG and actuation of safety boration, subcritical conditions are reached in the core.

Dilution during shutdown states: In the case of spurious dilution by the RCV [CVCS] in shutdown conditions, the dilution source is detected and isolated. Core power may rise up to a value which depends on the reactivity inserted into the core before isolation. Power excursion is limited by reactivity feedback. The power level will be finally stabilised at a level, which is lower or equal to the heat removal capacity of the available ASG [EFWS]. From the physical point of view, the reactivity is under control, since the negative reactivity feedback compensates the reactivity excess due to diluted coolant. Furthermore, the core coolant inventory is stable and the cooling chain ASG [EFWS] + MSRV removes heat produced in the core. This is a controlled state.

1.1.3.2. Safe shutdown state

The safe shutdown state is defined as:

- core subcritical even after xenon depletion (subcriticality margin to be defined),
- decay heat removed by the cooling chains LHSI/RRI [CCWS]/SEC [ESWS],
- activity release and barrier integrity within the limits of each PCC as defined in section 1.1 of this appendix.

1.3. INITIAL CONDITIONS

The initial conditions for the transient analysis correspond to a steady state.

Physical parameters are within the limits provided by the controls or by the limiting conditions of operation (LCO functions). A conservative combination of parameters is considered allowing for uncertainties, dead-bands and response times.

For each PCC event the most conservative case is analysed. All plant operating conditions, including shutdown states, are considered as potential initial conditions for the transient analysis.

1.4. RULES FOR OPERATOR ACTIONS

A distinction is made between two phases of the accident: the automatic phase and the manual phase. The automatic phase lasts from event occurrence to the first manual action. The manual phase lasts from the first manual action to the safe shutdown.

During the manual phase, manual actions may be taken into account in the safety analysis in addition to automatic actions.

In the safety analysis, a manual action from the main control room (MCR) may be credited, at the earliest, at 30 minutes after the first significant information is transmitted to the operator. A local manual action, i.e. a manual action performed outside the MCR, may be credited at the earliest at 1 hour after the first significant information is transmitted to the operator.

In the large majority of cases the controlled state can be reached relying only on automatic actions. However this is not mandatory. Manual actions are allowed in attaining the controlled state provided that the 30 minutes criterion is met.

Operators are assumed to act according to the operating procedures. Operator errors are not considered in the safety analysis of PCC-2 to PCC-4 events. Such errors are covered by the PSA on the basis of human reliability models.

1.5. MECHANICAL, ELECTRICAL AND I&C SYSTEMS MODELLED IN THE SAFETY ANALYSIS

1.5.1. F1 systems and functions

In the analysis of PCC-2 to PCC-4 events, pessimistic assumptions are made about the performance of safety systems, i.e. minimum system effectiveness is assumed (taking into account uncertainties in equipment characteristics, uncertainties in actuation set points and considering of the most adverse environmental conditions, etc.).

It must be shown in the analysis of PCC-2 to PCC-4 events that the controlled state can be reached relying only on F1A systems and that the transfer from the controlled state to the safe shutdown can be made relying only on F1A systems and/or F1B systems.

1.5.2. Assumptions regarding F2 and non-safety-classified (NC) systems

The following principles apply to F2 and NC systems used in the analysis of PCC-2 to PCC-4 events:

1) If the transient leads to an actuation of an F2 or NC system and if this system would have a beneficial effect with regard to the applicable safety criterion, the safety analysis must be performed disregarding this system.

   Example: Since the main pressuriser spray is not an F1 system, it cannot be claimed to reduce primary pressure following a safety demand (e.g. SGTR).

2) If the transient leads to an actuation of an F2 or NC system and if this system aggravates the consequences of the accident with regard to the applicable safety criterion, the safety analysis must be performed assuming that the system is operating normally.

   Example: The main pressuriser spray is considered to operate normally for calculation of minimum DNBR in the case of LOOP.

3) If the transient has no impact on the performance of an F2 or NC system (no change of state, no change of operating or environmental conditions), and if the system was operating prior to the accident, the system is assumed to continue operating normally. Spurious commands from the I&C system need not be assumed in these circumstances.

   Example: This applies to the grid in PCC-2, to the Reactor Coolant Pumps (including their seal injection system), and to I&C closed loop controls. No spurious orders from the I&C system are assumed in these conditions.

   The case of the SG level control provides an example of the application of this rule: the SG level control by the main feedwater valves is not safety classified. In case of SGTR, the I&C closed loop control is not impacted by the event. Therefore, this I&C function is assumed to continue working properly. In particular it does not generate spurious commands leading to a full opening or closing of the feedwater valves.

4) More generally, an F2 or NC system is either assumed to work correctly or not to work at all. Faulty operation is not considered in the analysis of the PCCs.

   Example: When the main pressuriser spray (or the main steam bypass) is actuated in the course of a transient, the valves are assumed to reclose normally and not to become stuck open.

5) The turbine isolation valves are not safety classified but they are assumed to close normally after a reactor trip. This is justified because they are in series redundant, they are operating under design conditions, and they are designed to be fail safe. The disconnection of the main power generator after turbine trip is also assumed to operate correctly.

1.6. APPLICATION OF THE SINGLE FAILURE CRITERION (SFC) IN THE SAFETY ANALYSIS

For the analysis of PCC events, the term single failure is understood to denote any active or passive failure, independent of the postulated initiating event, that affects all or part of an equipment item that is claimed in the transient being analysed. The SFC applies to equipment that must change of state to fulfil its function and that has beneficial effects on the transient.

- If an F1 function can be carried out using more than one safety system, including diverse or supporting systems, the single failure is applied to these systems only once.

- With regard to passive single failure, it must be verified in PCC analysis that a single failure in the form of a leak at any location in the pressure boundary and its consequential failure do not prevent performance of the required safety function.

  If the leak cannot be detected or isolated, it must be assumed to develop to a flow rate corresponding to a complete pipe break; in that case it must be confirmed that ability to perform its safety function is maintained.

The following additional rules are also applicable to the SFC in safety analysis:

a) The most pessimistic single failure must be assumed to occur anywhere in systems needed to perform the safety function – only one single failure is applied per initiating event studied in the safety analyses.

b) Consequential failures resulting from the assumed failure must be considered as part of the application of the single failure criterion.

c) If necessary, sensitivity studies must be performed for a given event by applying the SFC to different components, in order to determine the most adverse single failure with regard to the safety criteria.

d) An active single failure is assumed to take place at the start of the transient. A passive single failure need not to be considered until 24 hours after initiation of the transient.

e) Any exception with respect to the single failure shall be stated and justified.

f) A stuck RCCA shall be considered as an application of the SFC.

g) Spurious opening of a safety valve shall be considered as an initiating event.

h) The non-closure of a safety valve after actuation shall be considered as an application of the SFC.

1.7. PREVENTATIVE MAINTENANCE

1.7.1. Preventative maintenance during power operation

During the period of time of preventative maintenance the equipment shall be considered to be unavailable.

If the nature of preventative maintenance is such that the system can be restored to an operational state in due time (such that the necessary safety function can be fulfilled in case of demand), the system is considered available. Examples are short time maintenance activities such as change of oil, change of filters for some supporting systems etc.

1.8. LOSS OF OFFSITE POWER (LOOP)

LOOP must be assumed to coincide with PCC-3 and PCC-4 events in the at power state, if it is conservative. LOOP is considered to occur at the time of turbine trip.

LOOP need not to be assumed for events occurring in shutdown states.

2. SAFETY ANALYSES

2.10. UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY (RCCA) BANK WITHDRAWAL "AT POWER" (STATE A)

2.10.1. Identification of causes and accident description

A - Definition, causes and description of the transient

Uncontrolled control rod assembly bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generators lags behind the core power generation until the steam generator pressure reaches the relief or safety valve set-point, there is a net increase in the reactor coolant temperature and pressure.

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

In the event of a fast reactivity insertion transient, the nuclear power increases very rapidly, in contrast with the coolant temperature; this could eventually lead to fuel damage (due to DNB or high linear power density).

An uncontrolled control rod assembly bank withdrawal at power can occur due to:

- an operator error,
- a control unit parameter error,
- an equipment failure.

The transient can be divided into two distinct phases:

From the postulated initiating event to the controlled state:

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

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

The controlled state is the hot shutdown state defined as:

- Nuclear power = 0% full power
- Coolant temperature = 301.0°C
- Primary pressure = 155 bar
- RCCAs fully inserted
- Boron concentration corresponding to the initial power state
- Xenon level equal to the initial Xenon level

- Reactor Coolant Pumps are running

The shutdown margin guarantees core subcriticality after Reactor Trip (RT).

The controlled state for this transient is similar to that for the loss of condenser vacuum transient (see section 2.5.2 in this Appendix where more details are provided).

From the controlled state to the safe shutdown state

The safe shutdown state corresponds to a state where the LHSI is in RHR mode or where RHR connection conditions are reached. It is defined as:

- Nuclear power = 0% full power
- Hot leg coolant temperature = 180°C
- Primary pressure = 30 bar
- RCCAs fully inserted
- Boron concentration sufficient to ensure core subcriticality, even after the Xenon depletion
- Residual Heat is removed by the steam generators or by the LHSI in RHR mode

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

- RCP [RCS] cooldown and depressurisation (F1 classified)
- boration performed through the RBS [EBS] (F1 classified)

As regards activity release, uncontrolled RCCA bank withdrawal at power is covered by the loss of condenser vacuum (section 2.5.2).

As regards the capability for heat removal, uncontrolled RCCA bank withdrawal at power is covered by the feedwater line break transient (section 2.16) since the four steam generators remain available (the ratio between the heat flux produced in the RCP [RCS] and the number of available SGs is lower for an uncontrolled RCCA bank withdrawal at power).

As regards subcriticality, uncontrolled RCCA bank withdrawal at power is covered by uncontrolled boron dilution

The sequence of actions to be performed to reach the safe shutdown state is detailed in the corresponding section (section 2.16 within this Appendix).

The analysis presented in the following paragraphs (section 2.10.2 and 2.10.3) deals with the particular phase of the transient from the PIE to the RT.

B - Safety and decoupling criteria

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

The decoupling criteria, defined in terms of behaviour of the barriers, are:

- critical heat flux limit:

This limit is satisfied if the minimum DNBR during the transient remains above the DNBR design limit (see section 0.2.8 in this Appendix).

- fuel temperature limit:

This limit is satisfied if the maximum linear power density at the hot spot remains less than 590 W/cm.

C - Reactor protection system actions

The different trip setpoints of the reactor protection system that ensure protection of the core during the transient are the following:

- low DNBR,
- ex-core high neutron flux rate of change protection,
- high linear power density protection,
- high core power level protection,
- high pressuriser pressure protection,
- high pressuriser level protection.

Low DNBR and high linear power density protection provide effective core protection for most reactivity insertion transients, except for very rapid transients which require the actuation of protection with a very short response time provided by ex-core neutron flux measurements.

2.10.2. Methods and assumptions

A - Failure selection for the analysis

In accordance with the general safety rules defined in section 1, single failure and maintenance are applied to F1 systems in the most pessimistic way with respect to the design criterion to be met.

The transient is analysed assuming worst case failure of the reactor protection system, identified with the highest worth rod being stuck above the core on reactor trip.

The assumption of preventative maintenance does not increase the severity of the transient.

B - Method of analysis

This transient is simulated using the multi-loop THEMIS code using a point kinetics neutronics model. This code calculates the evolution of the following parameters during the transient:

- the nuclear power,
- the heat flux,

- the pressuriser pressure,
- the temperature in the loops and the core inlet temperature.

The evolution of the DNBR during the transient is then calculated using the thermal-hydraulic design code FLICA.

A global processing algorithm is used to determine the evolution of the on-line DNBR value calculated by the protection system.

These two calculations are performed with the same axial power distribution and the same nuclear $F_{\Delta H}$. The axial power shape and the nuclear $F_{\Delta H}$ are kept constant during the entire transient.

A simplified diagram of the low DNBR protection channel as simulated in the global processing algorithm is shown in Appendix 14B.2.10 - Figure 1. The time constants for the relevant modules (lead-lag modules and filter modules) are given in the same figure.

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

- the nuclear power distribution derived from the nuclear in-core instrumentation (Self Powered Neutron Detectors),
- the pressure derived from the primary pressure sensors,
- the core flow rate derived from the Reactor Coolant Pump speed sensors,
- the inlet temperature derived from the cold leg temperature sensors.

The SPND measurements are treated with a filtering module. The time constant of this module is chosen to take into account the delay between the variation of the nuclear power (parameter measured for the DNBR calculation) and the variation of the heat flux (relevant parameter for the value of the physical DNBR).

The cold leg temperature measurements are treated with a filtering module and a lead-lag module. This lead-lag module aims to compensate the delay due to the temperature sensor and the advance between the cold leg temperature measurement (used for the DNBR calculations) and the core inlet temperature (relevant parameter for the physical DNBR).

The calculated DNBR value is also treated with a lead-lag module, mainly to compensate the total delay between low DNBR threshold actuation and beginning of rod drop (RT).

The purpose of the analysis is to optimise the response times, time constants and protection thresholds to:

- provide effective protection at all reactivity insertion rate transients,
- limit the DNBR variation during the transient,
- provide margins for operational transients.

The on-line low DNBR protection channel ensures core protection for a wide range of reactivity insertion transients but its response time is not rapid enough to cope with very fast reactivity insertion transients. On the contrary, the ex-core high neutron flux rate of change protection channel provides effective protection when the reactivity insertion rate is high.

C - Initial conditions (see Appendix 14B.2.10 - Table 1)

The analysis of the transient is performed for two power levels: 100% and 10% of nominal power (NP).

The initial conditions are chosen in a conservative way with respect to DNBR. The initial values for power, average coolant temperature, and reactor coolant pressure are the extreme values compatible with operation in steady-state conditions:

. reactor power :	102% NP	12% NP
. average reactor coolant temperature :	311.25 + 2.5°C	303.1 + 2.5°C
. reactor coolant pressure :	155.0 – 2.5 bar	155.0 – 2.5 bar

For the 100% NP transients, the nuclear $F_{\Delta H}$ value is chosen in such a way that the initial DNBR is equal to the DNBR limiting value. This $F_{\Delta H}$ value is 1.85[(1)].

For the 10% NP transients, an initial decoupling nuclear $F_{\Delta H}$ value is considered, taking into account the insertion of control rods.

D - Core related assumptions

- Reactivity coefficients.

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

  - minimum reactivity feedback:
    - the moderator density coefficient is zero,
    - the Doppler coefficient has a minimum absolute value (see section 0.2)
    - the kinetics coefficients are assumed to be at their minimum values.
  - maximum reactivity feedback:
    - the moderator density coefficient is a maximum (see section 0.2),
    - the most negative Doppler coefficient is assumed (see section 0.2),
    - the kinetic coefficients are assumed to be at their maximum values.

- fuel to coolant heat transfer coefficient.

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

(1) value higher than the $F_{\Delta H}$ resulting from fuel management schemes examined in Chapter 4

- shutdown margin.

  The rod having the greatest worth is assumed to be stuck above the core. The negative reactivity insertion following the trip is thus minimised, giving a minimum final subcriticality margin; additionally, the most conservative negative reactivity insertion curve as a function of time is used (see Appendix 14B.0.2 - Table 8 without earthquake).

E - Reactivity insertion rate

To verify that the different means of reactor trip provide protection in all the possible situations, a wide range of reactivity insertion rate is considered, covering all possible cases of rod withdrawal over the whole set of fuel management cycles, starting at different initial power levels.

F - Protection actions

The low DNBR and ex-core high neutron flux rate of change trip thresholds and time delays to trip are given in Appendix 14B.2.10 - Table 1.

These threshold values include instrumentation and set-point uncertainties: maximum time delay values are assumed.

G - Control actions

To be conservative with respect to DNB, the pressure control system is assumed to be operational and the pressuriser spray flow rate is considered to be at its maximum value, to minimise the reactor coolant pressure increase during the transient.

2.10.3. Results and conclusions

From an initial power equal to 100% NP, the low DNBR channel ensures the most effective core protection up to 32 pcm/s for the case of minimum reactivity feedback, and for the whole range of reactivity insertion rates for the case of maximum reactivity feedback.

From 32 pcm/s to 90 pcm/s with minimum reactivity feedback, effective core protection is ensured by the ex-core high neutron flux rate of change protection.

From an initial power equal to 10% NP, the low DNBR channel ensures protection for the whole range of reactivity insertion rates and for both minimum and maximum reactivity feedback.

The sequence of events for the most onerous transient (100% NP, 32 pcm/s, minimum reactivity feedback) is given in Appendix 14B.2.10 - Table 2.

The results are presented in the following figures:

Appendix 14B.2.10 - Figure 2:	nuclear power and heat flux versus time (100% NP, 32 pcm/s, minimum reactivity feedback),
Appendix 14B.2.10 - Figure 3:	pressuriser pressure versus time (100% NP, 32 pcm/s, minimum reactivity feedback),
Appendix 14B.2.10 - Figure 4:	average coolant temperature versus time (100% NP, 32cm/s, minimum reactivity feedback),
Appendix 14B.2.10 - Figure 5:	algorithm DNBR calculation and physical DNBR versus time (100% NP, 32 pcm/s, minimum reactivity feedback),
Appendix 14B.2.10 - Figure 6:	nuclear power and heat flux versus time (100% NP, 1pcm/s, minimum reactivity feedback),
Appendix 14B.2.10 - Figure 7:	pressuriser pressure versus time (100% NP, 1 pcm/s, minimum reactivity feedback),
Appendix 14 B.2.10 - Figure 8:	average coolant temperature versus time (100% NP, 1 cm/s, minimum reactivity feedback),
Appendix 14B.2.10 - Figure 9:	algorithm DNBR calculation and physical DNBR versus time (100% NP, 1 pcm/s, minimum reactivity feedback),
Appendix 14B.2.10 - Figure 10:	nuclear power and heat flux versus time (100% NP, 50pcm/s, minimum reactivity feedback)
Appendix 14B.2.10 - Figure 11:	pressuriser pressure versus time (100% NP, 50 pcm/s, minimum reactivity feedback)
Appendix 14B.2.10 - Figure 12:	average coolant temperature versus time (100% NP, 50pcm/s, minimum reactivity feedback)
Appendix 14B.2.10 - Figure 13:	physical DNBR versus time (100% NP, 50 pcm/s, minimum reactivity feedback)
Appendix 14B.2.10 - Figure 14:	minimum physical DNBR versus reactivity insertion rate (100% NP, minimum reactivity feedback),
Appendix 14B.2.10 - Figure 15:	minimum physical DNBR versus reactivity insertion rate (10% NP, minimum reactivity feedback)

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

The analysis of reactivity insertion transients allows the conclusion that the minimum initial DNBR value compatible with the required DNBR design limit is 1.26.

APPENDIX 14B.2.10 – TABLE 1

Uncontrolled RCCA bank withdrawal at power
Parameter Values assumed

PARAMETER	VALUE
- Reactivity insertion rate (pcm/s)	up to 90
- Initial conditions	
. power level (% of nominal power)	100 + 2 = 102 | 10 + 2 = 12
. coolant average temperature (°C)	311.25 + 2.5 = 313.75 | 303.1 + 2.5 = 305.6
. reactor coolant pressure (bar)	155 - 2.5 = 152.5 | 155 - 2.5 = 152.5
- core related assumptions	
. moderator density coefficient $\Delta k/k/(g/cm^3)$	Minimum value : 0 Maximum value : 0.49
. shutdown margin (pcm)	4000
- protection system	
. low DNBR reactor trip setpoint low DNBR reactor trip delay (s)	1.00 0.9
. high flux rate of change reactor trip setpoint (%) high flux rate of change reactor trip delay (s)	13* 0.6

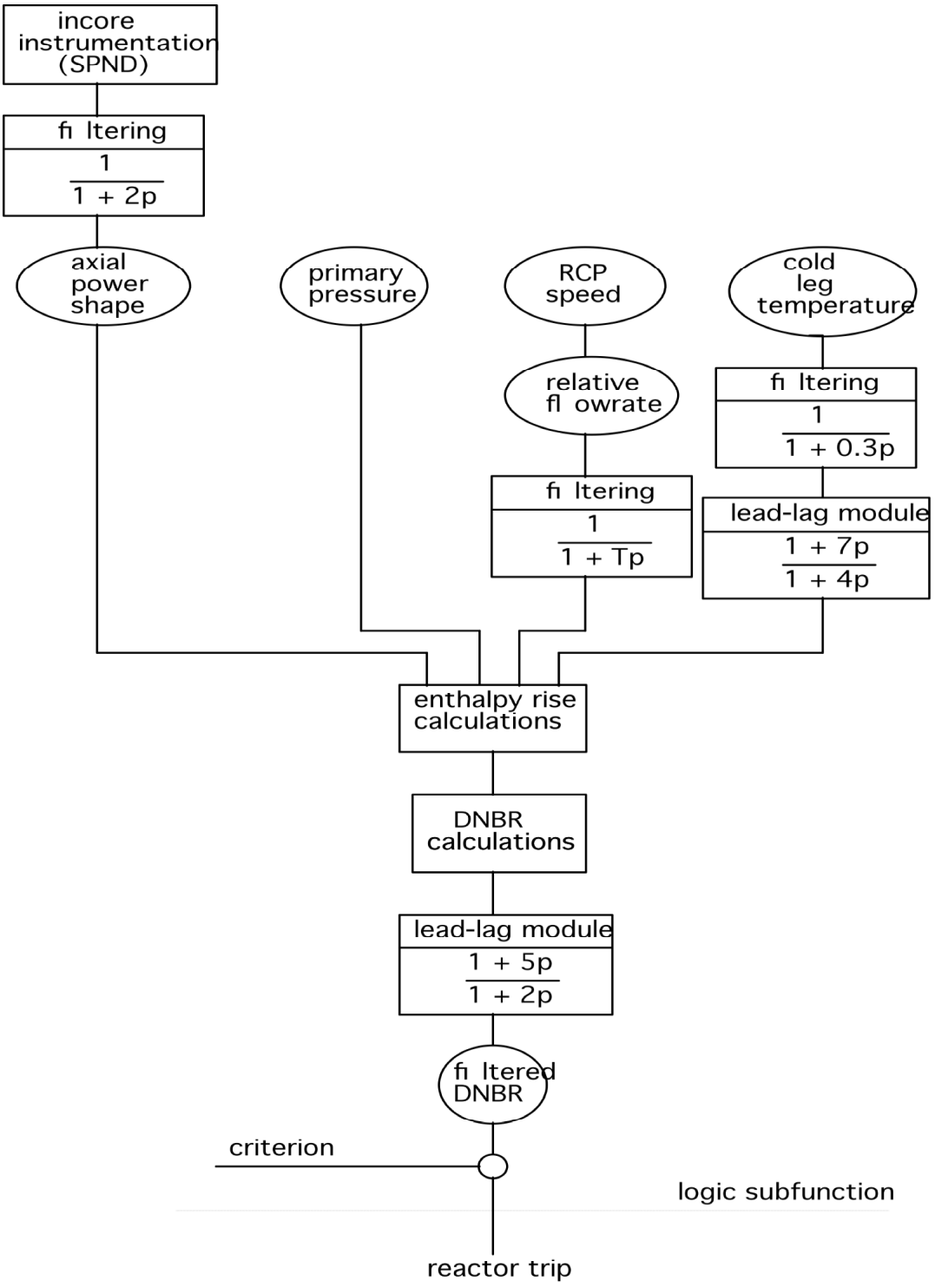
* Filtered flux derivative with a time constant equal to 30 seconds

APPENDIX 14B.2.10 – TABLE 2

Uncontrolled RCCA bank withdrawal at power
Sequence of events
(100% NP, 32 pcm/s, minimum reactivity feedback)

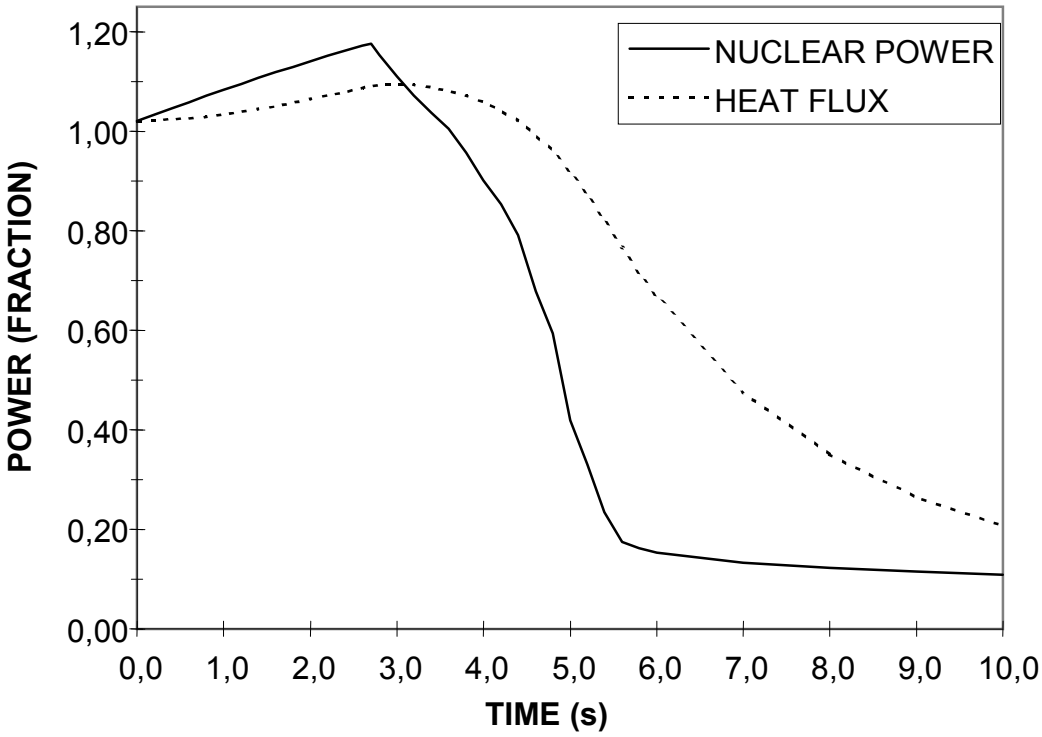
Event		Time (s)
Reactivity insertion begins		0
Low DNBR reactor trip		1.75
Rods begin to drop		2.65
Minimum DNBR (1.03) occurs		3.00
thermal power (%)	1.094	
core inlet temperature (°C)	294.8	
pressuriser pressure (bar)	154.2	

APPENDIX 14B.2.10 - FIGURE 1



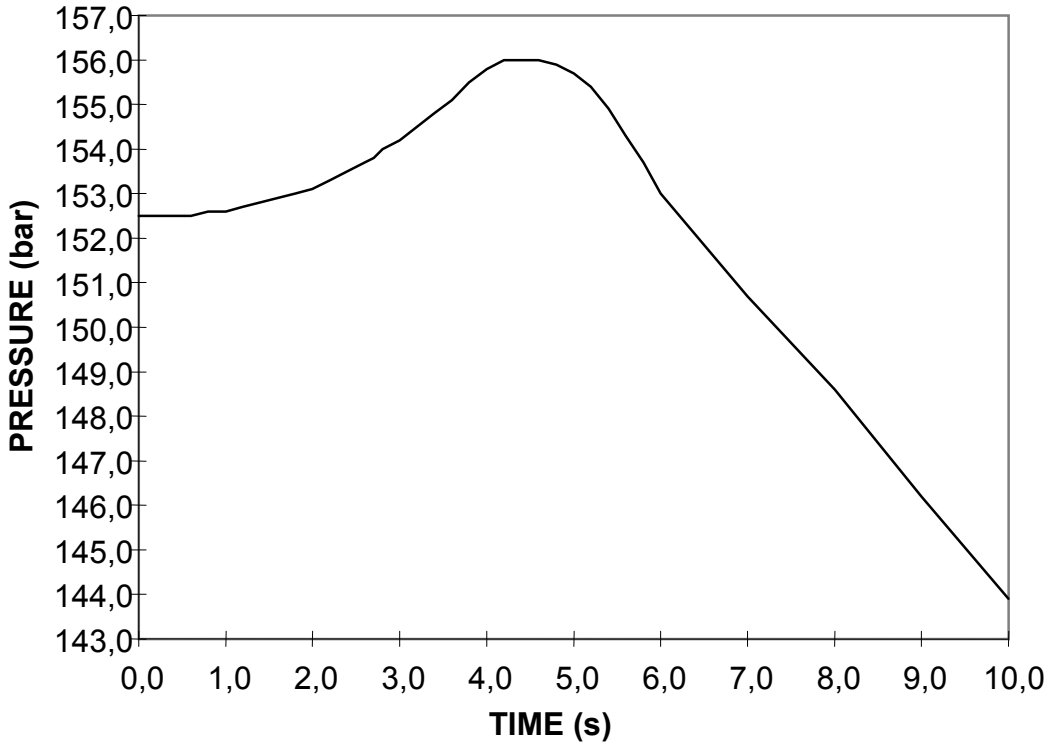
Uncontrolled RCCA bank withdrawal at power
Simplified diagram of the low DNBR protection channel

APPENDIX 14B.2.10 - FIGURE 2



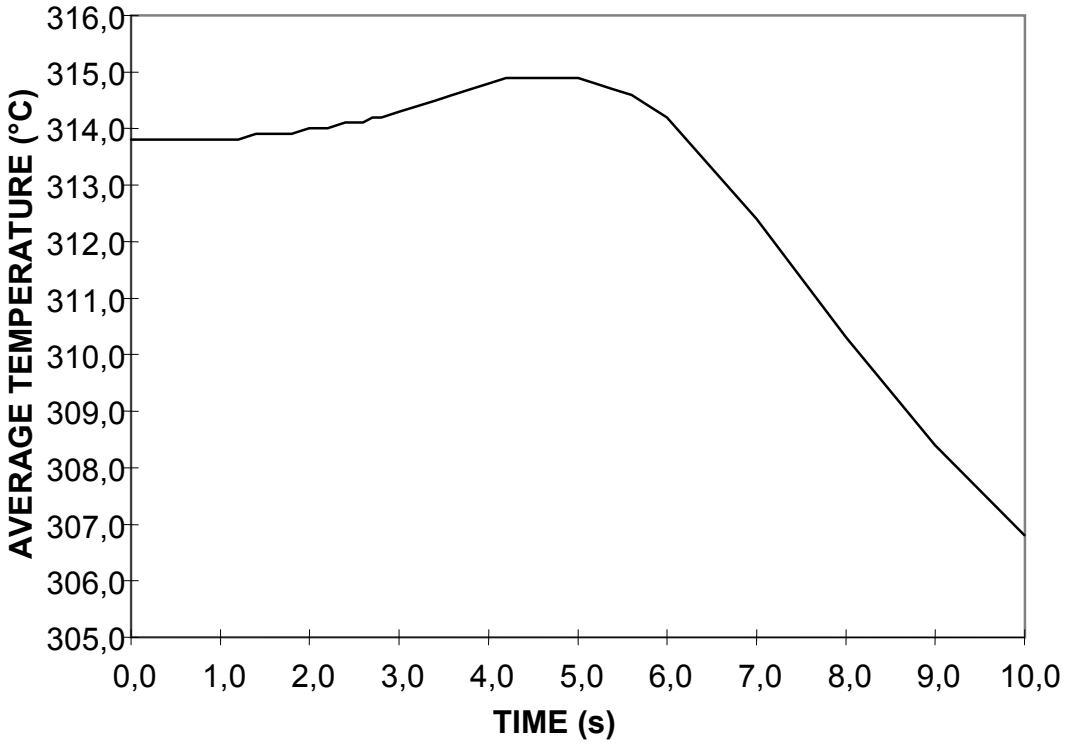
Uncontrolled RCCA bank withdrawal at power
Nuclear power and heat flux versus time
(100% NP, 32 pcm/s, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 3



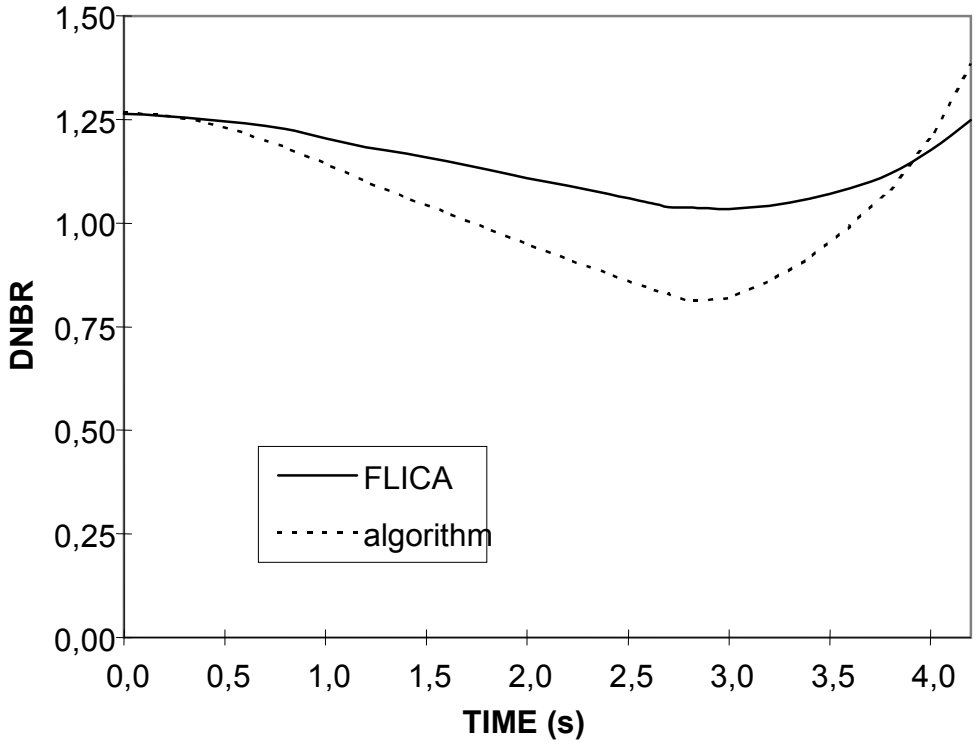
Uncontrolled RCCA bank withdrawal at power
Pressuriser pressure versus time
(100% NP, 32 pcm/s, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 4



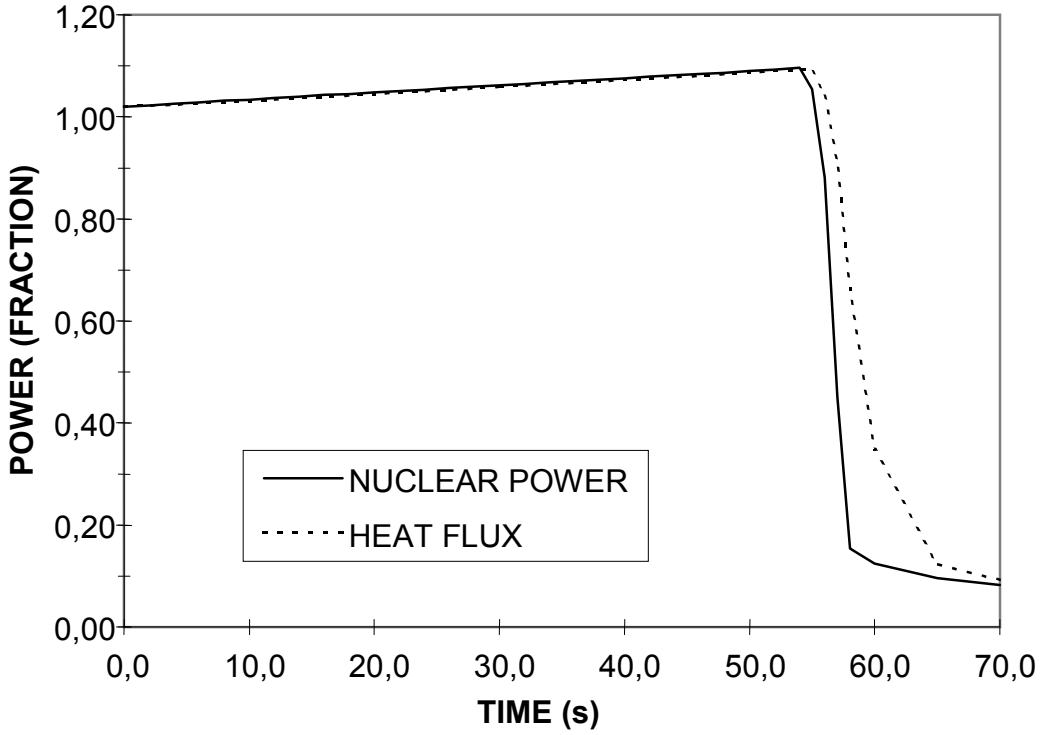
Uncontrolled RCCA bank withdrawal at power
Pressuriser pressure versus time
(100% NP, 32 pcm/s, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 5



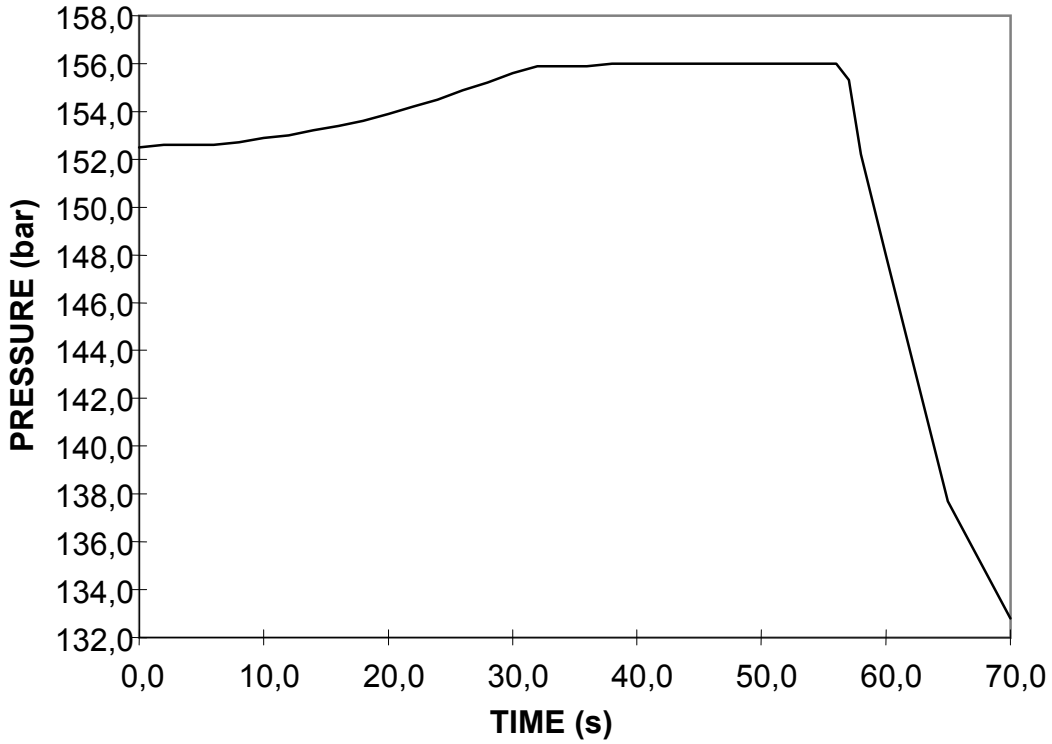
Uncontrolled RCCA bank withdrawal at power
Algorithm DNBR calculation and physical DNBR versus time
(100% NP, 32 pcm/s, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 6



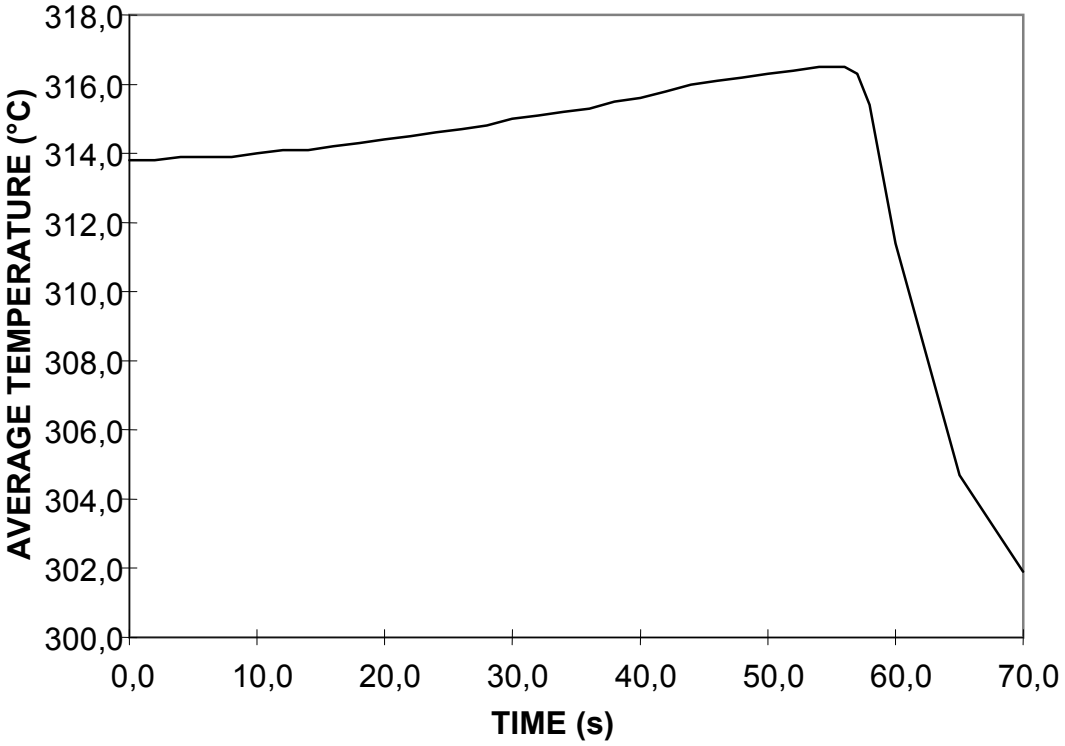
Uncontrolled RCCA bank withdrawal at power
Nuclear power and heat flux versus time
(100% NP, 1 pcm/s, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 7



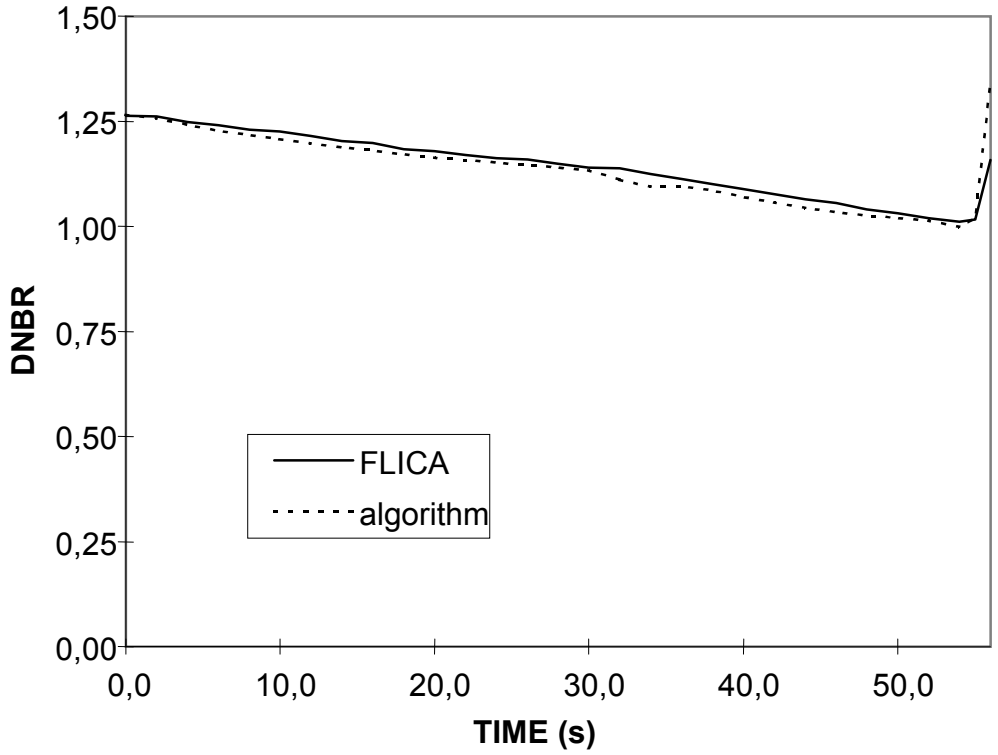
Uncontrolled RCCA bank withdrawal at power
Pressuriser pressure versus time
(100% NP, 1 pcm/s, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 8



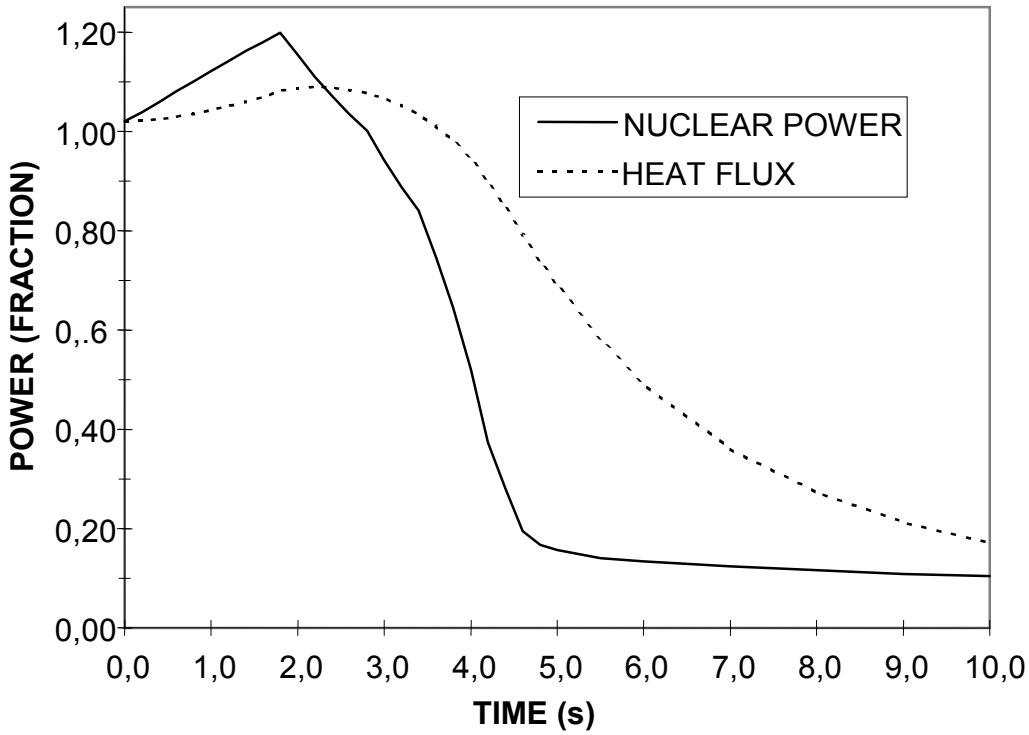
Uncontrolled RCCA bank withdrawal at power
Average coolant temperature versus time
(100% NP, 1 pcm/s, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 9



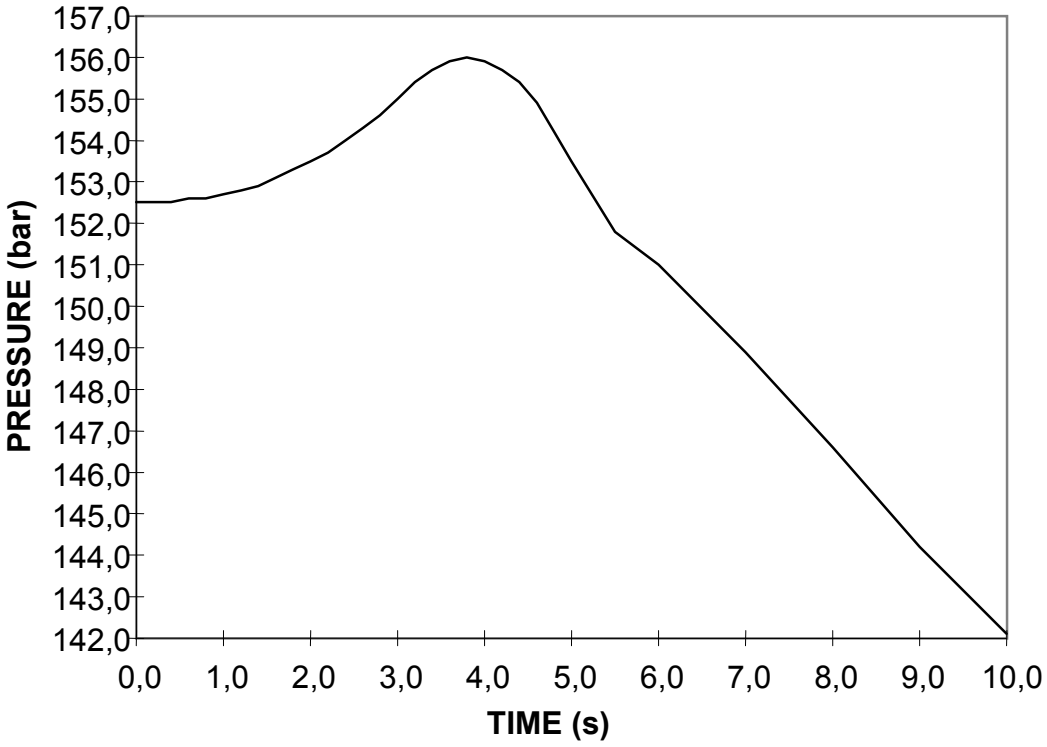
Uncontrolled RCCA bank withdrawal at power
Algorithm DNBR calculation and physical DNBR versus time
(100% NP, 1 pcm/s, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 10



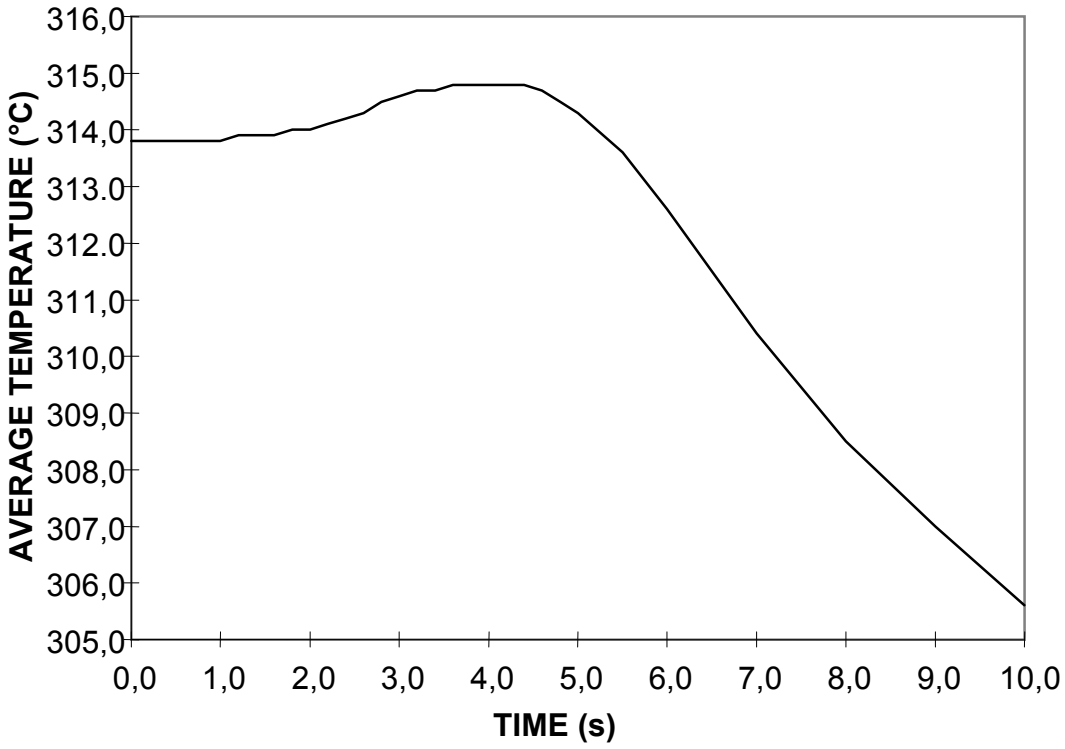
Uncontrolled RCCA bank withdrawal at power
Nuclear power and heat flux versus time
(100% NP, 50 pcm/s, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 11



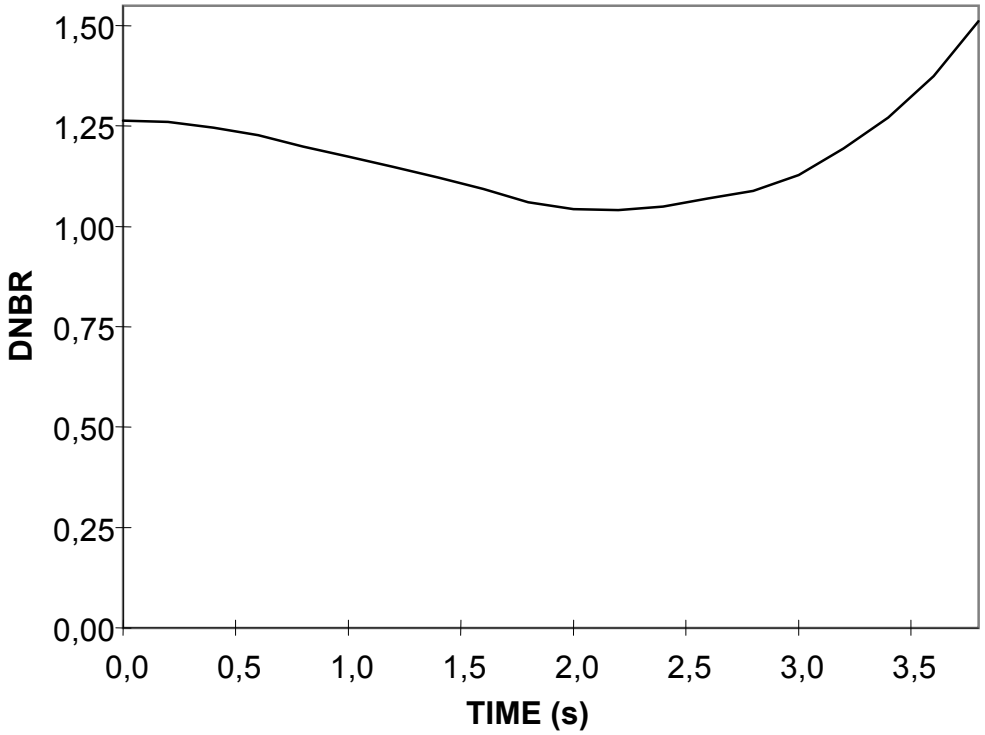
Uncontrolled RCCA bank withdrawal at power
Pressuriser pressure versus time
(100% NP, 50 pcm/s, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 12



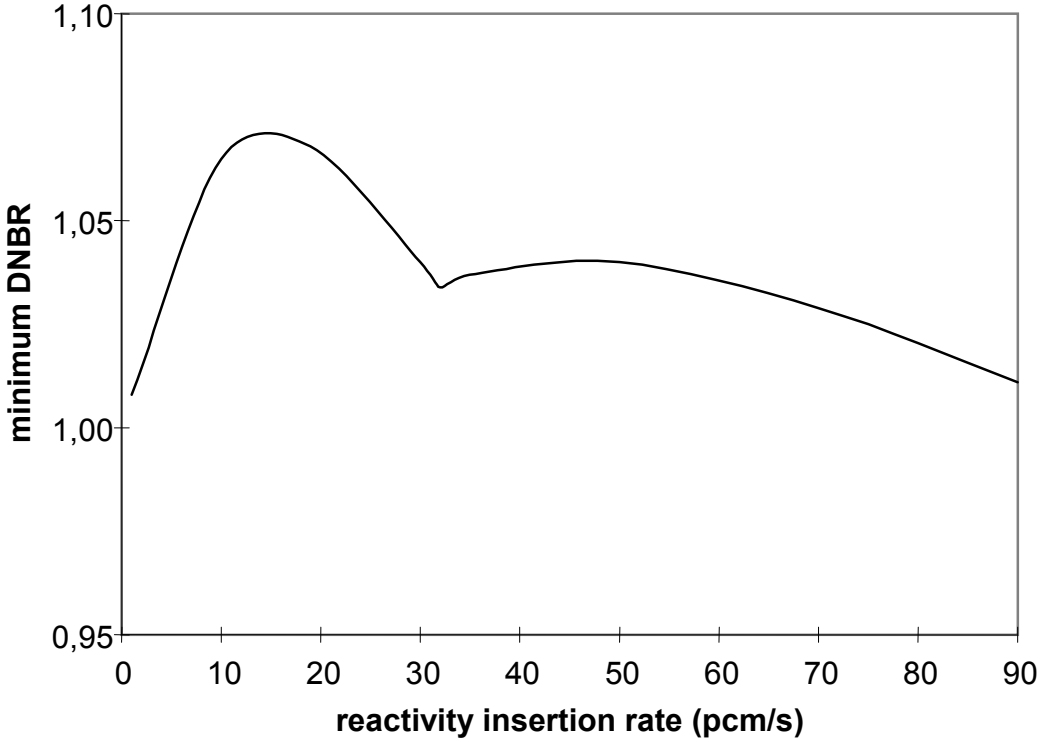
Uncontrolled RCCA bank withdrawal at power
Average coolant temperature versus time
(100% NP, 50 pcm/s, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 13



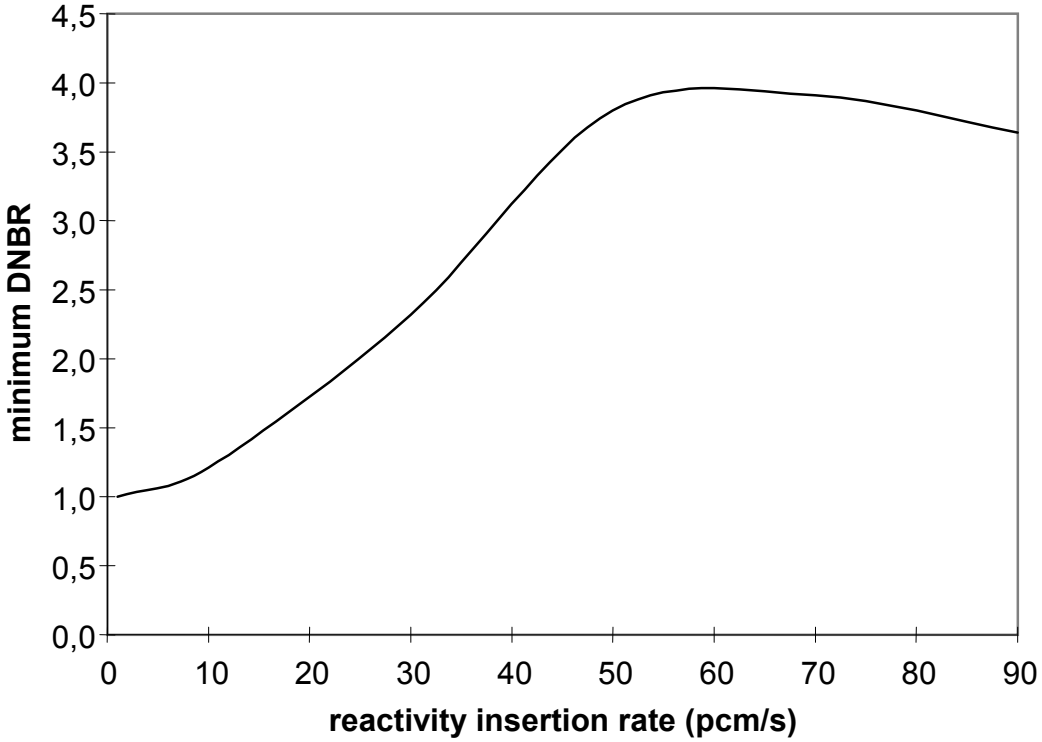
Uncontrolled RCCA bank withdrawal at power
Physical DNBR versus time
(100% NP, 50 pcm/s, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 14



Uncontrolled RCCA bank withdrawal at power
Minimum physical DNBR versus reactivity insertion rate
(100% NP, minimum reactivity feedback)

APPENDIX 14B.2.10 - FIGURE 15



Uncontrolled RCCA bank withdrawal at power
Minimum physical DNBR versus reactivity insertion rate
(10% NP, minimum reactivity feedback)

2.14. EXCESSIVE INCREASE IN SECONDARY STEAM FLOW / INADVERTENT OPENING OF A SG RELIEF TRAIN OR A SG SAFETY VALVE

2.14.1. Excessive increase in secondary steam flow (in state A, PCC-2) / Inadvertent opening of a SG relief train or a SG safety valve (in state A, PCC-3)

2.14.1.1. Identification of causes and accident description

2.14.1.1.1. General concern

The most severe core conditions for an accidental depressurisation of the main steam system due to a spurious or inadvertent valve(s) opening results from:

- spurious opening of a main steam bypass valve (PCC-2),
- spurious opening of a main steam relief or safety valve (PCC-3),
- spurious partial cooldown (PCC-2),
- failure to close of a main steam relief or safety valve after it is opened (PCC-2).

During power operation, core protection is achieved by a reactor trip initiated by the Reactor Protection System (including the DNBR signal). Automatic actuation of reactor trip prevents occurrence of core damage before completion of reactor shutdown.

After reactor shutdown, the overcooling transient continues as long until main steam system depressurisation is terminated, potentially resulting in a return to core criticality. The severity of the event depends on this potential return to critical conditions after reactor trip.

A discussion is presented below of the classification of different events resulting from spurious or inadvertent opening of the valve(s), and the severity of the resulting PCC-2 events, in order to define the most onerous PCC-2 event that should be analysed.

a) Spurious opening of a main steam bypass valve

   Spurious opening of a main steam bypass valve is classified as a PCC-2 event. The resulting cooldown is terminated when the steam lines are isolated (at the latest when the pressure reaches 50 bar). Therefore this inadvertent core cooling event is bounded by the case of 'failure to close of a VDA [MSRT] after a challenge' where steam isolation occurs later (at 40 bar steam pressure) - see item d below.

b) Spurious opening of a main steam relief or safety valve

   Spurious opening of a main steam relief train would arise due to the spurious opening of the corresponding main steam relief isolation valve. This would require spurious opening of the two solenoid-driven pilot valves in series. Spurious opening of a main steam relief train is consequently classified as a PCC-3 event.

The main steam safety valves use spring-load technology. Spurious opening of a main steam safety valve is therefore classified as a PCC-3 event.

As a consequence:

- there are no inadvertent core cooling events due to the spurious opening of one VDA [MSRT] or one MSSV in the PCC-2 event category,
- spurious opening of one VDA [MSRT] or one MSSV, classified as a PCC-3 event, is bounded by main steam line break (see section 2.15 of this appendix). During the corresponding uncontrolled core cooling transient, analysis shows that core DNB is prevented.

c) Spurious partial cooldown

Partial cooldown involves reducing the main steam relief valve (or the main steam bypass) setpoint from 93 bar a to 60 abs. bar (87.1 bar a to 55 abs. bar in the case of spurious operation of the main steam bypass) at a rate of 100°C/h.

Thus a spurious partial cooldown due to a spurious I&C command, results in an overcooling event which is terminated when the steam generator pressure reaches 60 bar a (55 bar a in the case of spurious operation of the main steam bypass).

The core is designed to so that subcritical conditions are maintained after a spurious partial cooldown, so that the resulting core overcooling has no consequence on the core power behaviour.

d) Failure to close of an MSSV or a VDA [MSRT] after a challenge

For the case of failure to close of an MSSV:

- In PCC-2 events MSSV actuation is either not possible (since VDA [MSRT] is designed to prevent MSSV actuation) or is due to failure of a VDA [MSRT] to open (in a secondary system overpressure event such as spurious closure of a VIV [MSIV]).

  Consequently, a stuck open MSSV need not be considered in PCC-2 analyses since it would require a double failure (failure of a VDA [MSRT] to open plus failure of a MSSV to close after a challenge).

- In PCC-3 and PCC-4 events the MSSV could be actuated despite operation of the corresponding VDA [MSRT] (no design requirement to prevent MSSV actuation, except in SGTR). However even if one MSSV opened and failed to close, safety criteria would still be met. The consequences of such an event would be bounded by those of a main steam line break (see section 2.15 of this appendix).

For the case of failure to close of a VDA [MSRT]:

- In PCC events, the VDA [MSRT] is assumed to be actuated after reactor / turbine trip (no credit taken for the GCT [MSB]): it must be assumed that the corresponding control valve sticks fully open due to the single failure principle. Sticking open of the MSRCV would lead to an SG depressurisation and hence to an uncontrolled RCP [RCS] overcooling event.

In this event all rods are considered to drop successfully as a single failure has already been taken into account.

When SG pressure reaches MIN3 (40 bar), the VDA [MSRT] is automatically isolated by closure of its isolation and control valve.

This case is analysed below. The event covers any PCC-2 event in which there is a failure of the MSRCV to close after a challenge.

2.14.1.1.2. Typical sequence of events

a) From the initiating event to attainment of the controlled state

Following any PCC-2 event, a reactor and a turbine trip occur leading to a challenge to the main steam relief trains on each steam generator (GCT [MSB] not credited). A single failure of a main steam relief control valve on one steam generator is then postulated, with the valve remaining fully open in its initial position.

The steam release which is a consequence of this accident results in an initial increase in steam flow and a subsequent decrease as the steam pressure falls. An SG pressure drop signal is generated which results in closure of all VIV [MSIV]s. After VIV [MSIV] closure, only the faulted steam generator continues to depressurise.

The energy removal from the RCP [RCS] results in a reduction in coolant temperature and pressure. With a negative moderator temperature coefficient, the cooldown results in a reduction in the core shutdown margin.

When SG pressure reaches MIN3 (40 bar), the VDA [MSRT] is automatically isolated by closure of its isolation and control valves.

Eventually, the controlled state is reached. This corresponds to stable conditions on the secondary side with heat removal via the unaffected steam generators (i.e. fed via the ARE [MFW] / AAD [SSS] if available or the ASG [EFWS], with steam discharge via the VDA [MSRT]).

Actuation of partial cooldown on an RIS [SIS] signal is highly probable (pressuriser pressure < MIN3). In that case, the controlled state is reached at the end of partial cooldown.

b) From the controlled state to the safe shutdown state

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

The sequence of actions to be performed (initiated by the operator) to reach LHSI/RHR operating conditions is as follows:

- RCP [RCS] boration:
    - During the cooldown, RCP [RCS] boration is performed via the RBS [EBS] (no credit is taken for the RCV [CVCS] since it is not F1 classified).
    - After completion of the required boration, the operator isolates the RBS [EBS].

- RCP [RCS] cooldown:
    - RCP [RCS] cooldown to LHSI/RHR connection conditions is performed via the secondary side by decreasing the VDA [MSRT] setpoints (the GCT [MSB] being unavailable as the VIV [MSIV]s are closed).
    - The RCP [RCS] cooling rate is consistent with the ASG [EFWS] tank capacity, which means that LHSI/RHR operating conditions are reached before the ASG [EFWS] tanks are empty.
    - The EPR design cooling rate is -50°C/h if 2 RBS [EBS] trains are available, or -25°C/h if only 1 RBS [EBS] train is available, provided it is not limited by the VDA [MSRT] capacity, assuming 2 Reactor Coolant Pumps are in operation (shut-off of 2 out 4 Reactor Coolant Pumps must be carried out by the operator).
- RCP [RCS] depressurisation:
    - After cooldown, if the RCP [RCS] pressure is greater than the LHSI/RHR connection pressure of 30 bar, operator momentarily opens the PSV, in order to depressurise the RCP [RCS].
    - During this depressurisation phase, the LHSI ensures a minimum RCP [RCS] pressure of about 20 bar, so that the RCP [RCS] subcooling margin is not impaired. LHSI/RHR connection conditions are achieved.

2.14.1.2. Safety criteria

The safety criteria are the radiological limits for normal operation.

The consequences of a main steam system depressurisation are analysed with respect to the following decoupling criteria:

- fuel cladding integrity,
- reactor coolant pressure boundary,
- amount of radioactive products released.

2.14.1.3. Definition of cases studied

Fuel cladding integrity:

- Analysis is performed to demonstrate that the following criterion for no core damage is satisfied : no DNB after reactor trip for a steam release equivalent to full opening of a main steam relief valve, with isolation on SG pressure < MIN3 (40 bar). As a single failure is, by the definition of the event, postulated to occur on one main steam relief control valve which stays stuck open, all RCCAs are assumed inserted.
- The accident phase from the initiation to attainment of the controlled state is analysed in detail in the following sections.
- Demonstration of attainment of the safe shutdown state is based on qualitative arguments, with reference to other analyses in section 2 of this appendix.

Reactor coolant pressure boundary:

- Reactor coolant system cooldown can cause thermal shock to the reactor vessel.

Radiological consequences:

- The safety criteria to be met are the dose equivalent limits for release to the atmosphere, as described in section 1 of this appendix.
- The bounding transient, with regard to radiological release is the loss of condenser vacuum analysed in section 2.5.2 of this appendix. The amount of steam released to the atmosphere is similar in both cases.

2.14.1.4. Methods and assumptions

2.14.1.4.1. Methods of analysis

The thermal-hydraulic analysis of this accident is carried out using the THEMIS code.

The analysis methodology is that described in section 2.15.1.4.1 in this appendix for 'steam system piping failure', which belongs to the same event family.

Since there is no return to critical conditions and hence no power excursion which might have an impact on the transient, the neutronic behaviour of the core has not been simulated.

Based on the thermal-hydraulic parameters calculated by the THEMIS code (core pressure, core inlet temperatures and core flow rate), the reactivity variation is computed using the SMART 3D code.

2.14.1.4.2. Protection and mitigation actions

The following F1A I&C functions provide protection for the case of accidental depressurisation of the main steam system, with regard to the DNBR criterion (for analyses relevant to other criteria, see section 2.14.1.3 within this appendix):

Reactor trip on:

- core power level > MAX3,
- DNBR < MIN3,
- pressuriser pressure < MIN2,
- steam generator pressure drop > MAX1,
- steam generator pressure < MIN1.

Safety injection, actuated when:

- pressuriser pressure < MIN3.

Closure of all main steam isolation valves on:

- steam generator pressure drop > MAX1,
- steam generator pressure < MIN1.

Closure of main feedwater full load isolation and control valves in all SGs on:

- steam generator pressure drop > MAX1,
- steam generator pressure < MIN1,
- steam generator level > MAX1.

Closure of main feedwater low load isolation and control valves of the affected steam generator on:

- steam generator pressure drop > MAX2,
- steam generator pressure < MIN2,
- steam generator level > MAX1.

Isolation of the main steam relief train of the affected steam generator on:

- steam generator pressure < MIN2.

The F1B functions required to transfer the plant from the controlled state to the safe shutdown state are described in section 2.5.2 of this appendix (loss of condenser vacuum event).

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

2.14.1.5.1. Choice of single failure and preventative maintenance

A single failure is postulated to be sticking open of a main steam relief control valve after being challenged due to operation of the corresponding main steam relief train.

No preventative maintenance condition is assumed since preventative maintenance does not have a significant impact on the transient, the amount of water injected by MHSI being negligible.

2.14.1.5.2. Initial state

Accidental opening of a main steam system valve is more serious when the unit is at hot shutdown conditions.

The RCP [RCS] boron concentration is assumed equal to zero, in order to maximise the reactivity insertion during the RCP [RCS] cooldown.

The initial conditions are presented on Appendix 14B.2.14 - Table 1.

2.14.1.5.3. Specific assumptions

a) Neutronic data and decay heat

In this study, it is assumed that the reactor is operating under the following conditions, which cover both UO2 and MOX fuel management schemes:

- End of life shutdown margin; hot full power equilibrium xenon level; all control and shutdown rod clusters inserted.

Core physics studies show that this margin is ensured even under the most unfavourable conditions, in particular at the end of the equilibrium cycle when the temperature coefficient reaches its highest absolute value.

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

The initial shutdown margin is 3000 pcm (all rods inserted). A moderator feedback coefficient corresponding to EOL with all rod clusters inserted is used.

No credit has been taken for the decay heat (terms B+C).

b) Assumptions related to non-F1 systems

- ARE [MFW]:
  - It is assumed that a maximum main feedwater flow is delivered to all steam generators until closure of the low load isolation and control valves.
  - The maximum flow assumed is 20% of the nominal feedwater flow, corresponding to the feedwater flow after reactor trip.
  - Other control systems are ignored as they have either a beneficial impact or no impact on the event.

c) Assumptions related to F1 systems:

- VIV [MSIV]s (F1A):
  - All VIV [MSIV]s are assumed to close on SG pressure drop > MAX1 (2 bar / min). The setpoint of this signal is adjusted to 8.5 bar (7 + 1.5 bar) below the initial SG pressure. The delay for steam lines isolation is assumed to consist of 0.9 seconds of channel delay plus 5 seconds for the valve closure time.
- ARE [MFW] low load isolation and control valves closure (F1A):
  - The ARE [MFW] low load isolation and control valves on the affected steam generator are closed on SG pressure drop > MAX2 (2 bar / min). The setpoint for this signal is adjusted to 18.5 bar (17 + 1.5 bar) below the initial SG pressure. The delay for feedwater isolation is assumed to consist of 0.9 seconds for channel delay plus 10 seconds for the valve closure time.
  - The ARE [MFW] low load isolation and control valves of the unaffected steam generators are closed on SG level > MAX1 (70.6 + 2% of the narrow range). The delay for feedwater isolation is assumed to consist of 1.5 seconds for channel delay plus 10 seconds for valve closure time.

- MHSI (F1A):
    - The minimum safety injection capability is assumed (see Appendix 14B.2.14 - Figure 1).
    - It is assumed for simplification purposes (conservative and bounding assumption) that the in-containment refuelling water storage tank contains unborated water.
    - Safety injection and partial cooldown are actuated on pressuriser pressure < MIN3 (115 – 1.5 bar).
    - The delay for MHSI injection is assumed to consist of 0.9 seconds for channel delay plus 10 seconds to the time taken to start the pumps.
- VDA [MSRT] (F1A):
    - The maximum capacity of the stuck open valve is conservatively assumed to be the maximum capacity of a main steam relief valve (1375 t/h or 55% of nominal steam flow at 97 abs bar).
    - The VDA [MSRT] on the affected steam generator is assumed to be completely isolated by closure of its isolation and control valves on SG pressure < MIN3 (40 – 1.5 bar).
    - The delay for VDA [MSRT] isolation is assumed to consist of 0.9 seconds for channel delay plus 5 seconds for valve closure time.

d) Other assumptions:

- A maximum heat transfer coefficient is used in the study.
- The steam flow through the stuck open valve is calculated using the Moody correlation at each calculation step. The back pressure is always assumed to be the atmospheric pressure.
- Perfect moisture separation in the steam generators is assumed.
- No credit is taken for heat stored in metal structures other than the fuel rods and steam generator tubes.

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[b].

- With respect to the criterion of no return to critical conditions, it is more pessimistic to assume forced circulation rather than natural circulation. Therefore the primary pumps are assumed to keep running during the transient.

The main assumptions are given in Appendix 14B.2.14 - Table 1.

2.14.1.5.4. Results

Appendix 14B.2.14 - Table 2 gives the sequence of events.

Appendix 14B.2.14 - Figures 2 to 4 show the evolution of the main parameters during the transient.

The stuck open main steam relief control valve (after the challenge to the main steam relief trains following turbine trip) results in steam generator depressurisation and RCP [RCS] cooldown via all 4 steam generators.

All steam lines are isolated on the first SG pressure drop signal: feedwater flow to the faulted SG is isolated on generation of the second SG pressure drop signal.

The primary pressure decreases slowly and safety injection into the core occurs after a long delay. The decrease in the reactor coolant temperature causes a reactivity insertion.

At about 400 seconds, the main steam relief train is automatically isolated and the steam leak is terminated.

Appendix 14B.2.14 - Table 3 gives the main thermal-hydraulic parameters at the time of main steam relief train isolation.

The reactivity calculated at the time of main steam relief train isolation is -630 pcm. It follows that there is no return to critical conditions.

The criterion for prevention of DNB is therefore met.

After a rapid stabilisation of the thermal-hydraulic parameters, the controlled state is reached using F1A function. The controlled state corresponds to the end of partial cooldown, with a RCP [RCS] temperature of 275°C (corresponding to a SG pressure of 60 bar) and a pressuriser pressure of 87 bar.

It follows that the reactivity is controlled, core power is being removed by the unaffected steam generators, and the core coolant inventory is stable.

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

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

The transition from the controlled state to the safe shutdown state is covered by the loss of condenser vacuum case (see section 2.5.2 of this appendix). Indeed, the controlled state after a main steam system depressurisation is less onerous than that after a loss of condenser vacuum (as the shutdown margin is the same but the RCP [RCS] pressure and temperature are lower). The capabilities of F1B systems required in both cases are identical (VDA [MSRT], RBS [EBS]).

2.14.2. Inadvertent opening of a SG relief train or of a SG safety valve (in state B, PCC-4)

The spurious opening of a VDA [MSRT] or MSSV in state B is classified as a PCC-4 event.

The cooldown resulting from such events is less severe than in case of main steam line break since the equivalent break area is smaller.

Therefore the consequences of such inadvertent core cooling events in state B are bounded by those following a main steam line break in state B (see section 2.15 of this appendix).

In the corresponding core cooling transient, analyses show that core DNB is prevented.

APPENDIX 14B.2.14 – TABLE 1

Stuck open main steam relief control valve
Main assumptions

Parameters		Limiting values used
Initial conditions		
- Reactor power	(% FP)	0
- Shutdown margin	(pcm)	- 3000
- RCP [RCS] boron concentration	(ppm)	0
- RCP [RCS] flow rate		Thermal-hydraulic
- Average RCP [RCS] temperature	(°C)	301
- Pressuriser pressure	(abs. bar)	155
- Pressuriser level	(% of MR)	28 + 5 = 33
- Steam generator water level	(% of MR)	56
Stuck open main steam relief valve : Maximum capacity	(t/h)	1375 at 97 abs. bar (55% of nominal steam flow)
Main feedwater flow rate per SG	(t/h)	500
SG pressure drop > MAX1 setpoint	(bar/mn)	2 Setpoint adjusted 8.5 bar below the initial value (7 + 1.5 = 8.5 bar)
SG pressure drop > MAX2 setpoint	(bar/mn)	2 Setpoint adjusted 18.5 bar below the initial value (17 + 1.5 = 18.5 bar)
pressuriser pressure < MIN3 setpoint	(abs. bar)	115 – 1.5 = 113.5
SG pressure < MIN2 setpoint	(abs. bar)	40 – 1.5 = 38.5
Steam line isolation delay	(s)	0.9 + 5 = 5.9
Main feedwater low load isolation and control valves closure delay :	(s)	
- on SG pressure drop > MAX2		0.9 + 10 = 10.9
- on SG level > MAX1		1.5 + 10 = 11.5
Main steam relief train isolation delay	(s)	0.9 + 5 = 5.9

APPENDIX 14B.2.14 – TABLE 2

Stuck open main steam relief control valve
Sequence of events

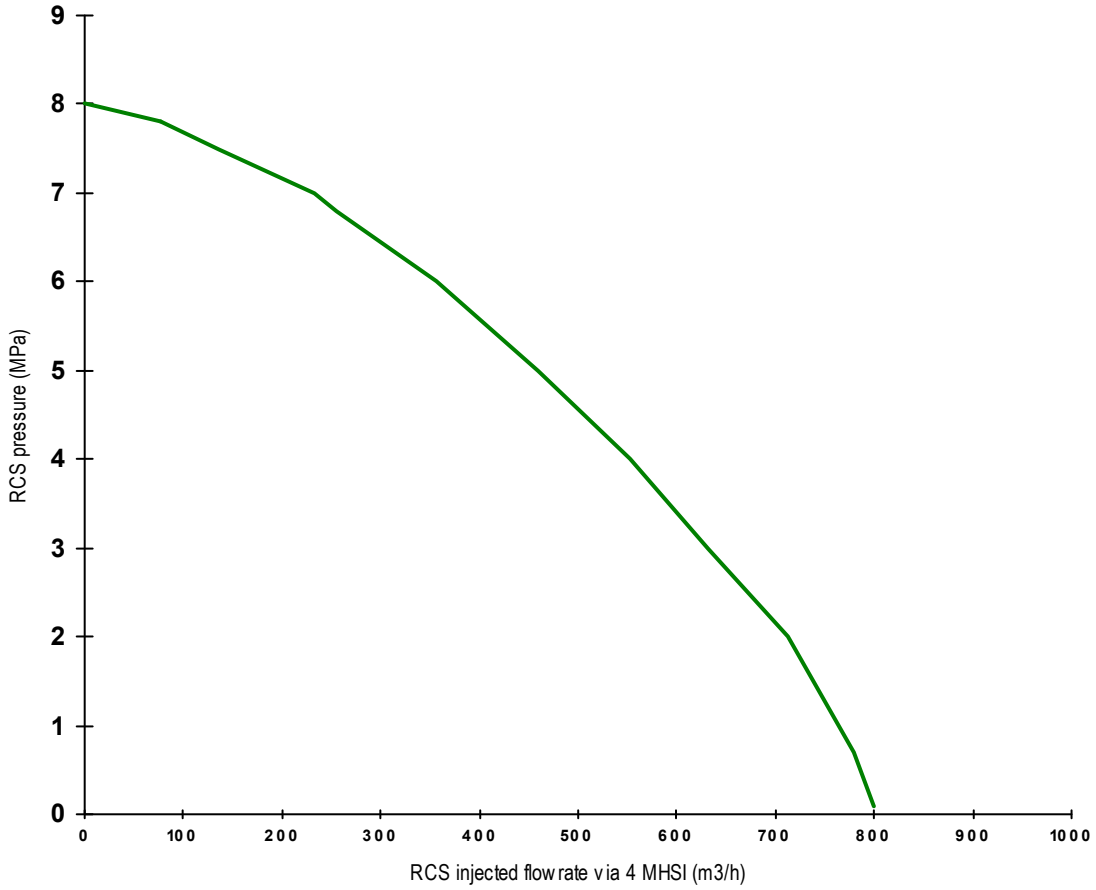
EVENT	TIME (seconds)
Stuck open main steam relief control valve	0
SG pressure drop > MAX1 setpoint is reached	139
Pressuriser is empty	143
Steam lines are isolated	145
Pressuriser pressure < MIN3 setpoint is reached	149
SG level > MAX1 setpoint is reached in unaffected SG	165
SG pressure drop > MAX2 setpoint is reached in the affected SG	176
Main feedwater low load isolation and control valves are closed in the affected steam generator	187
Safety injection flow enters the RCP [RCS]	208
Main feedwater low bad isolation and control valves are closed in the unaffected SG	265
SG pressure < MIN2	319
Main steam relief train is isolated	325

APPENDIX 14B.2.14 – TABLE 3

Stuck open main steam relief control valve
Main thermal-hydraulic parameters at the time of VDA [MSRT] isolation

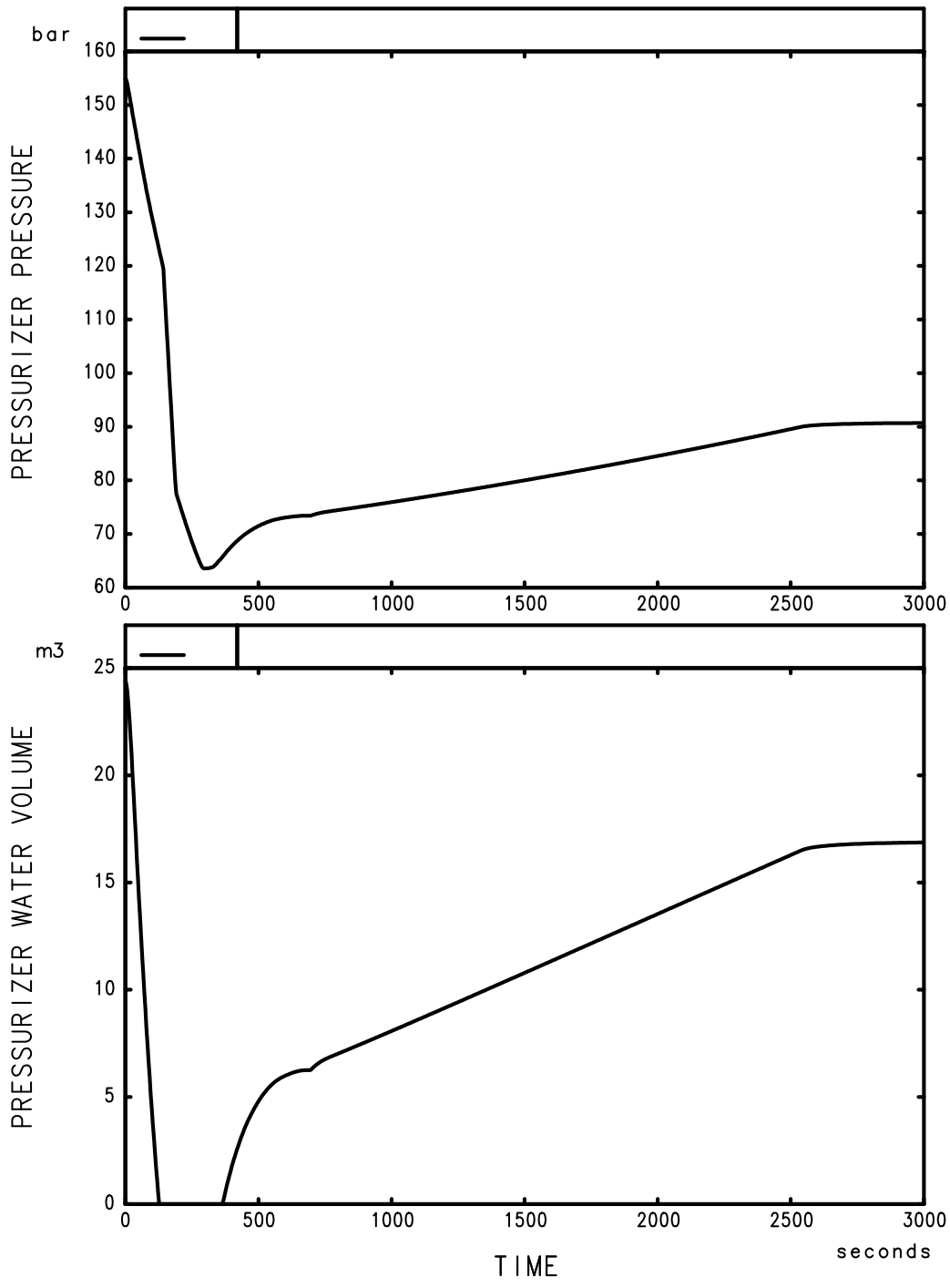
Time(s)	397
Core boron concentration (ppm)	0
Average core pressure (abs. bar)	65
Core inlet temperature of affected loop (°C)	254
Core inlet temperature of unaffected loops (°C)	262
Core flow rate (kg/s)	22230
Reactivity (pcm)	-630

APPENDIX 14B.2.14 – FIGURE 1



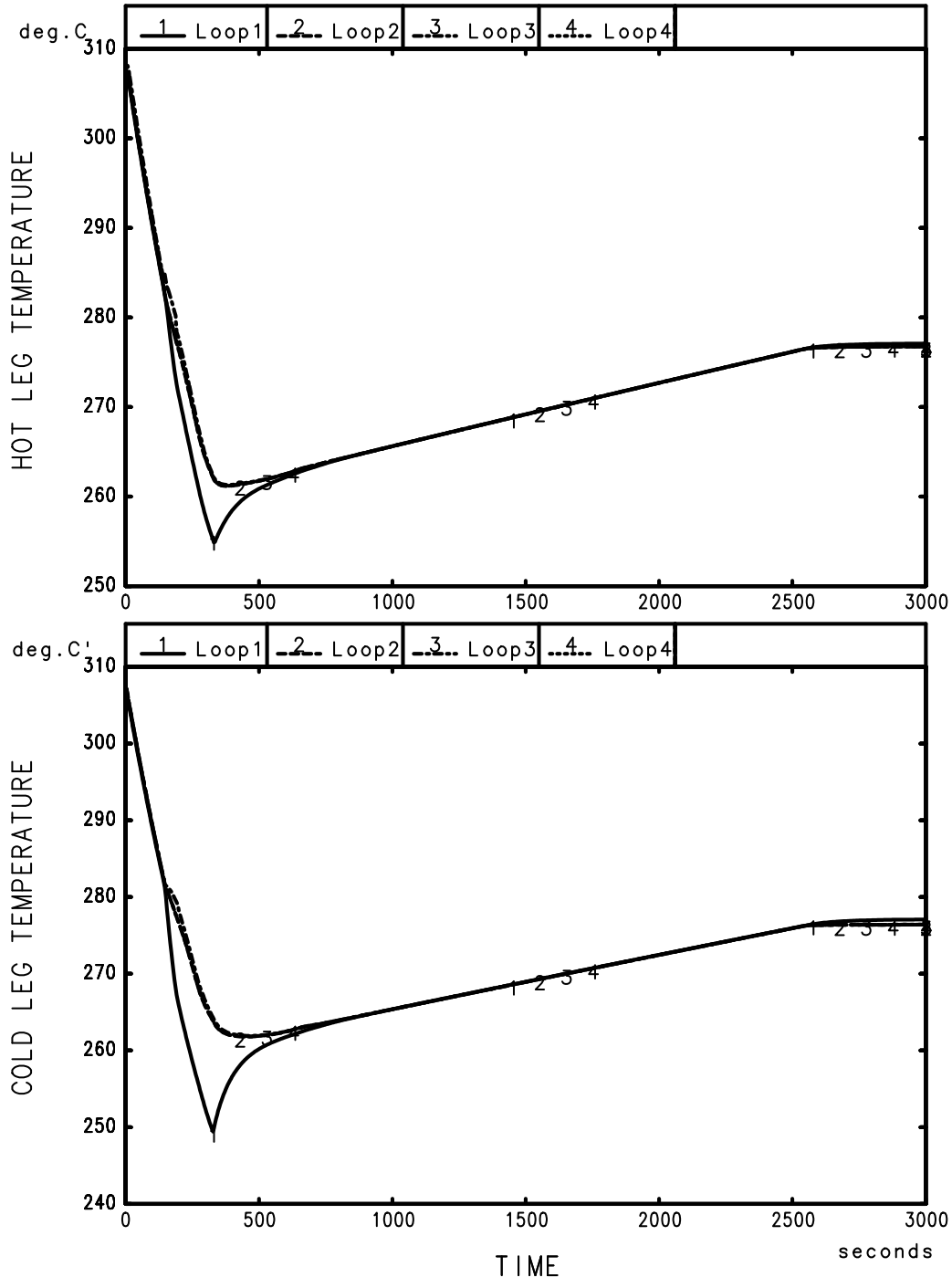
Stuck open main steam relief control valve
MHSI flow injected in the RCP [RCS]

APPENDIX 14B.2.14 – FIGURE 2



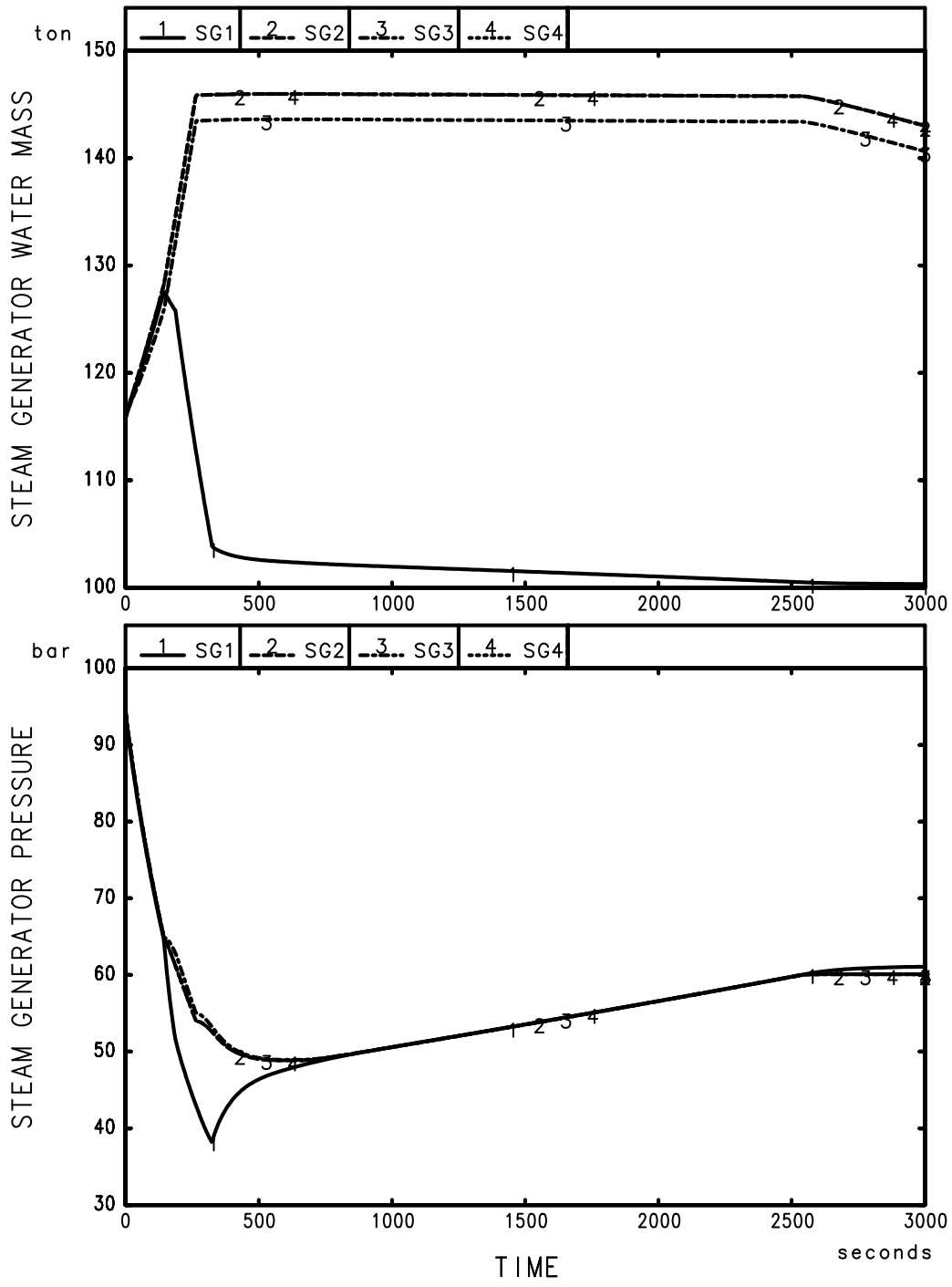
Stuck open main steam relief control valve
At hot shutdown conditions

APPENDIX 14B.2.14 – FIGURE 3



Stuck open main steam relief control valve
At hot shutdown conditions

APPENDIX 14B.2.14 – FIGURE 4



Stuck open main steam relief control valve
At hot shutdown conditions

2.15. STEAM SYSTEM PIPING FAILURE

2.15.1. Steam system piping failure (in state A, PCC-4)

2.15.1.1. Identification of causes and accident description

2.15.1.1.1. General concern

Steam release as a consequence of rupture of a main steam line results in an initial increase in steam flow which decreases during the accident as the steam pressure falls.

The energy removal from the RCP [RCS] causes a reduction of coolant temperature and pressure. In the presence of a negative moderator coefficient, the cooldown leads to an insertion of positive reactivity.

The core may become critical and return to power after reactor trip. This power increase is more significant when the highest worth Rod Cluster Control Assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip.

2.15.1.1.2. Typical sequence of events

a) From the initiating event to the controlled state

Following the break initiation, the secondary system depressurises. SG pressure drop or pressure low signals actuate the RT (if SLB at power), close all VIV [MSIV]s, and isolate the ARE [MFWS]/AAD [SSS] of the affected SG.

After this isolation, only the affected steam generator which has experienced the non-isolatable SLB (break either inside the containment, or outside the containment with failure to close the VIV [MSIV]), continues to depressurise. This SG is supplied by the ASG [EFWS] which has been automatically actuated on low SG level.

The energy removed from the RCP [RCS] causes a reduction of coolant temperature and pressure and actuation of MHSI and partial cooldown on pressuriser pressure < MIN3.

In the presence of a negative moderator coefficient, the RCP [RCS] cooldown results in an insertion of a positive reactivity. The reactor becomes critical with a power excursion.

The Doppler effect limits or stops this power increase.

When the affected steam generator is empty, the power rapidly reduces to a level depending on the ASG [EFWS] flow rate.

The controlled state is then reached with:

- the core just critical (i.e. reactivity equal to zero),
- the core power removed via the leak and the ASG [EFWS] supply to the affected steam generator,
- a stable coolant inventory.

b) From the controlled state to the safe shutdown state

The safe shutdown state is defined as a state where the core is subcritical, LHSI/RHR operating conditions are reached and the affected steam generator is isolated.

In this state the heat removal function is provided by the LHSI/RHR.

The sequence of actions to be performed (initiated by the operator) to reach LHSI/RHR operating conditions are as follows:

- Isolation of the affected steam generator:

In the controlled state, the affected steam generator is depressurised.

The first operator action is to complete and confirm the isolation of this affected steam generator with respect to steam discharge and feedwater supply. This is performed by closing the VIV [MSIV] and isolating MFW to the affected steam generator (if not yet done automatically in case of a small SLB) and by isolating the associated ASG [EFWS] line (there is no automatic isolation of ASG [EFWS]). The objectives of these actions are to:

  - prevent an unacceptable increase of containment pressure and temperature, if the steam line break is located inside containment,
  - terminate the RCP [RCS] cooldown caused by the affected steam generator,
  - prevent the associated ASG [EFWS] tank from completely draining through the break.

After isolation of the affected SG, the corresponding ASG [EFWS] flow can be sent to another SG via the ASG [EFWS] header, if necessary.

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 partial cooldown.

- RCP [RCS] boration:

Even without boration, core subcriticality is recovered after isolation of the affected steam generator. However, RCP [RCS] boration is needed when transferring the plant to LHSI/RHR operating conditions.

During the cooldown, RCP [RCS] boration is performed via the RBS [EBS] (the RCV [CVCS] not being credited since it is not F1 classified).

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

- RCP [RCS] cooldown:

RCP [RCS] cooldown to LHSI/RHR connecting conditions is performed using the secondary side by decreasing the VDA [MSRT] setpoints of the unaffected SGs (the GCT [MSB] being unavailable since the VIV [MSIV]s are closed).

The EPR cooling rate is designed to be consistent with the ASG [EFWS] tanks capacity: Hence LHSI/RHR operating conditions are reached before the ASG [EFWS] tanks are empty.

The EPR design cooling rate is -50°C/h if 2 RBS [EBS] trains are available, or -25°C/h if only 1 RBS [EBS] train is available, provided it is not limited by the VDA [MSRT] capacity.

- RCP [RCS] depressurisation:

After cooldown, if the RCP [RCS] pressure is greater than the LHSI/RHR connecting pressure of 30 bar, the operator will momentarily open the PSV (in some cases with the Reactor Coolant Pumps switched off), in order to depressurise the RCP [RCS].

During this depressurisation phase, the LHSI ensures a minimum RCP [RCS] pressure of about 20 bar so that the RCP [RCS] subcooled margin is not impaired. LHSI/RHR connection conditions are achieved.

2.15.1.1.3. Precautions limiting the event consequences

Each steam generator is equipped with integral multinozzle flow limiters. In the event of a main steam line break these limiters restrict the steam flow at the steam generator outlet, whatever the location of the break.

2.15.1.2. Safety criteria

The safety criteria are the radiological limits for PCC-4 (see section 1.1).

The consequences of a main steam line break are assessed against the following decoupling criteria:

- no degradation of the fuel cladding (no DNBR)
- reactor coolant pressure boundary integrity,
- reactor containment integrity,
- acceptable amount of radioactive products released.

In accordance with the safety analysis rules defined in section 1 in this appendix, it must be possible to reach the controlled state relying only on F1A functions, and safe shutdown state relying only on F1A and F1B functions.

2.15.1.3. Definition of cases studied

- Demonstration of avoidance of degradation of fuel cladding

  The analysis is performed to demonstrate that the following decoupling criterion is met to show avoidance of core damage: no DNB after reactor trip assuming the highest worth rod cluster control assembly stuck in its fully withdrawn position, for the case of a non-isolatable double-ended guillotine steam line break.

The accident phase from the initiating event to the controlled state is analysed in detail in the following sections. The analysis takes credit for F1A measures only.

The demonstration that a safe shutdown state can be achieved, relying only on F1A and F1B functions, is based on qualitative arguments, that the steam system piping failure is bounded by the feedwater system piping failure analysed in section 2.16.

- Integrity of the reactor coolant pressure boundary

  RCP [RCS] cooldown can cause thermal shock to the reactor vessel.

- Reactor containment integrity

  The effect of a steam line break on containment pressure and temperature is analysed in PCSR Sub-Chapter 6.2.

- Radiological consequences

  The safety criteria to be met are the dose equivalent limits applicable to releases to atmosphere.

  The bounding PCC-4 transient with regard to radiological releases is rupture of two steam generator tubes analysed in section 2.18.

2.15.1.4. Methods and assumptions

2.15.1.4.1. Method of analysis

The analysis of a main steam line break is performed using:

- the THEMIS code to describe the overall thermal hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), taking into account operation of F1 systems,
- the PANBOX/COBRA code for neutronic and thermal hydraulic behaviour of the core.

a) With respect to the system transient (THEMIS code), the analysis methodology is based on the following approach :

- identification of the dominant phenomena,
- verification of the adequacy of the code to simulate those phenomena,
- application of conservative PCC analysis rules.
    - The dominant phenomena in this transient are:
        - SG blowdown,
        - asymmetric RCP [RCS] cooldown,
        - mixing of loop flows within the RPV,
        - RCP [RCS] depressurisation and PZR emptying.

- All these phenomena are within the applicability range of the THEMIS code, whose qualification is based on :
  - the use of recognised and tested correlations (e.g. the MOODY correlation for critical steam-water flow rate through a pipe break under saturated conditions),
  - validation of specific models against test results from small-scale simulations (e.g. modelling of mixing between loop flows within the RPV, is based on use of pessimistic data representative of current 4xloop plants),
  - overall verification of the code by simulation of PWR plant transients (e.g. opening of a VDA [MSRT] on the PALUEL 3 4xloop plant was accurately simulated with THEMIS using the RPV mixing model, using the MOODY correlation for the VDA [MSRT] discharge flow with the discharge quality set to 1.0: calculated loop temperatures compared well with measured values, thus validating the entire RCP [RCS] hydraulic calculation. The SG secondary side depressurisation transient calculated by THEMIS also agreed closely with the measured values, demonstrating the applicability of the secondary side model for these conditions).
- The transient analysis utilises the conservative PCC analysis rules defined in section 1. These rules include the adoption of pessimistic boundary conditions relevant to the decoupling criteria under consideration. These pessimistic assumptions address as a minimum:
  - the characterisation of the initiating event (maximisation of the resulting impact),
  - the plant initial conditions (control dead band limits, maximum measurement uncertainties),
  - the effectiveness of protection and mitigation actions (maximum uncertainty on each I&C measurement and signal delay, and on each system response time and capacity).

The analysis methodology gives a conservative result which can be used directly for the assessment of the fulfilment of the decoupling criteria.

b) With regard to the core transient (PANBOX / COBRA code), the corresponding attributes are:

- Important phenomena and qualification of the models used in the PANBOX / COBRA code: The transient belongs to the family of transients involving secondary / primary overcooling with a fast reactivity increase.
- Core behaviour with regard to core power and DNBR:
  - Reactivity distribution and power increase in the stuck rod region after the reactor becomes critical.
  - Neutron flux, reactor power (integral) locally dependent on thermal hydraulic parameters.

- Coolant flow behaviour dependent on pressure drop and cross-flow.

- Qualification of the models in PANBOX / COBRA for DNBR related phenomena: Capability to model the above phenomena is verified using international MSLB benchmark calculations.

c) The analysis methodology is based on iterative calculations between THEMIS and PANBOX / COBRA based on the following principles (see Appendix 14B.2.15 - Figure 1):

- 1st THEMIS analysis:

  Initiation of the SLB (double-ended guillotine of the main steam line at SG outlet). In this first calculation there is no consideration of reactor power (core power nil).

  Minimum mixing between loop flows within the RPV is based on typical data from current 4 x loop tests (Lacydon tests).

  Conservative initial and boundary conditions are assumed.

- 1st PANBOX / COBRA analysis :

  Calculation of the reactor power transient, using the following time dependent parameters from the 1st THEMIS calculation:

  - core inlet temperature taken as equal to THEMIS cold leg temperatures, (without assuming any further mixing),
  - core inlet flow rate,
  - core pressure,
  - core boron concentration.

  Conservative initial neutronic data are assumed.

- 2nd THEMIS analysis:

  Repeat of 1st calculation using the time dependent reactor power from the PANBOX / COBRA calculation.

The iterative calculations are repeated until good agreement is reached between the results obtained with the two codes. Convergence is considered to be achieved when:

- the change in thermal hydraulic parameters between two successive THEMIS calculations is less than 1%,
- the change in core power between two successive PANBOX / COBRA calculations is less than 1%.

2.15.1.4.2. Protection and mitigation actions

- With regard to meeting the DNBR criterion, the following F1A I&C functions provide protection in case of a main steam line break (for other criteria, see the relevant sections in 2.15.1.3 within this appendix):

- Reactor trip on:
  - core power level > MAX3,
  - DNBR < MIN3,
  - pressuriser pressure < MIN2,
  - steam generator pressure drop > MAX1,
  - steam generator pressure < MIN1.
- Safety injection on:
  - pressuriser pressure < MIN3.
- Closure of all main steam isolation valves on:
  - steam generator pressure drop > MAX1,
  - steam generator pressure < MIN1.
- Isolation of all main feedwater full load lines on:
  - steam generator pressure drop > MAX1,
  - steam generator pressure < MIN1.
- Isolation of the main feedwater low load lines of the unaffected SG on:
  - steam generator level > MAX1.
- Isolation of the main feedwater low load line of the affected steam generator on:
  - steam generator pressure drop > MAX2,
  - steam generator pressure < MIN2,
  - steam generator level > MAX1.

  at low power level (typically power < 20% NP), the full load lines are closed, and the low load lines are open.

- In addition, emergency feedwater actuation in the affected steam generator has to be considered on steam generator water level < MIN2.

The F1B systems required to transfer the plant from the controlled state to the safe shutdown state are described in section 2.16 within this appendix (feedwater line break).

2.15.1.5. Description of cases analysed (from the initiating event to attaining the controlled state)

2.15.1.5.1. Choice of single failure and preventative maintenance assumptions

The most conservative single failure, for assessing the criterion of no DNB, is a rod cluster control assembly stuck in its fully withdrawn position following reactor trip.

The challenge to the loss of nucleate boiling criterion is mainly due to a high hot channel factor in the assembly which contains the stuck rod or in its neighbouring assemblies.

Preventative maintenance is assumed to not be underway as it has no significant impact on the transient:

- for the primary side, boron injection via MHSI is not credited in the accident analysis,
- for the secondary side, maximum overcooling occurs for maximum ASG [EFWS] flow rate.

2.15.1.5.2. Initial state

A main steam line break results in more severe reactor conditions when the unit is operating at hot zero power.

If the reactor is operating at full power, the RCP [RCS] contains more energy than at hot zero power, since there is additional energy stored in the fuel. This additional energy provides thermal inertia which delays the time at which the reactor temperature and reactivity shutdown margin corresponding to hot zero power are reached.

Moreover, since the initial mass of secondary fluid and the steam generator pressure are greater at hot zero power, the amplitude and duration of the reactor coolant system cooldown are greater.

This analysis therefore postulates a non-isolatable main steam line break at hot zero power.

A small initial nuclear power is more onerous, with respect to insertion of positive reactivity, and therefore a conservative value of 10^{-9} of nominal power is chosen.

The initial RCP [RCS] temperature and pressure correspond to the hot zero power state, without uncertainties. The impact of these uncertainties is included in the initial shutdown margin assumed in the analysis.

The initial RCP [RCS] boron concentration is equal to zero, in order to maximise the positive reactivity insertion during the RCP [RCS] cooldown.

The initial conditions are presented on Appendix 14B.2.15 - Table 1.

2.15.1.5.3. Specific assumptions

a) Neutronic data and decay heat

In this study, it is assumed that the reactor power is operating under the following conditions:

- The fuel management considered for the PANBOX analysis is UO2 - 18 m - IO for an equilibrium cycle, with a stuck rod at location K2.

- End of life shutdown margin, at hot full power equilibrium xenon level, with the rod cluster control assembly having the highest worth stuck in its fully withdrawn position.

The initial shutdown margin considered in this analysis is 2000 pcm (deterministic, conservative assumption for N-1 rods). It provides additional margins, but it does not represent the lowest acceptable limit with respect to SLB.

Core physics studies demonstrate that this margin exists even under the most unfavourable conditions, in particular at the end of an equilibrium cycle when the moderator temperature coefficient reaches its highest absolute value.

In PANBOX 3D core analysis, the neutronic coefficients are adjusted by input at the start of the transient with the following assumptions:

- A maximum moderator density coefficient (in absolute value) is chosen in the analysis.

In the initial condition, the moderator temperature coefficient is -83.9 pcm/°C, including and allowance of -3.6 pcm/°C for uncertainties. This value is calculated for 0% NP (UO_2 - 12M - IO), with all rods inserted except the rod at location M2 and equilibrium xenon for 100% NP.

- A minimum differential boron worth is chosen in the analysis.

In the initial condition, the differential boron worth is -4.9 pcm/ppm, including a -10% allowance for uncertainties. This value is calculated for 0% NP (MOX - 18M - OI), with all rods inserted except the rod at location M2 and equilibrium xenon for 100% NP.

- A minimum Doppler temperature coefficient, in absolute value, is assumed, to maximise the power increase.

Core physic studies show that a stuck rod at location M2 minimises the Doppler effect in cycle 1.

In the initial conditions, the Doppler temperature coefficient is -2.5 pcm/°C, including an allowance of -20% for uncertainties.

- No credit is taken for the residual heat, thus maximising the RCP [RCS] overcooling transient.

b) Assumptions related to non-F1 systems

- MFW:

  It is assumed that a maximum main feedwater flow is delivered into all steam generators. The respective flows are conservative values corresponding to MFW operation with only low load lines open (fully open in the affected SG, at their initial position in the unaffected SG).

  In the affected steam generator, the MFW flow is 900 kg/s, until MFW isolation occurs following a SG pressure drop > MAX2 signal.

  In the unaffected steam generators, the MFW flow is 500 kg/s/SG, till MFW isolation occurs following a SG level > MAX1 signal.

- All other control systems are not included in the analysis as they have either a beneficial impact or no impact on the event.

c) Assumptions related to F1 systems:

- VIV [MSIV] (F1A):

  All VIV [MSIV] are closed following a SG pressure drop > MAX1 (2 bar / mn) signal. The setpoint of this signal is assumed to be 8.5 bar (7 + 1.5 bar) below the initial SG pressure. The delay for steam lines isolation consists of 0.9 seconds for channel delay plus 5 seconds for valve closure time (modelled by a step in flow rate).

- MFW low load line isolation (F1A):

  The MFW low load line to the affected steam generator is isolated following a SG pressure drop > MAX2 (2 bar / min) signal. The setpoint of this signal is assumed to be 18.5 bar (17 + 1.5 bar) below the initial SG pressure. The delay for feedwater isolation consists of 0.9 seconds for channel delay plus 10 seconds for valves closure time (modelled by a step in flow rate) (2 valves in series).

  The MFW low load lines to the unaffected steam generators are isolated following a SG level > MAX1 (70.6 + 5% of the narrow range) signal. The delay for feedwater isolation consists of 1.5 seconds for channel delay plus 10 seconds for valves closure time (modelled by a step in flow rate) (2 valves in series).

- MHSI (F1A):

  A minimum safety injection system capacity is assumed, consistent with the 80 bar a minimum delivery pressure (see section 0.2 within this appendix).

  The temperature of MHSI is assumed to be the minimum IRWST temperature of 10°C.

  It is also conservatively assumed that the boron concentration in the in-containment refuelling water storage tank is 0 ppm.

The delay assumed before MHSI injection starts consists of:

- the time to generate the safety injection signal, consisting of the time to reach the limiting value for the safety injection signal setpoint of 112 bar (115 - 3 bar) plus the channel delay of 0.9 seconds to generate the signal,
- the time to start the MHSI pumps (10 seconds),
- the time for MHSI pumps to reach full flow (5 seconds).

- ASG [EFWS] (F1A):

  A maximum emergency feedwater flow rate is assumed, supplying only the affected steam generator, since the ASG [EFWS] actuation setpoint (SG level < MIN2) is not reached during the transient in the unaffected steam generators.

  This maximum flow of 200 t/h corresponds to the operation of one ASG [EFWS] pump at 1 bar a, with no credit claimed for the active flow limitation.

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

  In addition, it is conservatively assumed that the ASG [EFWS] is actuated in the affected steam generator at the beginning of the transient.

d) Other assumptions:

- Assumptions related to Reactor Coolant Pump trip:

  If there was a loss of offsite power at the beginning of the event, the resultant accident conditions would be less severe. This is because when offsite power is lost, coolant flow in the RCP [RCS] decreases, which reduces the temperature gradient across the steam generators and consequently the cooldown rate of the RCP [RCS].

  There is no automatic Reactor Coolant Pump trip implemented to mitigate the steam line break event. Nevertheless, in the case of a SLB inside containment, the containment isolation stage 2 signal may be generated following a containment pressure > MAX2 signal. The Reactor Coolant Pumps are then tripped via this signal as they are no longer cooled by RRI [CCWS].

  The most onerous time for Reactor Coolant Pump trip is a few seconds before the power peak reached for a case where offsite power is assumed available.

  The Reactor Coolant Pump trip will not impact the temperature gradient across the steam generators and hence the power peak. However, the reduction in the RCP [RCS] flow rate might result in a slight decrease in the DNBR.

  To assess Reactor Coolant Pump trip, two cases are analysed:

  - without Reactor Coolant Pump trip, base case (i.e. forced circulation is maintained),
  - with Reactor Coolant Pump trip occurring a few seconds before the power peak (power peak reached in the without Reactor Coolant Pump trip case).

- Steam flow through the break is computed by the Moody correlation at each calculation step. A perfect moisture separation in the steam generator secondary side is assumed (quality equal to 1 at steam generator outlet). Pure steam flow at the break conservatively maximises the RCP [RCS] cooling. The back pressure is conservatively assumed to remain at atmospheric pressure.

  During the first seconds of the transient (prior to steam lines isolation):

  - the steam flow from the affected steam generator is limited by the flow limiter, with a cross sectional area of 0.13 m²,
  - 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 sectional area of 0.32 m².

- A maximum SG-tubes heat transfer coefficient is used in the analysis.
- No credit is taken for heat stored in metal parts other than the fuel rods and the tubes of the steam generators.
- For mixing at the core inlet, it is assumed that a maximum of 65% of the flow entering through inlet nozzle remains in the associated core quadrant at core inlet. The minimum loop flow mixing within the RPV is a conservative assumption for the subsequent core power transient.

The main assumptions are given on Appendix 14B.2.15 - Table 1.

2.15.1.5.4. Results

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 at 4 seconds which results in steam lines isolation 5.9 seconds later. After VIV [MSIV] closure, only the affected steam generator continues to depressurise via the break.

The SG pressure drop > MAX2 signal setpoint is reached after 7.7 seconds in the affected SG. It results in the complete isolation of MFW (10.9 seconds later) to the affected SG which is then only supplied by ASG [EFWS].

The shutdown margin is eroded and the reactor becomes critical and consequently thermal power is increasing.

The Doppler effect limits the thermal power increase but does not stop it. The maximal thermal power is reached as the affected steam generator empties, in the case of Reactor Coolant Pump ON, or a few seconds after Reactor Coolant Pump trip in the case of Reactor Coolant Pump OFF.

Following steam generator dryout, the thermal power decreases down to a power level consistent with the boiling of the ASG [EFWS] flow rate.

After a quick stabilisation of the thermal hydraulic parameters, the controlled state is reached, assuming the following F1A means only:

- VIV [MSIV] and MFW for affected SG isolation,

- ASG [EFWS] and VDA [MSRT] for RCP [RCS] heat removal (required only in the unaffected SG),
- MHSI actuation on SI signal.

Consequently, the rapid secondary transient is finished with:

- the core just critical (i.e. reactivity equal zero),
- the core power removed via the break and ASG [EFWS] in the affected steam generator (core power stabilised at approximately 3% of full power),
- a stable primary coolant inventory.

Appendix 14B.2.15 - Table 2 gives the sequence of events, with and without Reactor Coolant Pump trip.

Appendix 14B.2.15 - Figure 2 to 6 show the evolution of the main parameters during the transient without Reactor Coolant Pump trip.

Appendix 14B.2.15 - Figure 7 to 11 show the evolution of the main parameters with Reactor Coolant Pump trip.

Appendix 14B.2.15 - Table 3 gives the main thermal hydraulic parameters at the time of minimum DNBR, with and without Reactor Coolant Pump trip.

The minimum DNBR is 2.2 without Reactor Coolant Pump trip and 2.1 with Reactor Coolant Pump trip.

The Reactor Coolant Pump trip does not result in a significant decrease of the minimum DNBR compared to a case without Reactor Coolant Pump trip.

Therefore the decoupling criterion of no core DNB (i.e. DNBR > 1.20 according to section 0.2.8 within this appendix) is met.

2.15.1.6. 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 subcritical, the LHSI/RHR operating conditions have been reached and the affected steam generator is isolated.

The transition from the controlled state to the safe shutdown is covered by that of the feedwater line break (see section 2.16 within this appendix). Following isolation of the affected steam generator, the RCP [RCS] and SG states are quite similar in the cases of the steam line break and feedwater line break accident. In addition, the capabilities of the F1B systems needed in both cases are identical (VDA [MSRT], ASG [EFWS], RBS [EBS], PSV).

2.15.2. Steam system piping failure (in state B, PCC- 4)

2.15.2.1. Accident definition

A steam system piping failure in state B is classified as a PCC- 4 event.

The steam release as a consequence of a rupture of a main steam line results in an initial increase in steam flow which decreases during the accident as the steam pressure falls.

The energy removal from the RCP [RCS] causes a reduction of coolant temperature and pressure. In the presence of a negative moderator density coefficient, the cooldown leads to an insertion of positive reactivity.

The main differences, in terms of protection and mitigation actions, between a steam line break in state A (analysed in section 2.15.1) and a steam line break in state B are as follows:

- The RIS [SIS] signal in state A is initiated following a "PZR pressure < MIN3" signal. This signal is vetoed in state B for normal depressurisation purpose when operating below hot shutdown conditions. In this case, the RIS [SIS] relies on the F1A classified "hot leg ΔP_{sat} < MIN" signal.

- The "SG pressure < MIN1 or MIN2" signals (leading to all VIV [MSIV] closure, and closure of the MFW low load line of the affected steam generator) have been vetoed for RCP [RCS] normal cooldown purpose, when operating below hot shutdown conditions.

  Therefore, in these conditions, isolations are only initiated following steam generator pressure drop signals. Consequently, there is no difference for the larger steam line breaks in state B, which actuate the pressure drop signal at -2 bar/mn.

  However the VIV [MSIV] closure and MFW isolation signals do not occur for the smaller breaks as the drop in secondary pressure is not sufficient to actuate the pressure drop signal.

- The accumulators are isolated during the normal plant cooldown, if the RCP [RCS] pressure is below 60 bar. However, the accumulators are not required for the fault initiated in state B as MHSI injection is sufficient to stop the RCP [RCS] depressurisation, as is the case in state A.

  Such a difference does not introduce any significant difference to the consequences of the steam line break in state B.

The main difference, in terms of core conditions, between a steam line break in state A (analysed in section 2.15.1) and a steam line break in state B is the following:

- All the control rods are inserted consistent with the definition of state B.

  This difference is highly beneficial for the steam line break transient in state B.

2.15.2.2. Safety criteria

The safety criteria and decoupling criteria are the same as those described in section 2.15.1.2 within this appendix, for a main steam line break in state A.

2.15.2.3. Definition of studied cases

- Fuel cladding integrity

  The analysis is performed to demonstrate that the following decoupling criterion of no core damage is satisfied: there will be no DNB for the case of a non-isolatable double-ended guillotine steam line break.

  The demonstration provided below relies on qualitative explanations.

- Reactor containment

  The analysis is performed to demonstrate that the containment pressure and temperature limits, are not exceeded.

  The demonstration provided below relies on qualitative explanations.

- Radiological consequences

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

  The bounding PCC-4 transient, with regard to radiological releases, is the steam generator tube rupture of 2 tubes analysed in section 2.18 within this appendix.

2.15.2.4. Protection and mitigation actions

- The following F1A I&C functions provide protection in case of a main steam line break:
  - Safety injection following a:
    - hot leg ΔP_{sat} < MIN signal.
  - Closure of all main steam isolation valves following a:
    - steam generator pressure drop > MAX1 signal.
  - All main feedwater full load lines are already isolated in state B.
  - Isolation of the main feedwater low load lines of the unaffected SG following a:
    - steam generator level > MAX1 signal.
  - Isolation of the main feedwater low load line of the affected steam generator following a:
    - steam generator pressure drop > MAX2 signal,
    - steam generator level > MAX1 signal.

- In addition, emergency feedwater actuation in the affected steam generator is assumed on a steam generator water level < MIN2 signal.

  The F1B systems required to transfer the plant from the controlled state to the safe shutdown state are described in section 2.16 within this appendix (feedwater line break).

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

2.15.2.5.1. Choice of single failure and preventative maintenance

No single failure is considered as it has no impact on the transient behaviour:

- all rods are inserted consistent with the definition of state B,
- the amount of water injected by the MHSI is negligible and boron injection via the MHSI is not credited in the accident analysis,
- if the single failure was postulated on the VIV [MSIV] of the affected SG (which fails to close) the other remaining three VIV [MSIV] would close isolating the unaffected SG and thus avoiding steam release from the unaffected SG.

In a similar way, preventative maintenance is not considered, as it has no impact on the transient behaviour.

2.15.2.5.2. Initial state

State B is an intermediate shutdown state above 120°C, as defined in section 0.1.2 within this appendix. The LHSI/RHR is not connected to the RCP [RCS] in normal operation above this temperature.

The initial conditions are chosen in order to maximise the RCP [RCS] cooldown following the steam line break:

- The initial RCP [RCS] boron concentration: RCP [RCS] boration is performed during the RCP [RCS] normal cooldown while transferring the plant from state A to state C. The RCV [CVCS] charging flow compensates for the RCP [RCS] contraction by the injection of borated water (at 7000 ppm). When the steam line break occurs, this boration will result in a shutdown margin greater than or equal to that for hot shutdown conditions (in state A).
- The PZR water level is identical to that at hot shutdown conditions.
- The RCP [RCS] temperature is assumed to that for hot shutdown conditions. This maximises the insertion of positive reactivity due to the RCP [RCS] cooldown.
- The RCP [RCS] pressure is assumed to be that for hot shutdown conditions consistent with the RCP [RCS] temperature according to the P/T diagram.

2.15.2.5.3. Results

a) Fuel cladding integrity

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

  Following the break initiation, the secondary system rapidly depressurises. The SG pressure drop signal setpoint is reached in the first few seconds of the transient. The resultant signals initiate all VIV [MSIV] closure and MFW isolation in the affected SG.

  After this isolation, only the affected steam generator, which is supplied by ASG [EFWS], continues to depressurise.

  The energy removed from the RCP [RCS] causes a reduction in the coolant temperature and pressure.

  The RCP [RCS] cooldown results in an insertion of a positive reactivity via the negative moderator coefficient. However, the initial RCP [RCS] boron concentration is sufficient to maintain a shutdown margin higher than that for hot shutdown conditions, with all rods inserted. Therefore the consequences of a steam system piping failure in state B are bounded by those of the fault in state A as the initial shutdown margin is higher (all rods are inserted in state B).

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

  Following the break initiation, the secondary system rapidly depressurises. However, the SG pressure drop signal setpoint is not reached and the « SG pressure < MIN1 or MIN2 » signals are vetoed. Consequently, the VIV [MSIV] are not closed and MFW is not isolated in the affected SG.

  All steam generators contribute to the RCP [RCS] cooldown at a maximum rate of 2°C per minute (setpoint of the SG pressure drop signal). Such a cooling rate should not lead to any core damage and DNB should be prevented, assuming an operator action at 30 minutes (manual isolation of the affected steam generator).

  However, should these conditions be shown to be unacceptable, a dedicated automatic F1A signal would be introduced in order to automatically close all VIV [MSIV] and the MFW line of the affected SG in state B (e.g. low cold leg temperature).

  Safety analyses in state B, performed later in course of the UK EPR design progress (as part of the detailed design), will provide the justification for the solution retained.

b) With regard to reactor containment

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

  In these cases, steam lines are isolated on the SG pressure drop signal, at the very beginning of the transient.

  The consequences of such an event in state B are hence bounded by those shown to be acceptable for the fault in state A.

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

  With the Steam lines not being isolated, the steam is released into the containment from all four steam generators. The high containment pressure reactor trip signal setpoint is then reached. Thirty minutes after this first signal, the operator will manually close all the VIV [MSIV], and isolate MFW and ASG [EFWS] in the affected SG (if already started). These small leaks should not lead to containment pressure peaks more onerous than those occurring following the fault in state A.

2.15.2.6. 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 subcritical, 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 2.15.1 within this appendix for a steam line break in state A.

APPENDIX 14B.2.15 – TABLE 1

Steam system piping failure
Main assumptions

Parameters		Values used
Initial conditions		
- Reactor power	(FP)	10^{-9}
- Shutdown margin	(pcm)	- 2000
- RCP [RCS] boron concentration	(ppm)	0
- RCP [RCS] flow rate		Thermal hydraulic
- Average RCP [RCS] temperature	(°C)	301
- Pressuriser pressure	(bar a)	155
- Pressuriser level	(% of MR)	28 + 5 = 33
- Flow limiter cross section	(m²)	0.13
- VIV [MSIV] cross section	(m²)	0.32
- Initial MFW flow rate in the affected SG	(kg/s)	900
- Initial MFW flow rate in the unaffected SG	(kg/s)	500
- ASG [EFWS] flow rate in the affected SG	(kg/s)	56
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)	0
. in the safety injection lines downstream from the connection of the pump mini-flow line returning into the IRWST	(ppm)	0
SG pressure drop > MAX1 setpoint	(bar/min)	2 Setpoint adjusted to 8.5 bar below the initial value (7 + 1.5 bar)
SG pressure drop > MAX2 setpoint	(bar/min)	2 Setpoint adjusted to 18.5 bar below the initial value (17 + 1.5 bar)
PZR pressure < MIN3 setpoint	(bar a)	115 - 3 = 113
Steam line isolation delay	(s)	0.9 + 5 = 5.9
Main feedwater low load line isolation delay on : - SG pressure drop > MAX2 - SG level > MAX1	(s)	 0.9 + 10 = 10.9 1.5 + 10 = 11.5

APPENDIX 14B.2.15 – TABLE 2

Steam system piping failure
Sequence of events

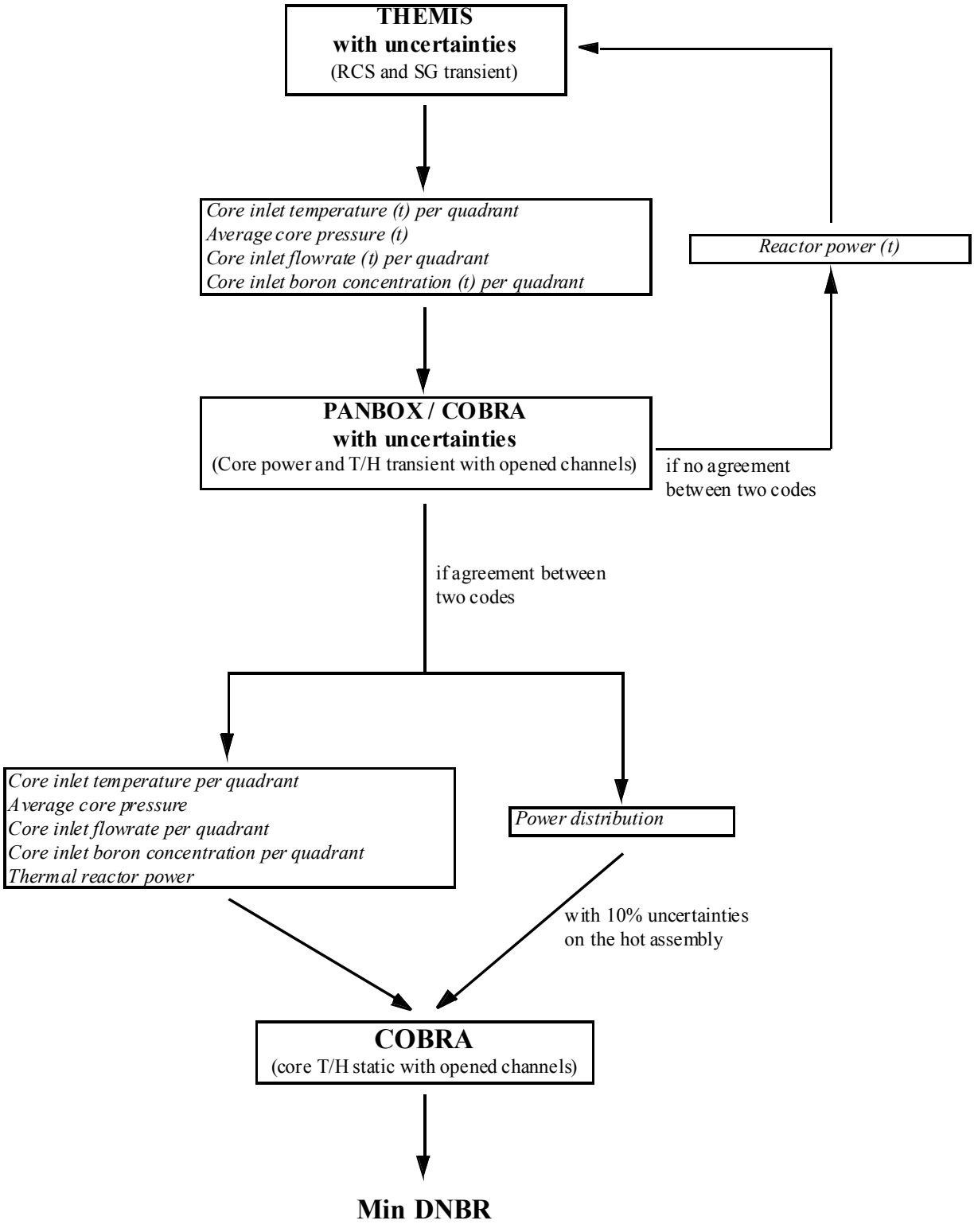
EVENT	TIME (seconds)	
	Without Reactor Coolant Pump trip	With Reactor Coolant Pump trip
Main steam line break	0	0
SG pressure drop > MAX1 setpoint is reached (actuation of RT)	4.0	4.0
SG pressure drop > MAX2 setpoint is reached in affected SG	7.7	7.7
Steam lines are isolated	9.9	9.9
Main feedwater low load line isolation in the affected SG	18.6	18.6
Reactor becomes critical	20.0	20.0
Pressuriser is empty	33.6	33.6
PZR pressure < MIN3 setpoint is reached (RIS[SIS] actuation signal)	35.2	35.2
MHSI begins to inject	57.7	57.7
SG level > MAX1 setpoint is reached in unaffected SG	60.5	60.5
MFW low load lines isolation in the unaffected SG	72	72
Reactor Coolant Pump trip	No	367.0
Minimum DNBR is reached	370.0	368.0
Affected steam generator is empty	387.0	397.0

Appendix 14B.2.15 – Table 3

Steam system piping failure
Reactor conditions at the time of minimum DNBR

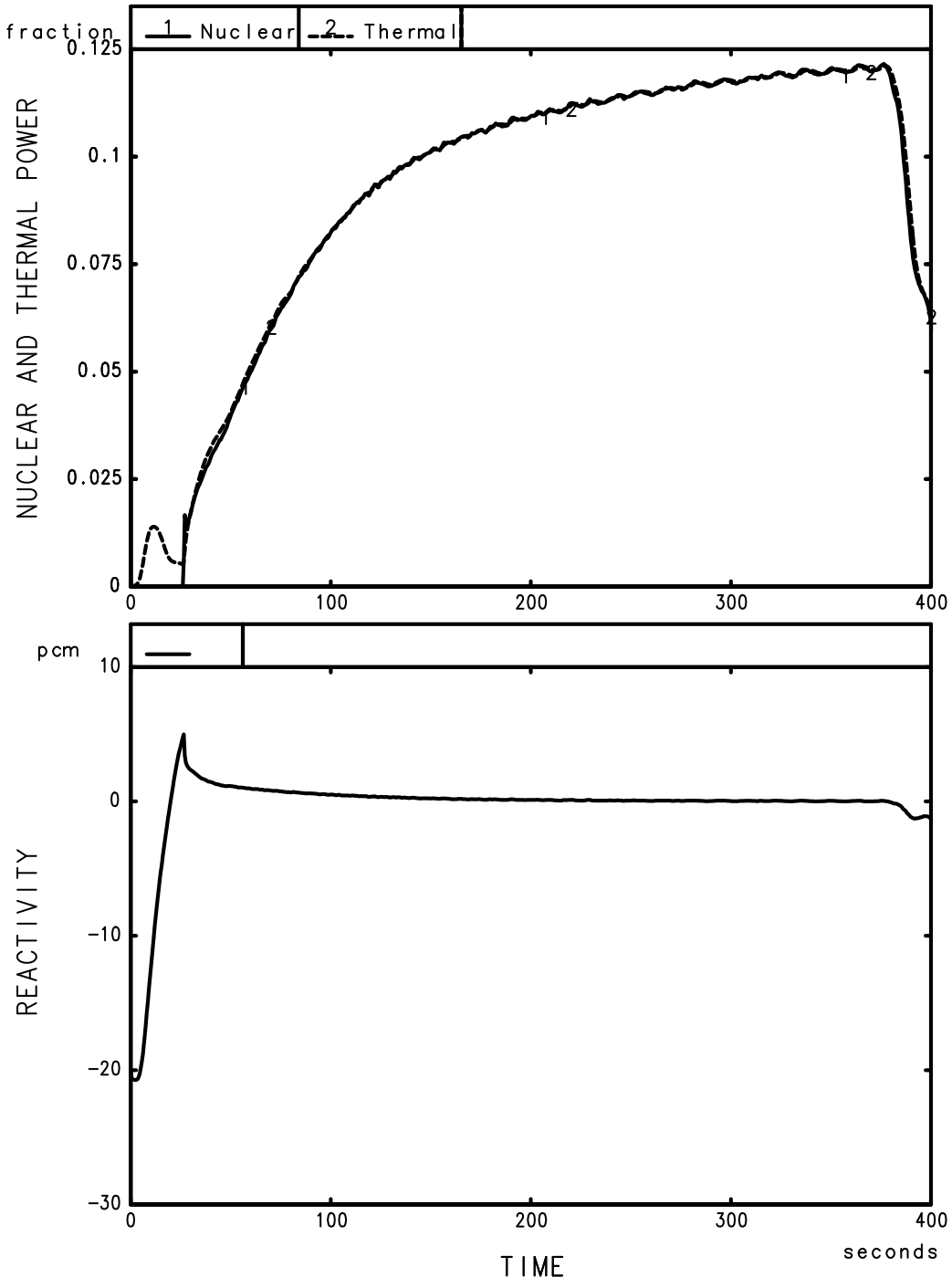
	Without Reactor Coolant Pump trip	With eactor Coolant Pump trip
Time (seconds)	370	368
Thermal power (%)	12.1	11.4
Concentration of boron in the core (ppm)	0	0
Average core pressure (bar a)	70.8	70.7
Cold leg temperature in affected loop (°C)	207.2	207.3
Cold leg temperature in unaffected loops (°C)	252.1	252.2
Core flow rate (fraction of nominal)	100	0.95
Minimum DNBR	2.2	2.1

APPENDIX 14B.2.15 - FIGURE 1



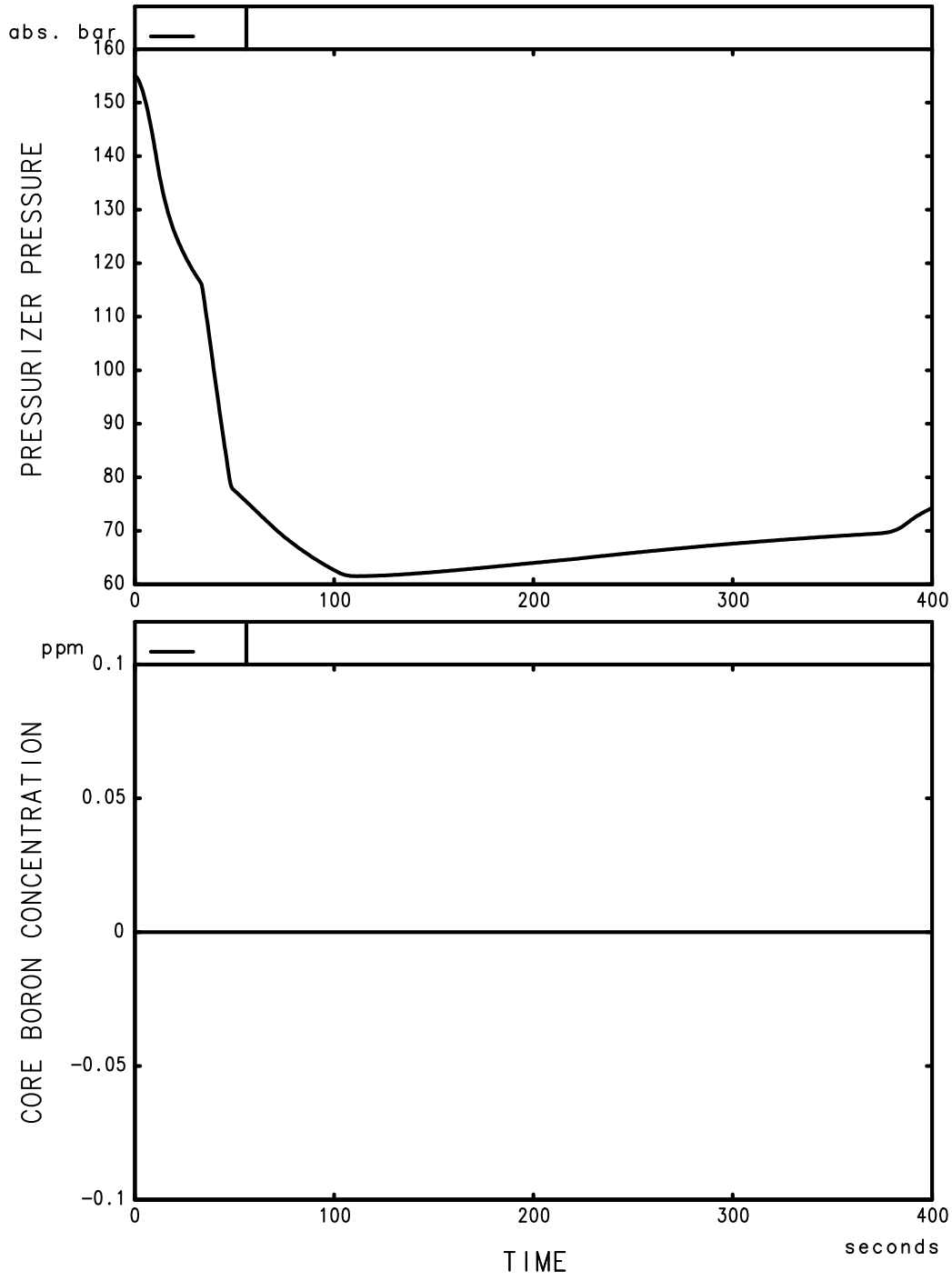
Steam system piping failure - Methodology of analysis

APPENDIX 14B.2.15 - FIGURE 2



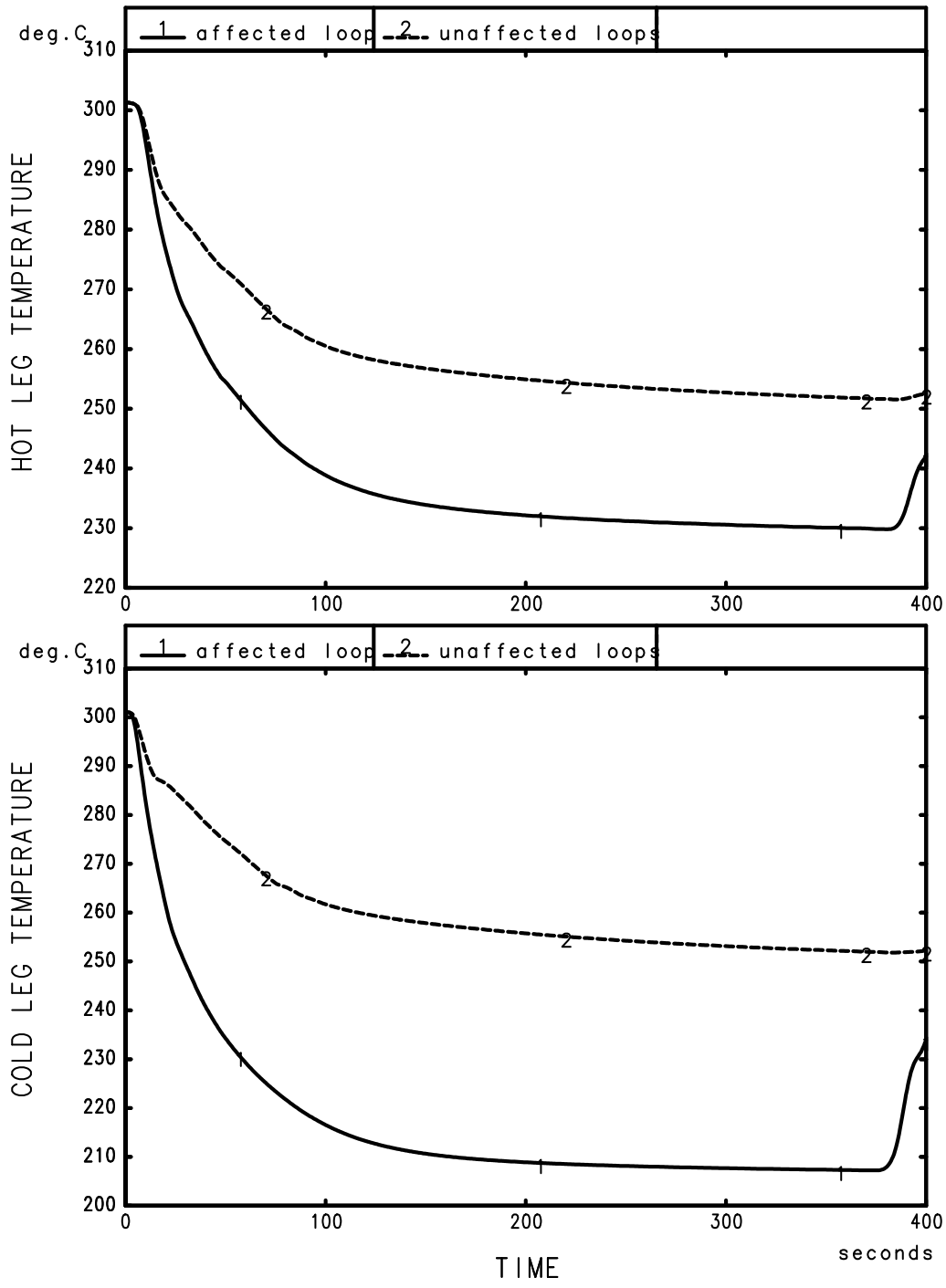
Steam system piping failure without Reactor Coolant Pump trip

APPENDIX 14B.2.15 - FIGURE 3



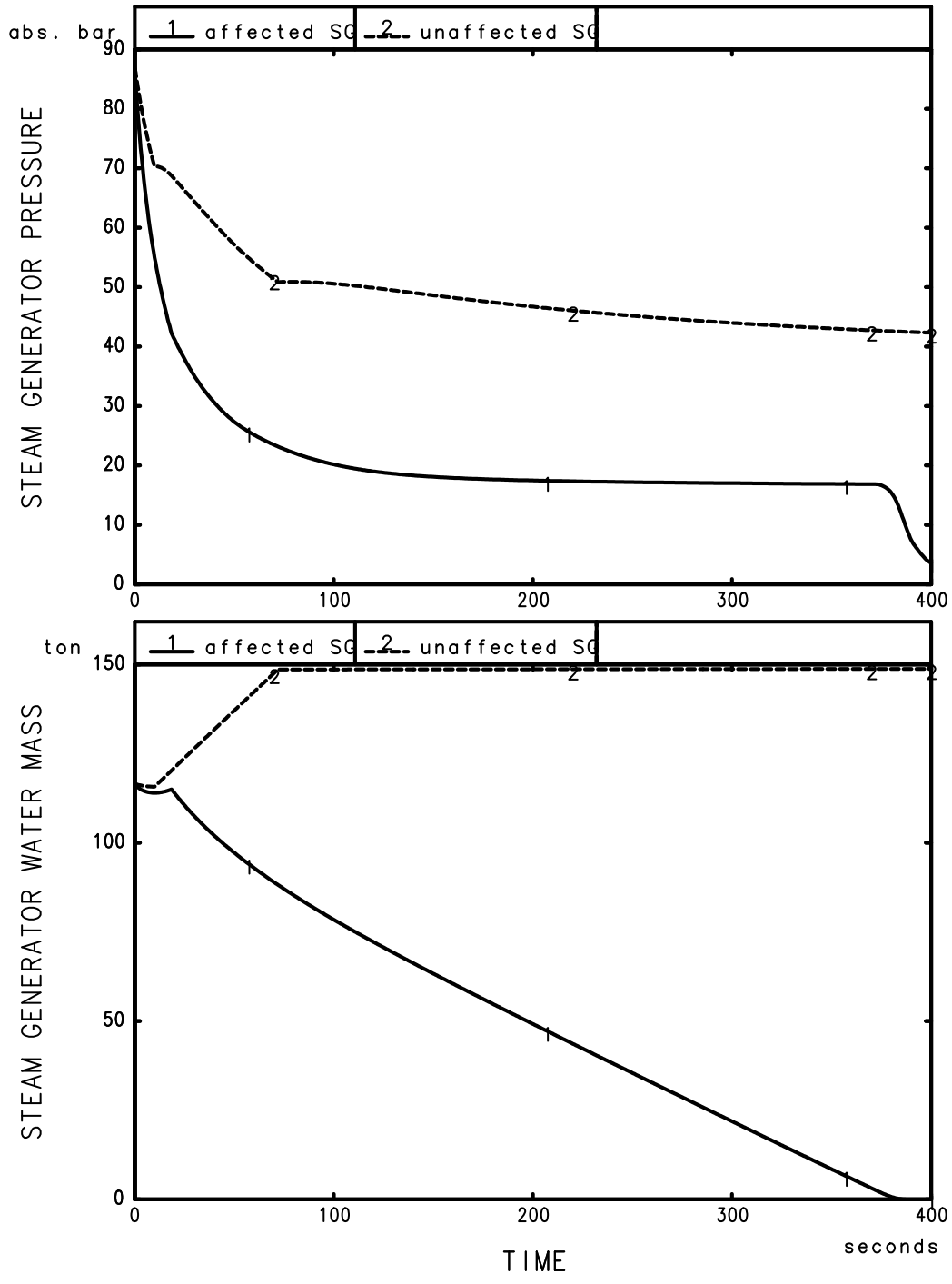
Steam system piping failure without Reactor Coolant Pump trip

APPENDIX 14B.2.15 - FIGURE 4



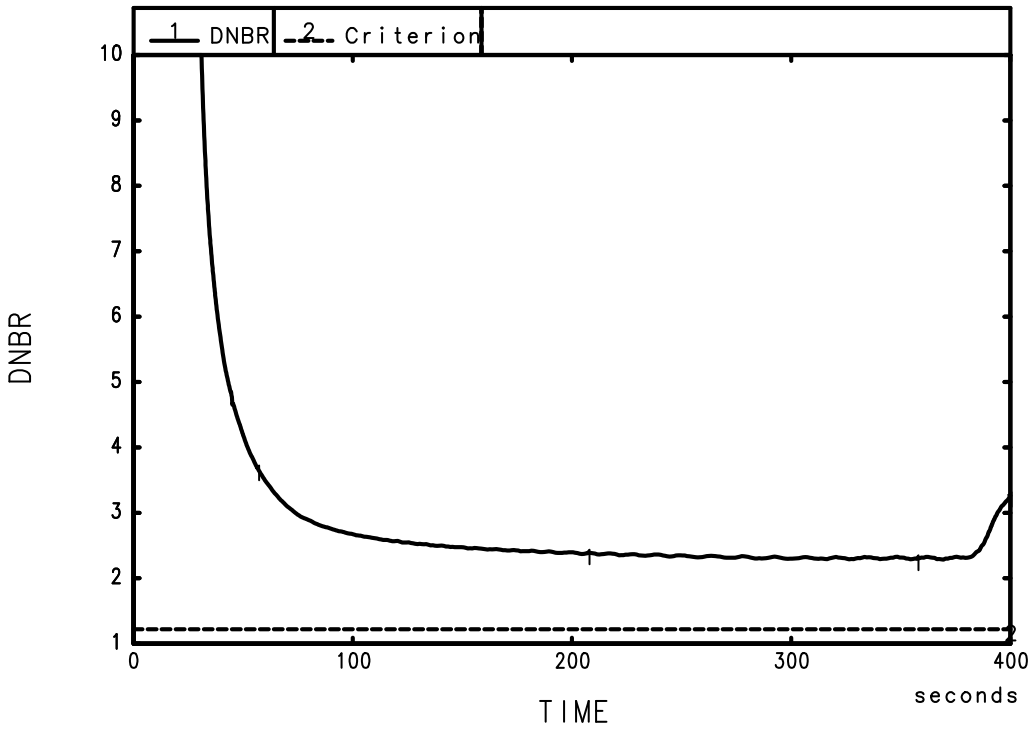
Steam system piping failure without Reactor Coolant Pump trip

APPENDIX 14B.2.15 - FIGURE 5



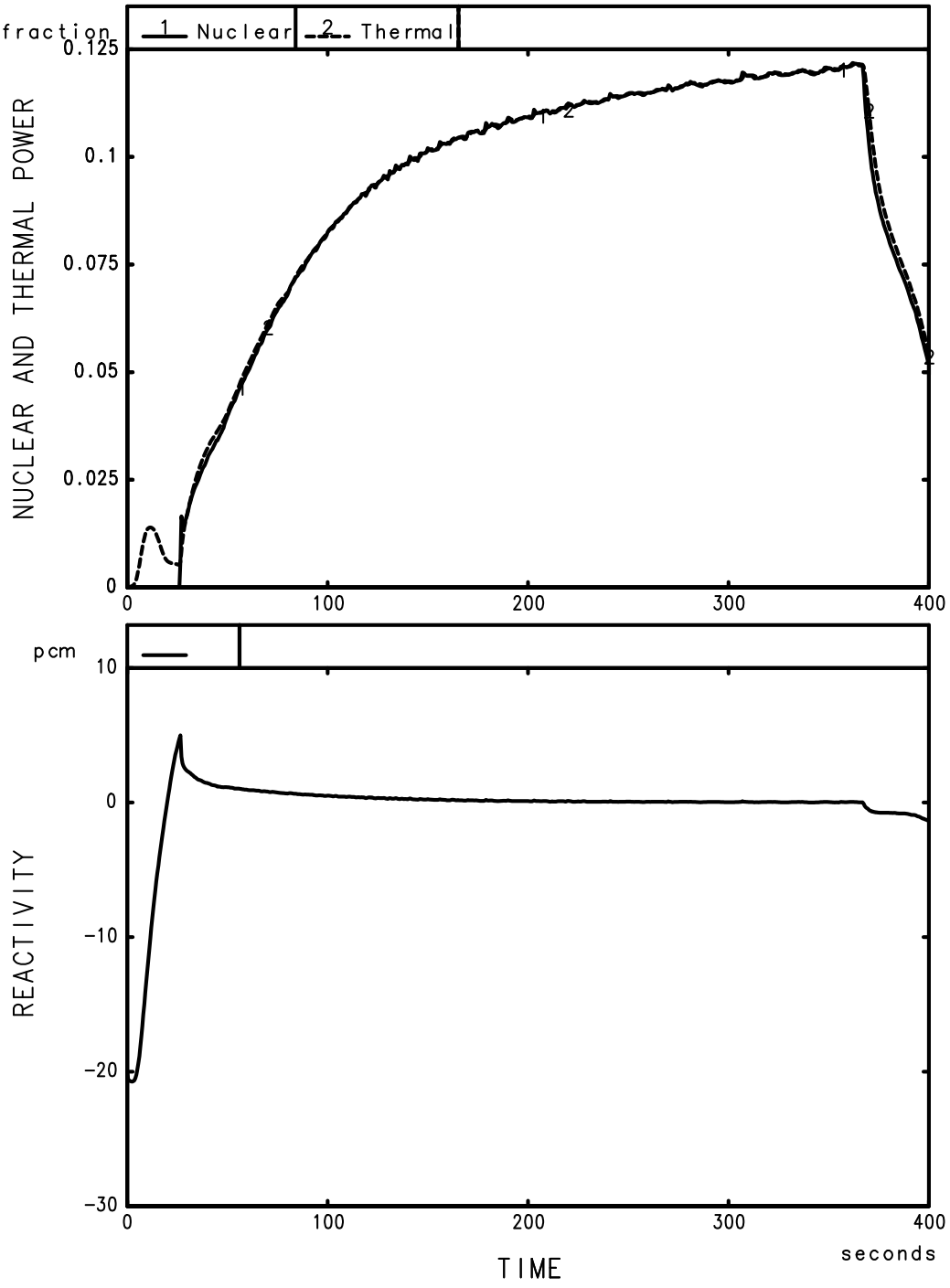
Steam system piping failure without Reactor Coolant Pump trip

APPENDIX 14B.2.15 - FIGURE 6



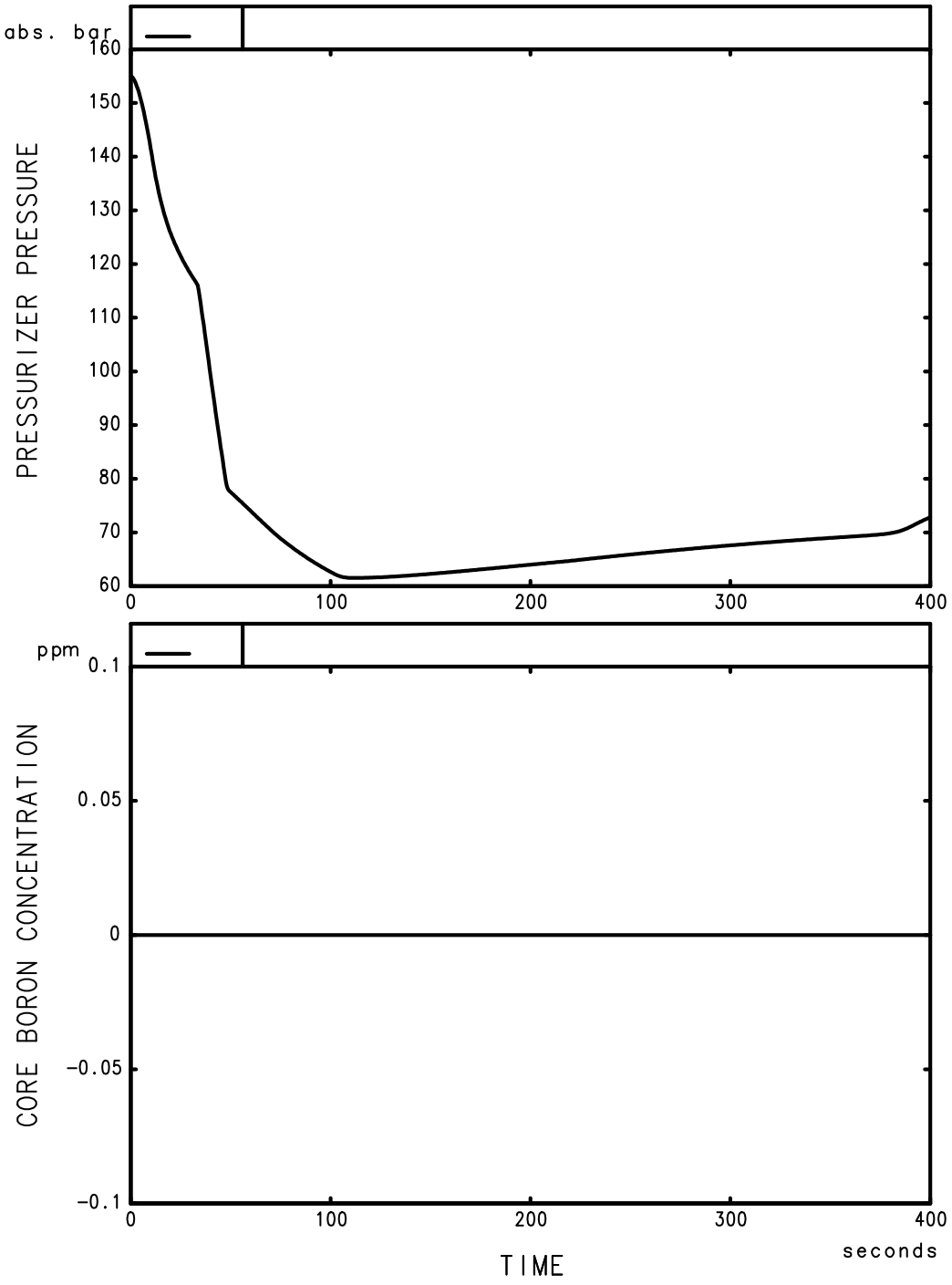
Steam system piping failure without Reactor Coolant Pump trip

APPENDIX 14B.2.15 - FIGURE 7



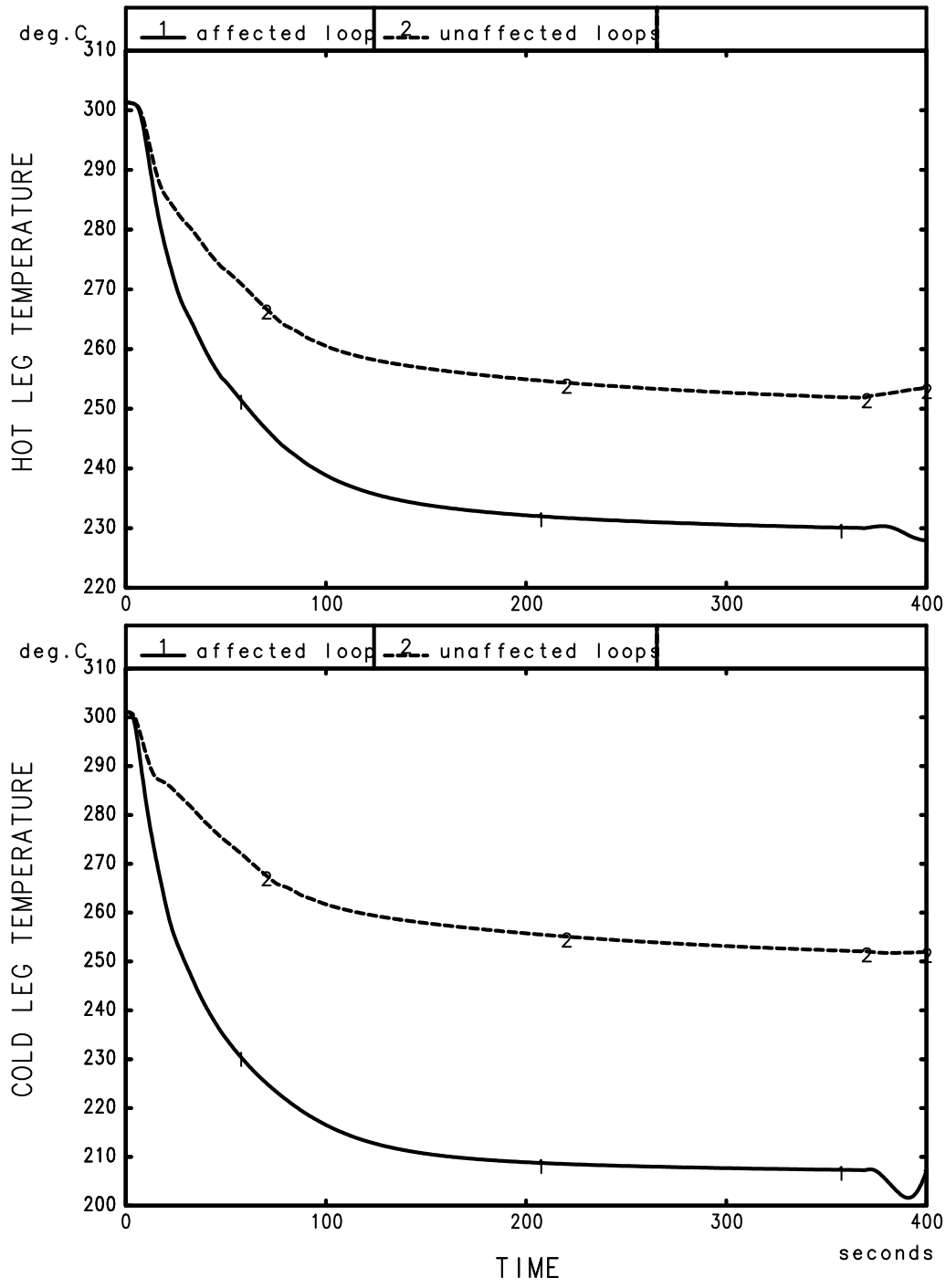
Steam system piping failure with Reactor Coolant Pump trip

APPENDIX 14B.2.15 - FIGURE 8



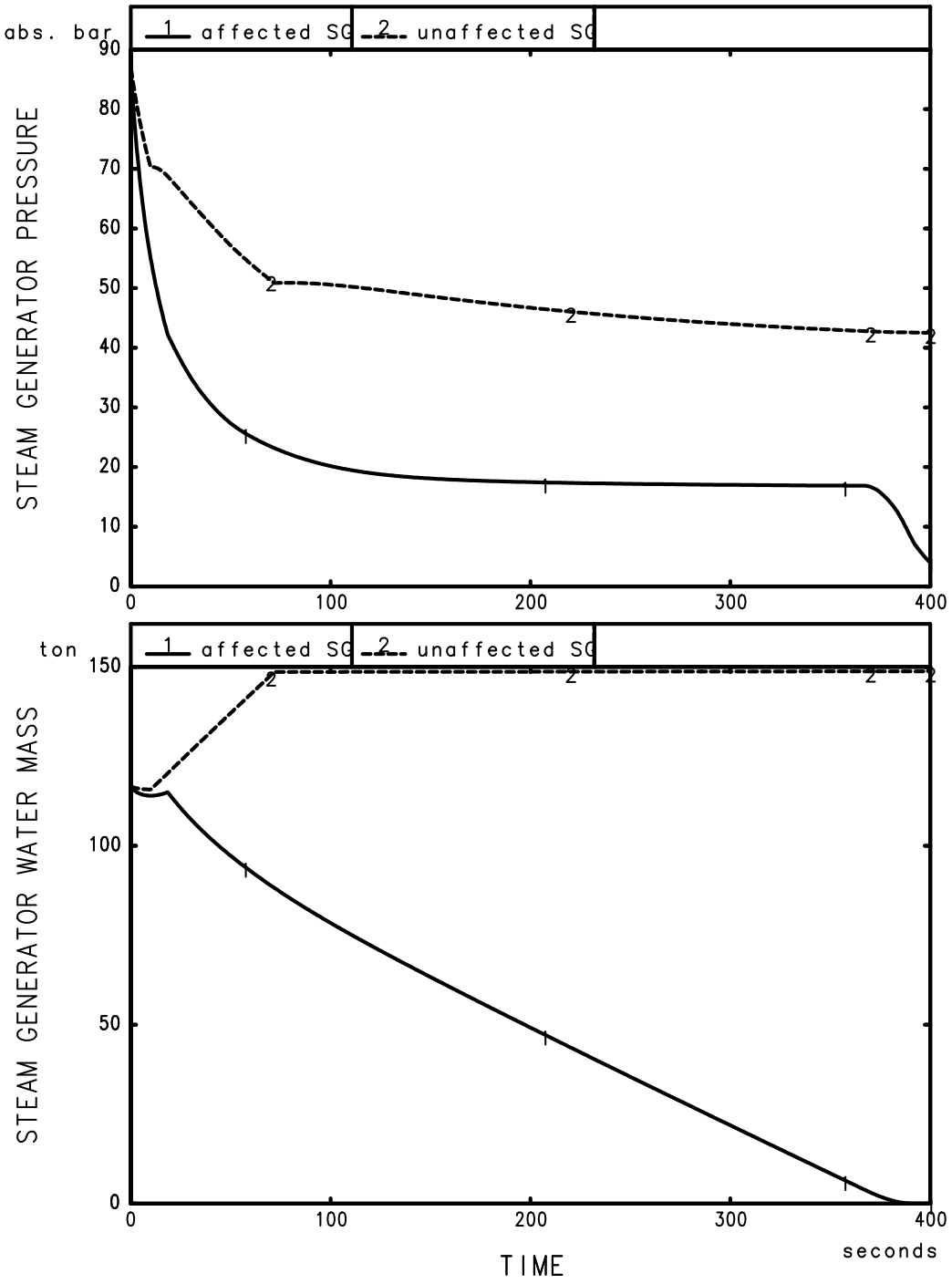
Steam system piping failure with Reactor Coolant Pump trip

APPENDIX 14B.2.15 - FIGURE 9



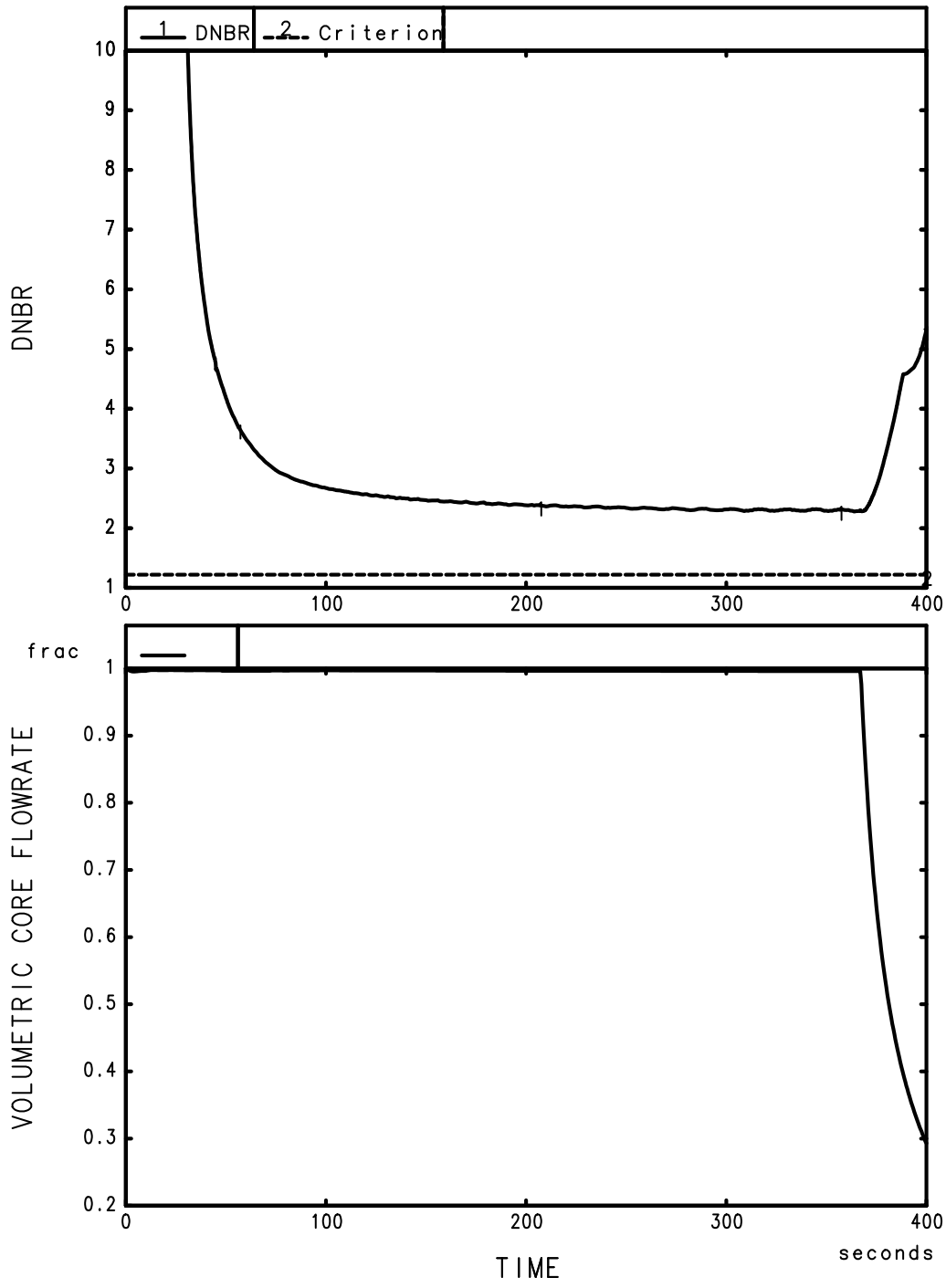
Steam system piping failure with Reactor Coolant Pump trip

APPENDIX 14B.2.15 - FIGURE 10



Steam system piping failure with Reactor Coolant Pump trip

APPENDIX 14B.2.15 - FIGURE 11



Steam system piping failure with Reactor Coolant Pump trip

2.16.1. Feedwater system pipe break (in state A, PCC4)

2.16.1.1. Identification of causes and accident description

2.16.1.1.1. General concern

The accident assessed is defined as a break located on a Main Feedwater System pipe, sufficiently large to prevent the supply of main feedwater to the Steam Generator.

This 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 a fast discharge of SG water) depending on the break size and the AAD [SSS] power level at the break initiation. The consequences of a RCP [RCS] overcooling are bounded by the 'Steam Line Break' analysis in section 2.15 of this appendix. In this section, only the RCP [RCS] overheating aspect is considered.

For a break located upstream of the non-return valve, the consequences of the accident are similar to those resulting from a 'loss of Main Feedwater System' accident.

The break size considered in this section considers the largest pipe located downstream of the ARE [MFWS] non-return valve corresponding to a PCC-4 event. This accident bounds the break of an ASG [EFWS] line or a SGBS line when assessing the RCP [RCS] heat-up aspect.

2.16.1.1.2. Typical sequence of events

The typical sequence of events following a Feedwater Line Break (FWLB) (excluding the impact from assessment of the single failure criterion) is as follows:

a) From the initiating event to the controlled state

Following break initiation, the secondary system depressurises and the affected SG liquid level decreases as liquid mass is lost via the break. Due to the loss of main feedwater flow via the break, the unaffected SG liquid contents also decrease but at a slower rate than in the affected SG.

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

During this phase a reactor trip occurs either on a "PZR pressure > MAX2" reactor trip signal or on a reactor trip signal generated from the secondary system, depending on the break size. The secondary side trip signal could be initiated from one of the following signals:

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

The reactor trip signal automatically trips the turbine and, if available, the steam dump valves open permitting steam dump to the condenser. If the GCT [MSB] is unavailable, the steam generator pressure rapidly increases resulting in steam discharge to the atmosphere through the MSRV. The unavailability of the GCT [MSB] could be due to either the VIV [MSIV] closure, failure as it is not an F1-classified system, or following a LOOP assumed to occur following the turbine trip),

For a FWLB size sufficient to remove the primary heat generated, the SG pressure falls until isolation of the main steam header following either a "SG pressure drop MAX1" or "SG pressure MIN1" signal. Following VIV [MSIV] closure the three unaffected SG experience a pressure increase, while the affected SG continues to depressurise via the break.

Following the generation of a "SG pressure drop MAX2" or "SG pressure MIN2" signal, the ARE [MFWS] line to the affected SG is isolated. This signal is only generated on the affected SG. This ensures that the ARE [MFWS] or AAD [SSS] water supply in the unaffected SG remains available. In the accident analysis, the ARE [MFWS] or AAD [SSS] are not credited as the system is not F1 classified.

Following reactor trip, the mismatch between the primary-to-secondary heat transfer and the feed supplied results in a continued decrease in liquid inventory in both the affected and unaffected SG. When the “SG level < MIN2” setpoint is reached in one SG, the associated ASG [EFWS] pump is started following a delay corresponding to the Diesel reloading sequence in the case of LOOP. Consequently, an ASG [EFWS] flow is sent to the corresponding SG to restore the liquid inventory. In the affected SG is conservatively assumed that the entire ASG [EFWS] flow sent towards this SG is lost via the break.

While the ASG [EFWS] flow rate entering the SG secondary sides remains insufficient to remove the entire heat transferred from the primary system, both decay heat and Reactor Coolant Pump heat, the temperature and pressure of the RCP [RCS] increase. When the pressuriser pressure reaches the PSV setpoints (PZR pressure MAX3, MAX4, MAX5 respectively), the corresponding PSV opens. This causes a RCP [RCS] discharge into the IRWST. In the accident analysis, the pressuriser spray is not claimed as the system is not F1 classified.

During this phase, from RT to the controlled state, both systems ASG [EFWS] and VDA [MSRT] limit the RCP [RCS] heat-up, whilst any RCP [RCS] overpressure is limited by the PSV.

The controlled state is reached when the RCP [RCS] temperature and pressure are stabilised with the primary side heat fully removed via the ASG [EFWS] supplied SG. The controlled state can be reached:

- either automatically before 30 minutes, relying only on F1 systems started automatically. This occurs when two unaffected SG are fed, the flow from two ASG [EFWS] pumps removing more than the RCP [RCS] heat load,

- or following an operator action claimed 1 hour(1) from RT. This action is required when only one unaffected SG is fed. This results from assuming a single failure and maintenance on the ASG [EFWS] delivering to the two remaining unaffected SG. In this case, the flow from one ASG [EFWS] pump is not sufficient to remove the entire RCP [RCS] heat. The operator actions to isolate the ASG [EFWS] line to the affected SG and realign the ASG [EFWS] pump delivering to the affected SG to one of those two remaining unaffected SG are necessary to ensure the removal of the entire RCP [RCS] heat load via the SG.

(1) Operator action is considered at RT + 30 min, if performed from the Main Control Room.

b) From the controlled state to the safe shutdown state

The safe shutdown state is defined as a state where the RIS/RRA [SIS/RHRS] is operating in RHR-mode. In this state:

- 1oo4 RIS/RRA [SIS/RHRS] train is sufficient to ensure the removal of the RCP [RCS] heat load.
- Connection conditions are:
  - RCP [RCS] hot leg pressure < 30 bar, and
  - RCP [RCS] hot leg temperature < 180°C, and
  - ΔT_{sat} [(2)] and RPVL consistent with RIS/RRA [SIS/RHRS] suction from the hot leg.

The sequence of actions to be performed by the operator in order for the plant to reach the safe shutdown state is as follows:

Affected SG isolation:

At the operator action time (assumed to be 30 minutes after reactor trip), the affected SG is depressurised. The first operator action is to completely isolate this SG for both the steam discharge and feed water supply. This is performed by closing the VIV [MSIV] if this has not occurred already and by isolating the associated ASG [EFWS] line. This will prevent the associated ASG [EFWS] tank from emptying via the break and will prevent an excessive pressure increase in the containment.

ASG [EFWS] passive header (pump discharge) opening:

Following isolation of the ASG [EFWS] to the affected SG, the associated ASG [EFWS] flow can be directed to another SG via the ASG [EFWS] header if required. As stated above, ARE [MFWS] and AAD [SSS] are not credited, as the systems are not F1 classified.

Reactor Coolant Pump shut down:

Before starting the RCP [RCS] cooldown to RIS/RRA [SIS/RHRS] connection conditions, the operator has to shut down two of the four Reactor Coolant Pumps, if all Reactor Coolant Pumps are still running. This is consistent with the ASG [EFWS] tanks design, which assumes no more than two of the four Reactor Coolant Pumps running during the cooling phase.

In the emergency procedures, the Reactor Coolant Pumps shut down will be undertaken based on indicated ASG [EFWS] tank water levels.

Operator action is considered at RT + 1 hour, if performed outside of the Main Control Room. 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

RCP [RCS] cooldown via unaffected SG:

The RCP [RCS] cooldown to RIS/RRA [SIS/RHRS] connection conditions is performed via the secondary side using the unaffected steam generators. The EPR cooldown rate is -50°C/h if two RBS [EBS] trains are available for boration duty or -25°C/h if only one RBS [EBS] train is available. This cooldown rate can be achieved unless it is limited by the VDA [MSRT] capacity. These cooling rates are consistent with the RBS [EBS] design which provides boration to match the reactivity insertion resulting from the RCP [RCS] cooldown.

The operator must initiate the RCP [RCS] cooldown within 2 hours of the RT, for the most onerous case where four Reactor Coolant Pumps were running until the end of the boration phase.

The RCP [RCS] cooldown is stopped when the RIS/RRA [SIS/RHRS] connection temperature is reached (180°C in hot leg) in the cooling loops. Should no Reactor Coolant Pumps be running, e.g. following LOOP, the unaffected SG depressurisation would be reduced to approximately 5 bar. This corresponds to a saturation temperature of 150°C. The RCP [RCS] cold leg temperature would also be 150°C, with the associated hot leg temperature below 180°C.

VIV [MSIV] bypass opening:

Should the VDA [MSRT] of one of the unaffected SG fail to open the operator would open the respective VIV [MSIV] bypass line (the VIV [MSIV] are already closed) and the VIV [MSIV] bypass line of another unaffected SG. This allows the under-pressurised SG to depressurise via a depressurised unaffected SG. This action accelerates the cooldown of the corresponding RCP [RCS] loop in the case of LOOP i.e. no Reactor Coolant Pumps running.

RCP [RCS] boration:

During the cooldown, the RCP [RCS] boration is performed by the RBS [EBS]. The RCV [CVCS] is not credited 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 take credit for all 4 ASG [EFWS] tanks water inventory. The ASG [EFWS] tank water supply is not being credited, as this part of the system is not F1 classified. The realignment of the ASG [EFWS] tanks previously not used to the suction of the operating ASG [EFWS] pumps is required as the sizing of the ASG [EFWS] tanks assumes the availability of all four tanks during this phase. The failure to deliver from these tanks initially is due to the assumed unavailability of the corresponding ASG [EFWS]-pump due to a single failure, preventative maintenance, or isolation of flow to the affected SG.

Depressurisation of RCP [RCS]:

Prior to or during the RCP [RCS] cooldown, the operator must isolate the accumulator injection lines, to prevent unwanted accumulator injection.

At the end of the cooldown phase, should the RCP [RCS] pressure be above the RIS/RRA [SIS/RHRS] connection pressure (30 bar), the operator must depressurise the primary side. This depressurisation can be performed by the pressuriser spray, although this system is not F1 classified. If the pressuriser spray is unavailable or not sufficient, e.g. in the case of FWLB with LOOP, depressurisation can be achieved by briefly opening 1 PSV. Once the RCP [RCS] pressure falls below 30 bar, the PSV is closed. During this depressurisation phase, the LHSI maintains a minimum RCP [RCS] pressure of approximately 20 bar. This ensures RCP [RCS] subcooling is not maintained, if only in the unaffected loops in case of LOOP.

RIS/RRA [SIS/RHRS] connection:

Once the RIS/RRA [SIS/RHRS] connection conditions are reached (RCP [RCS] pressure approximately 20 bar, hot leg temperature below 180°C, RPVL consistent with RIS/RRA [SIS/RHRS] suction from the hot leg), one LHSI train must be connected. Operation of one RIS/RRA [SIS/RHRS] train in RHR mode (suction from hot leg, injection into cold leg) is sufficient to remove the decay heat.

c) Precautions limiting the event occurrence

- A non-return valve prevents the SG from blowdown via the break if the break is located upstream of the valve. In this case, the break leads to a 'loss of main feedwater' accident and not to a 'Feedwater Line Break' accident.
- Main feedwater lines from and including their isolation valves outside containment and up to the SG are classified ETC-M class 2.

2.16.1.2. Safety criteria

The safety criteria are the radiological limits for PCC-4 events (see sub-section 1.1 of this appendix). The consequences of a Feedwater Line Break are analysed with respect to the following decoupling criteria:

- fuel cladding integrity,
- reactor coolant pressure boundary (RCPB) integrity,
- reactor containment overpressure and over temperature limits,
- amount of radioactive products released.

For the RCPB criteria, the FWLB analysis is covered by the primary overpressure analyses.

For the criteria associated with the reactor containment, the FWLB analysis is covered by the 'steam line break' analysis.

For the criteria associated with the amount of radioactive products released, the FWLB analysis is covered by the 'SGTR 2 tubes' analysis presented in sub-section 2.18 of this appendix.

As a consequence, the FWLB is analysed for the fuel cladding integrity criteria only in this Appendix it must be shown that the following two safe states are reached, with the application of the safety analysis rules defined in section 1 of this appendix:

- the controlled state, relying only on F1A means,
- the safe shutdown state, relying only on F1A and F1B means.

2.16.1.3. Methods and assumptions

2.16.1.3.1. Method of analysis

The FWLB accident analysis is performed with the CATHARE code in the frame of a **realistic deterministic methodology.**

The realistic deterministic methodology is characterised by the following 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 in:

- the phenomenological analysis of the accident scenario, and the identification of the key phenomena;
- the judgement of code adequacy for calculating the accident scenario, based on physical understanding of the conditions, 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 required), or a check of the bounding conservative approach to the calculation of key phenomena by the code. This relies on the assessment matrix of the code;
- the introduction, where necessary, of conservative biases as close as possible to the uncertainty on 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 blowdown and depressurisation,
- the asymmetric RCP [RCS] heating, and the resulting RCP [RCS] overpressure (with Reactor Coolant Pumps on or off),
- the asymmetric RCP [RCS] heat removal via the unaffected SG with potentially low water inventories (with Reactor Coolant Pumps on or off),
- the asymmetric RCP [RCS] cooling, and the RCP [RCS] depressurisation down to RIS/RRA [SIS/RHRS] 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 mock-up tests;

- the overall verification of the code by simulation of integral effect tests (IET), covering a wide range of representative PWR transients on small-scale facilities, e.g.:

  - test LOBI BT-06 'partial FWLB (45%)': CATHARE gave a good prediction of the faulted SG, primary to secondary heat transfer, RCP [RCS] pressure and temperature trends,

  - tests BETHSY 5.1a 'steady states with reduced SG mass inventories in forced and natural circulations': CATHARE trends were in good agreement with test results as RCP [RCS] temperatures, distribution of heat transfer between ascending and descending SG U-tubes parts, secondary side swell level, global primary to secondary heat transfer, provided a fine mesh was used in the SG riser and downcomer volumes of CATHARE. These are the conditions that arise in this type of transient analysis).

  - tests BETHSY 5.2c and e '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.

The transient analysis relies on the application of the conservative PCC analysis rules defined in section 1 of this appendix. Part of these rules is the use of conservative assumptions for all relevant boundary conditions, with respect to the decoupling criteria under consideration. These conservatisms address at least:

- the characteristics of the initiating event (maximising 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 capacity).

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

The CATHARE code is used, with explicit modelling of the four RCP [RCS] loops.

2.16.1.3.2. Main assumptions

a) Accident definition

The cases studied correspond to the double ended guillotine rupture of a main feedwater pipe at the inlet to the steam generator (PCC-4 event). This leads to the most rapid SG blowdown.

For accident analyses, a loss of offsite power (LOOP) is assumed in the accident, if this assumption is conservative.

b) Protection and mitigation actions

In the case of a FWLB event, automatic actions which are F1A classified both trip the reactor and ensure the decay heat removal to limit the RCP [RCS] heat-up.

In accordance with the rules defined for safety analyses in section 1 of this appendix, the controlled state is reached using only F1-A classified actions. The safe shutdown is reached using only F1-A and F1-B classified actions.

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

The potential reactor trip signals are:

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

To cover the complete break size spectrum, a set of conservative assumptions was used in the analysis. The largest break size of a main feedwater pipe is analysed with the following bounding hypotheses:

- no signal coming from the primary side is claimed,
- the signals "SG pressure drop > MAX1" or "SG pressure < MIN1", which might not occur or might occur later than the signal " SG level < MIN1 " for smaller break sizes, are not credited. As a consequence, the RT and VIV [MSIV] isolation occur on signal "SG level < MIN1" in the transient calculation. This assumption ensures that the SG water content claimed at RT is minimum, over the entire break spectrum [(1)].

For additional conservatism, no signal coming from the affected SG is claimed.

Consequently, only the "SG level < MIN1" signals in the unaffected SG are considered for the RT actuation. This is a conservative assumption for SG dryout.

The other automatic actions considered in the analysis are as follows:

- turbine trip (F1A)

  following the RT signal, the turbine trip is actuated,

- VIV [MSIV] isolation (F1A)

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(1)
- for small FWLB, the RT/TT occurs on “SG level low”
- for large FWLB, the RT/TT occurs on “SG Pressure drop high” or “SG pressure low”. At that time, the SG water content is higher than in case of small FWLB, which is beneficial.
- the BDR analysis considers a large FWLB with RT/TT on “SG level low”. It combines the rapid dynamics of the large FWLB, with the low SG water content of the small FWLB. As a result, it bounds the consequences of any break size within the FWLB spectrum.

following a "SG pressure < MIN1" or "SG pressure drop > MAX1", signal, VIV [MSIV] closure is initiated. As discussed above for these signals for RT actuation, they are not considered for VIV [MSIV] closure actuation.

- ASG [EFWS] actuation (F1A)

  following a "SG level < MIN2" signal the ASG [EFWS] is actuated to the corresponding SG (SG specific signal). The time delay between the setpoint being reached and the effective ASG [EFWS] flow injection is defined consistent with an assumption of LOOP, or no LOOP, to account for the EDG reloading sequence.

- VDA [MSRT] actuation (F1A)

  when the SG pressure reaches the VDA [MSRT] setpoint "SG pressure > MAX1", the VDA [MSRT] opens and provides the steam removal path required for heat removal.

- MSSV actuation (F1A, passive opening)

  should the SG pressure increase up to the MSSV setpoint, e.g. in case of VDA [MSRT] failing to open, the MSSV would open and perform the steam removal.

- PSV actuation (F1A, passive opening)

  should the RCP [RCS] pressure increase up to the PSV setpoints, the PSV would open to limit the RCP [RCS] overpressure, with any discharged primary coolant delivered into the IRWST.

c) Operator actions

No operator action is claimed before 30 minutes after RT. When operator action local to plant is necessary, this delay is increased to 1 hour following RT.

As required by the emergency procedures, the operator must perform the following actions, already described in sub-section 2.16.1.1.2b of this appendix).

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

<u>Affected SG isolation (F1B)</u>

- manual ASG [EFWS] isolation (ASG [EFWS] flow rate, SG pressure)
- manual VIV [MSIV] isolation, if not yet done (SG pressure)
- manual ARE [MFWS]/AAD [SSS] isolation, if not yet done (SG pressure)

<u>ASG [EFWS] passive header (pump discharge) opening</u>

- manual realignment of the ASG [EFWS] pump discharge to the unaffected SG, via the dedicated passive header (SG level, SG pressure, ASG [EFWS] flow rate).

<u>VIV [MSIV]-bypass valve opening (F1B)</u>

- manual opening of the VIV [MSIV]-bypass valve (SG pressure)

Reactor Coolant Pumps shutdown (F1B)

- manual shut down of Reactor Coolant Pumps (Reactor Coolant Pumps on/off status)

RCP [RCS] cooldown via 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] initiation (at the latest at RCP [RCS] cooldown actuation), with manual alignment to an unaffected loop
- manual RBS [EBS] shutdown (RBS [EBS] tank level)

ASG [EFWS] passive header (pump suction) opening (F1B)

- manual realignment of the ASG [EFWS]-tank to the ASG [EFWS] pump suction via the dedicated passive header (ASG [EFWS] tank level)

RCP [RCS] depressurisation (F1B)

- manual isolation of the 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)
- manual opening/closing of the PSV (RCP [RCS] pressure)

RIS/RRA [SIS/RHRS] connection (F1B)

- manual connection of the LHSI (in RHR-mode) to the RCP [RCS] (RCP [RCS] pressure, RCP [RCS] temperature, RPVL, ΔT_{sat})

2.16.1.4. Definition of studied cases

2.16.1.4.1. Studied cases from initiator up to controlled state

As described in sub-section 2.16.1.2 of this appendix, the decoupling criterion to be met is the fuel cladding integrity.

The objective of the FWLB analysis is to show that the secondary side as a whole remains available for primary side heat removal throughout the transient. This is despite of the loss of one SG due to the initiating event. This objective is significantly more restrictive than the fuel cladding integrity decoupling criteria.

This objective necessitates that at least one SG remains available for primary side heat removal during the post-accident phase. A temporary insufficiency in 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 primary coolant content remains sufficient to provide core cooling. The controlled state is reached when this temporary insufficiency ends.

The limiting time period is from the initiating event to the controlled state, with only F1A means assumed to be available. For the objective defined above, the FWLB transients considered are those which maximise the RCP [RCS] heat-up. Thus the following assumptions are made:

- the highest initial power level (i.e. 102% FP),
- the assumption of LOOP at RT, when this is conservative.

Two cases are analysed, in order to assess the most onerous single failure for the secondary side heat removal capability:

- Case 1 (102% FP, without LOOP):
  - Single failure applied on one ASG [EFWS] pump feeding one unaffected SG.

    The assumption 'without LOOP' is more onerous, assuming preventative maintenance combined with a single failure. In these conditions with only one ASG [EFWS]-pump remaining available, the secondary side does not remove all the primary heat. The primary heat not removed by the secondary side heats-up the primary coolant. This causes the PSV to open 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 onerous case occurs with the Reactor Coolant Pumps running (no LOOP).
- Case 2 (102% FP, with LOOP):
  - Single failure applied on 1 VDA [MSRT] related to one unaffected SG.

    The assumption 'with LOOP' is more limiting in this case, even when the combination of preventative maintenance with the single failure is included. In fact, during this transient two ASG [EFWS] pumps remain available for heat removal which is sufficient to remove the entire primary heat load. The RCP [RCS] heat-up does not, therefore, result from any temporary insufficiency in heat removal, even when the Reactor Coolant Pumps power input is included. The loss of Reactor Coolant Pumps is more onerous in this case because of the RCP [RCS] overpressure induced by the higher ΔT over the core.

The other specific assumptions related to these cases are described in sub-section 2.16.1.5.3 of this appendix.

2.16.1.4.2. Studied cases from controlled state up to safe shutdown

As described in Sub-section 2.16.1.2 of this appendix, demonstration that the safe shutdown is reached relying only on F1A and F1B means must be provided. The objective of the FWLB analysis is to demonstrate that: The secondary side heat removal remains available until the RIS/RRA [SIS/RHRS] is connected, relying only on F1-means. This necessitates an adequate secondary side water supply (ASG [EFWS] tanks and pumps), consistent with the other means needed for transferring the plant from controlled state up to a safe shutdown state (RCP [RCS] boration, RCP [RCS] cooling and RCP [RCS] depressurisation).

During that 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 a unsupplied unaffected SG if necessary, [1]
- boration of RCP [RCS],
- cooldown by unaffected SG,
- final depressurisation of RCP [RCS].

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

Case 3 (102% FP, without LOOP):

- This case is performed to demonstrate the capability of the F1 systems, especially the ASG [EFWS] and VDA [MSRT], to cool the plant to the LHSI/RHR connection conditions, with the maximum heat to be removed.
- The cooldown has to be completed with the available ASG [EFWS] tanks capacity.
- It is assumed that four Reactor Coolant Pumps run until the beginning of the cooldown. At this point, two Reactor Coolant Pumps are switched off.

Case 4 (102% FP, with LOOP):

- This case is performed to demonstrate the capability of the F1 systems to depressurise the RCP [RCS] to the RIS/RRA [SIS/RHRS] connection conditions.
- Natural circulation which results from the LOOP is conservative for the RCP [RCS] depressurisation due to the presence of hot uncooled loops (e.g., the loop connected to the affected SG).

[1] That realignment, if necessary, is needed to reach the controlled state. The relevant actions of ASG [EFWS] isolation to the affected SG, and ASG [EFWS]-pump alignment to one of the unfed unaffected SG, are part of the first phase "from initiating event to the controlled state".

2.16.1.5. Description of cases 1 and 2 (from initiating event to the controlled state)

2.16.1.5.1. Choice of single failure and preventative maintenance

As required by the general safety rules defined in section 1 of this appendix, single failure and preventative maintenance are applied to the F1 systems in the most conservative way for the criteria being assessed.

When considering the RCP [RCS] heat-up, the most conservative assumption is associated with lowering the secondary side heat removal capability. As a consequence, preventative maintenance assumed for the ASG [EFWS] pump connected to one unaffected SG (preventative maintenance has no impact on the VDA [MSRT] availability).

As indicated previously, 2 different cases are considered for the single failure:

Case 1 (102% FP, without LOOP): In addition to the preventative maintenance of one ASG [EFWS] pump connected to one unaffected SG, the single failure is applied to one ASG [EFWS] pump connected to another unaffected SG.

Similarly for the maintenance choice, it is postulated that the failure of one ASG [EFWS] pump associated with another unaffected SG reduces the secondary side heat removal capability, which is conservative for the RCP [RCS] heat-up.

Case 2 (102% FP, with LOOP): In addition to the preventative maintenance of one ASG [EFWS]-pump associated to one unaffected SG, the single failure of one VDA [MSRT] on another unaffected SG supplied with ASG [EFWS] water.

In this case, after VIV [MSIV] closure the SG pressure associated with the failed VDA [MSRT] increases to the MSSV setpoint. As the pressure in the other unaffected SG is at the VDA [MSRT] setpoint (lower than MSSV setpoint), the steam released from the SG affected by the VDA [MSRT] single failure decreases. Consequently, the decay heat removal is mainly provided by the two remaining unaffected SG (instead of 3 SG). This is another way to reduce the heat removal capability of the secondary side and is conservative for the RCP [RCS] heatup.

Appendix 14B.2.16.1 - Figure 1 shows the SG configurations for VDA [MSRT] relief and ASG [EFWS] feeding assumed in cases 1 and 2.

From initiator up to 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] heat-up	

The purpose of cases 1 and 2 is to identify the most onerous single failure of one ASG [EFWS]-pump or one VDA [MSRT]. This is assessed for RCP [RCS] heat-up, and the resulting PSV coolant discharge.

2.16.1.5.2. Initial state

The initial state conditions, given in Appendix 14B.2.16.1 - Table 1 are chosen to maximise the power to be removed and therefore the RCP [RCS] heat-up.

2.16.1.5.3. Specific assumptions

a) Neutronic data and decay heat

Core power is assumed constant at 102% of full power prior to reactor trip. Following RT, the maximum residual heat curve, as described in sub-section 0.2.4 of this appendix, is considered for terms B + C.

Term A (fission heat) is input to the CATHARE code, with the CATHARE point-kinetic model not being utilised. This term A results from a separate conservative RT-simulation, see Appendix 14B.2.2 – Table 1 of this appendix.

b) Assumptions related to control systems (not F1)

Turbine: turbine control is assumed and modelled as follows:

The flow to the turbine is modelled as a constant flow rate until the steam line pressure reaches 90 percent of the initial pressure. Below this pressure the steam flow rate decreases linearly with secondary system pressure until the turbine is tripped. This assumption maximises the turbine flow rate, and hence the SG water depletion, before TT. This assumes a potential 10% over capacity of the turbine inlet valves.

GCT [MSB]: not considered.

ARE [MFWS]/AAD [SSS]: the ARE [MFWS]/AAD [SSS] flow is assumed lost into the break via the ARE [MFWS] header. Thus, neither the affected SG nor the unaffected SG are supplied by ARE [MFWS]/AAD [SSS] from the time the FWLB occurs. Consequently, SG level control is lost as soon as the break occurs.

Heaters/spray: the pressuriser pressure control is not considered because it is not actuated in the case of the heaters or it is unavailable in the case of the spray following LOOP).

c) Assumptions related to F1 systems

Reactor trip (F1A): this occurs following a “SG level < MIN1” signal. The nominal setpoint for min 1 is 13.4 m above the tube sheet. A setpoint reduction corresponding to -5% of the narrow range span is assumed relative to the nominal value. This minimises the SG liquid inventory at RT. Only SG level signals coming from unaffected SG are assumed to lead to RT actuation. The initial level in the unaffected SG is maximised (nominal level + 5% of narrow range) in order to delay the signal.

Turbine trip (F1A): the turbine is tripped following the RT signal.

VIV [MSIV] (F1A): the VIV [MSIV] of all SG are automatically closed.
It is conservatively assumed that the VIV [MSIV] begins to close on the RT signal (SG level < MIN1), with a maximum closure delay of 5 seconds.

VDA [MSRT] (F1A): following RT, TT and VIV [MSIV] closure, the unaffected SG pressures increase up to the VDA [MSRT] setpoint. In the analysis, the minimum value is assumed for the VDA [MSRT] setpoint (93 – 1.5 bar) to increase the rate of emptying the SG.

MSSV (F1A): in the case of a single failure assumed in one VDA [MSRT], the pressure in the corresponding SG increases up to the MSSV actuation pressure. The MSSV setpoint value is maximised (102.5 + 1.5 bar).

PSV (F1A): as a consequence of the RCP [RCS] heatup, the RCP [RCS] pressure increases up to the PSV actuation pressure. The minimum pressure setpoints are assumed for PSV actuation in order to maximise the PSV coolant discharge (PSV1: 174 - 1.5 bar; PSV2: 178 - 1.5 bar; PSV3: 178 - 1.5 bar).Similarly, a maximum hysteresis of 10 bar for PSV closure is assumed.

ASG [EFWS] (F1A): ASG [EFWS] is actuated following a “SG level < MIN2” signal. The nominal setpoint for min 2 is at 8 m above the tube sheet. A setpoint reduction corresponding to - 5% of the wide range span is assumed relative to the nominal value. This minimises the SG liquid inventory at ASG [EFWS] actuation. In the case of LOOP, a delay of 50 s is assumed between the LOOP occurring and the ASG [EFWS] flow delivery to the SG. This accounts for the EDG reloading sequence.

The ASG [EFWS] flow rate is the minimum value of 93.5 te/h per pump, assumed to be constant within the pressure range 1 to 91.5 bar and decreasing to 30 te/h at an SG pressure of 104 bar.

Isolation of the ASG [EFWS] line assigned to the affected SG (F1A):

The isolation of the ASG [EFWS]-line assigned to the affected SG is performed by manual closing of the respective isolation and control valves. This operator action is assumed to be delayed for one hour after reactor trip as it is a local to plant action.,

Opening of EFW header at pump discharge (F1A):

the realignment of the ASG [EFWS] pump o the affected SG to an unaffected and non-supplied SG is performed by manual opening of the EFW header (including the opening of at least the two isolation valves connecting the two SG together). This operator action is assumed to be delayed for one hour after reactor trip as it is a local to plant action.,

2.16.1.5.4. Results

The sequence of events for case 1 (FWLB, from the initiating event to the controlled state, without LOOP, with a single failure of one ASG [EFWS], maintenance of 1 ASG [EFWS]) is given in Appendix 14B.2.16.1 - Table 2.

The sequence of events for case 2 (FWLB from the initiating event to the controlled state, with LOOP - SF on 1 VDA [MSRT], maintenance on 1 ASG [EFWS]) is given in Appendix 14B.2.16.1 - Table 3.

The figures showing the main thermal-hydraulic trends are provided:

- 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.

Appendix 14B.2.16.1 - Figures 3 to 8 are related to case 1.

Appendix 14B.2.16.1 - Figures 9 to 14 are related to case 2.

The time duration of the transient analysed for both cases is 1 hour following the RT.

The most onerous transient for the primary water loss via the PSV is case 1, without LOOP and with only one ASG [EFWS] pump available. The RCP [RCS] discharge via the PSV is 35 tons, compared to the initial RCP [RCS] water inventory of 305 tons. One PSV cycles open and closed intermittently throughout the transient. The RCP [RCS] loops remain full of liquid, thus continued core cooling is not challenged, with a large margin (see Appendix 14B.2.16.1 – Figure 9, RPV-upper plenum full).

In case 2, the RCP [RCS] discharge via the PSV is less than 3 tons.

In case 1 where one ASG [EFWS] pump only is available, all the SG dry out during the transient, including the fed SG. This occurs as the flow from one ASG [EFWS] pump is not sufficient to remove the whole primary heat. To reach the controlled state, the ASG [EFWS] realignment is needed to provide delivery from two ASG [EFWS] pumps for heat removal (see below).

In case 2 where two ASG [EFWS] pumps are available, the heat removal capacity is more than sufficient to remove the whole primary heat load. The controlled state is reached within 1 hour of the RT without any need for realignment of the ASG [EFWS] (see RCP [RCS] temperatures Appendix 14B.2.16.1 - Figure 10, and SG levels Appendix 14B.2.16.1 - Figure 12).

The ASG [EFWS] realignment is not modelled in the CATHARE calculation case 1. The following simple energy balance ensures that in the most onerous case 1, the controlled state is reached 1 hour after reactor trip. For this to be achieved, the secondary system must remove the primary heat load after realignment of the ASG [EFWS] pump connected to the affected SG to a non-supplied unaffected SG.

The decay heat curve provides the residual core power 1 hour after RT:

- maximum decay heat at 1 hour: 1.5%,
- maximum core power before shutdown: 102% x 4900 MWth,
- maximum core power: 75 MWth.

Four Reactor Coolant Pumps power transferred to RCP [RCS] coolant: 30 MWth:

- total primary heat: 105 MWth.

The heat removal capacity for a working SG is:

- ASG [EFWS]-pump flow rate: 26 kg/s or 93.5 t/h,
- latent heat plus specific heat from 50°C: 2525 KJ/kg,
- ASG [EFWS] pump heat removal (minimum): 65 MWth.

After the realignment the total heat removal capacity is 130 MWth.

Therefore the total heat removal is higher than the primary heat load at 1 hour after RT. The PSV no longer needs to open to partly remove the primary heat. The controlled state is therefore reached.

This statement is confirmed by the following long term cases 3 and 4 where the realignment is also assumed (see section 2.16.1.6 of this appendix).

The results of the analysis of cases 1 and 2 demonstrate that:

- With two ASG [EFWS] pumps available delivering to two unaffected SG (single failure or preventative maintenance), the controlled state is reached without need for operator action.
- With 1 ASG [EFWS] pump only available into 1 unaffected SG (single failure and preventative maintenance), the controlled state is reached following manual opening of the ASG [EFWS] passive header in order to take credit of a second ASG [EFWS] pump (pump assigned to the affected SG).
- In all cases, the controlled state is reached relying only on F1-A means, despite the most onerous single failure and of the most onerous preventative maintenance:
  - 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.

2.16.1.6. Description of cases 3 and 4 (from the controlled state to the safe shutdown state)

2.16.1.6.1. Case 3: without LOOP

a) Choice of single failure and preventative maintenance

Single failure and preventative maintenance are chosen to pessimise the assumed capacity of the F1 systems used to cooldown the RCP [RCS] to the RIS/RRA [SIS/RHRS] (LHSI in RHR-mode) connection conditions.

Preventative maintenance has no impact on the VDA [MSRT] availability, as preventative maintenance is not performed on the VDA [MSRT] mechanical part, and preventative maintenance on the VDA [MSRT] electrical part (e.g. on EDG in case of LOOP) does not prevent the VDA [MSRT] availability:

- during the 2 first hours, the VDA [MSRT] can be operated with power supply from batteries,
- after 2 hours, the VDA [MSRT] can be operated manually

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

The most onerous single failure for the RCP [RCS] cooldown is the failure of one VDA [MSRT]. Case 3 is performed without LOOP, as Reactor Coolant Pumps running increases the primary heat load, which is conservative for the ASG [EFWS] water consumption.

An additional failure is included, to limit the number of cases analysed: failure of 1 out of 2 RBS [EBS] trains is postulated with considering to the RCP [RCS] boration. The resulting RCP [RCS] cooldown rate is -25°C/h, the minimum one, as this is conservative for assessing the ASG [EFWS] water consumption.

Preventative maintenance has no impact on RBS [EBS], as preventative maintenance is not performed on the RBS [EBS] mechanical part, and the power supply is effective from the neighbour electrical division (via a dedicated cross-connection) in the case of preventative maintenance on the electrical division.

All these plant unavailabilities are assumed to continue throughout the transient.

Appendix 14B.2.16.1 - Figure 2 depicts the SG configurations for VDA [MSRT] relief and ASG [EFWS] feeding, in both phases (before and after ASG [EFWS] realignment), and indicates the additional single failure in the RBS [EBS].

Case 3 (102% FP, w/o LOOP)
min VDA [MSRT], min RBS [EBS]
From initiator up to controlled state: 1 broken SG 1 SG with relief and feeding 1 SG with relief 1 SG with feeding
From controlled state up to safe shutdown: 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] requirements (RCP [RCS] cooling concern)

Case 3 then considers the minimum relief capacity for RCP [RCS] cooldown and the minimum efficiency of RCP [RCS] boration. This maximises the ASG [EFWS] water consumption prior to RIS/RRA [SIS/RHRS] connection.

b) Initial state

The assumptions for the calculation of the first phase, from initiating event to the controlled state of case 3, are identical to the assumptions described for case 2 in sub-section 2.16.1.5.2 of this appendix (102% FP initial state). The only differences are for the case ‘without LOOP’ and are associated with increasing the ASG [EFWS] water consumption:

- no LOOP at RT,
- time delay between "SG level < MIN2" signal and ASG [EFWS] flow injection of 16.5 seconds (no EDG reloading sequence).

c) Specific assumptions

The assumptions specific to the transfer from the controlled state up to safe shutdown state are the following. The classification indicated refers to the mechanical system. The classification of the relevant I&C is F1B when specific to the post controlled state phase (see sub-section 2.16.1.3.2c of this appendix):

RBS [EBS] (F1B): 1/2 RBS [EBS] trains is manually actuated at the beginning of the RCP [RCS] cooldown phase, 2 hours after RT, and closed 6000 seconds later. The boration duration is 6000 seconds. The minimum RBS [EBS] capacity of 2.8 kg/s is considered, in a conservative manner (see sub-section 0.2 of this Appendix). The boron content injected is 17 tonne, corresponding to the required boration needed to achieve the safe shutdown state (core subcriticality at RIS/RRA [SIS/RHRS] connecting conditions).

A minimum enriched boron concentration of 7000 ppm has been assumed. This corresponds to a natural boron concentration of 11750 ppm, required for a MOX fuel management scheme.

VDA [MSRT] (F1A): as required by the emergency procedures, the operator uses the available VDA [MSRT] on the unaffected SG to perform the RCP [RCS] cooldown via the secondary side. This is attained by decreasing the VDA [MSRT] setpoint at a rate consistent with the required RCP [RCS] cooldown rate.

Reactor Coolant Pump trip (F1B): as required by the emergency procedures, the operator trips two of the four Reactor Coolant Pumps, before performing the RCP [RCS] cooldown using the unaffected SG.

The RCP [RCS] cooldown is achieved by decreasing the VDA [MSRT] setpoint for a cooldown rate of -25°C/h from 91.5 bar (93 – 1.5, T_{sat}: 304.5°C) to 5 bar (T_{sat}: 150°C) over 6 hours.

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

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

This bypass is not considered in case 3 without LOOP: with the Reactor Coolant Pumps running, the cooling of the RCP [RCS] loop associated with the SG affected by the VDA [MSRT] single failure is efficient enough (without MSIV bypass opening) for achievement of the RCP [RCS] boration and depressurisation functions.

ASG [EFWS] (F1A):

When the reference level is reached, the SG level (measured using the wide range) is controlled (not F1-classified) by ASG [EFWS] according to the following simple principle:

- ASG [EFWS] flow = 0 t/h if SG level > nominal level + 0.2 m,
- ASG [EFWS] flow = 93.5 t/h if SG level < nominal level – 0.2 m

This is included in the CATHARE modelling

Crediting the SG level control is conservative for the assessment of the ASG [EFWS] water consumption to reach the safe shutdown.

Isolation of the ASG [EFWS] line assigned to the affected SG (F1A):

The isolation of the ASG [EFWS]-line assigned to the affected SG is performed by the operator manually closing the relevant isolation and control valves.

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 by the operator manually opening the ASG [EFWS] header at the pump discharge (including 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 of the four ARE [MFWS] tanks water content is not explicitly simulated in the calculation. Only the ASG [EFWS] injection flow rates are modelled, although opening of the header it is implicitly considered.

RIS/RRA [SIS/RHRS] (F1B): RIS/RRA [SIS/RHRS] 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 RIS/RRA [SIS/RHRS] suction from the hot leg.

In the CATHARE calculations, when RIS/RRA [SIS/RHRS] connection is assumed, the hot legs related to the cooled loops are full of liquid in a subcooled condition.

2.16.1.6.2. Case 4: with LOOP

a) Choice of single failure and preventative maintenance

Single failure and preventative maintenance are chosen to pessimise the capacity of the F1-systems to provide RCP [RCS] boration and depressurisation during the long term phase of a FWLB transient during which a LOOP is assumed.

Preventative maintenance has no impact on the VDA [MSRT] and RBS [EBS] availability, as already discussed in sub-section 2.16.1.6.1a of this appendix) (case 3).

Consequently, assuming the most onerous preventative maintenance, maintenance of one safety division or one diesel, results in the unavailability of one ASG [EFWS] train.

The most onerous single failure for RCP [RCS] depressurisation is the failure of one diesel, causing the failure of one ASG [EFWS]-pump. Case 4 is performed with LOOP, as RCP [RCS] depressurisation is pessimised by the existence of hot uncooled 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, maximising the primary water mass lost via the PSV before reaching the RIS/RRA [SIS/RHRS] connection conditions.

As in case 3, an additional failure is assumed, only to limit the number of cases analysed: for RCP [RCS] boration, failure of 1/2 RBS [EBS] trains are assumed. The resulting RCP [RCS] cooldown rate is -25°C/h.

It is necessary to open a RCP [RCS] relief path for RCP [RCS] depressurisation for RIS/RRA [SIS/RHRS] connection at the end of the transfer to the safe shutdown. This was not required for case 3 due to the uncooled RCP [RCS] loop during RCP [RCS] natural circulation. This is performed using only 1/3 PSV, as this has sufficient capacity.

Appendix 14B.2.16.1 - Figure 2 depicts the resulting SG configurations for VDA [MSRT] relief and ASG [EFWS] feeding, in both phases (before and after ASG [EFWS] realignment), with indication of the additional single failure assumed on the RBS [EBS].

Case 4 (102%, with LOOP)
min VDA [MSRT], min RBS [EBS]
From the initiating event to the controlled state:
1 broken SG
1 SG with relief and feeding
2 SG with relief
From the controlled state 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)

Case 4 then represents the most onerous case for RCP [RCS] depressurisation: RCP [RCS] in natural circulation with two uncooled loops.

Note: Case 4 has been defined with only two SG are available for RCP [RCS] cooldown to safe shutdown. It shows that in the most onerous PCC for heat removal (FWLB), two SG only are needed to achieve the safe shutdown state (two ASG [EFWS] and two VDA [MSRT]).
However, the operator would have the possibility to use the third unaffected SG, via a realignment of one ASG [EFWS]-pump to two SG (for feeding) or / and use of the VIV [MSIV]-bypass in place of a failed VDA [MSRT] (for relief). It would result in only one uncooled loop as opposed of two, with less onerous RCP [RCS] conditions for the depressurisation. The consequences in terms of PSV water discharge are given later.

b) Initial state

The assumptions considered for the calculation of the first phase (from the initiating event to the controlled state) of case 4 are identical to the assumptions made for case 2 (102% FP initial state).

c) Specific assumptions

The assumptions specific to the transfer from the controlled state to the safe shutdown state are as follows:

RBS [EBS] (F1B): as discussed in sub-section 2.16.1.6.1.c of this appendix. Each RBS [EBS] train injects into two cold legs, with one isolation valve in each injection line. The operator will close the RBS [EBS] line(s) into uncooled loop(s) In case of FWLB with Reactor Coolant Pumps off, This fully aligns the RBS [EBS] to cooled loop(s) which experience natural circulation flow.

VDA [MSRT] (F1A): as discussed in sub-section 2.16.1.6.1.c of this appendix

PSV (F1A system, F1B operator action): at the end of the RCP [RCS] cooldown phase using the 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 in operation. When the RIS/RRA [SIS/RHRS] connection conditions are reached (P < 30 bar, T < 180°C in the hot leg), the operator closes the PSV by increasing its setpoint [1].

The minimum capacity of the PSV is considered (see sub-section 0.2 of this appendix).

[1] Other operating modes are possible, such as a push-button causing a full opening over a given period of time (e.g. 30 s or 1 min). The operator would actuate the push-button sequentially, as necessary to reach the required RCP [RCS] conditions.

ASG [EFWS] isolation to the affected SG (F1A): this is the same assumption as made in sub-section 2.16.1.6.1.c of this appendix.

ASG [EFWS] (F1A): this is the same assumptions as made in sub-section 2.16.1.6.1.c of this appendix

Opening of the ASG [EFWS] header at pump discharge (F1A) and at pump suction (F1B): these are the same assumptions as made in sub-section 2.16.1.6.1.c of this appendix.

RIS/RRA [SIS/RHRS] (F1B): these are the same assumptions as made in sub-section 2.16.1.6.1.c of this Appendix.

2.16.1.6.3. Results

The sequence of events for case 3 (FWLB - transfer to safe shutdown - without LOOP) is given in Appendix 14B.2.16.1. - Table 4.

The sequence of events for case 4 (FWLB - transfer to safe shutdown - with LOOP) is given in Appendix 14B.2.16.1 - Table 5.

The figures showing the trends for the main thermal-hydraulic parameters are provided:

- primary total mass / water liquid masses in SG,
- hot leg liquid temperature / hot leg liquid flow,
- ASG [EFWS] flow / integrated ASG [EFWS] flow,
- PSV flow / RBS [EBS] flow,
- pressuriser pressure / SG pressures.

Appendix 14B.2.16.1 - Figures 15 to 18 are related to case 3.

Appendix 14B.2.16.1 - Figures 19 to 22 are related to case 4.

- For case 3, a FWLB, covering transfer to the safe shutdown state without LOOP, the RCP [RCS] pressure and temperature have reached RIS/RRA [SIS/RHRS] connection conditions at the end of the long term RCP [RCS] cooldown phase using the unaffected SG. No additional RCP [RCS] depressurisation (i.e. via opening of the PSV) is needed.

  All RCP [RCS] loops, including the affected loop, are at a temperature lower than 180°C as the Reactor Coolant Pumps are operating. The RCP [RCS] pressure, as a result of RCP [RCS] cooldown, is lower than 30 bar. The boron concentration in the core at the end of the transient is approximately 620 ppm, the value required for the MOX fuel management at end of life.

  The ASG [EFWS] water consumption at the end of the cooldown is 1430 tons (including 100 tons lost via the break prior to isolation of the affected SG by the operator 1 hour after reactor trip). This value is below the ASG [EFWS] tanks capacity of 1500 tonne. In addition, 200 tonne of water remain in the two SG available for heat removal when RIS/RRA [SIS/RHRS] connection conditions are reached.

- For case 4, a FWLB covering transfer to the safe shutdown state with LOOP, the RCP [RCS] cooldown has been performed by lowering the available VDA [MSRT] setpoint pressure to 4 bar. The primary pressure is higher than the RIS/RRA [SIS/RHRS] connection conditions (P ~ 110 bar > 30 bar) when this cooldown is completed.

  This high pressure is caused by the pressurisation effect of the hot legs associated with the affected SG and unfed intact SG. In these hot legs saturation conditions have been reached. The operator must briefly open one PSV to depressurise the RCP [RCS] and cooldown the RCP [RCS] loops.

  Once the RIS/RRA [SIS/RHRS] connection conditions are reached in the heat removing loops the PSV is closed. The primary water discharged by the PSV is approximately 50 tons, compared to an actual RCP [RCS] water inventory of approximately 320 tons. At the end of the transient the boron concentration has stabilised around 650 ppm, a little higher than the value required for the MOX fuel management at end of life of 620 ppm.

Note: Cooling three loops instead of two, whilst not needed to achieve the safe shutdown is of benefit to minimise the PSV water discharge for RCP [RCS] depressurisation. To cover the single failure of one VDA [MSRT], the MSIV-bypass could be used in that case.

A complementary CATHARE calculation has been performed in which the PSV opening is preceded by changing the ASG [EFWS] alignment; the SG affected by the preventative maintenance is no longer fed (SG4 Appendix 14B.2.16.1 - Figure 2) and the ASG [EFWS] pump of the broken SG is aligned to an empty intact SG (SG1). By this action the associated saturated loop is cooled down (loop 1), which has a perceptible effect on the primary pressure. In this case the loss of primary water is considerably decreased to less than 5 tons.

Appendix 14B.2.16.1 - Figure 25 is related to this calculation.

- The results of the analysis of both cases 3 and 4, demonstrate that use of the F1-systems is sufficient to reach the safe shutdown state RCP [RCS] boron concentration and to cool and depressurise the RCP [RCS] to the RIS/RRA [SIS/RHRS] connection conditions. This can be achieved within a time period being consistent with the ASG [EFWS]-tanks capacity. This is despite the most onerous single failure and of the most onerous preventative maintenance.

The F1-systems involved in the achievement of the safe shutdown 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.

Note: the VIV [MSIV] bypass is not required to reach the safe shutdown. However it is of benefit in lowering the PSV water discharge for RCP [RCS] depressurisation, since it limits the number of uncooled loops to the one containing the broken SG in the most onerous case (single failure on 1 VDA [MSRT] with LOOP).

2.16.1.7. Conclusion

The analysis presented of the FWLB accident shows that even with the most onerous single failure and preventative maintenance:

- the controlled state is reached, relying only on F1-A means:
  - 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, relying only on F1-A and F1-B means:
  - ASG [EFWS] and VDA [MSRT] for RCP [RCS] cooling,
  - RBS [EBS] for RCP [RCS] boration,
  - PSV (and VIV [MSIV]-Bypass of benefit but not required) for RCP [RCS] depressurisation in the case of LOOP,
  - LHSI for long term heat removal,
- sufficient RCP [RCS] water inventory is maintained to maintain core cooling. The fuel cladding integrity decoupling criterion is met, with a large margin.

APPENDIX 14B.2.16.1 – TABLE 1

FWLB (state A, PCC-4)
Initial conditions for all cases

Parameters	Limiting values used
Reactor coolant system	
- Initial reactor power (% of nominal power)	100 + 2 = 102
- Initial average RCP [RCS] temperature (°C)	311.2 + 2.5 = 313.7
- Initial reactor coolant pressure (bar)	155 + 2.5 = 157.5
- Reactor coolant flow (kg/s)	21035 (T/H)
- Pressuriser water volume / level (m^3 / m)	43 / 7.4 (nominal + 5% MR)
Steam generators	
- Initial steam pressure (bar)	76.7 (consistent with RCP [RCS] temperature)
- Initial SG level (m) in unaffected SG	16.53 (nominal + 5% NR)
- Initial SG level (m) in affected SG	15.85 (nominal - 5% NR)
- Initial SG water liquid mass (tons) in unaffected SG	89.4
- Initial SG water liquid mass (tons) in affected SG	82.3
Feedwater	
- Main feedwater flow (% of nominal flow)	100 + 2 = 102
- Initial ARE [MFWS] temperature (°C)	232

APPENDIX 14B.2.16.1 – TABLE 2

FWLB (state A, PCC-4)
Sequence of events - case 1
(FWLB – from the initiating to the controlled state - without LOOP - SF on one ASG [EFWS])

0. s	FWLB occurs in SG2 All ARE [MFWS] flows are lost via the break
18. s*	Signal dP/dt = MAX1 in SG2
43. s*	Signal dP/dt = MAX2 in SG2
60.4 s	"SG level < MIN1" in unaffected SG
61.9 s	Beginning of VIV [MSIV] closure
62.2 s	- RT - Turbine trip
65.4 s	End of VIV [MSIV] closure
70 s	The VDA [MSRT] setpoints are reached in all unaffected SG, causing the opening of the VDA [MSRT].
98 s	ASG [EFWS] actuation signal on "SG signal < MIN2" in SG1
117 s	ASG [EFWS] flow injection into SG1 (15 s delay between actuation signal and injection in cases without LOOP). Maintenance is assumed on the ASG [EFWS] pump delivering to SG4, SF is applied to the ASG [EFWS] pump delivering to SG3. ASG [EFWS] flow delivered to the affected SG2 is assumed fully lost via the break.
3660 s	At this time, the operator performs the realignment of the ASG [EFWS] pump connected to the affected SG (SG2) towards one unaffected and non-supplied SG (SG3). Following that action the plant has reached the controlled state.

* signal not credited in the accident analysis

APPENDIX 14B.2.16.1 – TABLE 3

FWLB (state A, PCC-4)
Sequence of events - case 2
(FWLB - from the initiating event to the controlled state - with LOOP - SF on one VDA [MSRT])

0. s	FWLB occurs in SG2 All ARE [MFWS] flows are lost via the break
18. s*	Signal dP/dt = MAX1 in SG2
43. s*	Signal dP/dt = MAX2 in SG2
60.4 s	« SG level < MIN1 » in unaffected SG
61.9 s	Beginning of VIV [MSIV] closure
62.2 s	- RT - Turbine trip - LOOP, the 4 Reactor Coolant Pumps are tripped
66.9 s	End of MSIV closure
75 s	The VDA [MSRT] setpoints are reached in all unaffected SG But only VDA [MSRT] of SG1 and SG4 open. SF is applied on the VDA [MSRT] of SG3 (stuck closed). The pressure in SG3 increases up to the MSSV setpoint (reached at t = 90 s).
124 s	ASG [EFWS] actuation signal on "SG level < MIN2" in SG1)
177 s	ASG [EFWS] flow delivery to SG1 (50 s delay between actuation signal and injection in cases with LOOP). Maintenance is postulated on the ASG [EFWS] pump connected to SG4 ASG [EFWS] flow delivered to the affected SG2 is assumed fully lost via the break
206 s	ASG [EFWS] flow delivery to SG3
~600 s	The controlled state is reached

* signal not credited in the accident analysis

APPENDIX 14B.2.16.1 – TABLE 4

FWLB (state A, PCC-4)
Sequence of events - case 3
(FWLB - transfer to the safe shutdown state - w/o LOOP)

0. s	FWLB occurs in SG2 All ARE [MFWS] flows are lost via the break
18. s*	Signal dP/dt = MAX1 in SG2
43. s*	Signal dP/dt = MAX2 in SG2
61.9 s	"SG level < MIN1" in unaffected SG
60.4 s	Beginning of VIV [MSIV] closure
62.2 s	- RT, TT
65.4 s	End of MSIV closure
70 s	The VDA [MSRT] setpoints are reached inducing VDA [MSRT] opening on SG1 and SG4 (failure is assumed of the VDA [MSRT] of SG3)
101 s	ASG [EFWS] actuation signal on "SG level < MIN2" in unaffected SG
115 s	ASG [EFWS] actuation signal on "SG level < MIN2" in SG3
118 s	ASG [EFWS] flow injection into SG1 (maintenance is assumed on the ASG [EFWS] connected to SG4)
132 s	ASG [EFWS] flow delivery to SG3
3660 s (1 h)	Operator action time → first operator actions : - complete isolation of affected SG2 (isolation of steam and water flows) - Realignment of the ASG [EFWS] pump assigned to the affected SG2 to deliver to one unaffected SG not supplied with ASG [EFWS] water (i.e. SG4, The ASG [EFWS] pump of which is ASG [EFWS] water assumed to be in maintenance)
7260 s (2 h)	- Beginning of the cooldown at -25°C/h - Trip of 2 Reactor Coolant Pumps (on loop 1 and loop 4) - Beginning of the boration by one RBS [EBS] train (failure is postulated on the second one)

* signal not credited in the accident analysis

APPENDIX 14B.2.16.1 – TABLE 4

FWLB (state A, PCC-4)
Sequence of events - case 3
(FWLB - transfer to the safe shutdown state - w/o LOOP)

13200 s (3.7 h) - End of RBS [EBS] boration

29185 s (8.1 h) - End of RCP [RCS] cooldown phase
$T_{RCP\ [RCS]}$ < 180°C in all hot legs
$P_{RCP\ [RCS]}$ < 30 bar
- The RIS/RRA [SIS/RHRS] connection conditions are reached

APPENDIX 14B.2.16.1 – TABLE 5

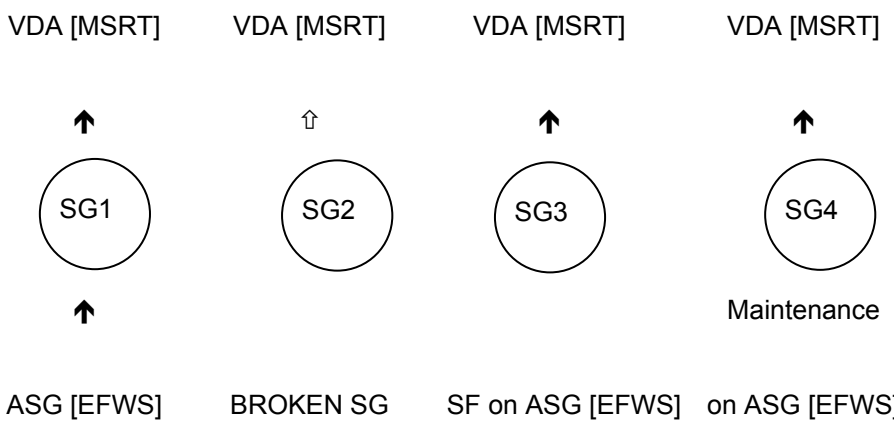
FWLB (state A, PCC-4)
Sequence of events - case 4
(FWLB - transfer to the safe shutdown state- with LOOP)

0. s	FWLB occurs in SG2 All ARE [MFWS] flows are lost via the break
18. s*	Signal dP/dt = MAX1 in SG2
43. s*	Signal dP/dt = MAX1 in SG2
60.4 s	"SG level < MIN1" in unaffected SG
61.9 s	Beginning of VIV [MSIV] closure
62.2 s	- RT, TT - LOOP = The 4 Reactor Coolant Pumps are tripped
65.4 s	End of VIV [MSIV] closure
72 s	The VDA [MSRT] setpoints are reached demanding the opening of the VDA [MSRT] on SG1, SG3 and SG4
122 s	ASG [EFWS] actuation signal on "SG level < MIN2" in unaffected SG
173 s	ASG [EFWS] flow delivery to SG3 (maintenance is assumed on ASG [EFWS] connected to SG4 and failure is assumed on ASG [EFWS] connected to SG1);
3660 s (1 h)	Operator action time → first operator actions : - complete isolation of affected SG2 (isolation of steam and water flows) - realignment of the ASG [EFWS] pump connected to the affected SG2 to one unaffected SG not supplied with ASG [EFWS] water (i.e. SG4, the ASG [EFWS] pump of which is assumed to be on maintenance)
7260 s (2 h)	- Beginning of the cooldown at -25°C/h - Beginning of the boration by one RBS [EBS] train (failure is assumed on the second train)
13200 s (3.7 h)	- End of RBS [EBS] boration
30370 s (8.4 h)	- End of RCP [RCS] cooldown phase (VDA [MSRT] setpoint = 4 bar) - Decreasing of 1 PSV setpoint to 10 bar → $T_{RCP\ [RCS]}$ < 180°C in the heat removing loops → $P_{RCP\ [RCS]}$ < 30 bar

* signal not credited in the accident analysis

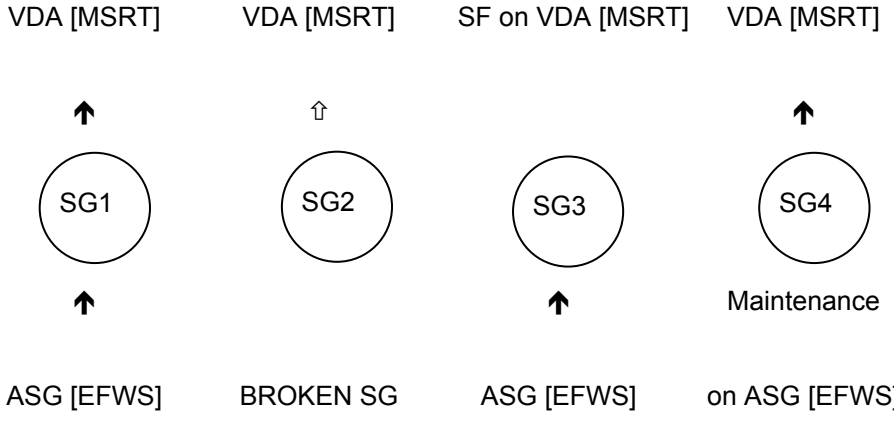
APPENDIX 14B.2.16.1 – FIGURE 1

CASE 1 : From the initiating event to the controlled state (without LOOP)



Note : AT t_{RT} + 1 hour, realignment of the ASG [EFWS] pump connected to SG2 to deliver to SG3

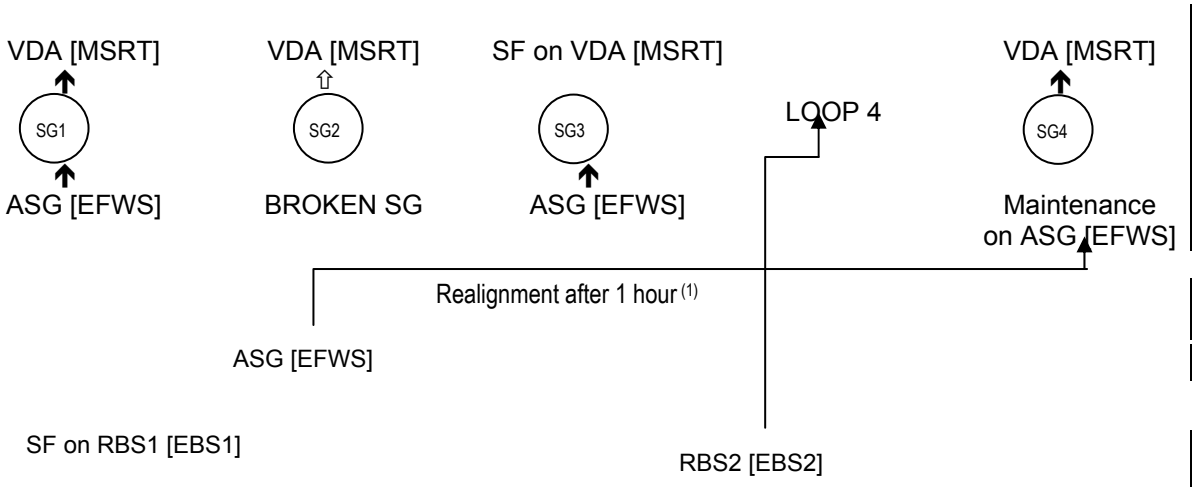
CASE 2 : From the initiating event to the controlled state (with LOOP)



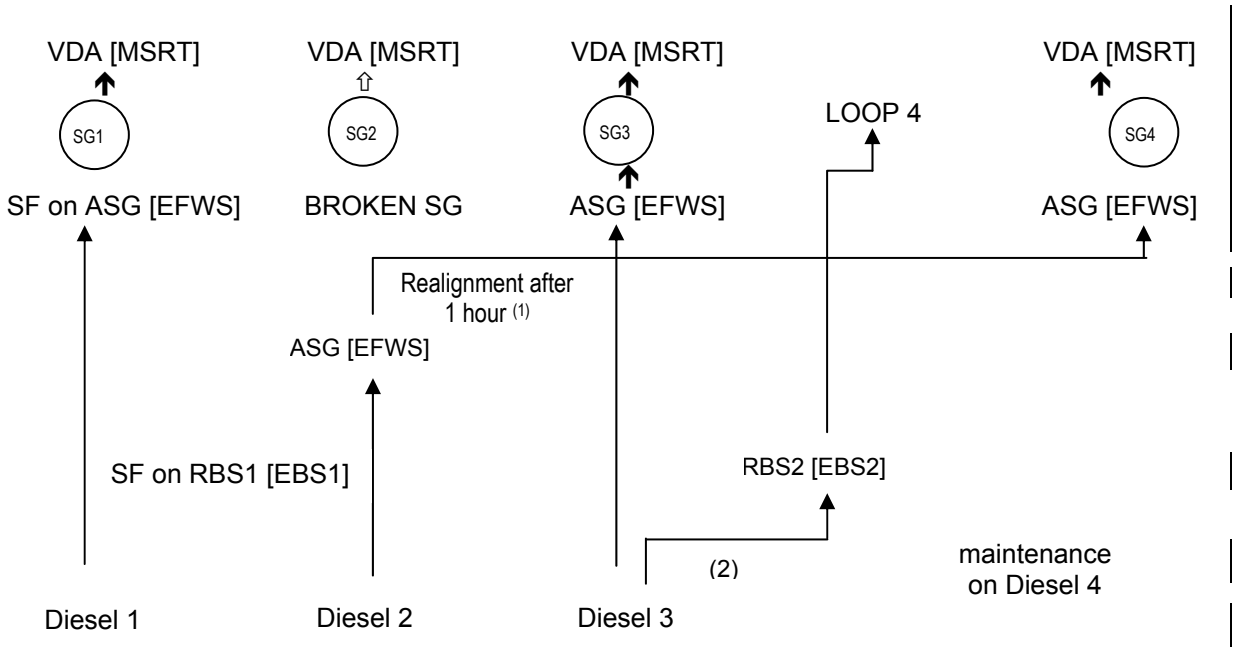
FWLB (state A, PCC-4)
Illustration of the different FWLB - cases analysed
Cases 1 and 2 (objective = max RCP [RCS] overheating)

APPENDIX 14B.2.16.1 – FIGURE 2

CASE 3 : From the initiating event to the controlled state (without LOOP)



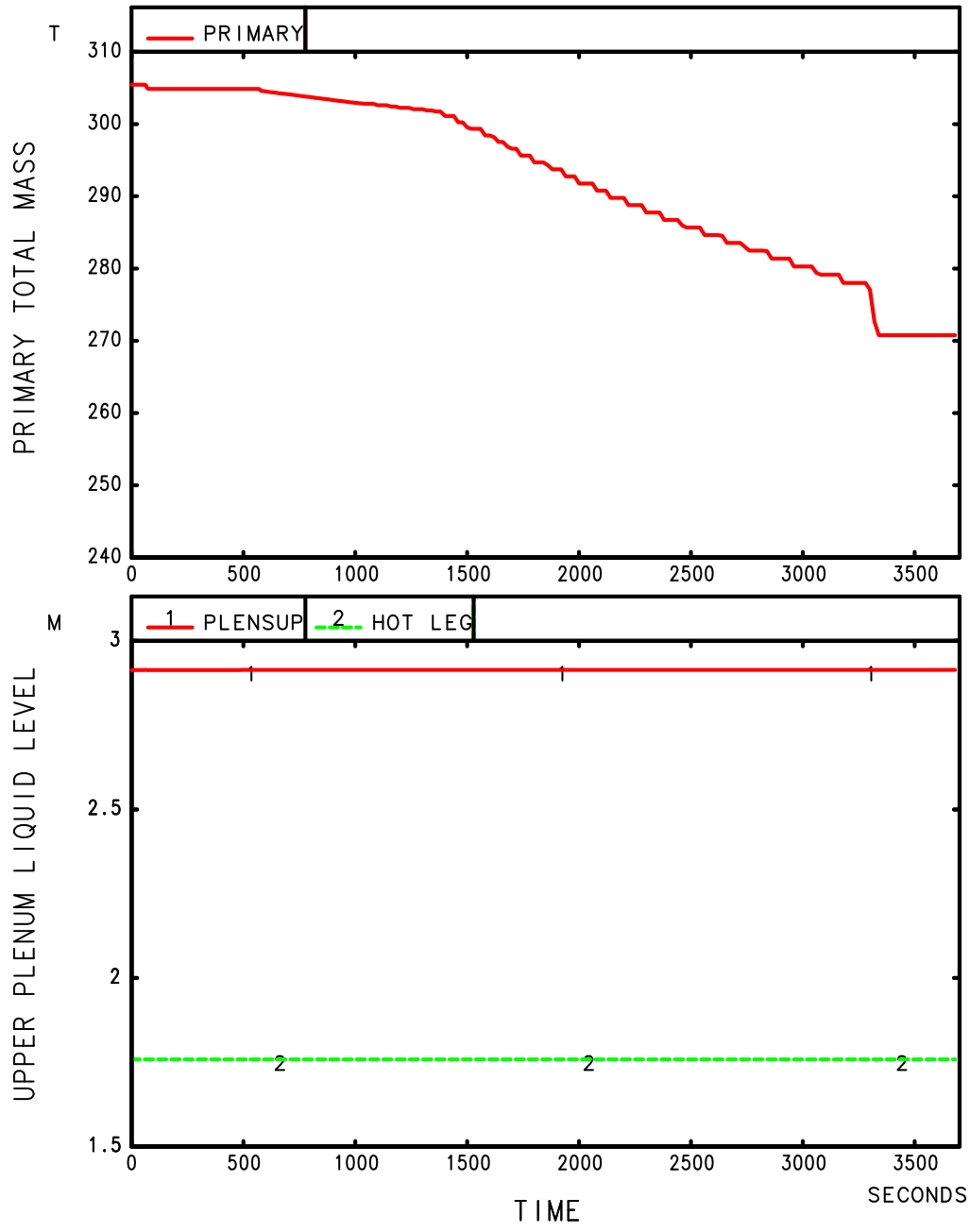
CASE 4 : From the initiating event to the controlled state (with LOOP)



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

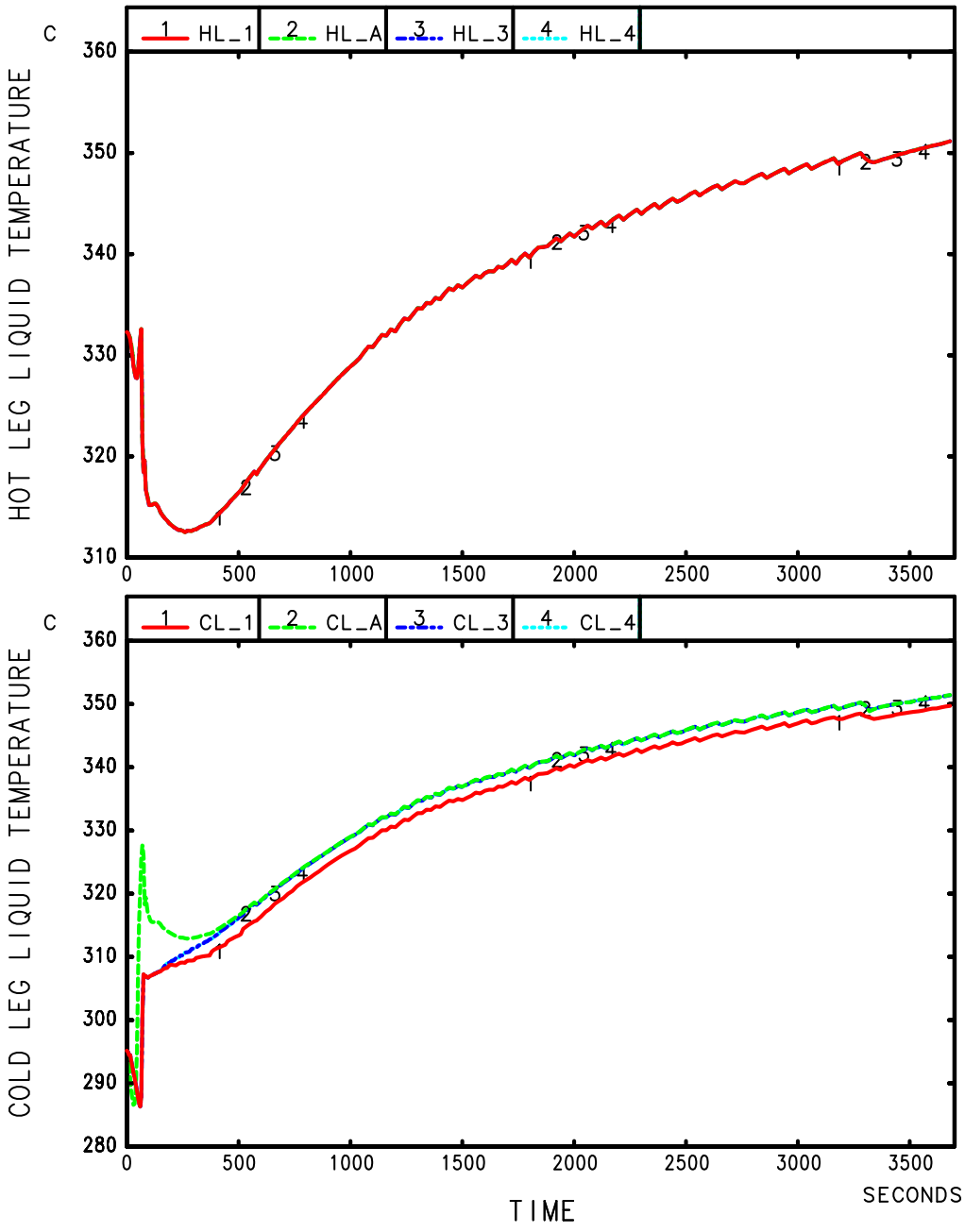
FWLB (state A, PCC-4)
Illustration of the different FWLB-cases analysed
Case 3 (criterion = ASG [EFWS] tanks capacity)
Case 4 (criterion = RCP [RCS] boration and depressurisation)

APPENDIX 14B.2.16.1 – FIGURE 3



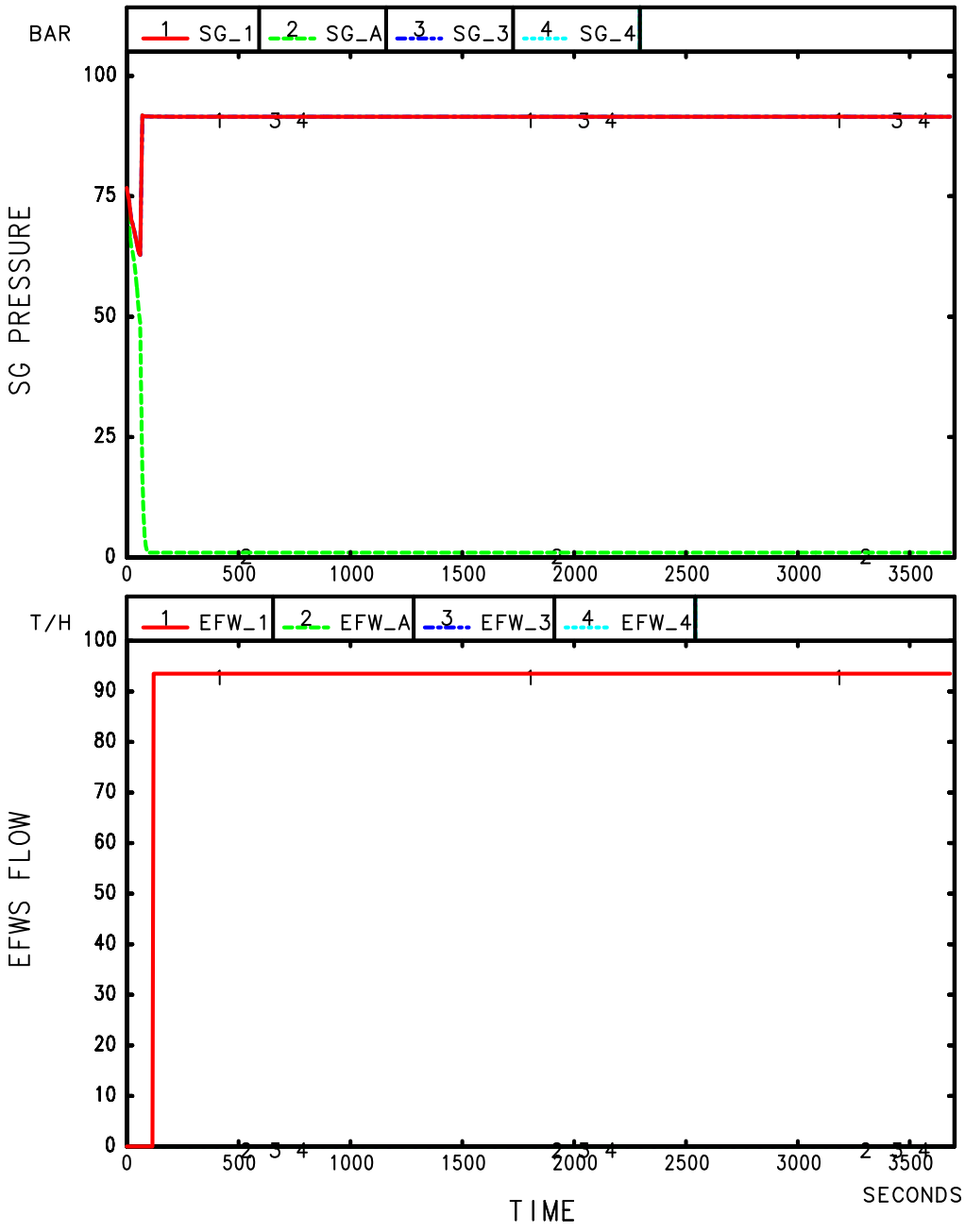
FWLB (state A, PCC-4)
Case 1 - from the initiating event to the controlled state - without LOOP - SF on one ASG [EFWS]

APPENDIX 14B.2.16.1 – FIGURE 4



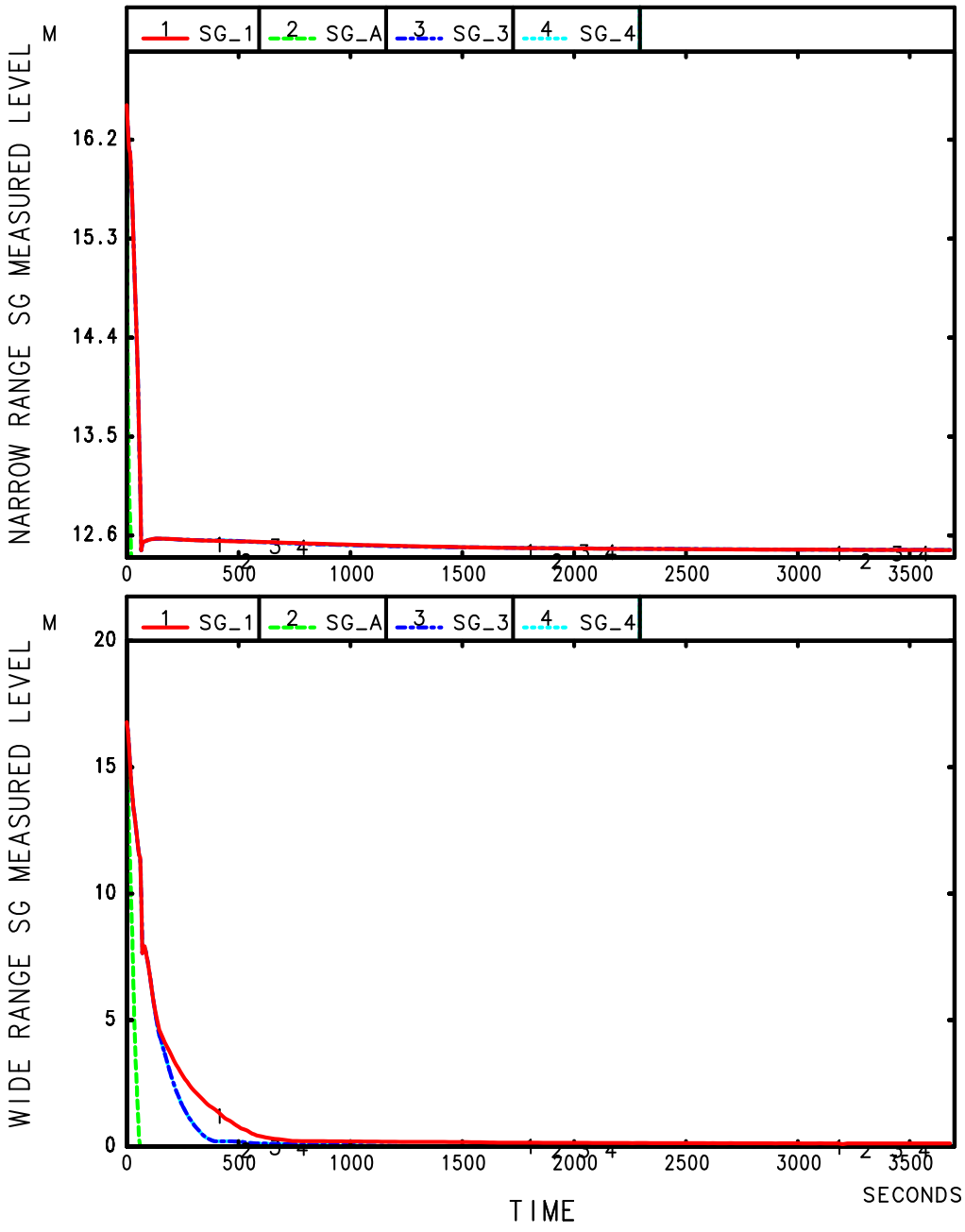
FWLB (state A, PCC-4)
Case 1 - from the initiating event to the controlled state - without LOOP - SF on one ASG [EFWS]

APPENDIX 14B.2.16.1 – FIGURE 5



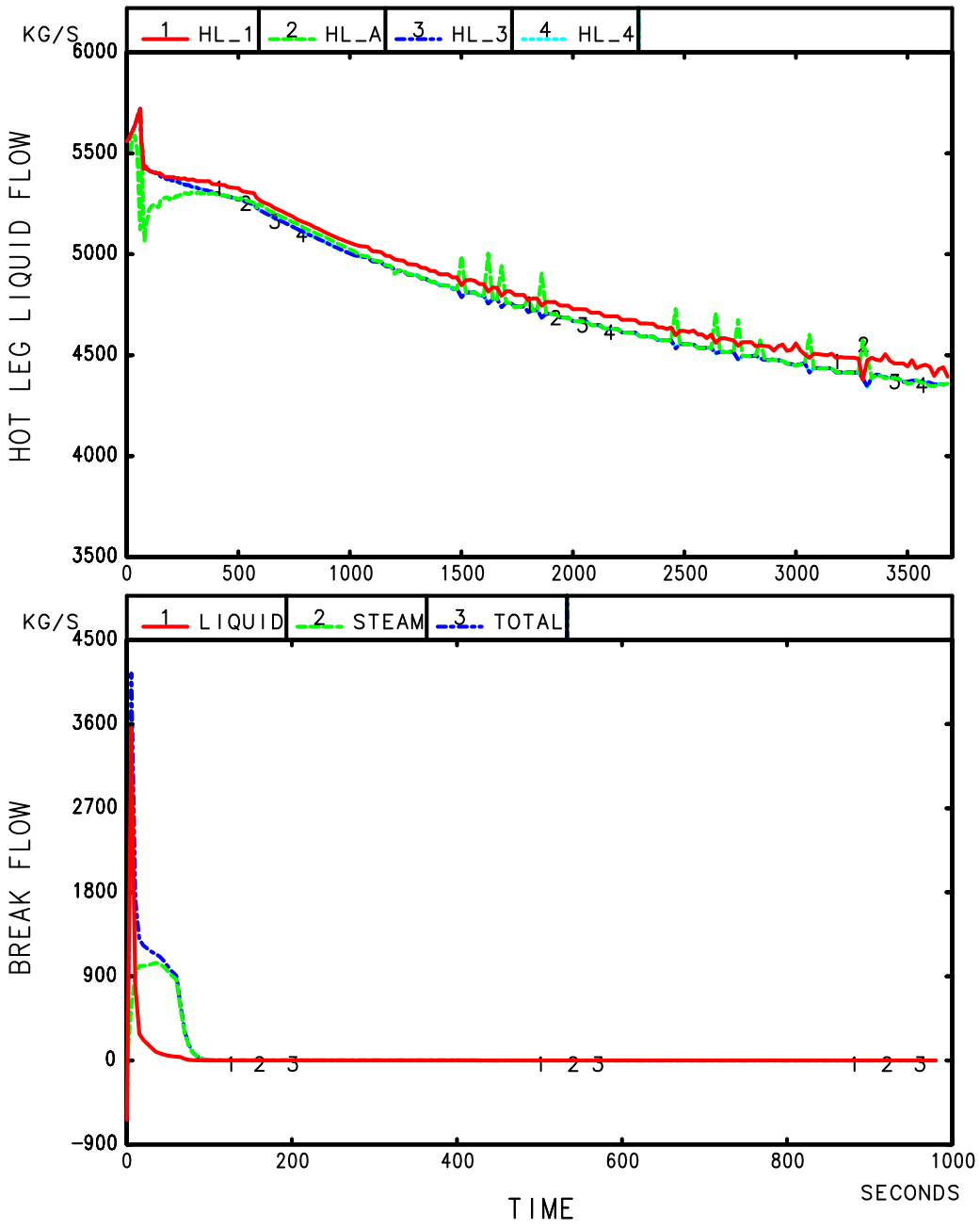
FWLB (state A, PCC-4)
Case 1 - from the initiating event to the controlled state - without LOOP - SF on one ASG [EFWS]

APPENDIX 14B.2.16.1 – FIGURE 6



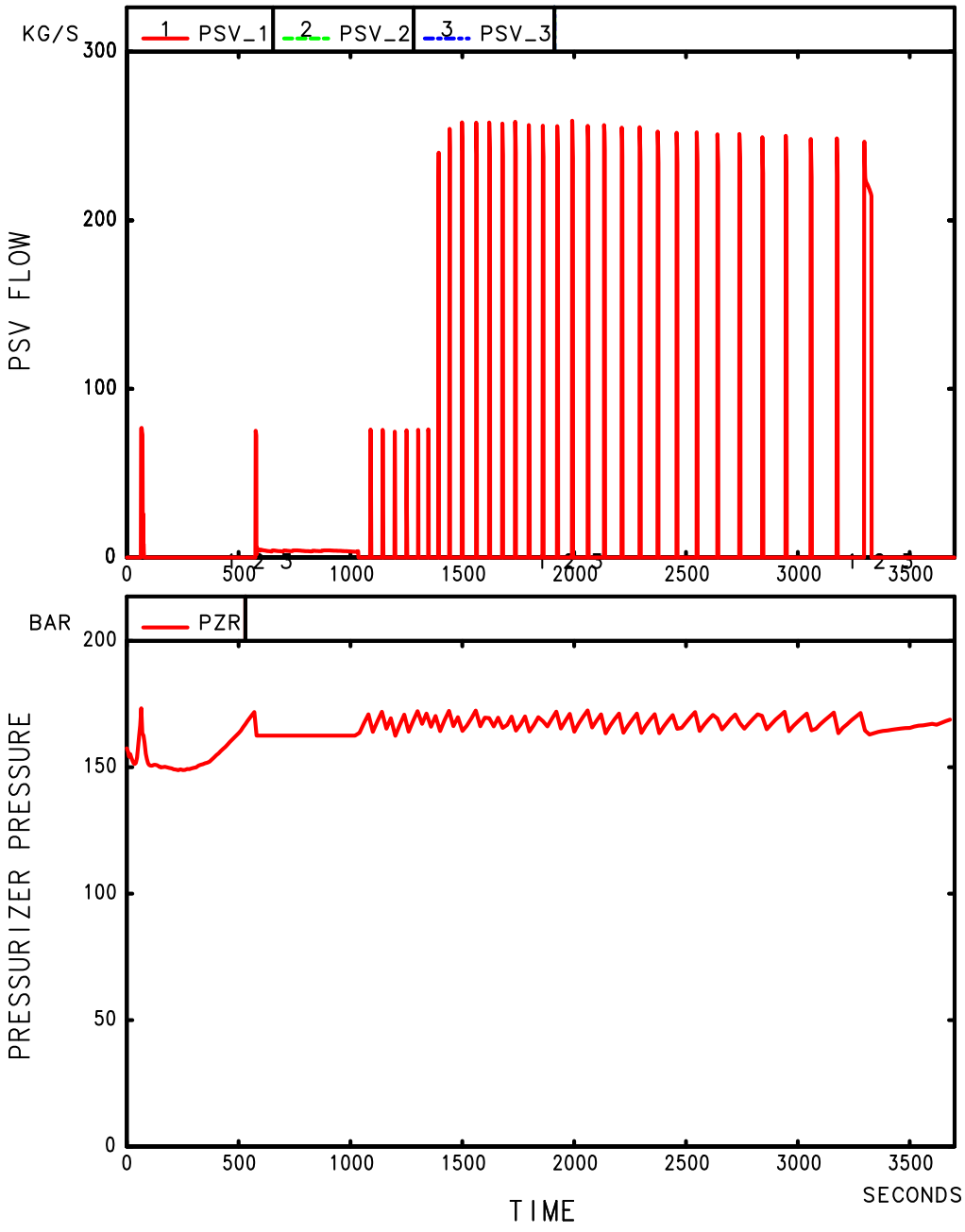
FWLB (state A, PCC-4)
Case 1 - from the initiating event to the controlled state - without LOOP - SF on one ASG [EFWS]

APPENDIX 14B.2.16.1 – FIGURE 7



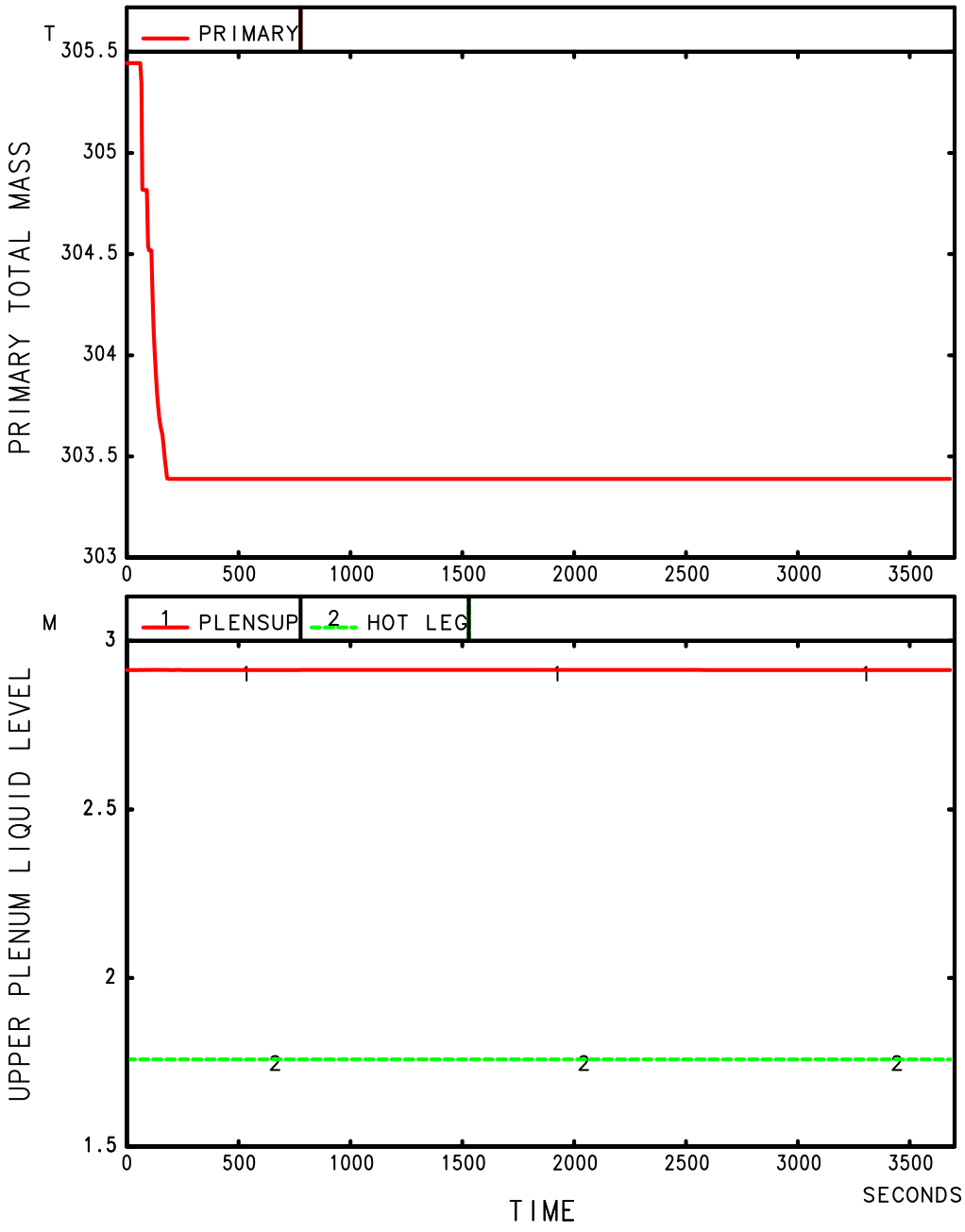
FWLB (state A, PCC-4)
Case 1 – from the initiating event to the controlled state - without LOOP - SF on one ASG [EFWS]

APPENDIX 14B.2.16.1 – FIGURE 8



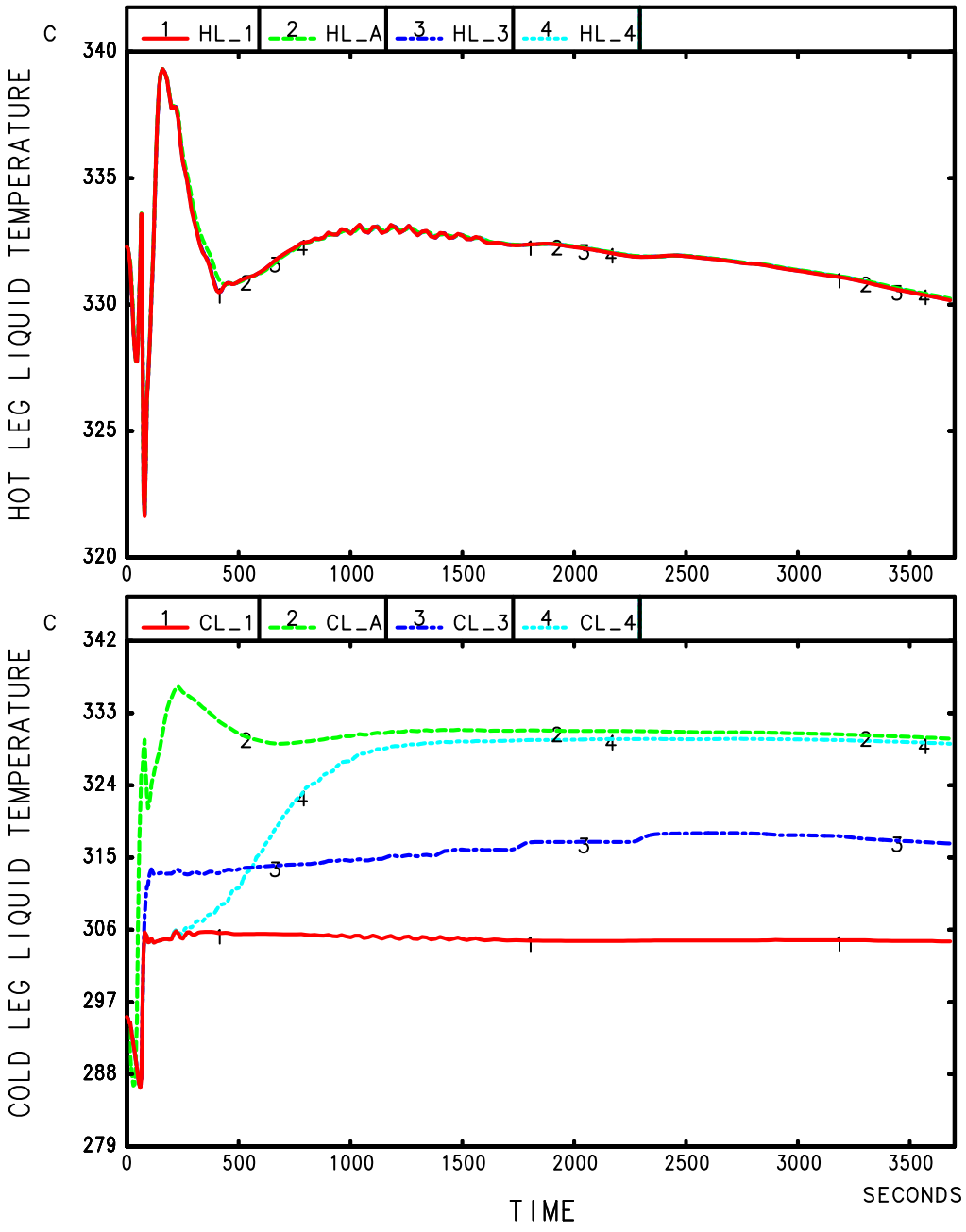
FWLB (state A, PCC-4)
Case 1 - from the initiating event to the controlled state - without LOOP - SF on one ASG [EFWS]

APPENDIX 14B.2.16.1 – FIGURE 9



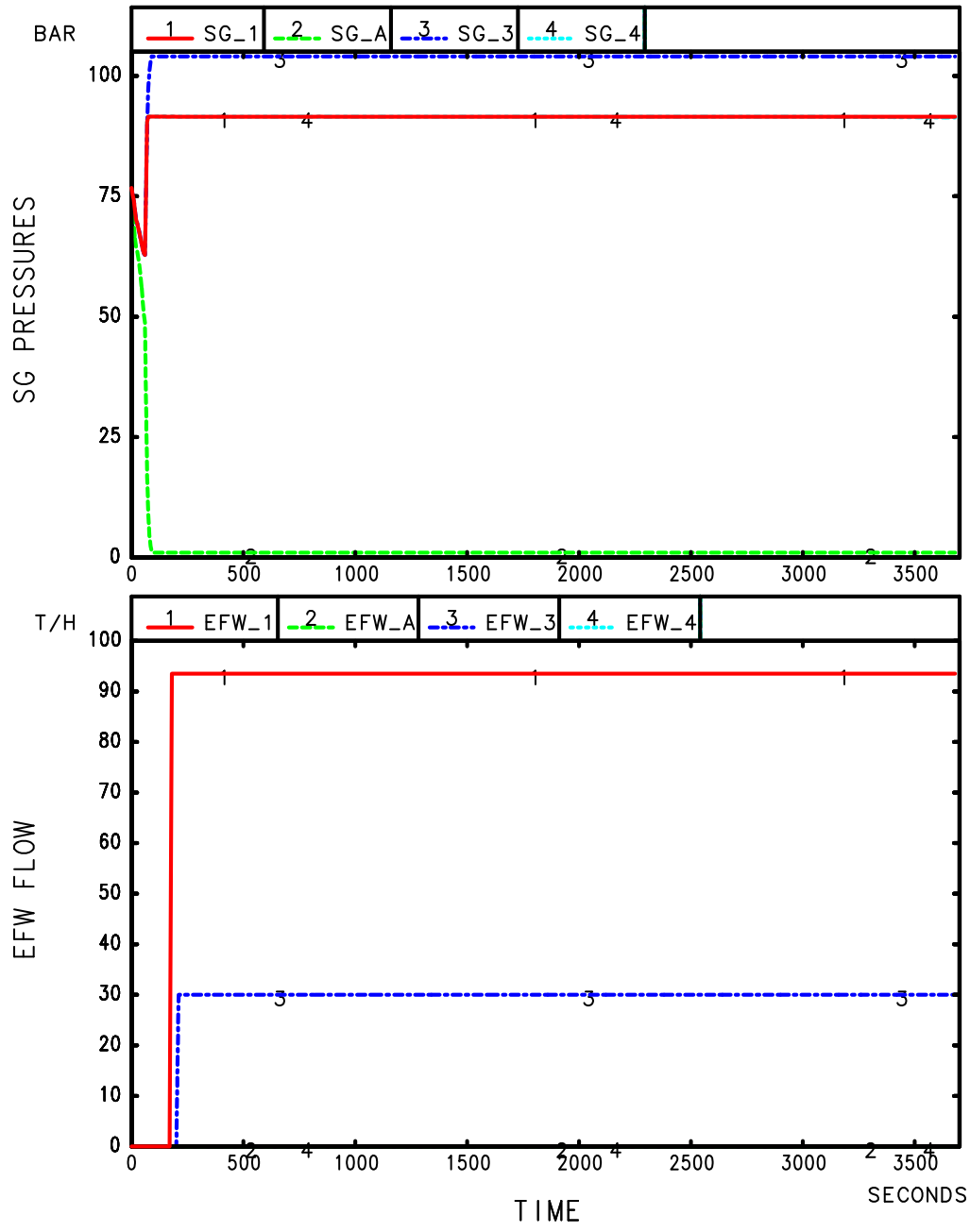
FWLB (state A PCC-4)
Case 2 - from the initiating event to the controlled state - with LOOP - SF on one VDA [MSRT]

APPENDIX 14B.2.16.1 – FIGURE 10



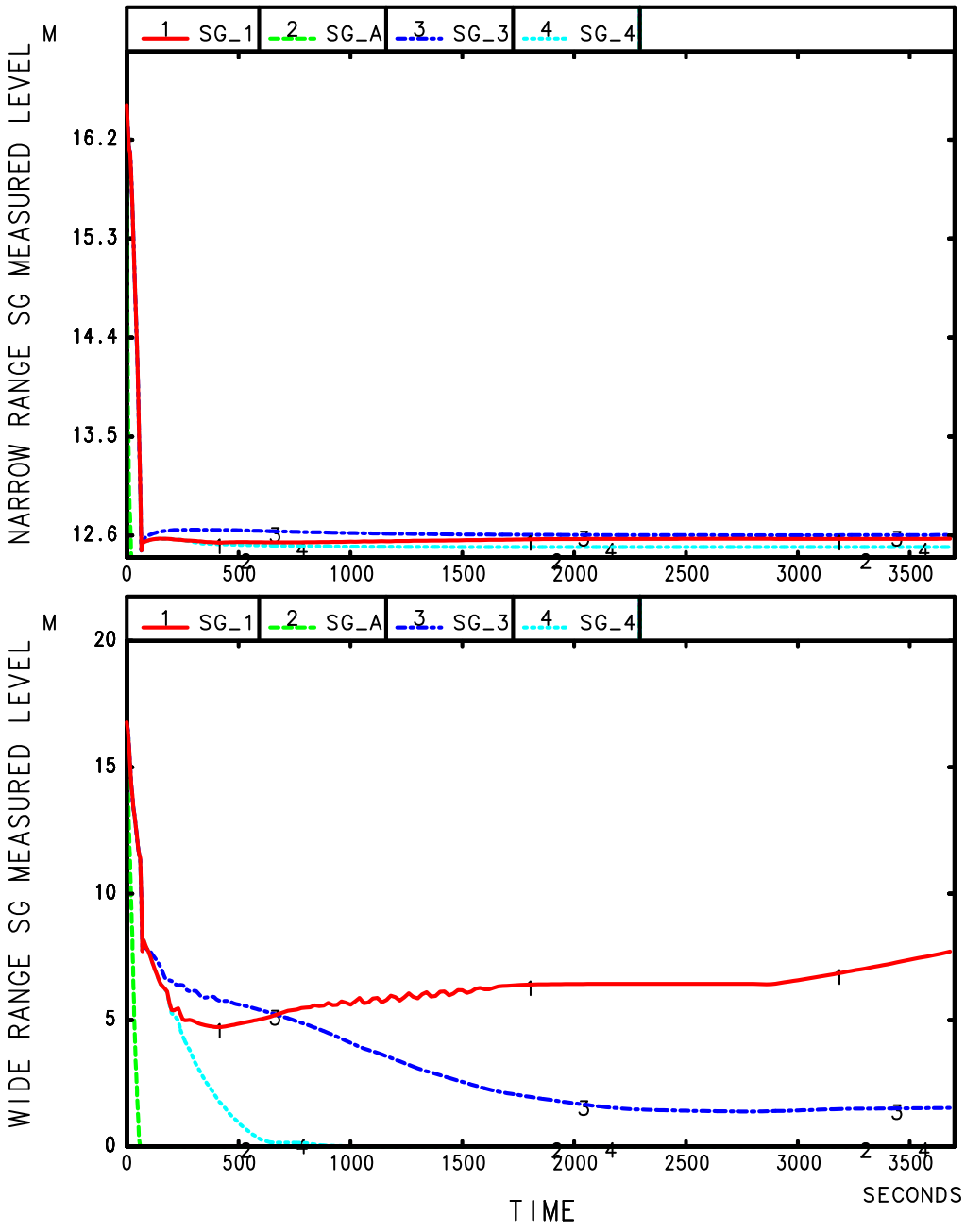
FWLB (state A PCC-4)
Case 2 - from the initiating event to the controlled state - with LOOP - SF on one VDA [MSRT]

APPENDIX 14B.2.16.1 – FIGURE 11



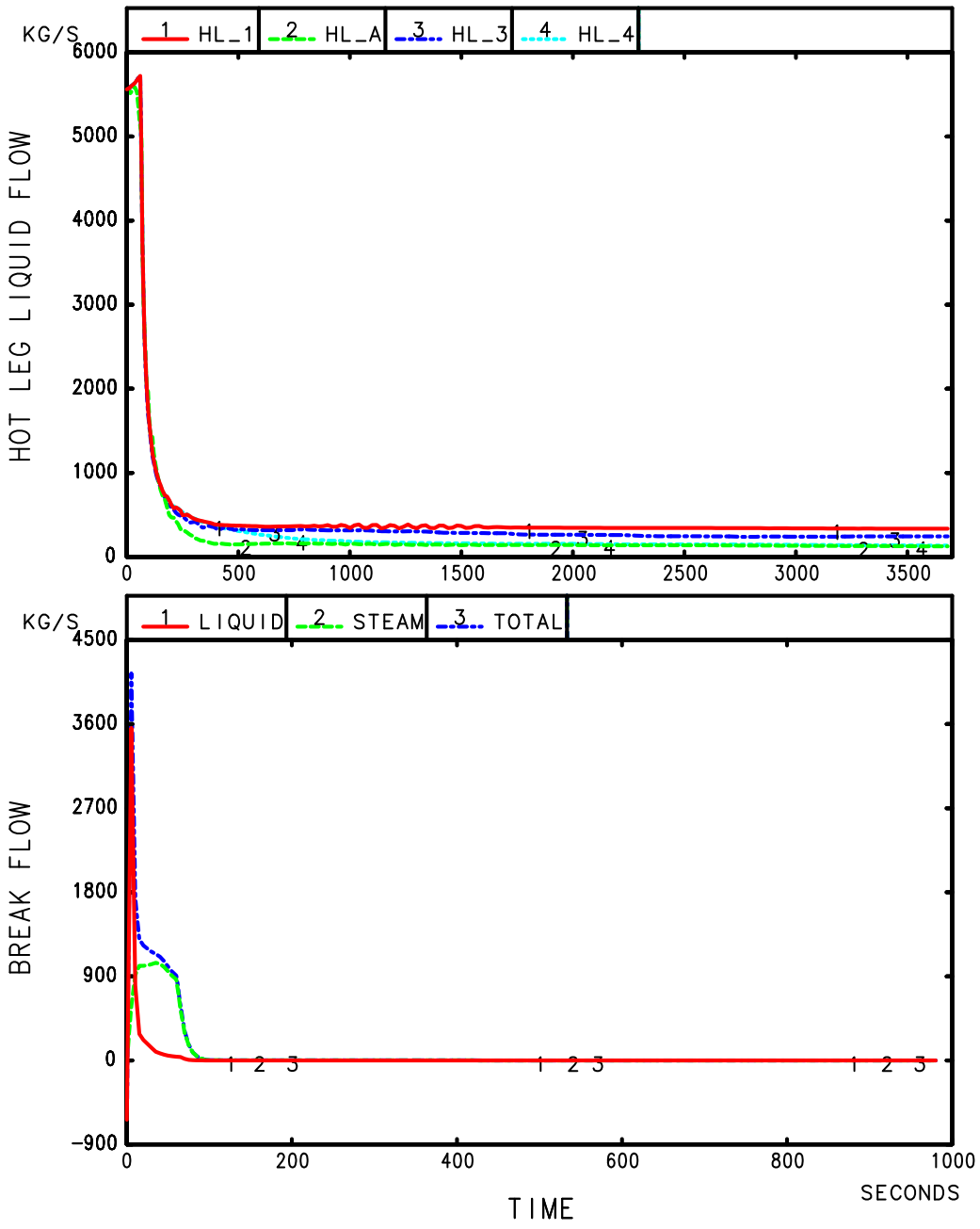
FWLB (state A PCC-4)
Case 2 – from the initiating event to the controlled state - with LOOP - SF on one VDA [MSRT]

APPENDIX 14B.2.16.1 – FIGURE 12



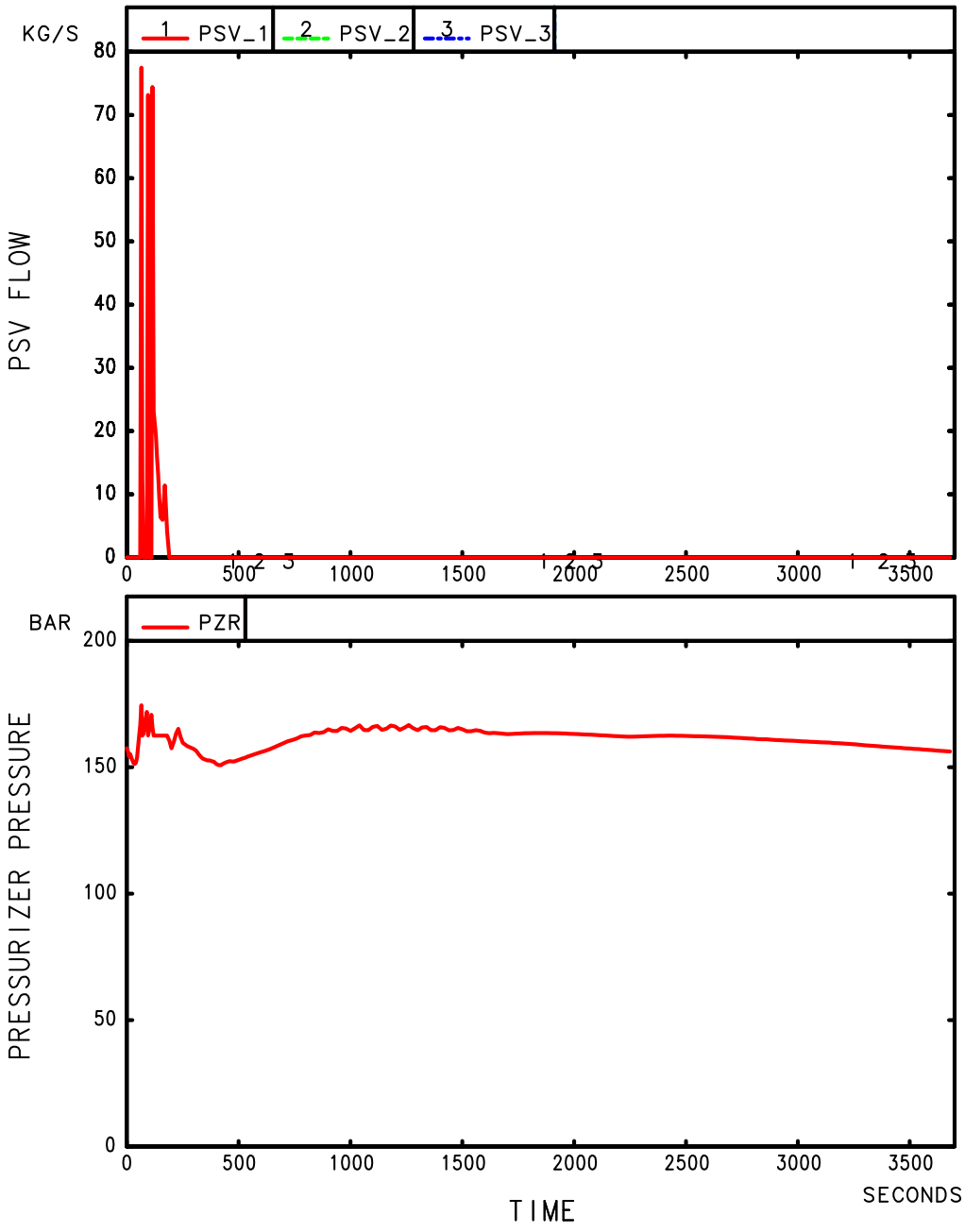
FWLB (state A PCC-4)
Case 2 - from the initiating event to the controlled state - with LOOP - SF on one VDA [MSRT]

APPENDIX 14B.2.16.1 – FIGURE 13



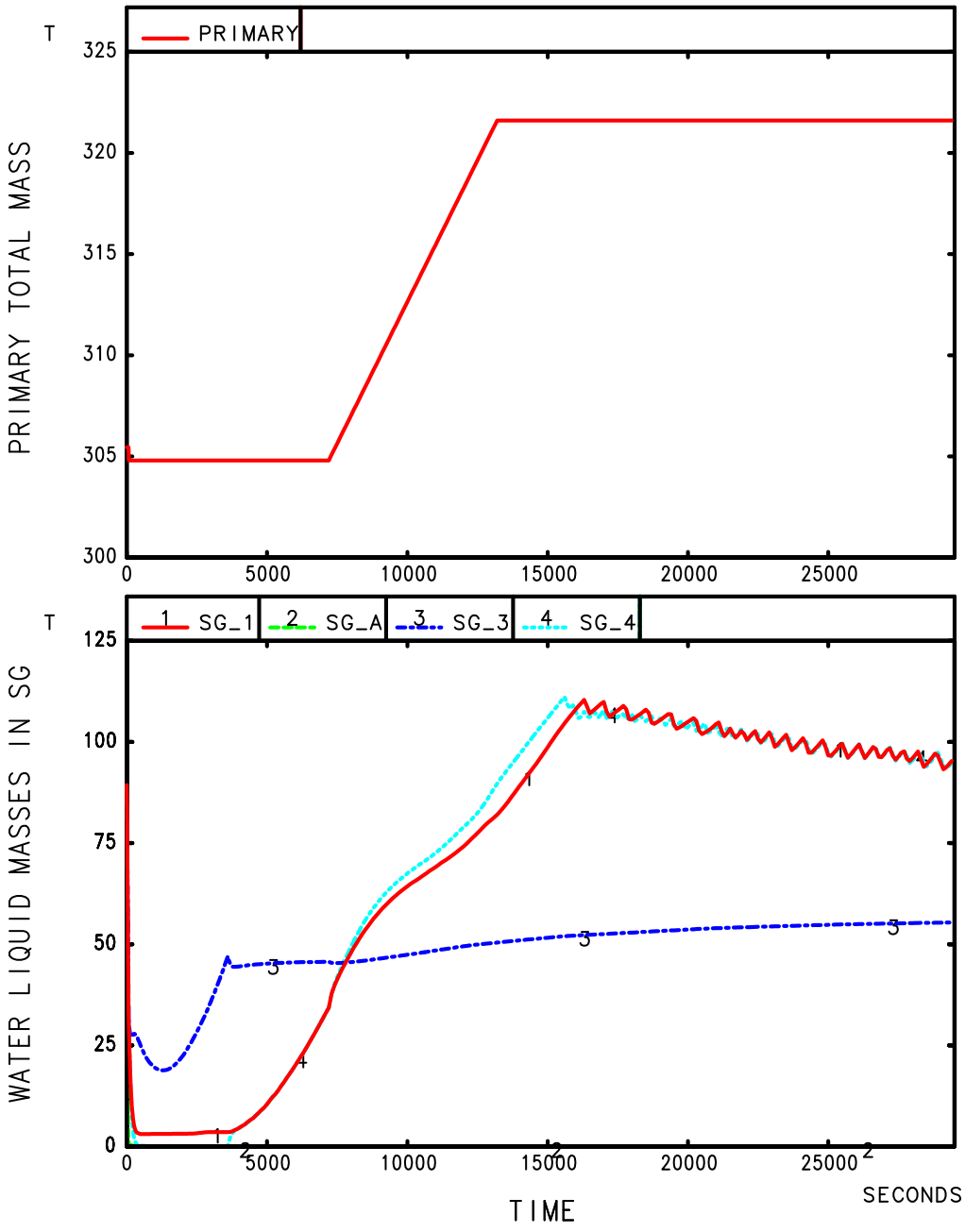
FWLB (state A PCC-4)
Case 2 - from the initiating event to the controlled state - with LOOP - SF on one VDA [MSRT]

APPENDIX 14B.2.16.1 – FIGURE 14



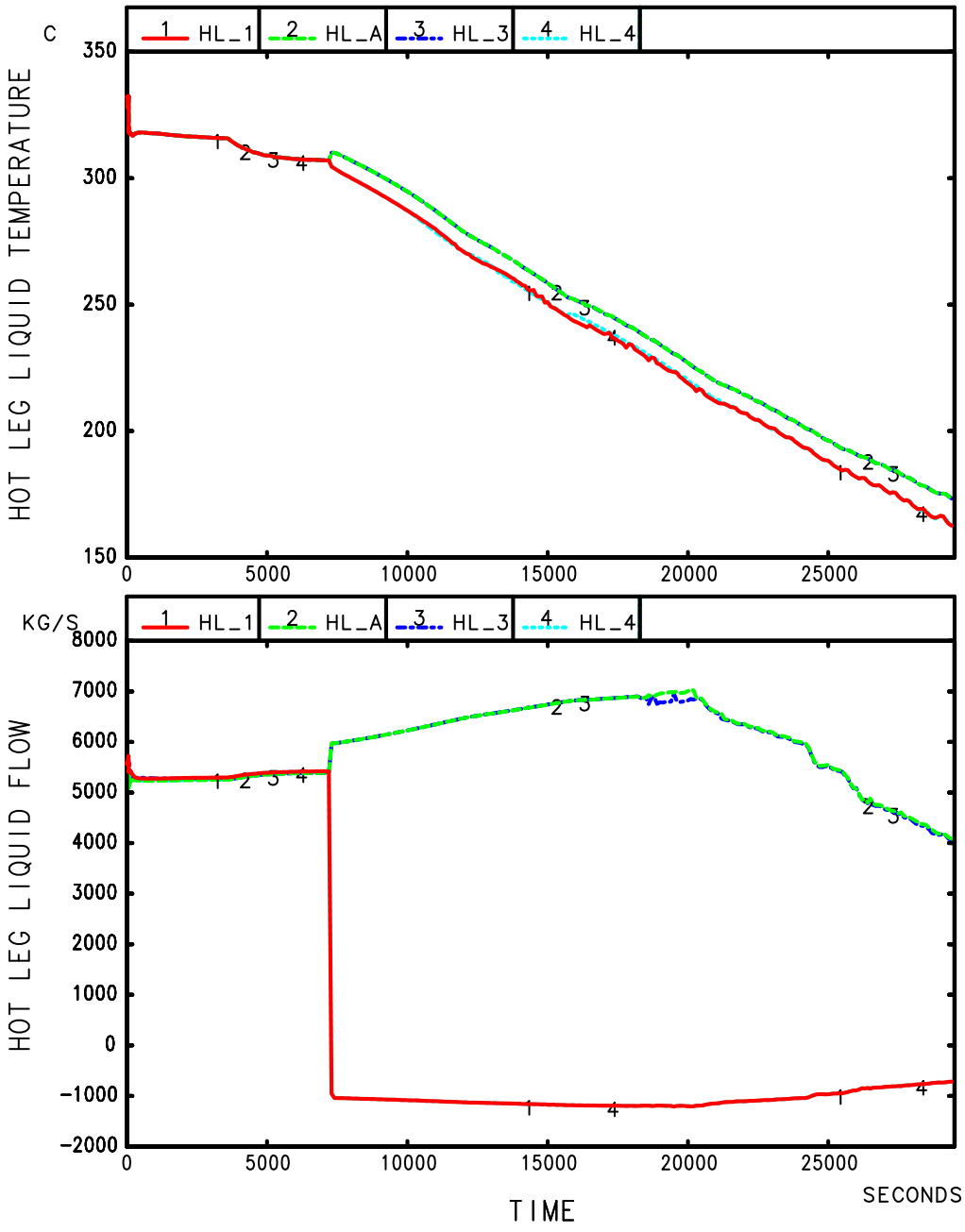
FWLB (state A PCC-4)
Case 2 – from the initiating event to the controlled state - with LOOP - SF on one VDA [MSRT]

APPENDIX 14B.2.16.1 – FIGURE 15



FWLB (state A, PCC-4)
Case 3 - Transfer to the safe shutdown state - without LOOP - ASG [EFWS] tanks capacity

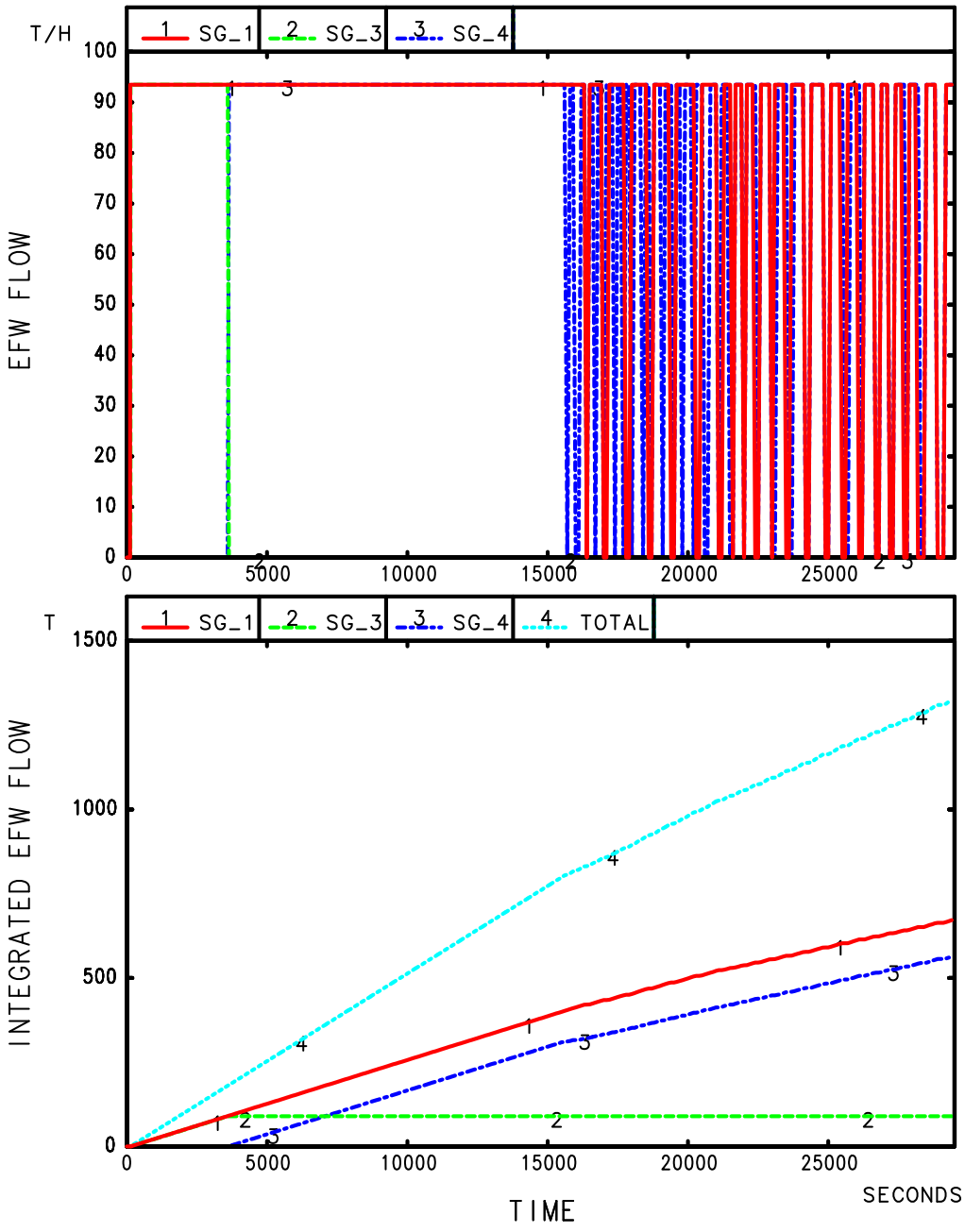
APPENDIX 14B.2.16.1 – FIGURE 16



FWLB (state A, PCC-4)
Case 3 - Transfer to the safe shutdown state - without LOOP - ASG [EFWS] tanks capacity

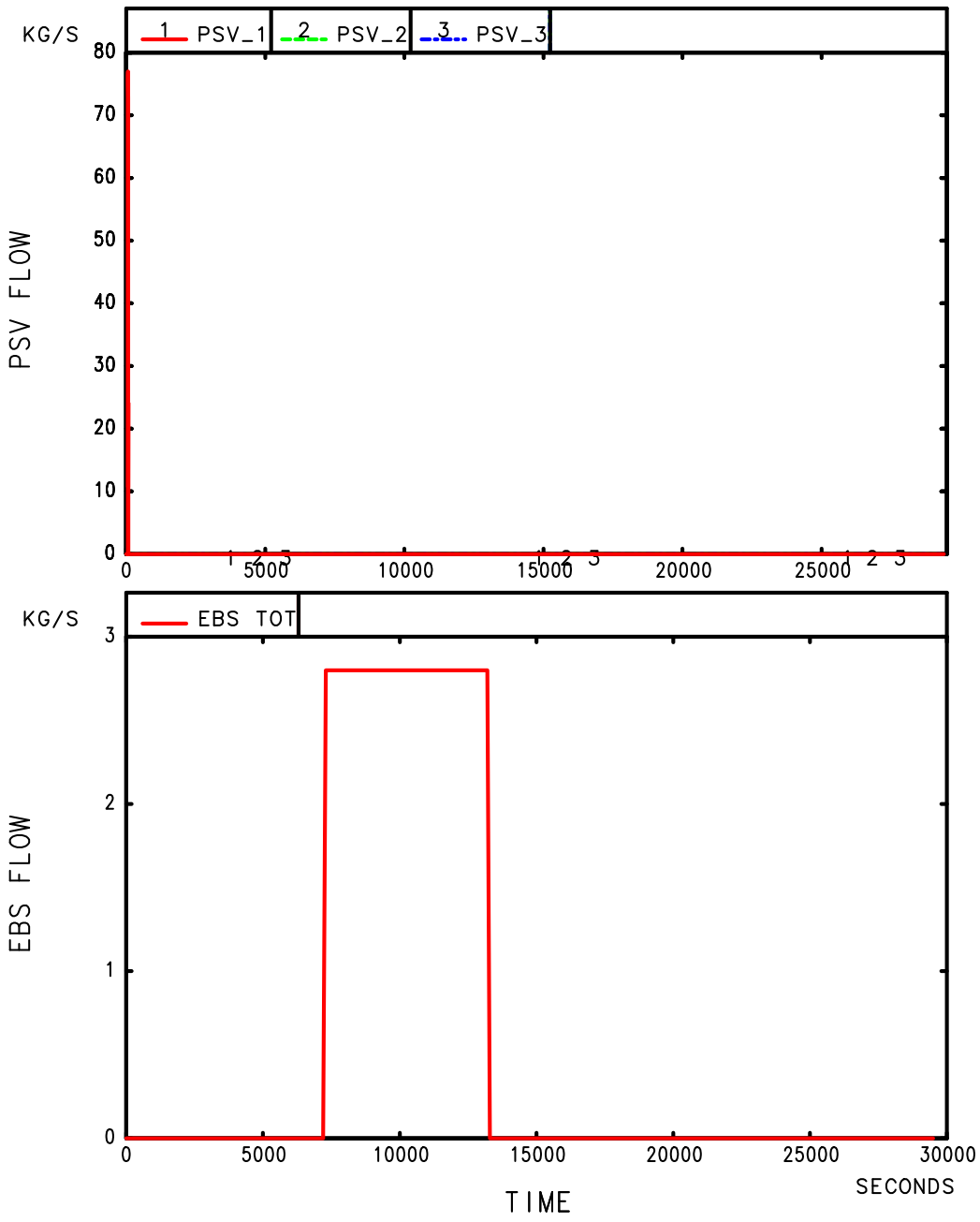
APPENDIX 14B.2.16.1 – FIGURE 17

7



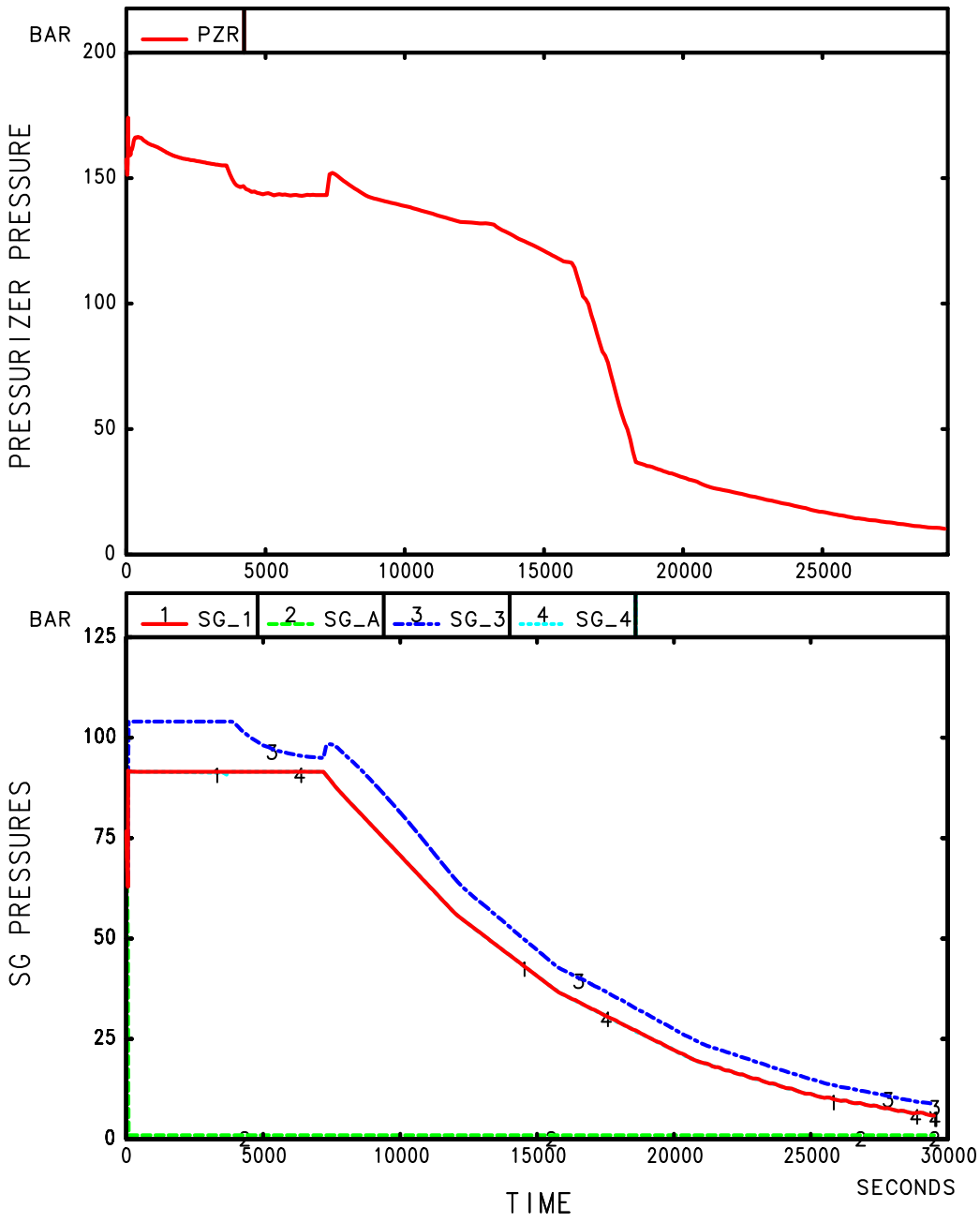
FWLB (state A, PCC-4)
Case 3 - Transfer to the safe shutdown state - without LOOP - ASG [EFWS] tanks Capacity - ASG [EFWS] flow / water liquid masses in SG

APPENDIX 14B.2.16.1 – FIGURE 18



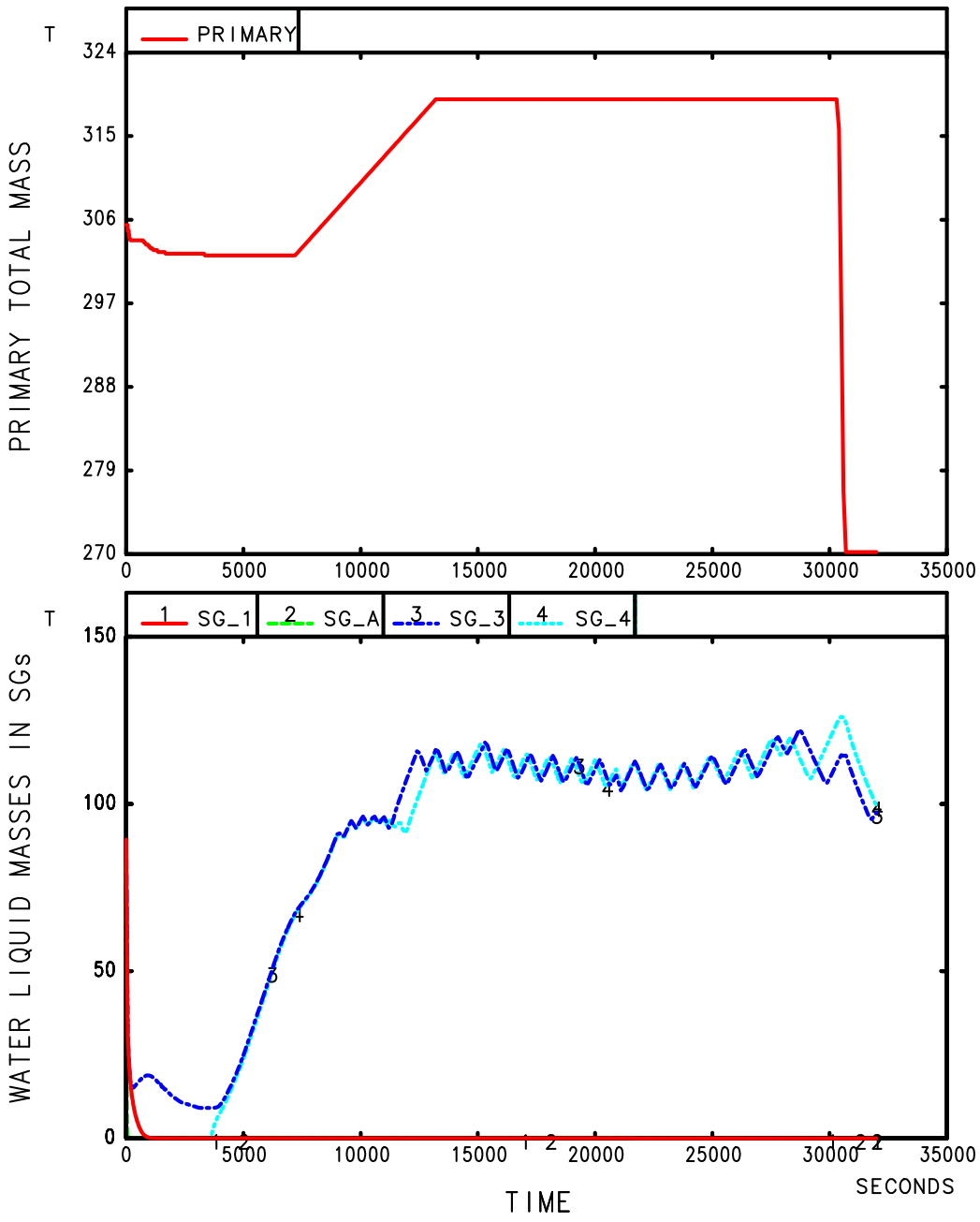
FWLB (state A, PCC-4)
Case 3 - Transfer to the safe shutdown state - without LOOP - ASG [EFWS] tanks capacity

APPENDIX 14B.2.16.1 – FIGURE 19



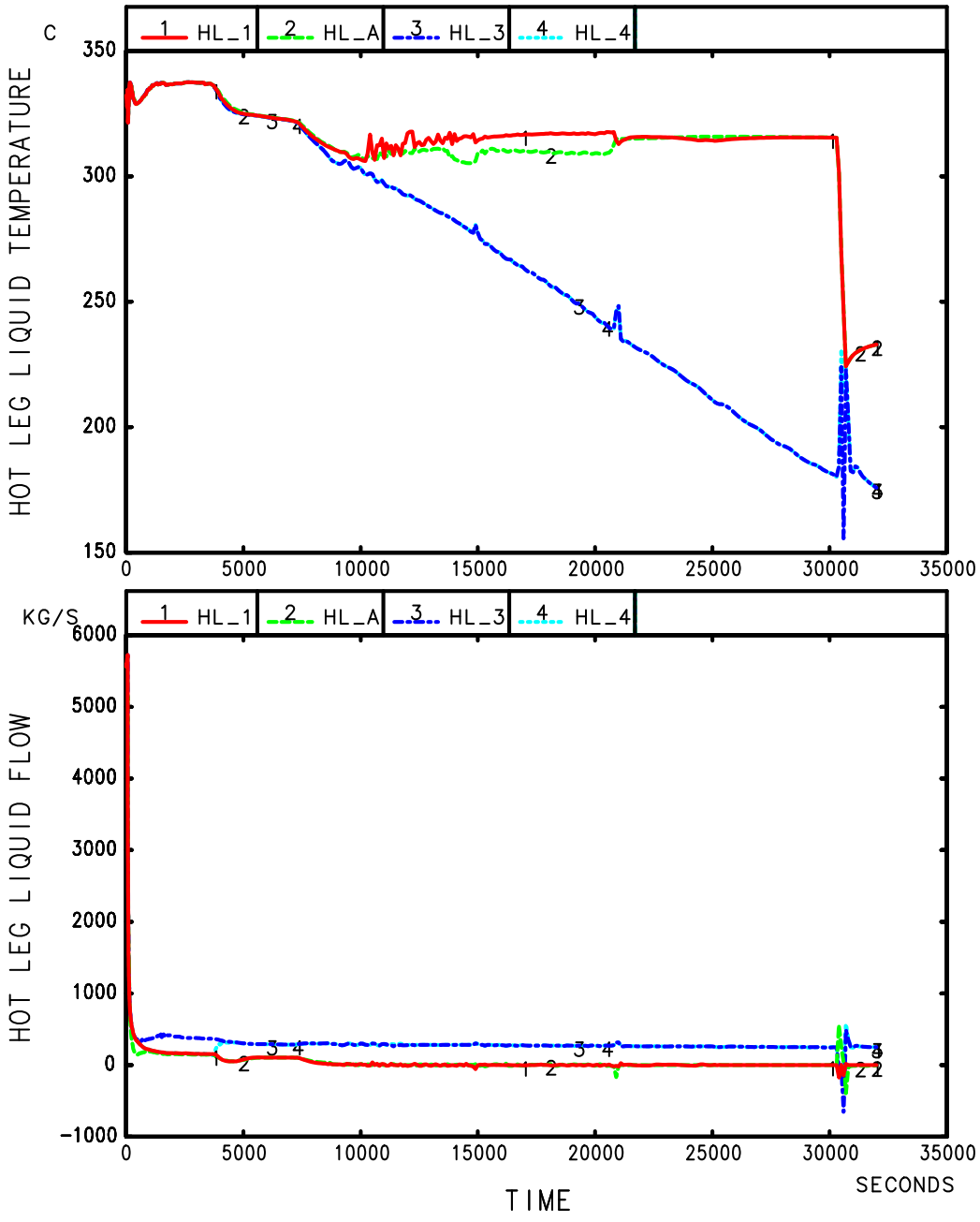
FWLB (state A, PCC-4)
Case 3 - Transfer to the safe shutdown state - without LOOP - ASG [EFWS] tanks capacity

APPENDIX 14B.2.16.1 – FIGURE 20



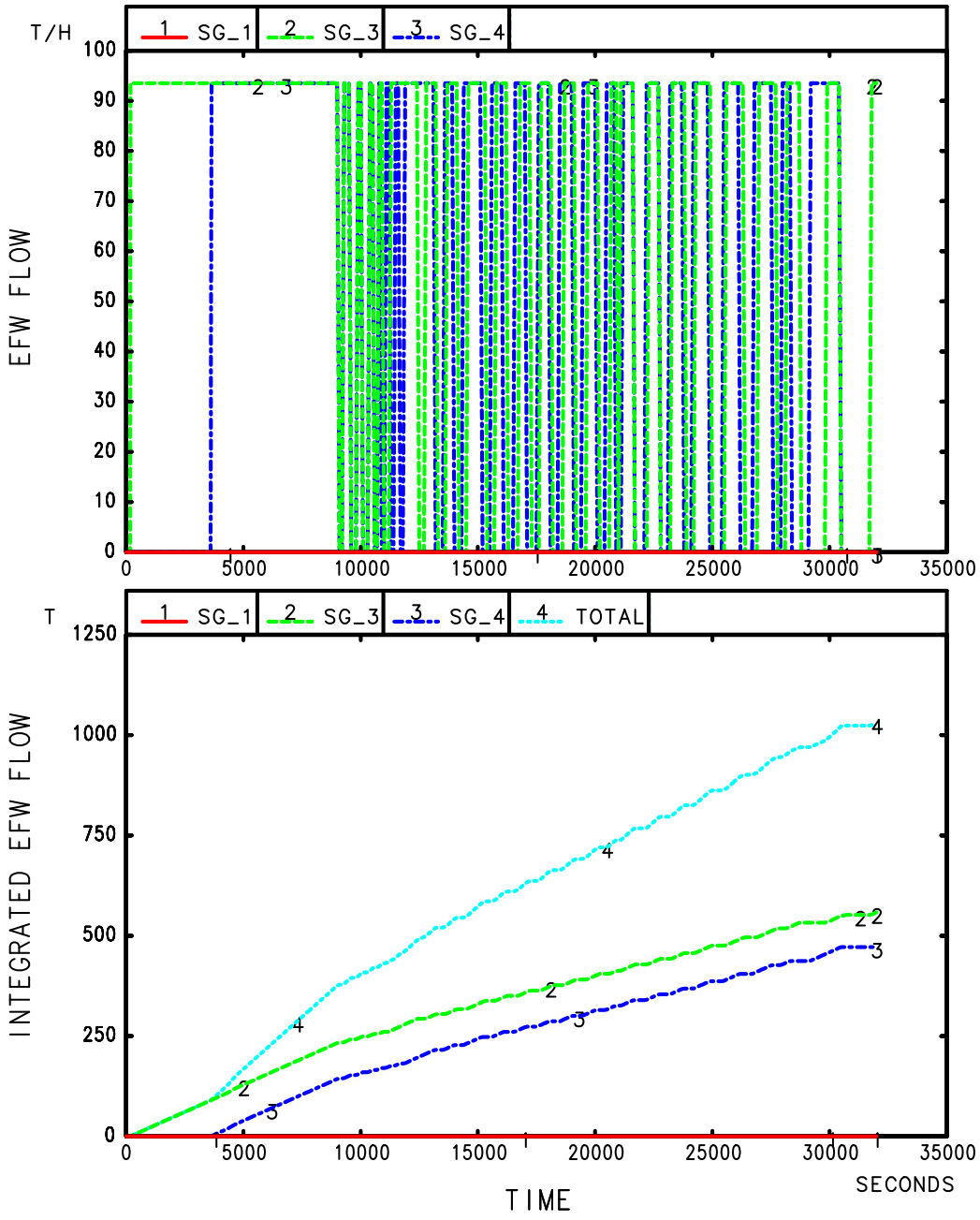
FWLB (state A, PCC-4)
Case 4 - Transfer to the safe shutdown state - with LOOP - RCP [RCS] boration and depressurisation

APPENDIX 14B.2.16.1 – FIGURE 21



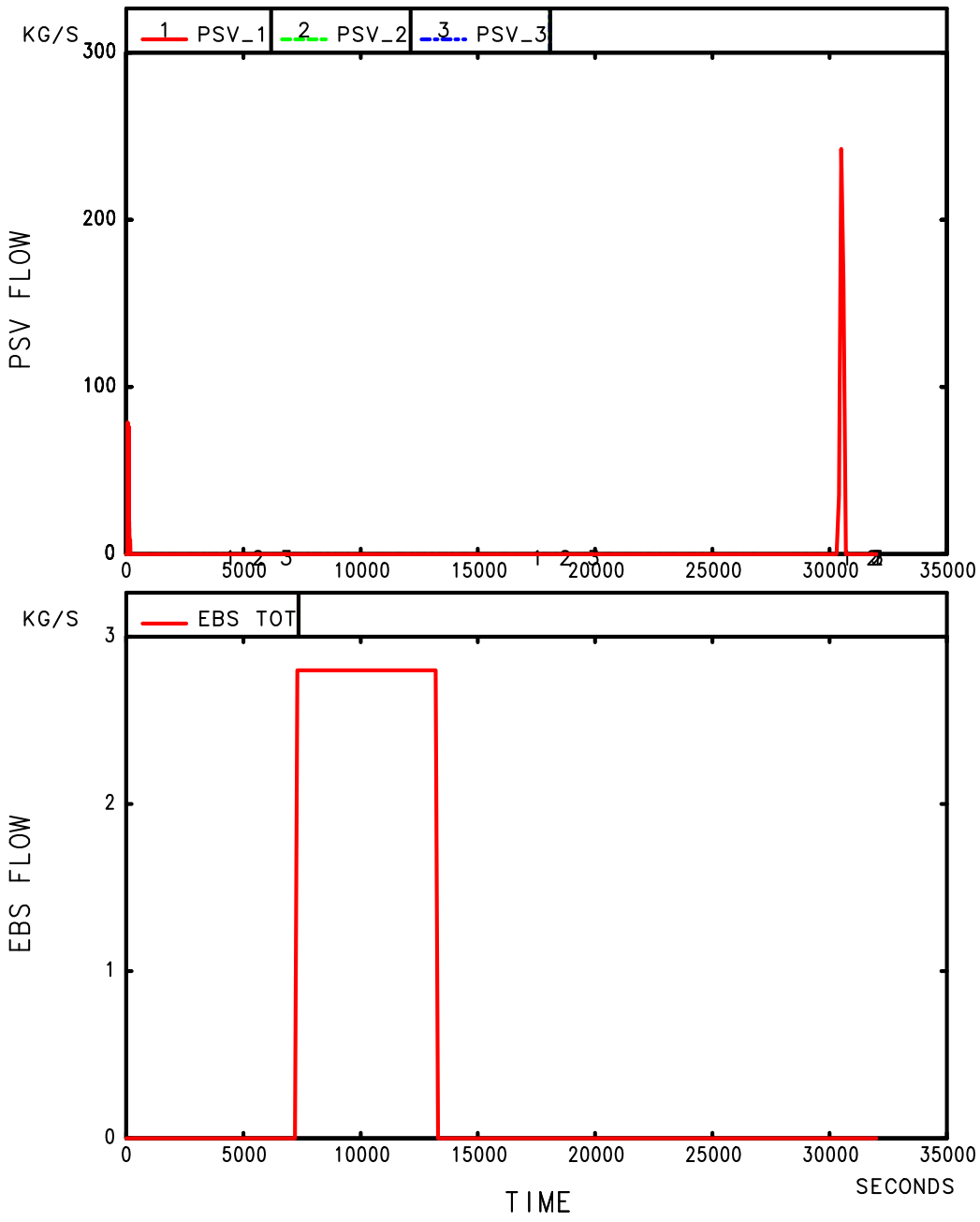
FWLB (state A, PCC-4)
Case 4 - Transfer to the safe shutdown state - with LOOP - RCP [RCS] boration and depressurisation

APPENDIX 14B.2.16.1 – FIGURE 22



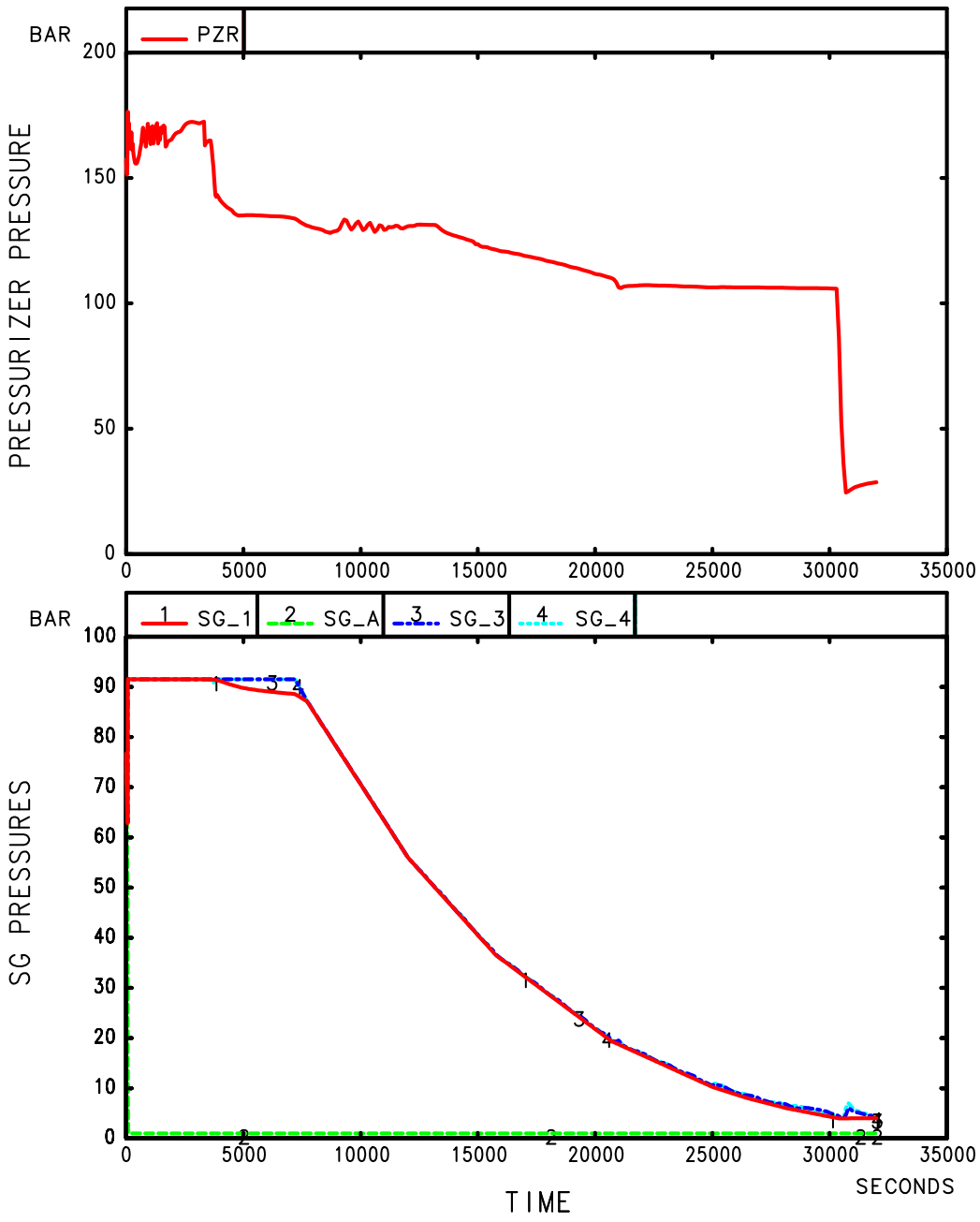
FWLB (state A, PCC-4)
Case 4 - Transfer to the safe shutdown state - with LOOP - RCP [RCS] boration and depressurisation

APPENDIX 14B.2.16.1 – FIGURE 23



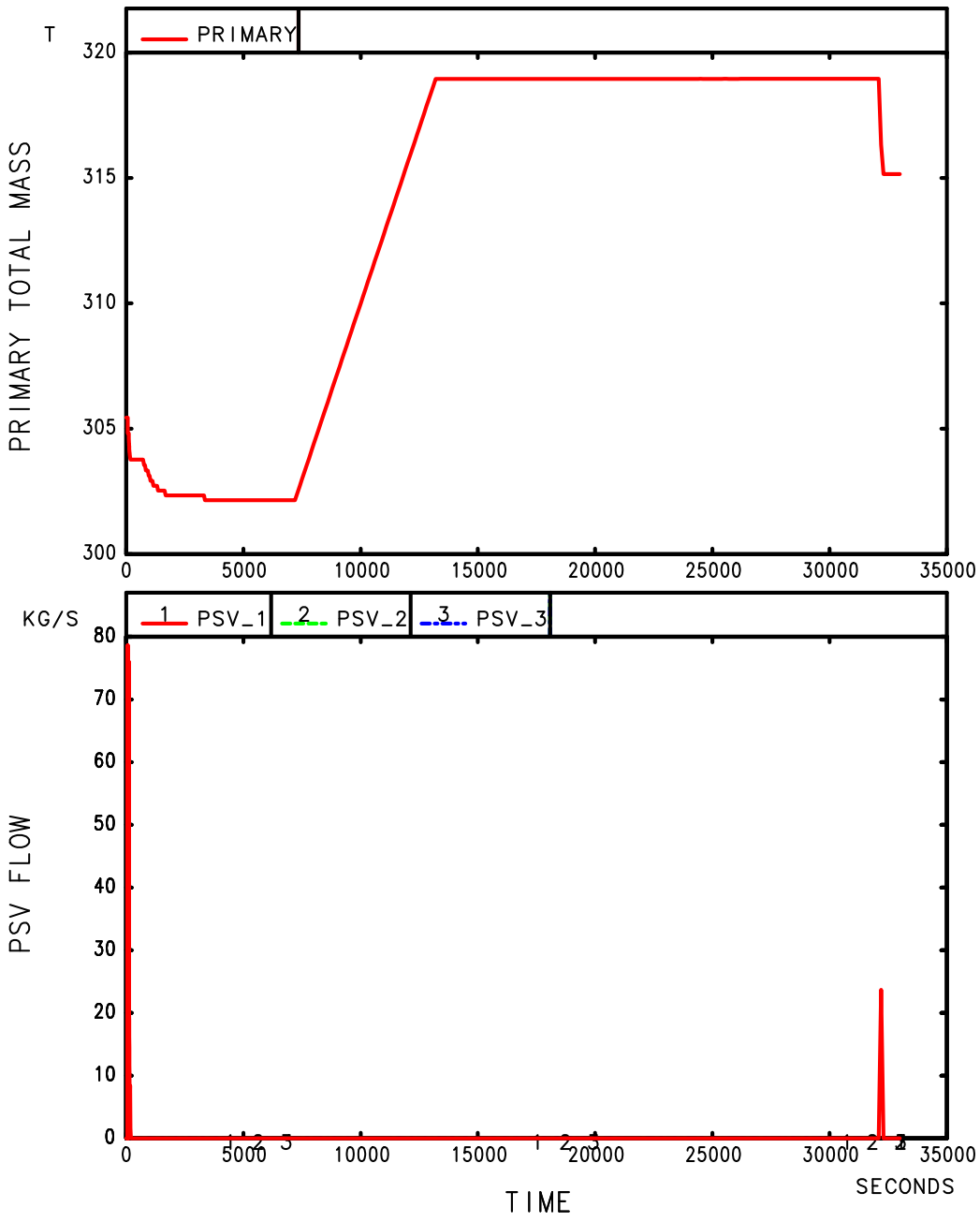
FWLB (state A, PCC-4)
Case 4 - Transfer to the safe shutdown state - with LOOP - RCP [RCS] boration and depressurisation

APPENDIX 14B.2.16.1 – FIGURE 24



FWLB (state A, PCC-4)
Case 4 - Transfer to the safe shutdown state - with LOOP - RCP [RCS] boration and depressurisation

APPENDIX 14B.2.16.1 – FIGURE 25



FWLB (state A, PCC-4)
Transfer to the safe shutdown state - with LOOP - RCP [RCS] boration and depressurisation
Realignment of ASG [EFWS] to an empty SG before PSV opening

2.17. STEAM GENERATOR TUBE RUPTURE (SINGLE TUBE IN STATE A, PCC-3)

This section deals with steam generator tube rupture of a single tube occurring from state A conditions as defined in section 1 of this appendix.

2.17.1. Identification of Causes and Accident Description

2.17.1.1. General Concern

The accident examined is either due to small continuous leaks or to the complete severance of one single steam generator tube (SGTR - 2A).

The main consequences of this accident are associated with the risks of contamination of the secondary side (mainly affected SG) inventory by leakage of radioactive coolant from the primary side. The primary coolant may be contaminated by corrosion and fission products associated with continuous operation with a limited amount of defective fuel rods. Subsequent to the tube rupture there is the possibly of discharge of activity, either in the steam or liquid phase, to the atmosphere via the MS safety and/or relief trains.

The descriptions of the course of the transient are subdivided into the short term and the long term phase to clearly separate the phases of activity release to the atmosphere. The short term phase is defined as the period up to the isolation of the affected SG, i.e. termination of the activity release. It includes the controlled state, which is defined as the SGTR leak flow rate matched by RCV [CVCS] or MHSI injection.

In the long term phase the plant is transferred to RHR conditions with a possible additional activity release if the affected SG is depressurised via the Main Steam (MS) relief valve or VIV [MSIV] bypass valve.

2.17.1.2. Typical Sequence of Event

The typical sequence of events in the case of a single tube SGTR, excluding the effects of additional events that could occur because of the single failure consideration, is as follows:

2.17.1.2.1. From the Initiating event to leak termination (Short Term)

The general course of the transient to controlled state is described in two ways:

a) as a best estimate approach, considering all systems and the availability of activity measurements, e.g. in the main steam line, which detect that a SG tube rupture has occurred automatically and invoke special SGTR countermeasures.

b) for the safety approach, without an early automatic identification of the tube rupture, and disregarding the special SGTR countermeasures initiated following the activity signal and only taking credit for F1 systems and functions.

The SG tube rupture leads to a loss of primary coolant, which is discharged to the SG affected by the rupture. The break results in a decrease of the primary pressure and contamination of the secondary side by the SGTR flow.

a) following a high activity signal which occurs approximately 15 seconds after the rupture occurs for a guillotine break at full power, a set of special SGTR countermeasures are actuated:

- reactor trip and turbine trip
- Pressuriser (PZR) spray with all relevant systems that are available to reduce the RCP [RCS] pressure level to just above the MS pressure level to reduce the SGTR flow and maintain a level of subcooling in the primary circuit
- shut off the pressuriser heaters to prevent a pressure increase by the primary pressure control system
- start up of both RCV [CVCS] makeup pumps and reduction of letdown flow to the minimum level, to counteract the SGTR flow

The activity monitors provide a rapid identification of even minor SG tube leaks significantly smaller than a guillotine break, and the countermeasures cause a significant reduction in the leak flow. This occurs because the RCP [RCS] pressure is rapidly reduced to match the secondary side pressure. In addition, the pressuriser water level is maintained close to the operating range for flows for breaks up to break sizes corresponding to the double-ended rupture of one SG tube (2A). This is because the RCV [CVCS] is able to compensate for this leak flows for this break size. Thus, that the controlled state is reached.

The affected steam generator can be identified by the operator and isolated on the steam side by closing the associated VIV [MSIV]. The secondary side heat removal is provided by the unaffected SG and the CGT [MSB]. If, for any reason, the CGT [MSB] is not available, the heat removal is performed by the VDA [MSRT] on the unaffected SG. The VDA [MSRT] of the affected SG does not operate as its setpoint is raised by the operator. Later in the transient the pressure in the affected SG will increase to match the primary pressure and the leak flow is terminated.

b) If the activity signal is not claimed the reactor trip occurs either following either a “pressuriser pressure < MIN2” or a “SG level > MAX1” signal from the affected SG.

As a consequence of the operation of SG level control the level increase caused by the SGTR leak flow is compensated for by a minor reduction of feedwater flow and the SG level does not increase to the MAX1 reactor trip setpoint.

If the RCV [CVCS] works properly the loss of primary coolant via the leak can be matched by the RCV [CVCS] makeup for a break size up to nearly that for a double guillotine break without reaching the “MIN2 pressuriser pressure” setpoint. In practice for a 2A break this trip setpoint would not be reached for about 35 minutes. In this case, the controlled state is reached just after the generation of the of reactor trip signal when the RCV [CVCS] is able to compensate for the guillotine break flow.

If the RCV [CVCS] does not work, e.g. because it is not a F1 qualified system, the SI signal is automatically actuated on “MIN3 pressuriser pressure”. Following the generation of the SI signal the partial cooldown is initiated in all four SG. In this case, the controlled state is reached during the partial cooldown, when the MHSI flow matches the SGTR leak flow.

The reactor trip signal automatically trips the turbine and with the assumption of unavailability of the CGT [MSB], the steam generator pressure rapidly increases resulting in steam discharge to the atmosphere through the VDA [MSRT]. The unavailability of the CGT [MSB] could arise as it is not an F1 qualified system or it is lost following a LOOP assumed to occur following turbine trip.

The continuous loss of RCP [RCS] coolant inventory via the SGTR and the possible plant cooldown after reactor trip for a case initiated at the full power state, leads to the depressurise of the primary side.

Either on emergency core cooling signal or on SG level > MAX 2 in the affected SG a partial cooldown is initiated. This will occur unless the partial cooldown has already been initiated by the safety injection signal. The partial cooldown is performed either by the CGT [MSB], if available, from 87.1 bar down to 55 bar, or by the VDA [MSRT] from 93 bar down to 60 bar. In either case the cooldown is implemented at a rate of 100°C/h and performed using all SG, including the faulted one. If the RCV [CVCS] works correctly, the loss of primary coolant via the break is not sufficient to depressurise the RCP [RCS] to the SI signal setpoint for a considerable period of time. In that case a partial cooldown could be manually initiated by the operator on a F1B activity signal, after a 30 minute operator action time, thus allowing MHSI.

At the end of the partial cooldown, the primary pressure is held at the MHSI shut off head and a contaminated SGTR flow still enters the affected SG. This leads to an increase in its level. At this stage, the controlled state (SI and SGTR mass flow balance) has already been reached.

Using the "SG level > MAX 2" signal, occurring after the end of the partial cooldown, the affected SG is identified and automatically isolated. This is performed by closure of the associated VIV [MSIV] and by increasing the MSRV pressure setpoint to above the MHSI delivery pressure and below the MSSV pressure setpoint.

Where there is a manual actuation of partial cooldown, the operator could manually isolate the affected SG, after the end of the partial cooldown, without waiting for the "SG level > MAX2", using a F1B activity signal.

Following the isolation, the pressures between the primary side and affected SG are equalised and the leak is finally terminated. This is the end of the short term phase.

2.17.1.2.2. From Leak termination to the Safe Shutdown state (Long Term)

The safe shutdown state is defined as a state where the LHSI is connected in RHR mode and the affected SG is isolated.

- 1/4 LHSI trains in RHR mode is sufficient to provide the heat removal. The connection conditions are "RCP [RCS] pressure < 30 bar" and "hot leg temperature < 180°C" with subcooled conditions. This requires an RCP [RCS] pressure > 10 bar, compared to, for example ≈ 20 bar provided by the LHSI in injection mode.

The sequence of actions to be performed by the operator to reach the safe shutdown state can be divided into 2 successive phases.

Boration and Cooldown

The boration is performed using the RBS [EBS] as the RCV [CVCS] not claimed, as it is not a F1 classified system. To compensate for the RBS [EBS] injection the plant is cooled down simultaneously by the unaffected SG. This avoids any primary pressure increase caused by the high pressure RBS [EBS] pumps. The RBS [EBS] is sized to match the volume contraction of a 50°C/h cooldown with both RBS [EBS] pumps in operation. If one pump failure is assumed as a single failure, the operator reduces the cooldown rate to 25°C/h. As an additional measure to maintain the pressure balance between primary side and the affected SG, the MHSI is kept operating.

Depressurisation of the RCP [RCS] and Affected SG

The primary side and faulted SG depressurisation is performed by opening the MSRV on the affected SG, following isolation of the MHSI, Accumulators and the RBS [EBS]. If the MSRV fails to operate, the depressurisation can be done using the bypass valve of the VIV [MSIV], classified F1B, without causing a leak back flow.

If the water level in the affected SG is too high, some inventory reduction can be performed by opening the blowdown line between the affected SG and an unaffected one, the blowdown line is F1B classified, before depressurising. This removes the risk of water entrainment when the SG start to reboil).

2.17.1.2.3. Radiological Releases

Before reactor trip contaminated steam goes through the turbine 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 the continuous activity control system and regular measurements.

Following reactor trip, if the condenser is not available, the main steam bypass valves remain closed. The resultant pressure rise leads to the opening of the VDA [MSRT]. Steam is released to the atmosphere and in the long term depressurisation of the affected SG an additional discharge may occur.

2.17.1.2.4. Precautions Limiting the Event Occurrence

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

- the SG tube material is a highly ductile material (either Inconel 690 or Incoloy 800),
- the blowdown system located at the bottom part of SG tube bundle is designed to prevent the accumulation of solid deposits on the tube plate,
- the secondary water is chemically conditioned. Its chemical properties are frequent monitored, which protects the SG tubes from corrosion,
- on the mechanical side, the steam generators are designed to prevent any object coming from the main feedwater system impacting one or several tubes. Following the double-ended guillotine rupture of SG tubes, whipping phenomena and the rupture of neighbouring tubes are limited by the presence of 9 SG tube support plates,
- SG tube support plates supporting the tube bundle are designed to prevent from general phenomena of corrosion tube-corrosion of type denting. This is implemented by controlling the type of material and geometry of holes,
- activity control of the secondary side SG water (SG blowdown) allows a continuous check of compliance with the limits established in the technical specification for minor leaks between the primary and secondary. This limit is significantly below the flow through one broken tube,

- activity in the secondary side SG steam is subject to continuous monitoring in the main steam lines using dedicated activity monitors. An accumulation of minor leaks which exceed the limits established in the technical specification is not permitted during unit operation,

- a balance between RCV [CVCS] letdown and charging flow is continuously performed to identify the leakage between the primary and secondary circuits. The maximum allowable leakage is defined as 3l/hour/SG during normal operation.

2.17.2. Safety Criteria

The safety criteria to be met are the dose equivalent limits following a release to the atmosphere, as discussed in section 1 of this appendix.

To meet these criteria the following decoupling criteria are used:

- no core damage (fuel cladding integrity).

- no MSSV actuation, to prevent any risk of MSSV failure in the open position.

- attain RHR conditions, via boration, depressurisation and heat removal, to reach safe shutdown conditions, using F1 means only. This must assume the bounding set of safety analysis rules, defined in section 1 of this appendix.

The prevention of core damage by keep the core coolable and preventing higher activity concentrations in the primary side, is covered by the LOCA studies of section 2.2 of this appendix.

The SGTR mitigation procedure, involving automatic and manual actions, is defined to fulfil the following two objectives

- prevent SG overfilling, to avoid increased activity release by liquid blowdown to the atmosphere.

- minimise SGTR leak back flow, to avoid problems with a plug of low borated water in the primary circuit. This is relevant to cases with no Reactor Coolant Pumps operating, e.g. following LOOP.

2.17.3. Methods and Assumptions

2.17.3.1. Methods of Analysis

The CATHARE code is used for case 1 (see section 2.17.4) - without LOOP.

The S-RELAP5 code coupled with the I&C routines of the NLOOP code is used for cases 2/3 (see section 2.17.4) - with LOOP.

Important phenomena and the qualification of the models used in S-RELAP5

The family of transients is SGTR

Phenomena

Primary Side

Small break LOCA behaviour as primary pressure and PRZ level decrease until halted by operation of the MHSI. The collapsed level is not in practice lowered to the upper plenum i.e. a steam bubble may be formed in the RPV head only during the emergency power mode. A SG isolated on the steam side during the emergency power mode and secondary side cooldown with the intact SG can lead to a very low circulation rate in the affected SG.

Secondary side

The MS pressure increase following the closure of the turbine/CGT [MSB] valves. Fast opening of the MS-relief train and reclosing to maintain a constant MS-pressure at zero load.

Filling of the affected SG after reactor trip to above the top edge of the separators. Complete flooding of the affected SG may occur. Heat transfer and circulation in a filled SG.

Qualification of the models in S-RELAP5 for primary and secondary side phenomena

Generally for S-RELAP5 a wide range of validation examples are provided. Integral or separate effects tests have been performed at different test facilities for the phenomena apparent in an SGTR. These tests are applicable for the verification of all primary side phenomena. Using results from the PKL and THTF test facilities a wide range of validation for smaller leaks is performed. This includes the thermal-hydraulic phenomena in transients including the phenomena on the secondary side. Additionally an SGTR incident which occurred at the Doel 2 PWR was analysed successfully.

In all cases the transient analyses are performed according to the conservative PCC-analysis rules defined in section 1 of this appendix.

2.17.3.2. Main Assumptions

2.17.3.2.1. Accident Definition

The cases studied here correspond to the double ended guillotine rupture of a single steam generator tube. This assumes unrestricted blowdown from both ends of the severed tube.

For accident analyses, a loss of offsite power (LOOP) is included in the accident, if this pessimises the transient.

2.17.3.2.2. Protection and Mitigation Actions

In a SGTR event, automatic protection systems and operator actions aim at tripping the reactor, removing the residual heat, terminating the primary to secondary leak flow, limiting the mass of contaminated SG water released to the atmosphere, and finally taking the reactor to a safe shutdown state. These actions are all performed using F1 classified systems.

The different automatic protections and alarms which could occur in the SGTR event mitigation process are linked either to the reactor coolant depressurisation, to the level increase in the affected SG or to the activity increase on the secondary side.

At least one of the following reactor trip signals is reached:

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

The other automatic protection signals are as follows:

- normally, on reactor trip, the full load Main Feedwater (ARE [MFWS]) trains are closed in all loops
- following a “SG level > MAX1” the low load MFW train in the affected loop is isolated.
- following a “pressuriser pressure < MIN3” signal the RIS [SIS] is actuated and a partial cooldown using all SG (including the affected one) is initiated
- following a “SG level > MAX2*” signal, a partial cooldown using all SG (including the affected one) is initiated if partial cooldown has not already been actuated
- following a “SG level > MAX2*” signal occurring after the end of the partial cooldown, the affected SG is isolated. This is performed by closure of the associated VIV [MSIV] and by lifting the MSRV pressure setpoint to above the MHSI shut off head but below the MSSV pressure setpoint.

For long term mitigation, operator actions are aimed at reaching the safe shutdown state. In accordance with the procedures, the operator performs boration and cooldown in parallel, to depressurise both the RCP [RCS] and the affected SG. This then allows the connection of the LHSI in RHR-mode.

2.17.3.2.3. Operator Actions

No operator action is considered before 30 minutes after the Reactor Trip (RT), except if it is more onerous for the transient. The operator is not instructed to isolate the affected SG before the end of the partial cooldown. This minimises the likelihood of SG overfilling allows safety injection and minimises the potential for leak back flow.

2.17.4. Definition of Studied Cases

2.17.4.1. Short Term Cases

The short term phase, as assessed in the SGTR study, is the time period between the SGTR initiation and the leak termination.

This phase includes the controlled state, corresponding to the state where the SI flow rate, or the RCV [CVCS] flow rate if the RCV [CVCS] works correctly, matches the SGTR flow rate.

During this phase the aim of the study is to quantify the maximum amount of fluid from the affected SG released to the atmosphere. This is done in two steps:

- to quantify the maximum amount of activity released to the atmosphere it is conservative to maximise the power to remove via the SG. In this case before isolation of the SG there is only steam release. Starting at 102% full power results in maximum decay heat and no LOOP means that Reactor Coolant Pump heat has to be removed in addition to the decay heat),

- to verify that no SG overfilling occurs, and thus no liquid release to the atmosphere before termination of the leak a minimum heat removal assumption is conservative. Thus, the transient is started at 2% reactor power with LOOP assumed to occur

Two cases are thus analysed, for each of these two objectives, with different assumptions (see in addition Appendix 14B.2.17 - Table 5):

Case 1: Without LOOP

To derive the maximum steam released from the affected SG, it is appropriate to maximise the heat removed via the SG.

The most onerous case is thus:

- with a maximum initial power (i.e. 102% FP),

- without LOOP.

It is assumed that THE Reactor Coolant Pumps are running throughout the transient.

The other specific assumptions related to this case are described in section 2.17.5.1 of this appendix.

Case 2: With LOOP

To derive the maximum inventory in the affected SG, it is appropriate to maximise the initial water inventory and to minimise the steam released. This is attained by using a low initial power level at which the SG mass is higher and to minimise the heat to be removed via the SG).

The most onerous case is thus:

- with a low initial power (i.e. 2% FP),

- with an assumption of LOOP when the turbine trip signal is generated.

The other specific assumptions related to this case are described in section 2.17.5.2 of this appendix.

2.17.4.2. Long Term Case

The long term phase is the period between the leak termination and reaching the safe shutdown state, i.e. reaching RHR conditions. It includes the phases of boration and cooldown of the RCP [RCS] in parallel using the unaffected SG and the final depressurisation of the RCP [RCS] and affected SG. Only one case is studied for this phase. (See in addition Appendix 14B.2.17 – Table 5).

Case 3: With LOOP

This case, with of the assumption of LOOP, is performed to demonstrate the capacity of the F1 systems to sufficiently borate the affected loop and to depressurise the RCP [RCS] and the affected SG without significant leak backflow, while the Reactor Coolant Pumps are not operating.

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

Comment: A long term case with a maximum initial power and without LOOP would be of interest to demonstrate the capacity of F1 systems, especially of the ASG [EFWS] tank, to cool down the plant to RHRS conditions, with a maximum power to remove. However no analysis is presented here since as one is provided in section 2.18.6 of this appendix covering an SGTR of two tubes.

2.17.5. Short Term Study

2.17.5.1. Case 1: Without LOOP

2.17.5.1.1. Choice of Single Failure and Maintenance

Each single failure or maintenance which could be assumed would lead to the earlier generation of a reactor trip signal. Thus assuming neither a single failure nor maintenance, with the RCV [CVCS] working properly, is the most onerous case. This leads to maximum steam release during the long time period from 30 minutes to the end of the partial cooldown. This is conservative for the radiological consequences due to the impact of the iodine spiking effect.

In practice, the SG level MIN2 setpoint is not reached in the affected SG during this transient phase. Thus, assuming the loss of one Emergency Feedwater (EFW) pump connected to the affected SG would have no effect.

2.17.5.1.2. Initial conditions

The initial conditions, given in Appendix 14B.2.17 - Table 1 are chosen to maximise the power to remove and to maximise the pressure difference between the RCP [RCS] and the SG.

2.17.5.1.3. Specific Assumptions

a) Neutronic data

Core power is assumed constant at 102% of full power until the time of reactor trip. Following RT, the maximum residual heat curve as described in section 0.2.4 of this appendix is considered for terms B+C (fission products).

Term A (fission heat) is input to the CATHARE code, with the CATHARE point-kinetic model turned off. This term A is derived from a conservative RT simulation, see Appendix 14B.2.2 – Table 1.

b) Assumptions related to control systems

Turbine: Turbine control is assumed to maintain turbine flow at 102% until turbine trip is initiated on reactor trip

CGT [MSB]: Not considered.

ARE [MFWS]: SG level control is in operation when the accident occurs and operates correctly until isolated on turbine trip. By the assumption of an early reactor trip actuation a “SG level > MAX1” signal is avoided. This maximises the activity transferred to the affected SG.

RCV [CVCS]: pressuriser level control is assumed to operate correctly to maximise the pressure difference between the RCP [RCS] and the affected SG. The charging flow rate is maximised: 2 charging pumps are in operation and letdown is isolated following a “pressuriser level < 2.09 m” signal. RCV [CVCS] charging is isolated manually by the operator at the end of the partial cooldown.

Heaters/spray: Pressuriser pressure control is assumed to work to maximise the pressure difference between the RCP [RCS] and the affected SG. The heater power is maximised and all heaters are actuated while any spray flow rate is ignored. The heaters are shut-off following emptying of the pressuriser.

c) Assumptions related to F1 systems

Reactor trip (F1A) This occurs following a “pressuriser pressure < MIN2 (135 – 1.5 bar)” signal with a maximum delay, to maximise the activity content of the affected SG at the time steam is released to atmosphere, i.e. at the time of MSRV actuation.

MSRV (F1A): The MSRV setpoint on the affected SG is minimised (93 -1.5 bar) to maximise the pressure difference between RCP [RCS] and affected SG. The MSRV setpoints of the unaffected SG are maximised (93 + 1.5 bar). After RT, a constant pressure difference occurs between the RCP [RCS] and the affected SG. The RCV [CVCS] flow matches the SGTR flow, preventing the “pressuriser pressure MIN 3” setpoint being reached. Partial cooldown is thus manually actuated by the operator 30 minutes after RT. After initiation of partial cooldown, the MSRV setpoints are automatically decreased to 60 - 1.5 bar in the affected SG, and to 60 +1.5 bar in the unaffected SG. The operator manually increases the setpoint for the MSRV on the affected SG to 98 bar at the end of partial cooldown.

VIV [MSIV] (F1A): The VIV [MSIV] of the affected SG is manually closed by the operator at the end of partial cooldown.

MHSI (F1A): The Safety injection signal is actuated on a “pressuriser pressure < MIN3 (115 + 1.5 bar)” signal following manual actuation of partial cooldown by the operator 30 minutes after RT. The MHSI injects with a maximum flow rate. This maximises the pressure difference between the RCP [RCS] and the affected SG. This is performed once the RCP [RCS] pressure goes below 92 bar.

ASG [EFWS] (F1A): the ASG [EFWS] is actuated to all SG following a "SG level < MIN2 (8 m) - 2% WR" with a minimum flow rate. The "Low level MIN2" setpoint is not reached in affected SG during the short term.

MSSV (F1A): To confirm the non actuation of the MSSV during the transient, their setpoint is minimised (102.5 - 1.5 bar).

2.17.5.2. Case 2: With LOOP

2.17.5.2.1. Choice of Single Failure and Maintenance

Assuming neither a single failure nor maintenance is the most onerous case. Assuming a loss of a F1 system leads to reduced filling of the affected SG which parameter of interest in this case.

2.17.5.2.2. Initial Conditions

The initial conditions, given in Appendix 14B.2.17 - Table 2 are chosen to minimise the power to be removed, to maximise the pressure difference between the RCP [RCS] and the SG and to maximise the water inventory in the SG.

2.17.5.2.3. Specific Assumptions

a) Neutronic data

The initial core power is assumed to 2% of full power, decreasing according to the best estimate residual heat curve.

b) Assumptions related to control systems

Turbine: The turbine is assumed to be not operating due to initial hot standby conditions.

CGT [MSB]: It is assumed that the CGT [MSB] is lost at the start of the transient to reduce the steam release from the affected SG.

ARE [MFWS]: No main feedwater supply by either the ARE [MFWS] or the AAD [SSS] is assumed at the start of the transient. This increases the amount of subcooled ASG [EFWS] injection and thus the water inventory.

RCV [CVCS]: Pressuriser level control is assumed to work correctly. This maximises the pressure difference between the RCP [RCS] and the affected SG. If the pressuriser level decreases below the "pressuriser level < 2.09 m" setpoint, the second charging pump is actuated and letdown is isolated. The RCV [CVCS] charging is assumed to be isolated manually by the operator when starting the cooldown/boration phase. This is because boration via the RCV [CVCS] is assumed to be unavailable.

Heaters/spray: Pressuriser pressure control is in operation. After emergency power mode occurs 300 KW of emergency power supplied heaters are assumed. The heaters are shut off when the pressuriser empties.

c) Assumptions related to F1 systems

Reactor trip (F1A): This occurs on a “MS pressure > MAX1 (93 - 1.5 bar)” signal with a minimum delay to reduce the time to emergency power mode which is assumed to occur with reactor trip / turbine trip signal.

MSRV (F1A): MSRV setpoint on the affected SG is minimised (93 -1.5 bar) to maximise the pressure difference between the RCP [RCS] and the affected SG. A calculation with an increased actuation pressure for the affected loop and a reduced actuation pressure for the other loops to reduce steam removal from the affected SG shows the opposite effect. In this case, performed to assess the maximum SG inventory, only slightly lower maximum water content occurs in the SG. After reaching a water level of 18.1 + 0.14m in the affected SG the partial cooldown with 100°C/h is performed automatically to 60 bar. When partial cooldown is finished the set value for the relief valve of the affected loop is manually set to 98 bar.

VIV [MSIV] (F1A): The VIV [MSIV] of the affected SG is manually closed by the operator at the end of the partial cooldown.

MHSI (F1A): The safety injection signal is actuated following a “pressuriser pressure < MIN3 (115 + 1.5 bar)” signal. The MHSI injects with a maximum flow rate to maximise the pressure difference between the RCP [RCS] and affected SG once the RCP [RCS] pressure falls below 92 bar.

ASG [EFWS] (F1A): This is actuated manually, a conservative assumption in this case, immediately the emergency power mode occurs. The ASG [EFWS] control is not considered leading to maximum emergency feedwater injection and an isolation of the system at a SG water level of 17.2 + 0.38 m.

MSSV (F1A): To check the non actuation of the MSSV during the transient, their setpoint is minimised (102.5 – 1.5 bar).

2.17.5.3. Results

The sequence of events for case 1 is given in Appendix 14B.2.17 - Table 3.

The sequence of events for case 2 is given in Appendix 14B.2.17 - Table 4. (Case 2 - short term - ends at leak termination – 6600 seconds.)

The figures showing of the main thermal-hydraulic parameters evolutions are:

- Case 1
  - Pressuriser and SG pressures,
  - RCP [RCS] inlet and outlet flow rates,
  - Cold/hot leg temperatures,
  - Affected SG level.

- Case 2
    - Pressuriser and SG pressures,
    - Cold/hot leg temperatures,
    - RCP [RCS] inlet and outlet flow rates,
    - SG water level
    - Cold/hot leg boron concentration

Appendix 14B.2.17 - Figures 1 to 4 are related to case 1.

Appendix 14B.2.17 - Figures 5 to 8 are related to case 2. (Case 2 - short term - ends at leak termination - 6600 seconds).

The results of case 1 show that a maximum of 93 tons of steam is discharged from the affected SG to atmosphere during the short term phase. This assumes the use of the F1B activity signal for the operator to initiates partial cooldown 30 minutes after RT, and a manual isolation of the affected SG at the end of the partial cooldown. This is performed before reaching “level MAX2” in the affected SG).

In addition, case 1 shows that, even for the fault initiated at power, the MSSV setpoints are not reached.

In case 1, claiming operation of the non-classified RCV [CVCS], the controlled state is reached when the RCV [CVCS] flow matches the SGTR flow. This occurs just after reactor trip. However, if operation of RCV [CVCS] injection is not assumed, RCP [RCS] depressurisation occurs until the “pressuriser pressure MIN3” SI injection and partial cooldown signal setpoint is reached. Thus the controlled state is reached using only F1A systems, RT, partial cooldown via VDA [MSRT] and MHSI.

The results of case 2 show that an overfeeding of the affected SG is prevented by automatic F1A actions and systems in the short term phase. However, beyond 30 minutes the RCV [CVCS] pumps must be shut out in order to exclude any risk of increased radiological releases by water entrainment in the relief valve flow.

The initial conditions of case 2 are chosen to minimise the steam release from the affected SG and maximise the SG inventory. Thus the radiological release is of 34.8 tons of steam discharge, being much lower than for case 1.

2.17.6. Long Term Study

2.17.6.1. Case 3 - With LOOP

2.17.6.1.1. Choice of Single Failure and Maintenance

The single failure is chosen to pessimise the capacity of F1 systems to cooldown the RCP [RCS] and the SG to RHR conditions. This failure can be either on:

- one RBS [EBS] pump to pessimise boration
- one MSRV to pessimise cooldown and SG depressurisation
- one ASG [EFWS] pump to pessimise cooldown
- the valves in one SG blowdown line which could prevent emptying of the affected SG into an unaffected SG before final depressurisation, if necessary

For assessing the radiological consequences during the long term phase, no failure on the secondary side, either feedwater or steam release, could impact the RCP [RCS] and affected SG depressurisation.

The redundant means to depressurise the affected SG if the MSRV fails is to open the VIV [MSIV] bypass (F1 classified) of the affected SG and one or more unaffected SG. This depressurises the affected SG via a MSRV of an unaffected SG and results in reduced activity release.

The connection of the steam generators via the blowdown lines is single failure proof assuming no maintenance.

Therefore the loss of one RBS [EBS] pump is chosen as the single failure to pessimise the capacity of the F1 systems to borate the primary side and to prolong the boration and cooldown phase. This is a 25°C/h cooldown is necessary if only one RBS [EBS] pump is available.

There is no maintenance case identified to additionally penalise cooldown, boration or activity release for the transient.

2.17.6.1.2. Conditions at Beginning of Long Term Phase

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

2.17.6.1.3. Specific Assumptions

The assumptions for the systems described in section 2.17.5.2.3 (case 2) remain valid. The additional assumptions and characteristics for the long term phase are as follows:

MHSI (F1A): Maintained until the end of RCP [RCS] cooldown phase (once RCP [RCS] temperatures in unaffected loops $\leq$ 180°C) and then manually shut down by the operator. This allows the depressurisation of the RCP [RCS] and the affected SG or inventory reduction of SGa via blowdown system to be initiated.

Accumulators (F1A): Are isolated manually with MHSI shut down (no accumulator injection during the transient).

LHSI (F1A): Injects in the RCP [RCS] once RCP [RCS] pressure reaches 20 bar.

RBS [EBS] (F1B): 1/2 RBS [EBS] pumps is started by the operator at 2500s when starting plant cooldown for boration and is stopped once cooldown is finished. Or if a connection from the affected SG to another one via blowdown line is opened to reduce the inventory in the affected SG. The guaranteed content of the RBS [EBS] tank is sufficient to borate the RCP [RCS] from 10 ppm initial concentration to 620 ppm, assuming natural boron concentration. This value of 620 ppm is required for RHR connection conditions using EOC MOX assumptions.

VDA [MSRT] (F1A): For the RCP [RCS] cooldown phase the VDA [MSRT] setpoint of the unaffected SG is reduced corresponding to a 25°C/h cooldown rate. This is required if only one RBS [EBS] pump is available. With the VDA [MSRT] capacity of 50% of nominal flow rate at 97 bar this gradient can be maintained to a secondary pressure of approximately 3.5 bar. Subsequently the VDA [MSRT] are fully open and the cooldown gradient decreases.

To depressurise the RCP [RCS] and the affected SG at the end of cooldown phase, after having reduced the inventory of the affected SG via blowdown, the VDA [MSRT] of the affected SG is opened by 20%.

ASG [EFWS] (F1A): In the long term phase the ASG [EFWS] of the unaffected SG is manually controlled by the operator to a 15m wide range level measurement. In preparation for the level reduction of the affected SG by connecting it via the blowdown line with another SG, the water level of SG4 is lowered to about 8m by manual operator action.

Blowdown line (F1B): To lower the water level in the affected SG, before the final depressurisation, a blowdown line connection from the affected SG (SG1) to another one (SG4) is manually opened by the operator. The capacity of this line is assumed to be 7 Kg/s (at a pressure difference from 93 bar to 4 bar, assuming a temperature of 281.5°C water temperature in the affected SG).

2.17.6.2. Results

Boration: The boration target at EOC, for a MOX core, the most onerous case, is to reach a boron concentration of 620 ppm in the primary coolant. This ensures subcriticality on reaching safe shutdown conditions. This concentration is reached after 81 minutes (approximately 40 minutes after the start of the RBS [EBS]). The RBS [EBS] pump which injects into the affected loop and one other pump are assumed to have failed and be unavailable. The flow rate in this loop stagnates after SG isolation, as the affected loop is borated via coolant mixing in the RPV only. Therefore, this loop only reaches the target boron concentration after 4 hours of the boration/cooldown phase. Therefore the total duration of the boration/cooldown phase is five hours. By this time the unaffected loops have reached a born concentration close to 2000 ppm.

Radiological releases: Before depressurisation of the RCP [RCS] and the affected SG at the end of the plant cooldown/boration phase, the water level in the affected SG is lowered by opening the blowdown line to another SG. This prevents water being discharged from the affected SG to atmosphere during the depressurisation via the MSRV. In the calculation the depressurisation is not stopped at < 30 bar to initiate the RHR as these results in a conservative calculation of the total steam discharge. The affected SG reaches 5.8 bar at the end of calculation. In this phase a total of 56.6 tons of steam is released to the atmosphere. If, for example, the depressurisation is halted at a SG pressure 20 bar, sufficient for actuation of the RHR, only 41.2 tons of contaminated steam would be discharged through the VDA [MSRT] of the affected SG.

SG overfilling: Approximately 20 minutes after the SGTR occurs, the narrow range level measurement of the affected SG reaches the MAX2 setpoint (18.1 + 0.34 m) and a partial cooldown is initiated. During the partial cooldown the water inventory of the SG decreases. Once the partial cooldown has been completed, when the operator isolates the affected SG after a 30 minute delay, the level is has fallen to 17.5 m. Once the steam side of the SG has been isolated, the SG pressure increases slowly due to the SGTR leakage until pressure equilibrium with the primary side is reached. During this phase the level in the affected increases to 20.8 m. Subsequently, small condensation effects in the SG lead to a further steady level increase. By 5.6 hours, the steam volume has fallen to 5 m^3 at which point the cooldown/boration phase is finished. The operator is assumed to open a blowdown line connection to SG4 at this time to reduce the water content of the affected SG. Within 2.8 hours (10000s) the water mass of the affected SG has been lowered from 180 tons to approximately 140 tons via the blowdown line. This reduces the collapsed water level to approximately 17.5 m. The final depressurisation of the affected SG and the RCP [RCS] is then performed via the relief valve of the affected loop without the risk of SG overfilling. Overfilling would lead to water entrainment in the flow to the atmosphere higher than the residual steam moisture of 0.25%.

2.17.7. Conclusion

The short term study shows that the controlled state, where SI and SGTR flow balance, can be reached with the use of F1A systems only. This target is reached independent of any failure assumptions (SF, maintenance).

The maximum amount of contaminated fluid released to the atmosphere is given by:

- case 1 - without LOOP for the short term: 93 tons of steam are released from the affected SG to the atmosphere,
- case 3 - with LOOP for the long term: 56.6 tons (41.2 tons if the depressurisation is halted at 20 bar) of steam are released from the affected SG to the atmosphere

During the transient no liquid is released to the atmosphere. Thus for calculation of the activity released only a residual steam moisture of 0.25% has been assumed.

APPENDIX 14B.2.17 – TABLE 1

Initial Conditions - Cases 1
(Short Term - Without LOOP)

Parameters	**Limiting values used**
Reactor coolant system	
• Initial reactor power (% of nominal power)	100 + 2 = 102
• Initial average RCP [RCS] temperature (°C)	311.25 – 2.5 = 308.75
• Initial reactor coolant pressure (bar)	155 + 2.5 = 157.5
• Reactor cooling flow (kg/s)	22240 (thermal-hydraulic)
• Pressuriser water volume / level (m^3 / m)	43.4 /7.41 (nominal + 5% MR)
Steam generators	
• Initial steam pressure (bar)	70.1 bar
• Initial SG level (m)	16.54 (nominal + 5% NR)
Feedwater	
• Main feedwater flow (% of nominal flow)	100 + 2 = 102
• Initial ARE [MFWS] temperature (°C)	232

APPENDIX 14B.2.17 – TABLE 2

Initial Conditions - Cases 2/3
(Short & Long Term - With LOOP)

Parameters	**Limiting values used**	
Reactor power	2%	decreasing in line with decay heat curve
Thermal reactor power	98 MW	
Reactor Coolant Pump thermal power	31.6 MW	
Total reactor coolant flow	22102 kg/s	Thermal design (22240 kg/s) recalculated from 292.5°C/155 bar to 299°C/157.5 bar
Total core bypass	5.5%	
Coolant mixing in RPV	100%	basic option in S-RELAP
Average coolant temperature	299°C	301.5°C (- 2.5°C uncertainty)
Pressuriser pressure	157.5 bar	155 bar (+ 2.5 bar uncertainty)
Pressuriser water level	4.5 m	
Pressuriser water volume	24.9 m^3	21.7m^3 + 3.2m^3 uncertainty (corresponds to 5% of measurement span 0.72 – 11.72 m)
Feedwater flow rate	14.35 kg/s	
Initial boron concentration	10 ppm	EOC
Steam flow rate	14.35 kg/s	
Feedwater temperature	120°C	
SG water level	16.54 m	16.2 m + 0.34 m uncertainty corresponds to 5% of measurement span 12.4 – 19.2 m
SG water mass	118.8 Mg	
Pressure at SG exit	84.1 bar	87.1 bar – 3.0 bar adjusted to primary temperatures

APPENDIX 14B.2.17 – TABLE 3

Sequence of events - Case 1
(1 SGTR Short term – Without LOOP)

0 s	SGTR occurs, 1 tube (double guillotine break - 2A) 102% FP, Pressuriser level control on Full heater power on (2500 kW).
941 s	Low pressuriser level (< 2.09 m) Isolation of RCV [CVCS] letdown
1463 s	Pressuriser is empty Heaters shut off
2005 s	Pressuriser pressure MIN2 setpoint reached (135 – 1.5 bar)
2006 s	Reactor trip Turbine Trip ARE [MFWS] isolation => beginning of short term radiological release to atmosphere
~ 2300 s	SGTR leak flow matched by RCV [CVCS] injection **Controlled state reached**
3485 s	SG level MIN2 setpoint reached in SG2 (on pressuriser loop)
3500 s	ASG [EFWS] actuation in SG2
3510 s	SG level MIN2 setpoint reached in SG1 and SG4
3530 s	ASG [EFWS] actuation in SG1 and SG4
3805 s	Operator manual action at RT + 30 min: actuation of Partial Cooldown
3920 s	SI signal on pressuriser pressure < MIN3 setpoint (115.0 + 1.5 bar)
4900 s	End of Partial Cooldown
4910 s	Operator manual action :RCV [CVCS] isolation MSRV opening setpoint on SG1 (98 bar) VIV [MSIV] closure on SG1 => end of short term radiological release to atmosphere (93 tons steam released to atmosphere from the affected SG during the short term)
~ 15000 s	**Leak termination** => end of short term calculation

APPENDIX 14B.2.17 – TABLE 4 – PAGE 1/2

0.0 s	SGTR occurs, guillotine break of area 2A initial leak flow rate 25.1 kg/s feedwater flow rate steps to zero main steam bypass flow rate steps to zero
237 s	Main steam pressure > 93.0-1.5 bar; reactor trip turbine trip signal actuation of emergency power mode, coast down of Reactor Coolant Pumps, actuation of MS relief valves, actuation of emergency feedwater pumps (105 tons/h per SG, 50°C)
265 s	Measured pressuriser water level more than 80 cm lower than set point; start of second RCV [CVCS] pump (20 kg/s injection, 10 kg/s letdown)
598 s	Water level in affected SG (wide range) > 17.2 + 0.38m; isolation of ASG [EFWS] of affected loop
862 s	Measured pressuriser water level < 2.09m (10 m^3); isolation of RCV [CVCS] letdown turn off of pressuriser heaters
1236 s	Water level in affected SG (narrow range) > 18.1 + 0.14m; actuation of partial cooldown (100°C/h from 93.0 – 1.5 bar to 60 bar)
1810 s	SGTR leak flow matched by RCV [CVCS] makeup **controlled state reached**
1815 s	Pressuriser pressure < 115.0 + 1.5 bar; ECC signal, actuation of MHSI / LHSI
2385 s	Partial cooldown finished
2500 s	Start of manual actions • RCV [CVCS] pumps halted (boration via RCV [CVCS] not available) • steam side isolation of affected SG by closing the VIV [MSIV] and increase of MSRV setpoint to 98 bar • start of RBS [EBS] (1 pump with 2.8 kg/s available) • start of cooldown with 25°C/h
4830 s	Boron concentration in all parts of RCP [RCS] > 620 ppm (required safe shutdown concentration for EOC MOX)
6500 s	SGTR leak flow halted due to pressure equilibrium between RCP [RCS] and SG1, i.e. **End of short-term and beginning of long-term phase**
16000 s	Manual isolation of ASG [EFWS] supply to SG4 (decrease of water level in SG4) to prepare emptying of affected SG1 via blowdown line

APPENDIX 14B.2.17 – TABLE 4 – PAGE 2/2

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

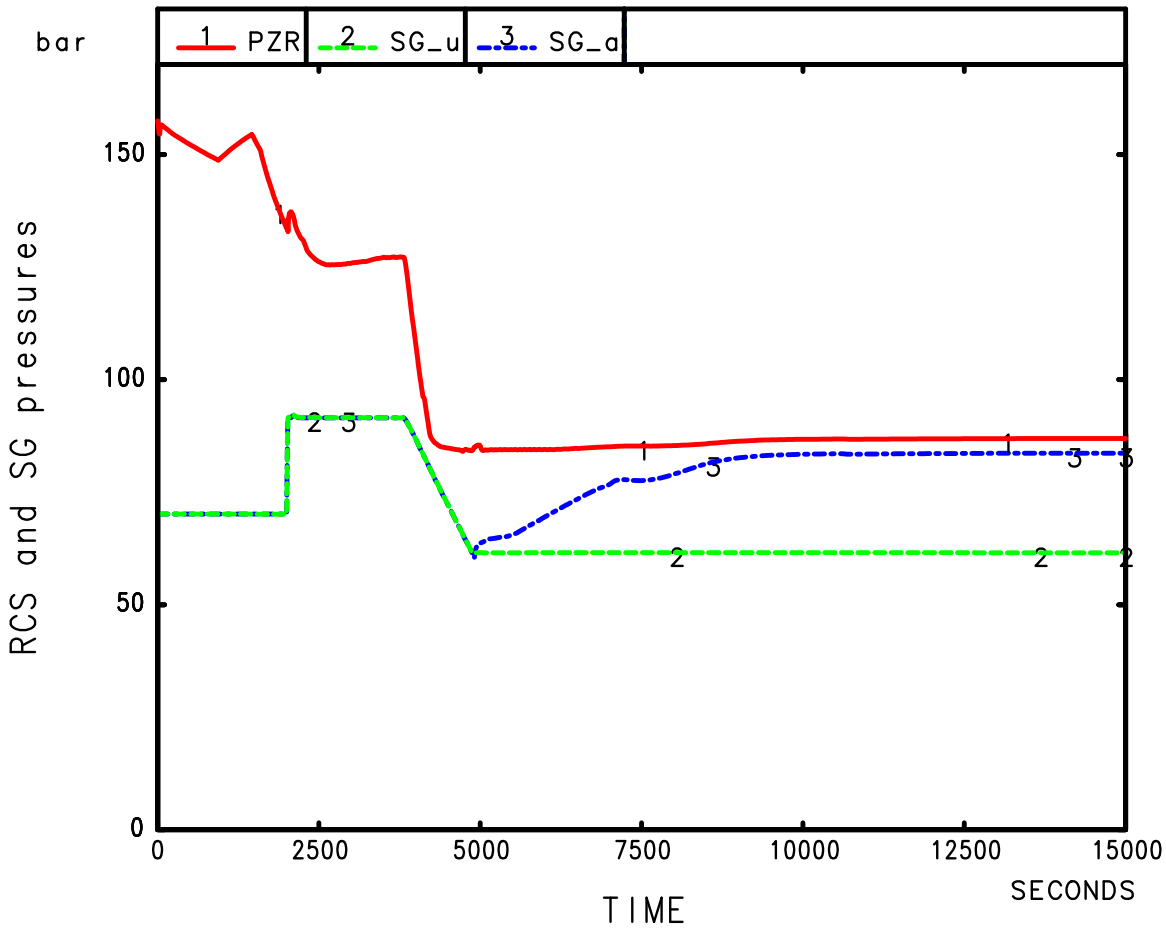
20000 s	Steam side isolation of SG4 and opening of blowdown line between affected SG1 and SG4
20000 s	Manual isolation of MHSI and accumulators
23600 s	RBS [EBS] halted to finally stop SGTR leak rate
30000 s	Water level in affected SG 17.5 m (SG4 13 m); start of final depressurisation by 20% opening of MS relief valve on the affected SG
31710 s	Primary pressure < 20 bar; LHSI starts loop injection
35000 s	End of calculation; RCP [RCS] pressure 20 bar, RCP [RCS] temperatures < 180°C RHR-Conditions reached
	safe shutdown state reached

APPENDIX 14B.2.17 – TABLE 5

Definition of studied cases

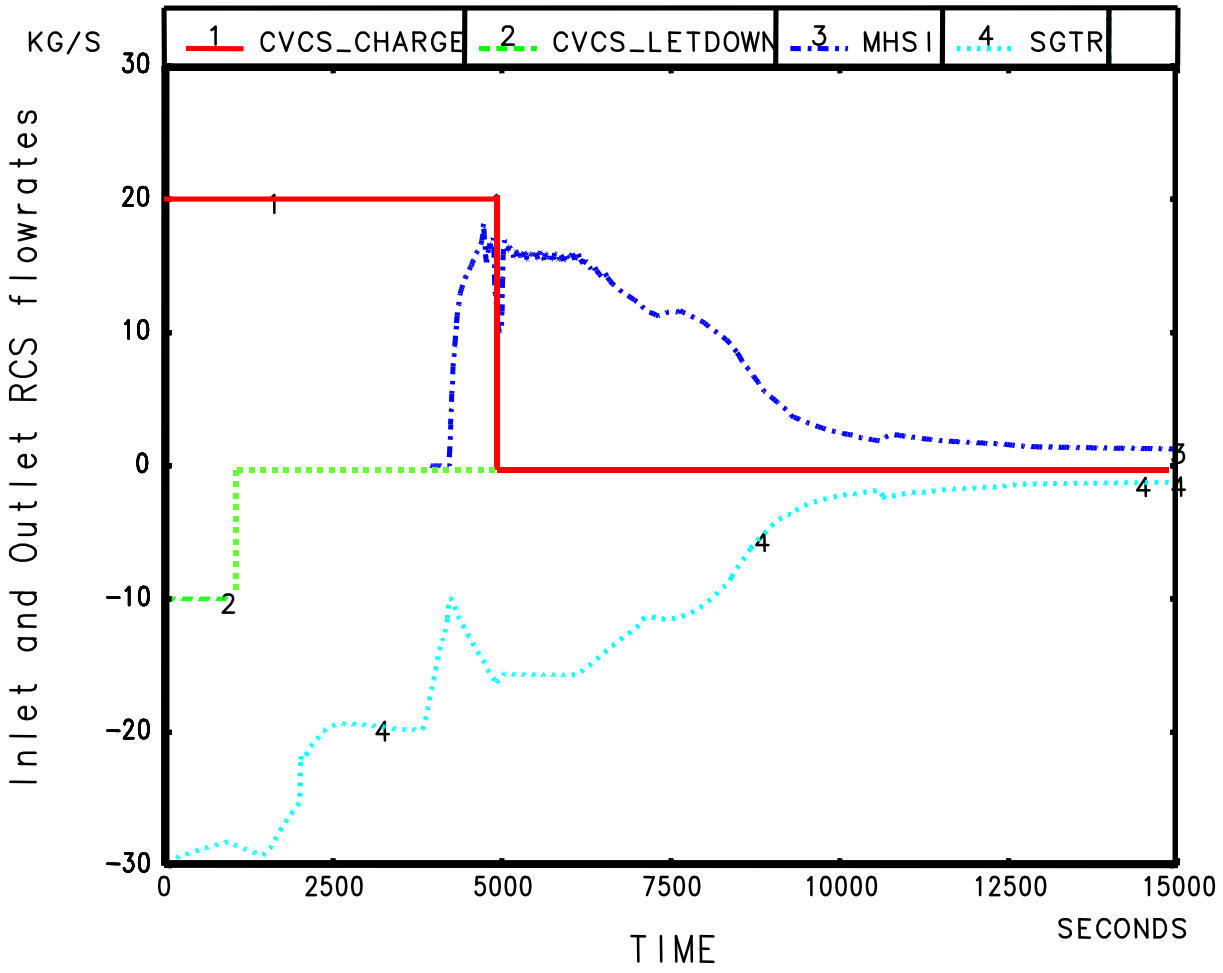
	SGTR 1 TUBE (2A)			
	Initiator	Controlled State	Safe Shutdown	Comments
Case 1	102% FP Without LOOP Calculation of maximum steam release	SG1 isolation		Calculation until reaching SG1 isolation in order to evaluate max. short term steam release (activity) conservative assumption: max. power
Case 2	2% FP with LOOP Verification of no overfilling of affected SG	Leak determination		Calculation until reaching leak termination in order to demonstrate no overfilling bounding assumptions: max. water content at t=0 min. power for min. release
Case 3		Restart of case 2 (with LOOP) RCP [RCS] and SG1 depressurisation Max. liquid and/or steam release Verification of no overfilling		Restart of case 2 Calculation until reaching RHR condition in order to evaluate max. long term release conservative assumptions: - max. SG water content - min. short term release - LOOP with respect to SG1 depressurisation (need for MSRV opening)

APPENDIX 14B.2.17 - FIGURE 1



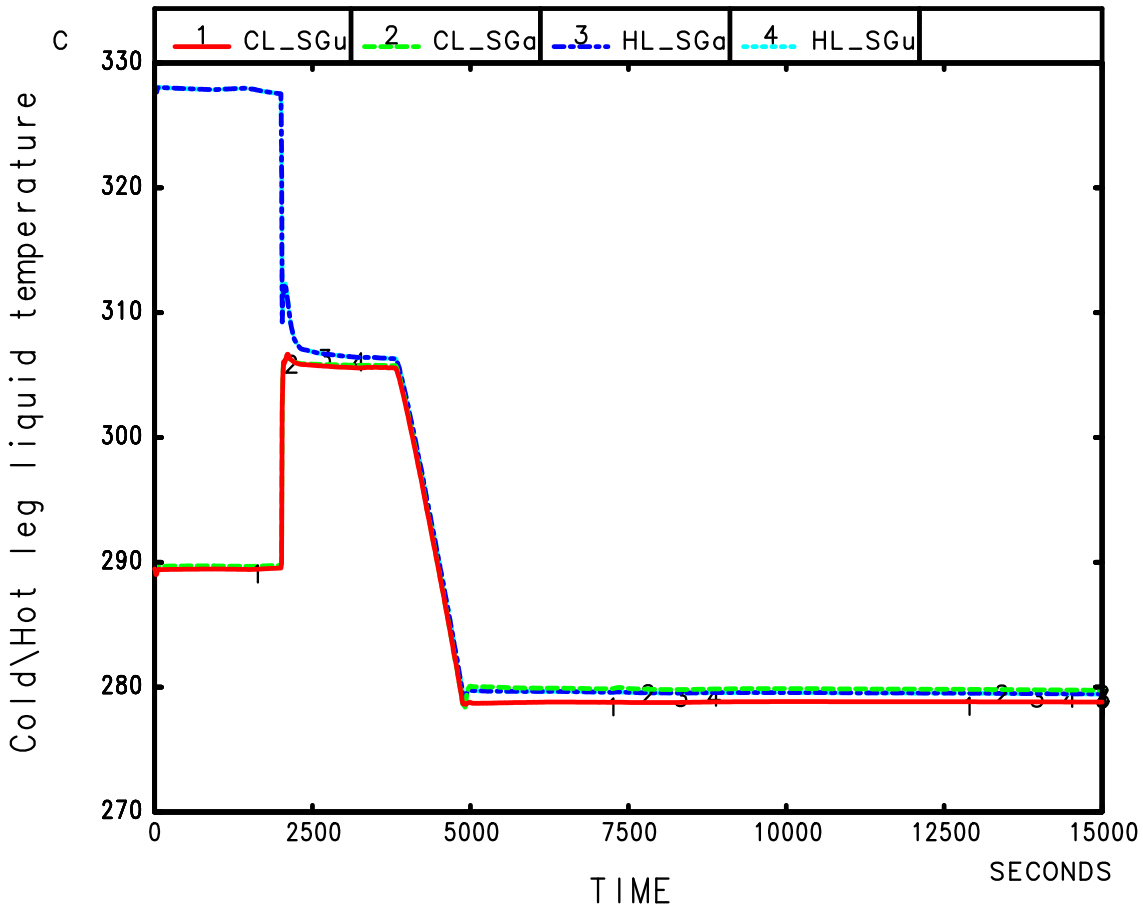
Pressuriser and SG Pressures
Case 1: Short Term - Without LOOP

APPENDIX 14B.2.17 - FIGURE 2



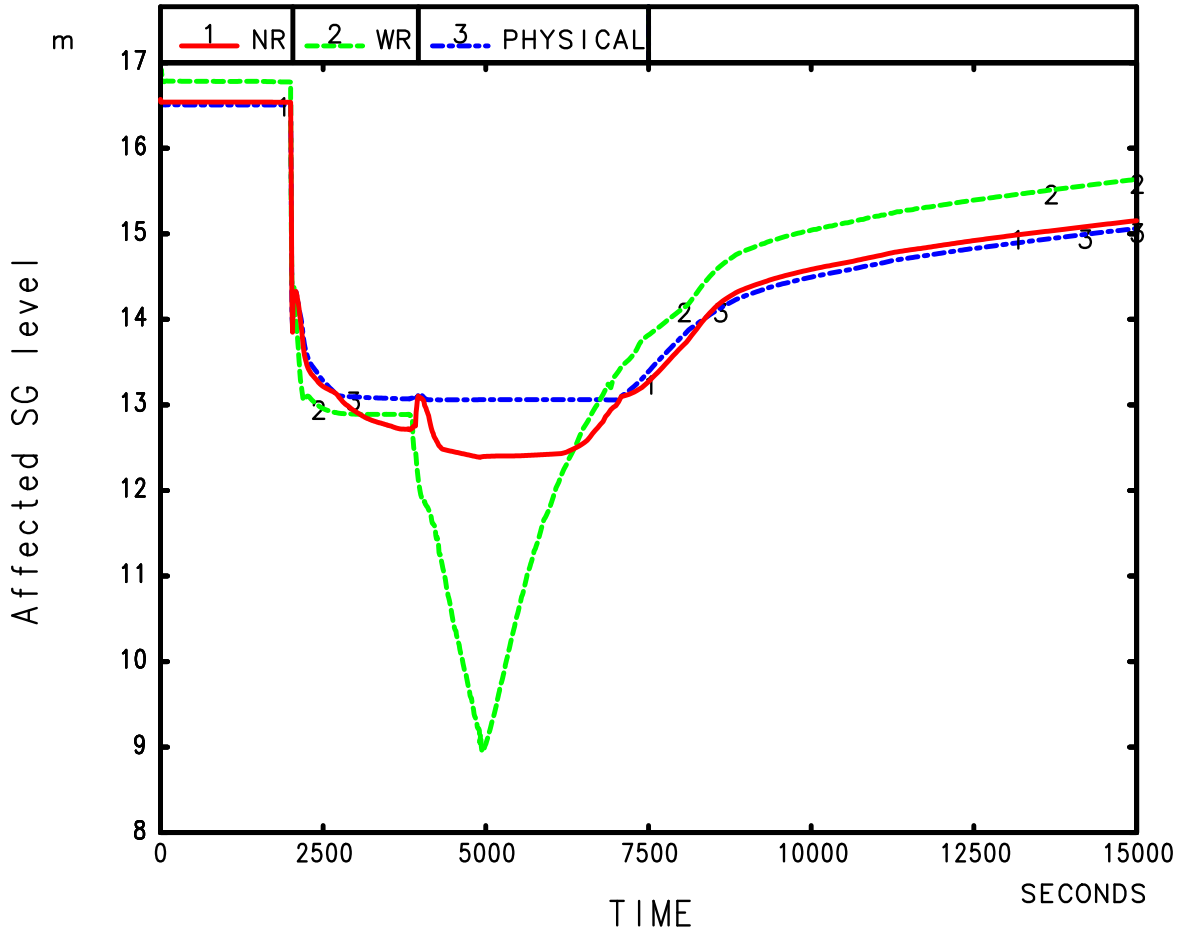
RCP [RCS] Inlet and Outlet Flow Rates
Case 1: Short Term - Without LOOP

APPENDIX 14B.2.17 - FIGURE 3



Cold / Hot Leg Temperatures
Case 1: Short Term - Without LOOP

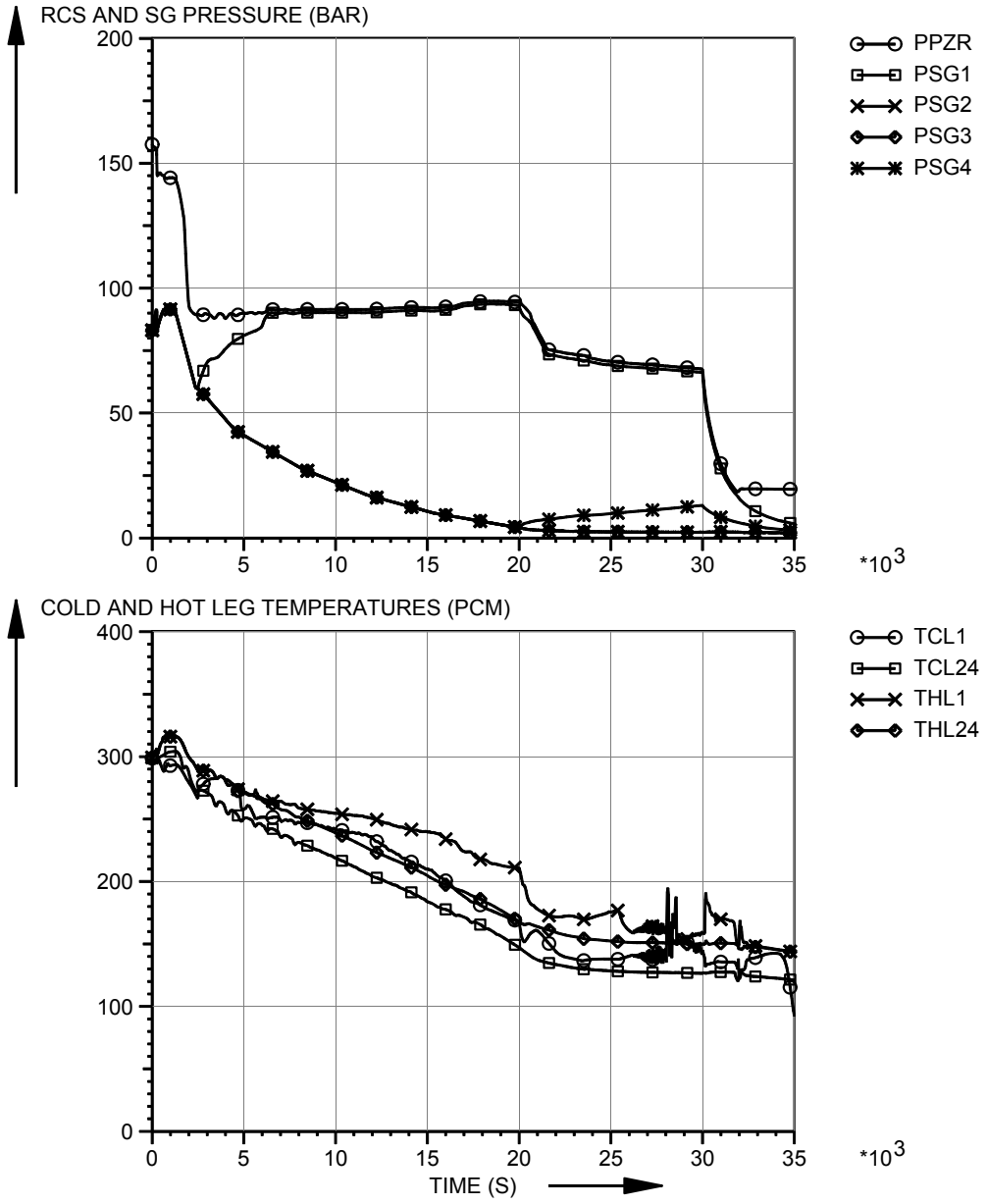
APPENDIX 14B.2.17 - FIGURE 4



Affected SG Level
Case 1: Short Term - Without LOOP

APPENDIX 14B.2.17 - FIGURE 5

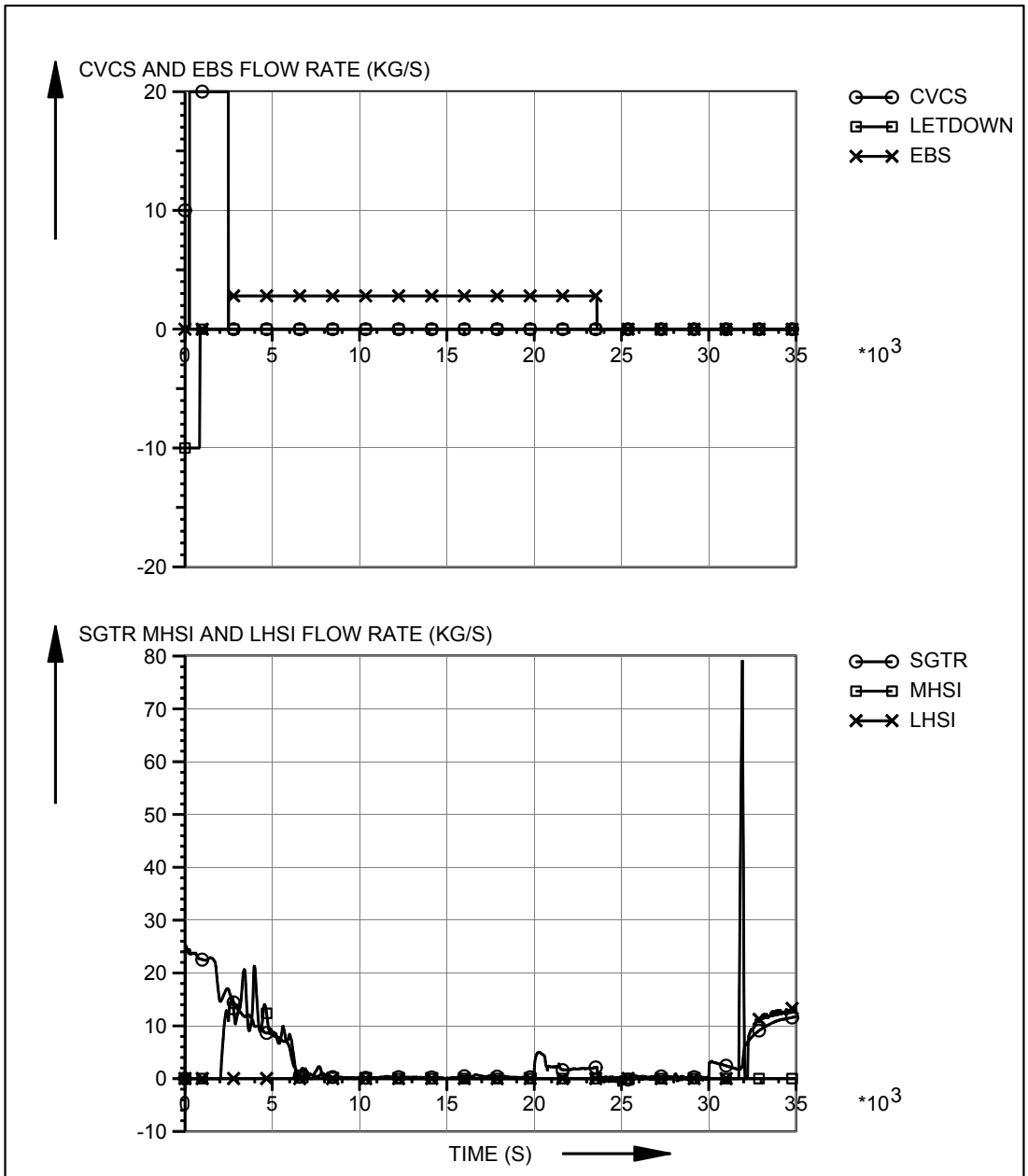
3.11.98 14:18: 4 TAPE4_DOKU.981027.1440.10057 Page 1



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

APPENDIX 14B.2.17 - FIGURE 6

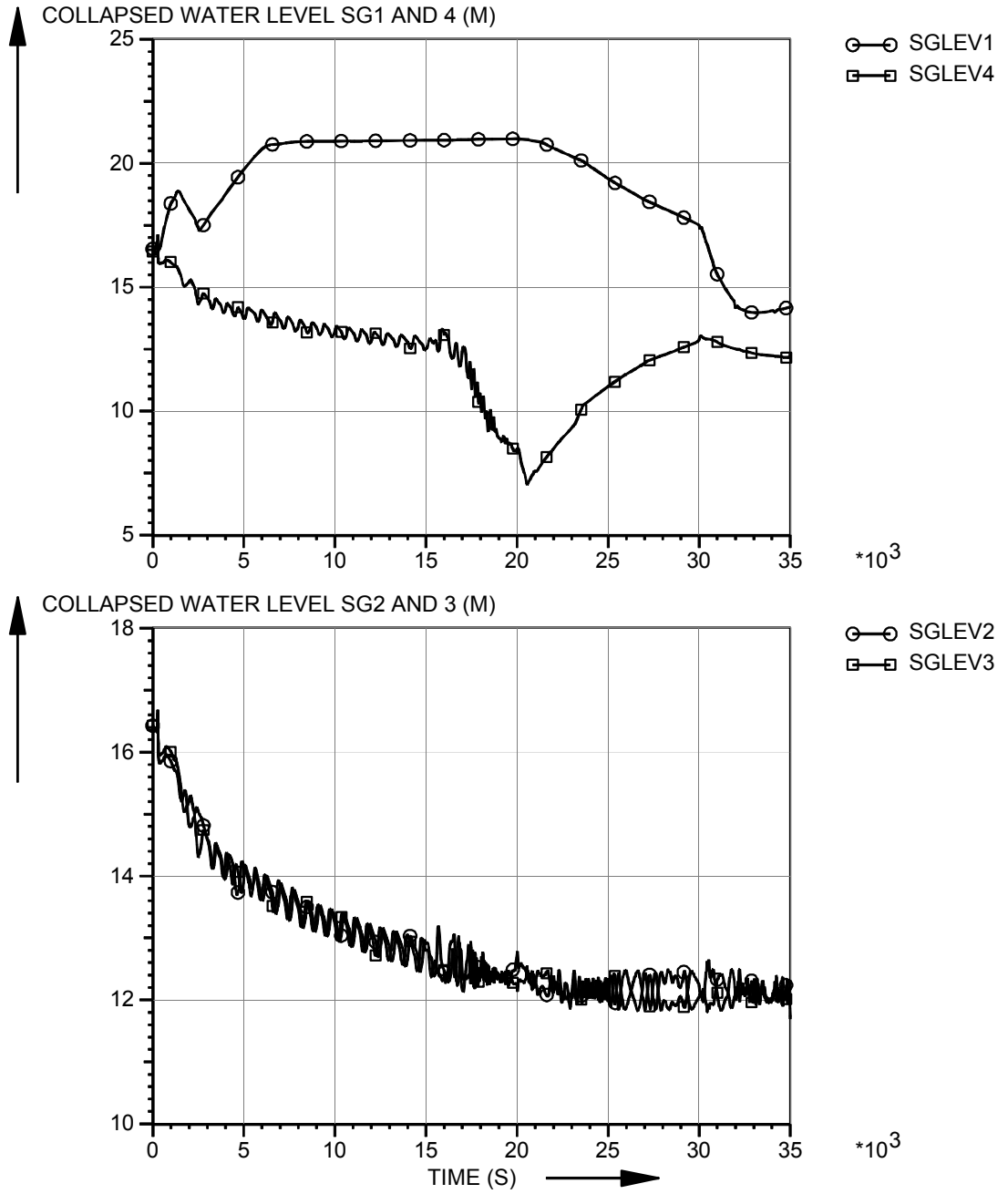
3.11.98 14:18:28 TAPE4_DOKU.981027.1440.10057 Page 2



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

APPENDIX 14B.2.17 - FIGURE 7

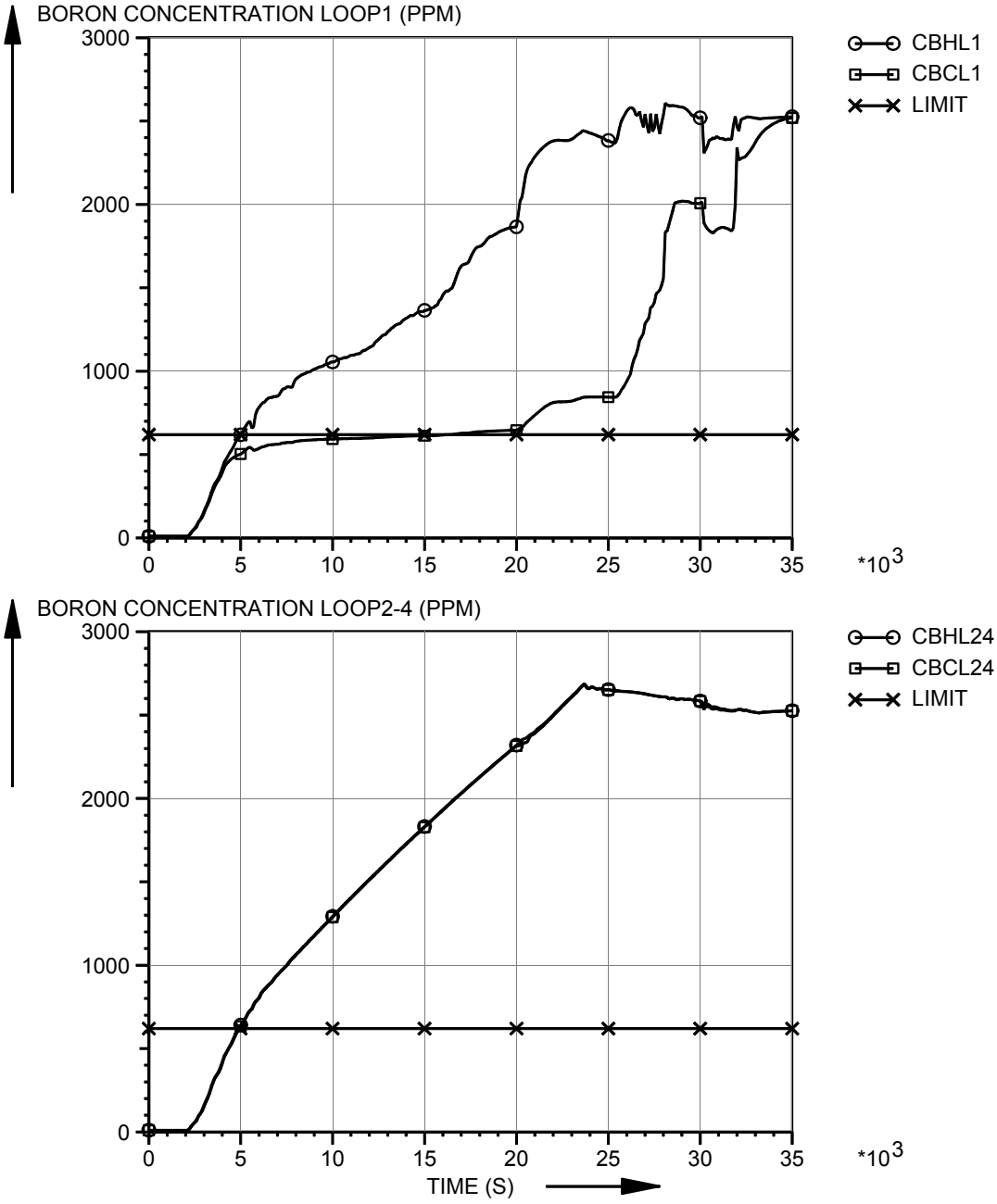
3.11.98 14:18:29 TAPE4_DOKU.981027.1440.10057 Page



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

APPENDIX 14B.2.17 - FIGURE 8

3.11.98 14:18:28 TAPE4_DOKU.981027.1440.10057 Page 4



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

2.18. STEAM GENERATOR TUBE RUPTURE (2 TUBES IN STATE A, PCC-4)

This section deals with Steam Generator Tube Rupture of 2 tubes in 1 Steam Generator, occurring from state A conditions as defined in section 0.1 of this appendix.

2.18.1. Identification of causes and accident description

2.18.1.1. General concern

The accident examined is either due to small continuous leaks or to the complete severance of 2 individual steam generator tubes located in the same steam generator. This results in an SGTR of break area four times the cross sectional area of the tubes (4A). It is classified as a PCC-4 event, as discussed in section 0.1 of this appendix.

There are two main consequences of this accident. The first is associated with the risk of contamination of the secondary side inventory, mainly the affected SG, by leakage of radioactive coolant from the primary side via the SGTR. The primary coolant is assumed to be contaminated by corrosion and fission products arising from continuous operation with a limited amount of defective fuel rods. The second is the possible discharge of the activity, either in the steam or liquid phase, to the atmosphere via the steam generator safety and/or power operated relief valves.

The description of the transient evolution is subdivided into short term and long term phases to clearly separate the phases of activity release to the atmosphere. The short term phase is defined as the period up to the isolation of the affected SG, i.e. termination of the activity release. This includes the controlled state when the SGTR leak flow is matched by MHSI injection. In the long term phase, the plant is transferred to LHSI/RHR conditions with a possible additional activity release if the affected SG is depressurised via the MS relief valve or VIV [MSIV] bypass valve.

2.18.1.2. Typical sequence of events

The typical sequence of events in the case of a 2 tubes SGTR (apart from additional events that could occur because of the inclusion of a single failure) is as follows:

2.18.1.2.1. From the initiating event to leak termination (short term)

a) From initiating event to the controlled state

The controlled state is defined as a state where the SI flow rate, or the RCV [CVCS] flow rate when the RCV [CVCS] is available, matches the leak flow rate due to the tube rupture, and where the decay heat is removed from the RCP [RCS] by the SG using the VDA [MSRT] and ASG [EFWS], or GCT [MSB] and ARE [MFWS]/AAD [SSS] when available.

Note : In the case of a 4A break, the RCV [CVCS] is not able to compensate the SGTR flow rate for this leak size. The controlled state cannot be reached using only RCV [CVCS] injection. Therefore, MHSI injection is also required.

The tube rupture leads to a loss of primary coolant, which is discharged into the SG affected by the rupture. The break causes a decrease in primary pressure and a contamination of the secondary side due to SGTR flow rate.

The reactor trip occurs either on a “pressuriser pressure < MIN2” signal, or on a “SG level > MAX1” signal from the affected SG.

During power operation, the operation of the SG level control system compensates for the SG level increase by the SGTR leak flow. This is performed by a minor reduction in feedwater flow to the affected SG and the SG level does not increase to the MAX1 setpoint.

After RT, or for hot shutdown initial conditions, the primary heat transferred to the secondary side becomes insufficient to boil all the SGTR leak flow rate. Therefore, the rupture flow results in an increase in the affected SG level. Following a “SG level > MAX1” signal, feed to the affected SG is isolated by closure of the associated ARE [MFWS]/AAD [SSS] isolation valves, and of the associated ASG [EFWS] isolation valves if the ASG [EFWS] has already been actuated).

Should the SG level increase further above the “MAX1” setpoint and reaches the “SG level > MAX2 “setpoint, it would be a result of only the SGTR leak rate. The “SG level MAX2” signal is used as the F1A detection and isolation signal for the affected SG (see the discussion below of "from the controlled state to the safe shutdown state"). It is the most appropriate signal for initiating actions for the prevention of SG overfilling.

The reactor trip signal automatically trips the turbine the steam generator pressure rapidly increases to the VDA [MSRT] setpoint, resulting in steam discharge to the atmosphere. This is a consequence of the assumed unavailability of the GCT [MSB] as it is not F1 qualification or subsequent to a LOOP occurring at turbine trip),

The continuous loss of RCP [RCS] coolant inventory via the SGTR and a possible plant cooldown after reactor trip in the case from full power initial conditions, lead to emptying of the pressuriser and a consequent depressurisation of the primary side.

Following either a SI signal on a “pressuriser pressure < MIN3” signal, or on a “SG level > MAX2” signal from the affected SG, a partial cooldown is initiated unless the partial cooldown has already occurred together with a safety injection signal). This is performed either by the GCT [MSB] if available, from 87.1 bar to 55 bar, or by the VDA [MSRT]s from 93 bar to 60 bar, at a cooldown rate of -100 ^{o}C/h), being performed by all SG, including the faulted one.

During the partial cooldown, once the MHSI injection matches the SGTR leak flow rate, the controlled state is reached. Having reaching the controlled state, the affected SG is still filling up with contaminated water as the SGTR flow has not been terminated, and continuous releases to the atmosphere are still occurring.

b) From the controlled state up to leak termination

At the end of the partial cooldown, the primary pressure is given by the MHSI shut off head and a contaminated SGTR flow is still entering the affected SG, leading to a continued level increase.

Following a “SG level > MAX2 (after end of partial cooldown)” signal, the affected SG is identified and automatically isolated on the steam side by closure of the associated VIV [MSIV] and the raising of the associated VDA [MSRT] pressure setpoint to above the MHSI delivery pressure and below the MSSV pressure setpoint. The VIV [MSIV] of the 3 unaffected SG remain open.

The pressure in the affected SG increases, primarily due to the SGTR flow rate. When the pressures in the primary side and affected SG equalise, the leak is finally terminated, without any overfilling of the affected SG. This is the end of the short term phase. Only steam has been discharged to atmosphere during this phase.

The short term phase, the period up to SGTR leak termination, relies only on automatic F1 means (I&C signals, systems). Of course, it is possible for the operator to anticipate automatic actions, provided he acts in accordance with the emergency guidelines, but operator action is not required during this phase for the fulfilment of the safety/decoupling criteria.

2.18.1.2.2. From leak termination up to the safe shutdown state (long term)

The safe shutdown state is defined as a state where at least one LHSI/RHR train is connected to the RCP [RCS]. The pump is operating in RHR mode with suction from a RCP [RCS] hot leg and re-injection through the LHSI heat exchanger into the cold leg of the same RCP [RCS] loop. In addition, the affected SG has been isolated.

- 1/4 LHSI/RHR trains is sufficient to provide the required heat removal. The connection conditions are:
  - RCP [RCS] hot leg pressure < 30 bar and,
  - RCP [RCS] hot leg temperature < 180°C and,
  - ΔT_{sat} [(1)] and RPVL consistent with LHSI/RHR suction conditions from the hot leg.
- Should 2 LHSI/RHR trains be unavailable (single failure and preventative maintenance), 1/2 remaining LHSI/RHR trains aligned in RHR-mode would be sufficient to ensure the heat removal below 180°C RCP [RCS] hot leg temperature. The other pump aligned in SI mode will be sufficient to support the required LHSI operating conditions in RHR-mode, and Reactor Coolant Pump operation if any.

The sequence of actions to be performed by the operator to reach the safe shutdown can be divided into 3 successive phases:

Boration

The RCP [RCS] boration is performed using t he RBS [EBS] during the cooldown. It is actuated by the operator. The RCV [CVCS] is not credited, because it is not a F1-classified system. Once the required boration has been completed, the operator stops the RBS [EBS].

The allowed cooldown rate depends on the number of available RBS [EBS] trains:

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

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

These cooling rates are defined so that the achievable RBS [EBS] boration matches the reactivity insertion resulting from the RCP [RCS] cooldown. The calculated rates are conservative with respect to the boration requirement as they do not take credit for the boron injected by the MHSI during the short term phase.

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

Cooldown by unaffected SG

The RCP [RCS] cooldown is performed using the unaffected steam generators, without affecting the pressure difference between the primary side and the affected SG. The primary pressure is maintained by the MHSI which is kept operating.

The RCP [RCS] cooldown is actuated by the Operator. The RCP [RCS] cooldown is conservatively assumed to commence 2 hours after RT.

The duration of this cooldown phase has been defined to be consistent with the VDA [MSRT] capacity and the ASG [EFWS] tank capacity. This ensures the LHSI/RHR connecting conditions are reached before emptying of the ASG [EFWS] tanks and the unaffected SG. The ASG [EFWS]-tanks design assumes 2 Reactor Coolant Pumps remain in operation during the RCP [RCS] cooling phase.

In the case of no Reactor Coolant Pump running, the secondary side depressurisation is performed down to a SG pressure corresponding to a cold leg temperature of 150°C, saturation for 5 bar, and a hot leg temperature lower than 180°C, including temperature measurement uncertainties.

Depressurisation of the RCP [RCS] and the affected SG

At the end of the RCP [RCS] cooldown phase using the unaffected SG, the RCP [RCS] temperature is compatible with LHSI/RHR connecting conditions. However, the RCP [RCS] pressure is close to the MHSI delivery pressure, higher than the LHSI/RHR maximal connecting pressure of 30 bar.

The operator switches off all the MHSI pumps, to perform the required RCP [RCS] depressurisation, while keeping the LHSI pumps on, or starting the LHSI if not already on, to prevent RCP [RCS] saturation conditions being reached.

The RCP [RCS] is then depressurised as follows, depending on the Reactor Coolant Pump operation:

- At least 1 Reactor Coolant Pump operating[(1)] : the RCP [RCS] and affected SG depressurisation is performed by the primary side, using the pressuriser (PZR) spray and if necessary RCV [CVCS] letdown, although this is not a F1 classified system, or if these are unavailable by opening 1 F1 classified PSV.

  The depressurisation via the primary side induces reverse flow via the SGTR. This is acceptable since a Reactor Coolant Pump is operating. This promotes mixing in the primary circuit and prevents any unacceptable heterogeneous dilution occurring due to an unborated water slug.

- Reactor Coolant Pump off: the RCP [RCS] and affected-SG depressurisation is performed using the secondary side, to minimise the SGTR reverse flow.

(1) Reactor Coolant Pumps in the unaffected-loops are preferably kept on, while Reactor Coolant Pump in the affected-loop is shut down

In most SGTR cases, the affected SG will be almost full of water at the end of the RCP [RCS] cooling phase. The level will be above the SG level "MAX2" setpoint, the bottom of dryers. In these conditions, a direct opening of the VDA [MSRT], or the VIV [MSIV] by-pass line, would lead to liquid release into the atmosphere. This resulting from the mixture level increase when the SG reaches saturation conditions. In order to avoid such a consequence, it is necessary to decrease the SG level before opening the VDA [MSRT]. This decrease is performed using the SG blowdown system (SGBS). This takes water from the affected SG into a storage tank located outside the containment, but is not a F1 classified system., The other option available is to transfer water from the affected SG to the unaffected adjacent SG, by opening the dedicated SGBS connection line, a F1B classified system. After the opening of this dedicated line, the affected SG and the unaffected SG reach a pressure balance. Depending of the resulting pressure level, two cases can be envisaged:

- no need for steam release: The opening of the dedicated line is sufficient to lower the RCP [RCS] pressure to a level compatible with LHSI/RHR connection. No additional action is needed.

- a need for steam release: The opening of the dedicated line is not sufficient to lower the RCP [RCS] pressure to a value compatible with LHSI/RHR connection. RCP [RCS] depressurisation is completed by opening the VDA [MSRT] of the affected SG, or the F1B classified VIV [MSIV] bypass line, with use of the VDA [MSRT] of the unaffected SG if failure of the VDA [MSRT] of the affected SG). As the level in the GS has been lowered, there is no risk of liquid entrainment.

Note : in the case with the Reactor Coolant Pumps operating, with pressuriser spray used to depressurise, assuming the RCV [CVCS] letdown is unavailable, the depressurisation by the secondary side may be preferred to the opening of PSV. The solution retained in the following CATHARE analysis (case 3) is the depressurisation by the secondary side.

2.18.1.2.3. Radiological releases

Before reactor trip, contaminated steam goes through the turbine 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 monitoring and periodic sampling.

After reactor trip, if the condenser is not available, the main steam bypass valves remain closed, resulting in the opening of the VDA [MSRT]. Steam is subsequently released to the atmosphere.

2.18.1.2.4. Precautions limiting the event occurrence

Please refer to 2.17.1.2.4 of this appendix for additional discussion.

2.18.2. Safety criteria

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

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

- No core damage (fuel cladding integrity).
- No MSSV demand, to prevent any risk of MSSV failure in the open position.
- Attainment of LHSI/RHR connecting conditions, RCP [RCS] boration, depressurisation and heat removal) with achievement of safe shutdown conditions, using F1 means only, with the conservative safety analysis rules defined in section 1 of this appendix.

The prevention of core damage, by keeping the core coolable, and preventing increased activity levels on the primary side, is covered by the LOCA studies, see section 2.2 of this appendix).

The SGTR recovery procedure involving automatic and normal actions is defined so that 2 following objectives can be fulfilled:

- Prevention of activity release by liquid blowdown to the atmosphere.
- Minimising the SGTR back flow. This avoids problems with a low-borated water slug in the primary circuit, in cases where the Reactor Coolant Pump are not operating, e.g. following LOOP.

2.18.3. Methods and assumptions

2.18.3.1. Methods of analysis

The SGTR accident analysis is performed with the CATHARE code in the frame of a **realistic deterministic methodology.**

The realistic deterministic methodology is characterised by the following two main features:

- key code model 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 the adequacy of the code 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 bounding conservative approach of key phenomena by the code, relying on the validation matrix of the code.

- the introduction, when necessary, of conservative biases as close as possible to the uncertainty on 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 SGTR transient are:

- the SGTR flow rate and resulting SG overfilling,

- the moderate RCP [RCS] draining (pressuriser) and depressurisation, in equilibrium with the affected-SG pressure,

- the asymmetric RCP [RCS] heat removal via the unaffected SG in subcooled RCP [RCS] conditions (with Reactor Coolant Pump on or off),

- the RCP [RCS] cooling, and the RCP [RCS] depressurisation down to LHSI/RHR connecting conditions (with Reactor Coolant Pump 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, e.g.

- CATHARE SGTR model for accurate SGTR break flow prediction,

- the validation of the axial SG model, with an economiser of the N4 SG-type, using MEGEVE small scale model tests,

- the overall validation of the code by simulation of integral effect tests (IET), covering a wide range of representative PWR transients on small-scale facilities, e.g.

    - test BETHSY 3.4b ‘1 SGTR’: CATHARE accurately calculated the RCP [RCS] pressure and SGTR break flow rate, the pressuriser level, the SG parameters (both affected and non-affected SG),

    - test BETHSY 4.3b ‘6 SGTR’: CATHARE accurately predicted the mass discharged via the faulted SG relief valve, the faulted SG mass inventory, the RCP [RCS] mass inventory and distribution (with formation of the large steam space inside the faulted SG tubes, the slow RCP [RCS] depressurisation during the draining of the faulted SG, the restart of the loop circulation after the Reactor Coolant Pump start up in the affected loop, and the consequent fast depressurisation due to the condensation and collapse of the faulted SG tubes steam space.

- The transient analysis relies on the application of the conservative PCC analysis rules defined in section 1 of this appendix. Part of these rules is the deterministic pessimising of all relevant boundary conditions, relative to the decoupling criteria under consideration. These pessimisms 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 efficiency of the protection and mitigation actions (maximal uncertainty on each I&C measurement and signal delay, and on each system response time and capacity).

  This analysis methodology provides conservative results which can be directly used for confirming the decoupling criteria are met.

2.18.3.2. Main assumptions

2.18.3.2.1. Accident definition

The cases studied in section 2.18 of this appendix correspond to the double ended guillotine rupture of 2 tubes in the same steam generator, which allows unimpeded blowdown from both ends of the severed tubes.

For the accident analyses, a loss of offsite power (LOOP) could be added to the accident scenario is that is more onerous.

2.18.3.2.2. Protection and mitigation actions

In the case of a SGTR event, automatic protection systems and operator actions, using F1-classified systems, aim at performing the following:

- tripping the reactor,
- removing the residual heat,
- terminating the primary to secondary flow rate,
- limiting contaminated SG water mass released to the atmosphere
- bringing the reactor to a safe shutdown state.

The different automatic protection actions and alarms which could occur in the SGTR event mitigation are linked either to the reactor coolant depressurisation or to the level increase in the affected SG.

The possible F1A reactor trip signals are:

- "pressuriser pressure < MIN2",
- “SG level > MAX1".

The other F1A automatic protections are the following:

- Turbine trip: on RT signal, the turbine trip is actuated,
- SI & Partial Cooldown: on a "pressuriser pressure < MIN3" signal, the RIS [SIS] is actuated and a partial cooldown performed using all SG including the affected one. This is initiated on the signal unless it has already been actuated,
- Partial Cooldown: following a "SG level > MAX2(*)" signal, a partial cooldown is initiated using all SG (including the affected one) except if partial cooldown has already been actuated),
- Affected SG isolation (feed side): following a "SG level > MAX1" signal, the affected SG is isolated by isolating the ARE [MFWS]/AAD [SSS] line and isolating the ASG [EFWS] line if the ASG [EFWS] has already been actuated. This signal is SG specific.
- Affected SG isolation (steam side): following a "SG level > MAX2(*)" signal, after the end of the partial cooldown, the affected SG is isolated. This is performed by closure of the VIV [MSIV] and raising the VDA [MSRT] pressure setpoint to above the MHSI shut off head but below MSSV pressure setpoint.
- ASG [EFWS] actuation: following a "SG level < MIN2" signal, the ASG [EFWS] train is actuated in the corresponding SG (SG specific). The time delay between the setpoint being reached and the effective ASG [EFWS] flow injection is defined consistent with the assumption of LOOP, or no LOOP, and includes the delay associated with the EDG reloading sequence,
- VDA [MSRT] actuation: when the SG pressure reaches the VDA [MSRT] setpoint "SG pressure > MAX1", the VDA [MSRT] opens and provides heat removal with pressure control,
- VIV [MSIV] isolation: following a "SG pressure < MIN1" or "SG pressure drop > MAX1" signal, the VIV [MSIV] closure is initiated, all the main steam lines are isolated as the signal is not SG specific),
- VDA [MSRT] isolation: when the SG pressure reaches the VDA [MSRT] setpoint "SG pressure < MIN2", the VDA [MSRT] of the corresponding SG is isolated, by closure of the associated MSRIV as this signal is SG specific).

For long term mitigation of the SGTR, the operator actions aim at transferring the plant to the safe shutdown state. In accordance with the written procedures, the operator must perform boration and cooldown using the unaffected SG, before depressurising the RCP [RCS] and the affected SG to reach the LHSI/RHR connecting conditions.

2.18.3.2.3. Operator actions

Operator action is not considered until 30 minutes after RT. When operator action is needed local to plant, the delay is assumed to be 1 hour from RT. In any case, the operator will not isolate the affected SG before the end of the partial cooldown. This minimises any leak back flow which could be detrimental for RCP [RCS] boration when LOOP is assumed.

(*) This level MAX2 is only reached in affected SG because of isolation of all SG feedwater injection means (ARE [MFWS], AAD [SSS], ASG [EFWS]) on SG level > MAX1.

In accordance with the emergency procedures, the operator performs the following actions, (already described in section 2.18.1.2.2 of this appendix).

The list of F1B operator actions related to the SGTR detection and SGTR flow termination, with indication of the main F1B information needed, are:

Affected SG isolation (F1B)

- manual actuation of partial cooldown, if not automatic (SG level, secondary side activity)
- manual ASG [EFWS] isolation, if not automatic (SG level, secondary side activity)
- manual ARE [MFWS]/AAD [SSS] isolation, if not automatic (SG level, secondary side activity)
- manual VIV [MSIV] isolation, if not automatic (SG pressure)
- manual VDA [MSRT] setpoint lift up, if not automatic (SG pressure)

RCV [CVCS] charging line isolation (F1B)

- manual isolation of the charging line, if not automatic (SG level, secondary side activity)

The list of F1B operator actions related to the transfer to the safe shutdown state, with indication of the main F1B information needed, are[(1)]:

ASG [EFWS] passive header (pump discharge) opening (F1B)[(2)]

- manual realignment of the ASG [EFWS] pump discharge to the unaffected SG, via the dedicated passive header (ASG [EFWS] flow rate, SG level).

VIV [MSIV] bypass valve opening (F1B)

- manual opening of the VIV [MSIV] bypass valve (SG pressure)

Reactor Coolant Pump shutdown (F1B)

- manual shutdown of the Reactor Coolant Pump (Reactor Coolant Pump on/off status)

RCP [RCS] cooldown by 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)

(1) Refer to section 2.16 "Feedwater line break" for detailed information
(2) ASG [EFWS] passive header opening only required if preventative maintenance

RCP [RCS] boration (F1B)

- manual start-up of the RBS [EBS] pump (at RCP [RCS] cooldown start-up)
- manual alignment to an unaffected loop (at RCP [RCS] cooldown start-up)
- manual shutdown of RBS [EBS] pump or manual closing of the isolation valve (RBS [EBS] tank level)

ASG [EFWS] passive header (pump suction) opening (F1B)

- manual realignment of the ASG [EFWS] tank to the ASG [EFWS] pump suction via the dedicated passive header (ASG [EFWS] tank level)

RCP [RCS] and affected SG depressurisation (F1B)

- manual isolation of the accumulators (RCP [RCS] pressure)
- manual MHSI pump shutdown, if previously actuated (RCP [RCS] pressure, MHSI flow rate)
- manual LHSI actuation, if not previously actuated (before LHSI/RHR connection)
- manual closing of the SGBS-tank isolation valves, if not yet done (before initiating the water transfer)
- manual opening of the dedicated SGBS-line for water transfer from the affected SG to the neighbour unaffected SG (SG pressure and wide range level)
- manual opening of VDA [MSRT] or VIV [MSIV] by-pass valve (SG pressure)

LHSI/RHR connection (F1B)

- manual connection of the LHSI (in RHR-mode) to the RCP [RCS] (RCP [RCS] pressure, RCP [RCS] temperature, RPVL, ΔT_{sat}).

2.18.4. Definition of studied cases

2.18.4.1. Short term cases

The short term phase, as defined for the SGTR study, is the time period between the SGTR initiation and the leak termination.

This phase includes the controlled state which represents the state where the SI flow rate matches the SGTR flow rate.

During this short term phase, the purpose of the study is to quantify the maximum amount of fluid released from the affected SG to the atmosphere. This is done in two steps:

- To quantify the maximum amount of activity released to the atmosphere, it is conservative to maximise the power to be removed. This is because before isolation of the affected SG there is only steam release. Starting at 102% full power leads to maximum decay heat. Assuming no LOOP means that Reactor Coolant Pump power has to be removed from the RCP [RCS] in addition to decay heat.

- To verify that no SG overfilling occurs, and thus no liquid is released to the atmosphere before leak termination, a minimum primary power is conservative. Therefore, the analysis is started from 2% reactor power with an assumption of LOOP).

Two cases are therefore analysed, for each of these two objectives, with different assumptions (see in addition Appendix 14B.2.18 - Table 6):

Case 1: without LOOP

To calculate a maximum steam release from the affected SG, it is appropriate to maximise the power to be removed via the SG.

The most onerous case is therefore:

- with a maximum initial power of 102% FP,
- without LOOP.

It is assumed that the Reactor Coolant Pump remain operating throughout the transient.

The other specific assumptions related to this case are described in section 2.18.5.1 of this appendix.

Case 2: with LOOP

To calculate a maximum liquid inventory in the affected SG, it is appropriate to maximise the initial water inventory in the SG. This is a maximum at low power level. It is also appropriate to minimise the steam released by minimising the power to be removed via the SG.

The most onerous case is thus:

- with a low initial power of 2% FP,
- with LOOP assumed coincident with the turbine trip signal.

The other specific assumptions related to this case are described in section 2.18.5.2 of this appendix.

2.18.4.2. Long term cases

The long term phase is the time period between the leak termination and reaching the safe shutdown state, i.e. reaching LHSI/RHR connecting conditions.

It includes the phases of boration of the RCP [RCS], cooldown of the RCP [RCS] using the unaffected SG and the final depressurisation of the RCP [RCS] and the affected SG.

During this long term phase, the purpose of the study is to quantify the maximum amount of fluid released from the affected SG to the atmosphere. This is done in two steps:

- To verify the capability of the plant to reach LHSI/RHR connecting conditions, of boration, cooldown, depressurisation and heat removal, to reach safe shutdown conditions, relying on F1 means only with the conservative set of safety analysis rules defined in section 1 of this appendix. In particular, the adequacy of the ASG [EFWS] tanks capacity is confirmed.

- To calculate the maximum amount of activity release to the atmosphere, mainly during the final depressurisation phase of the RCP [RCS] and the affected SG.

Two cases are therefore analysed, for each of these two objectives, with different assumptions (see in addition Appendix 14B.2.18 – Table 6):

Case 3: without LOOP

This case, without LOOP is performed in to demonstrate the capability of the F1 systems, especially the adequacy of the ASG [EFWS] tanks water content, to cooldown the plant to LHSI/RHR connecting conditions with a maximum power to be removed.

For the long term, the final depressurisation of the RCP [RCS] and the affected SG is performed by the mass transfer from the affected SG to the adjacent unaffected SG. If the affected SG draining is sufficient for the depressurisation, there will be no VDA [MSRT] opening (or no VIV [MSIV] by-pass opening) thus preventing any release from the affected SG to atmosphere.

Case 4: with LOOP

This case, with the assumption of LOOP, is performed to demonstrate the capacity of the F1 systems to adequately borate the RCP [RCS], and to depressurise RCP [RCS] and the affected SG without leak backflow, while the Reactor Coolant Pumps are off.

For the long term, the final depressurisation of the RCP [RCS] and the affected SG is performed by the mass transfer from the affected SG to the adjacent unaffected SG. If the affected SG draining is not sufficient for the depressurisation, there will be a transitory VDA [MSRT] opening (or VIV [MSIV] by-pass opening) with a temporary steam release, but without any liquid release from the affected SG to atmosphere.

2.18.5. Short term study

2.18.5.1. Case 1: without LOOP

2.18.5.1.1. Choice of single failure and preventative maintenance

The single failure is assumed to be on the Main Steam Relief Control Valve (MSRCV) of the affected SG, which remains stuck in its initial position, fully open. This means that, at first demand on the MSRIV, following turbine trip, the VDA [MSRT] of the affected SG remains stuck open.

The preventative maintenance is assumed on 1 ASG [EFWS] pump injecting into the affected SG to maximise the SG level reduction and the steam release.

The assumptions for single failure (SF) and preventative maintenance (PM) are summarised in Appendix 14B.2.18 - Table 7.

2.18.5.1.2. Initial state

The initial state conditions, given in Appendix 14B.2.18 – Table 1 are chosen in order to maximise the power to be removed and to maximise the pressure difference between the RCP [RCS] and the SG.

2.18.5.1.3. Specific assumptions

a) Neutronic data

Core power is assumed constant at 102% of full power prior to reactor trip. After RT, the maximum residual heat curve as described in section 0.2.4 of this appendix is considered for terms B+C (fission products).

Term A (fission heat) is provided as an input to the CATHARE code, with the CATHARE point kinetic model turned off. This term A is derived using a decoupled conservative RT-simulation, see Appendix 14B.2.2 – Table. 1.

b) Assumptions related to the control systems (NC)

Turbine: Turbine control is assumed to maintain turbine flow rate at 102% nominal flow rate until turbine trip on reactor trip.

GCT [MSB]: Not considered.

ARE [MFWS]: SG level control is in operation at when the SGTR occurs and operates correctly until isolation on turbine trip.

RCV [CVCS]: Pressuriser level control is assumed to work correctly, maximising the pressure difference between the RCP [RCS] and the affected SG. The charging flow rate is maximised with two charging pumps in operation and letdown isolated on a “pressuriser level < 2.09 m” signal. The RCV [CVCS] charging line is isolated manually by the operator at 30 minutes after RT, based on signals which uniquely identify the SGTR, SG levels, activity in the secondary side.

Heaters/spray: Pressuriser pressure control is assumed to operate, maximising the pressure difference between the RCP [RCS] and the affected SG.

The heater power is assumed to be at the maximum value with all heaters actuated, while spray flow rate is not considered. The heaters are turned off when the pressuriser empties.

c) Assumptions related to F1 systems

Reactor trip (F1A): This occurs on a “pressuriser pressure < MIN2 (135 - 1.5 bar)” signal with a maximum delay. This maximises the activity contained in the affected SG when steam is released to atmosphere, i.e. at VDA [MSRT] actuation.

VDA [MSRT] (F1A): The VDA [MSRT] setpoint is minimised on the affected SG and maximised on the other SG to maximise the steam release from the affected SG.

On the affected SG, the VDA [MSRT] is actuated after RT at a pressure setpoint of 93.0 - 1.5 bar and remains stuck open (SF) until isolation of the MSRIV on a “SG pressure < MIN2” signal with a setpoint of 40 - 1.5 bar.

On the unaffected SG, the VDA [MSRT] are actuated after RT at a pressure setpoint of 93.0 + 1.5 bar and remain at this pressure level until the beginning of the partial cooldown which is performed following a SI signal. The VDA [MSRT] setpoints are then decreased from 94.5 bar to 61.5 bar by the end of the partial cooldown.

VIV [MSIV] (F1A): the VIV [MSIV] on all the SG are closed either on a DP/DT signal, "SG pressure drop < max1" or on a SG low pressure signal "SG pressure < 50.0 + 1.5 bar" These arise following the depressurisation of the affected SG by the VDA [MSRT]. The DP/DT signal and SG low pressure signal uncertainties and delays are chosen to initiate an early MS isolation. This maximises the amount of steam released to atmosphere from the affected SG.

MHSI (F1A): a safety injection signal is actuated on "pressuriser pressure < MIN3" with a setpoint of 115 + 1.5 bar. This signal initiates a partial cooldown, and MHSI injection with a maximum flow rate. This maximises the pressure difference between the RCP [RCS] and the affected SG as soon as the RCP [RCS] pressure goes below the maximum MHSI shutoff head of 92 bar (see section 0.2 of this appendix).

ASG [EFWS] (F1A): the ASG [EFWS] is actuated on the unaffected SG on a "SG level < MIN2" signal with a setpoint of 8m - 2% Wide Range span. A minimum flow rate is assumed. The ASG [EFWS] is assumed to not actuate on the affected SG because of maintenance.

MSSV (F1A): in order to confirm the non actuation of the MSSV during the transient, a minimum setpoint of 102.5 - 1.5 bar is assumed.

2.18.5.2. Case 2: with LOOP

2.18.5.2.1. Choice of single failure and preventative maintenance

This case is studied to verify that no SG overfilling, and thus no liquid release, occurs before the rupture flow is terminated.

For the overfilling assessment, the choice is governed by the long term study as the single failure and the preventative maintenance has a negligible impact on the filling of the affected SG, (see section 2.18.6.2.1 of this appendix):

- SF on 1 diesel : loss of 1 safety division (division assigned to one unaffected SG)
- PM on 1 diesel : loss of 1 safety division (division assigned to another unaffected SG)

The assumptions for SF and PM are summarised in Appendix 14B.2.18 - Table 8.

2.18.5.2.2. Initial conditions

The initial conditions given in Appendix 14B.2.18 - Table 2 are chosen to maximise the initial water content in the affected SG, to maximise the pressure difference between the RCP [RCS] and the SG, and to minimise the power to be removed via the affected SG.

2.18.5.2.3. Specific assumptions

a) Neutronic data

The initial core power is 2%, decreasing following the conservative decay heat curve presented in section 0.2 of this appendix.

b) Assumptions related to control systems (NC)

GCT [MSB]: GCT [MSB] is assumed to be available to remove residual heat, maximising the pressure difference between the RCP [RCS] and the affected SG, until LOOP occurs. The LOOP leads to the loss of condenser after a short delay. It is conservatively assumed in this study that the GCT [MSB] is lost at the time of LOOP occurs, i.e. with a zero delay.

ARE [MFWS]/AAD [SSS]: SG level control by the ARE [MFWS] or AAD [SSS] is in operation when the SGTR occurs and continued to work correctly until LOOP occurs following reactor trip.

RCV [CVCS]: The RCV [CVCS] is assumed to work correctly as for case 1 section 2.18.5.1.3 of this appendix). RCV [CVCS] charging is isolated following a "SG level > MAX2" signal with a setpoint of 18 m + 2% Narrow Range (NR) span if partial cooldown has finished.

Heaters/spray: Pressuriser pressure control is assumed to work as for case 1, section 2.18.5.1.3 of this appendix).

c) Assumptions related to F1 systems

Reactor trip (F1A): a reactor trip signal occurs following a “SG level > MAX1” signal with a setpoint of 17.2 m + 2% NR, but initiates no specific action, except that LOOP is assumed to occur at that time.

VDA [MSRT] (F1A): the VDA [MSRT] setpoint is minimised on the affected SG and maximised on the other SG in order to maximise the pressure difference between the RCP [RCS] and the SG.

On the affected SG, the VDA [MSRT] is actuated after RT at a pressure setpoint of 93.0 - 1.5 bar. It remains at this pressure until the beginning of the partial cooldown, which is initiated following a “SG level > MAX2” signal with a setpoint of 18.1 m + 2% Narrow Range span, which, occurs earlier than the SI signal. The VDA [MSRT] setpoint is then decreased from 91.5 bar to 58.5 bar by the end of the partial cooldown.

On the unaffected SG, the VDA [MSRT] are actuated after RT, at a pressure setpoint of 93.0 + 1.5 bar, and remain at this pressure until the beginning of the partial cooldown. The VDA [MSRT] setpoints are then decreased from 94.5 bar to 61.5 bar by the end of the partial cooldown.

The VDA [MSRT] setpoint on the affected SG is automatically raised to 98 bar on the signal:

- SG level > MAX2 and,
- Partial Cooldown already finished.

VIV [MSIV] (F1A): The VIV [MSIV] on the affected SG is closed on the same signal as the VDA [MSRT] setpoint increase discussed above.

MHSI (F1A): the same assumption as discussed in section 2.18.5.1.3, except that only 2 MHSI pumps inject into the RCP [RCS] because of the assumption of a single failure and preventative maintenance. Assuming 4 MHSI pumps operate as opposed to 2 would have a negligible impact on the results as only a slightly higher RCP [RCS] pressure would occur, resulting in a slightly increased SGTR flow rate. This would lead to a slightly earlier "SG level MAX2" signal for partial cooldown and isolation of the affected SG.

ASG [EFWS] (F1A): The ASG [EFWS] is assumed to be actuated manually by the operator when LOOP occurs, as soon as the SG level falls below the nominal level. Delivery only occurs to the affected SG and one unaffected SG, the other ASG [EFWS] trains being affected by the assumed SF and PM. SG level is then maintained at the reference value by the operator. This operation is undertaken on a SG specific basis.

This assumption has little impact on the transient as the SG level in the affected SG always remains higher than max 1 level following the LOOP.

MSSV (F1A): the same assumptions as case 1 discussed above.

2.18.5.3. Results

The sequence of events for case 1 is given in Appendix 14B.2.18 - Table 3.

The sequence of events for case 2 is given in Appendix 14B.2.18 - Table 4.
(Case 2 short term calculation ends at leak termination, approximately 2660 seconds).

The figures describe the transient variation of the following main thermal-hydraulic parameters:

- Pressuriser and SG pressures,
- RCP [RCS] inlet and outlet flow rates,
- Cold/hot leg temperatures,
- Affected SG level.

Appendix 14B.2.18 - Figures 1 to 4 are related to case 1.

Appendix 14B.2.18 - Figures 5 to 8 are related to case 2.
(Case 2 short term calculation ends at leak termination, approximately 2660 seconds).

The results of case 1 show that a maximum of 79 tons of steam is discharged from the affected SG to atmosphere during short term phase.

In addition, case 1 shows that, even at power, the MSSV setpoint is not challenged. This justifies the assumption that the single failure is on the VDA [MSRT] rather than the MSSV.

The results of case 2 show that at the end of short term phase, no SG overfilling occurs and thus, no liquid is discharged to the atmosphere from the affected SG. The maximum SG level at end of the short term phase at approximately 2660 seconds corresponds to a physical level of 19.3 m. This should be compared to the 21.2 m level for the top of SG, leaving a steam volume of 30 m^3.

The initial conditions of case 2 are chosen to minimise the steam release from the affected SG and maximise the filling of the. In this case, about 35 tons of steam is discharged, much lower than for case 1.

In both cases, only F1A systems, RT, partial cooldown via VDA [MSRT] and the MHSI, are used to reach the controlled state.

2.18.6. Long term study

2.18.6.1. Case 3 - without LOOP

2.18.6.1.1. Choice of single failure and preventative maintenance

The single failure is chosen to pessimise the capacity of the F1 systems which borate the RCP [RCS], and which cooldown the RCP [RCS] and the affected SG to LHSI/RHR connection conditions. This failure can be either:

- one RBS [EBS] pump to pessimise the RCP [RCS] boration, and consequently to pessimise the cooldown rate, which would be limited to 25°C/h,
- the VDA [MSRT] of one unaffected SG to pessimise the heat removal capacity for the RCP [RCS] cooldown, with potential reduction of RCP [RCS] cooldown rate at the end of the cooling phase when low SG pressures are reached.

Those 2 single failures are combined in the same accident analysis, simply to limit the number of cases analysed. This is possible because of the small impact of the VDA [MSRT] failure on the ASG [EFWS] tanks capacity assessment.

The preventative maintenance has no impact on the availability of the RBS [EBS] or VDA [MSRT] (see section 2.16.1.6.1 within this appendix). The preventative maintenance results in the unavailability of one ASG [EFWS] train associated to one unaffected SG (bounding for steam release).

The assumptions for SF and PM are summarised in Appendix 14B.2.18 - Table 9.

2.18.6.1.2. Conditions at beginning of long term phase

The assumptions made for the calculation of the short term transient in this study are identical to those described for case 1 in sections 2.18.5.1.2 and 2.18.5.1.3 of this appendix (102% FP initial state) with the exception of the assumptions on SF and PM.

The SF of one VDA [MSRT], due to the MSRIV failing to open, is assumed on an unaffected SG in case 3, instead of the VDA [MSRT] of the affected SG in case 1, due to the MSRCV stuck fully open. This is conservative for the long term study.

The PM is assumed to be on 1 ASG [EFWS] pump associated with one unaffected SG in case 3 instead of 1 ASG [EFWS] pump associated to the affected SG in case 1. This is conservative for the long term study.

2.18.6.1.3. Specific assumptions

The assumptions regarding the systems described in section 2.18.5.1.3 of this appendix remain valid (except SF and PM). The additional assumptions for the long term phase are the following:

MHSI (F1A): maintained in operation (4/4 trains) until the end of the RCP [RCS] cooldown phase until the RCP [RCS] temperatures in the unaffected loops $\leq$ 180°C. They are then shut down by the operator to initiate the depressurisation of the RCP [RCS] and the affected SG.

LHSI (F1A): 4/4 trains inject to the RCP [RCS] once the RCP [RCS] pressure reaches 20 bar.

VDA [MSRT] (F1B):
Unaffected SG: the VDA [MSRT] setpoint remains constant at 60 + 1.5 bar until the beginning of the cooldown initiated by the operator. To perform RCP [RCS] cooldown, the VDA [MSRT] setpoints are decreased to provide a RCP [RCS] cooldown rate of - 25°C/h until a pressure of 5 bar is reached in unaffected SG.

Affected SG: the VDA [MSRT] setpoint remains constant at 98 bar throughout the long term phase. This ensures that the VDA [MSRT] on the affected SG remain closed throughout the cooldown.

ASG [EFWS] (F1A): the ASG [EFWS] is assumed to be manually controlled by the operator during the long term phase. The operator maintains SG levels in the unaffected SG at their nominal value. In the affected SG, the level remains above the actuation setpoint for the ASG [EFWS].

RBS [EBS] (F1B): the RBS [EBS] is manually actuated by the operator 2 hours after RT, at the beginning of the cooldown. The system is aligned to provide injection to an unaffected loop. The RBS [EBS] injection is provided at a minimum flow rate of 2.8 kg/s with a concentration of 7000 ppm of enriched boron. This is for MOX fuel management, see section 0.2 of this appendix. This is equivalent to a concentration of 11750 ppm of natural boron).

The RBS [EBS] train is shut down after 6000 seconds. During this period, 7 tons of boric acid has been injected. This boron injection is sufficient to reach the required boron concentration for the safe shutdown state. It also assumes no boron injection by the MHSI during the short term phase. This is a conservative assumption for the assessment of the decoupling criterion.

SGBS SG-connecting line (F1B): after the completion of the RCP [RCS] cooldown and the shutdown of all the MHSI pumps, the operator opens the dedicated SGBS SG connecting line. This allows a water transfer from the affected SG, which remains at approximately 90 bar, to the adjacent unaffected SG, which is at approximately 5 bar.

Before the beginning the transfer, the operator isolates the unaffected SG used for the transfer by closing the VIV [MSIV] and increasing the VDA [MSRT] setpoint.

The SGBS dedicated line remains opened until the end of the transient.

2.18.6.2. Case 4 with LOOP

2.18.6.2.1. Choice of single failure and preventative maintenance

The single failure and the preventative maintenance are chosen to pessimise the capacity of the F1 systems used to borate the RCP [RCS], and to cooldown the RCP [RCS] and the affected SG to LHSI/RHR connection conditions :

- SF on 1 diesel : loss of 1 safety division (division associated with one unaffected SG)
- PM on 1 diesel : loss of 1 safety division (division associated with another unaffected SG)

The simultaneous loss of 2 diesels leads to the loss of 1 RBS [EBS] pump which is conservative as the cooldown rate is limited to 25°C/h.

As the loss of 1 VDA [MSRT] on 1 unaffected SG is conservative for the cooldown rate at the end of the cooldown, this additional single failure is combined with the diesel one in the same accident analysis. The only purpose for this combination is to limit the number of cases analysed. This is possible because of the small impact of the VDA [MSRT] failure on the assessment of the ASG [EFWS] tanks capacity.

The assumptions for SF and PM are summarised in Appendix 14B.2.18 - Table 8.

In the case of the failure of the VDA [MSRT] on the affected SG, the redundant means to depressurise the RCP [RCS] and the affected SG is to open VIV [MSIV] bypass on the affected SG, which is F1B classified. This depressurises the affected SG via the VDA [MSRT] on an unaffected SG. In order to cover all cases, the opening of the VDA [MSRT] on the affected SG in this phase is assumed. This gives a flow rate equivalent to the one that would have been obtained via VIV [MSIV] bypass. A VDA [MSRT] opening of approximately 25% of the total VDA [MSRT] capacity is required to provide this flow rate.

2.18.6.2.2. Conditions at the beginning of the long term phase

The long term phase starts at the end of the short term phase presented in case 2 above. Thus, the conditions at the beginning of the long term phase correspond to a maximum level in affected SG with maximum activity transferred through the leak.

2.18.6.2.3. Specific assumptions

The assumptions on the systems described in section 2.18.5.2.3 of this appendix remain valid.

The additional assumptions, related to the long term phase are as follows:

MHSI and LHSI (F1A): operate as in case 2 (section 2.18.5.2.3 of this appendix).

VDA [MSRT] (F1B): operates as in case 3 (section 2.18.6.1.3 of this appendix).

ASG [EFWS] (F1A): operates as in case 3 (section 2.18.6.1.3 of this appendix).

The operator re-aligns the ASG [EFWS] train associated with the affected SG to deliver to an unaffected SG. This action is performed one hour after the event identifying signal.

RBS [EBS] (F1B): operates as in case 3 (section 2.18.6.1.3 of this appendix).

SGBS SG-connecting line (F1B): operates as in case 3 (section 2.18.6.1.3 of this appendix).

VIV [MSIV] bypass (F1B): is opened to complete the depressurisation of the affected SG to 30 bar. This also allows the RCP [RCS] to further depressurise after the first depressurisation resulting from the SGBS SG-connecting line opening, when failure of the VDA [MSRT] to open is assumed.

2.18.6.3. Results

The sequence of events for case 3 is given in Appendix 14B.2.18 - Table 5.

The sequence of events for case 4 is given in Appendix 14B.2.18 - Table 4.

The figures representative of the main thermal-hydraulic parameter transient are:

- Pressuriser and SG pressures,
- RCP [RCS] inlet and outlet flow rates,
- Cold/hot leg temperatures,
- Affected SG level,
- Unaffected SG level for the SG used for the transfer.

Appendix 14B.2.18 - Figures 10 to 14 are related to case 3.

Appendix 14B.2.18 - Figures 5 to 9 are related to case 4.

The main results are as follows:

Boration:
The most onerous case for RCP [RCS] boration is case 4 "with LOOP". The target, at EOC for MOX core, the most conservative case, requires a concentration of 620 ppm at the safe shutdown state conditions of 150°C. The minimum boron concentration in the core at the LHSI/RHR connection condition is 1100 ppm. This is the minimum concentration during the transient from the end of RBS [EBS] injection and exceeds the required value.

ASG [EFWS]-tanks capacity:
The most onerous case for the ASG [EFWS] water content consumed is case 3 "without LOOP". In this case, 1420 tons of ASG [EFWS] have been used in reaching the safe shutdown state. This is below the total ASG [EFWS] tanks guaranteed capacity of 1500 tons. For case 3 the ASG [EFWS] tank depletion is very conservatively calculated because the cooldown is performed assuming 4 Reactor Coolant Pumps in operation, instead of the 2 assumed in the ASG [EFWS] design.

Radiological releases:

Reactor Coolant Pump on:

There is no release during the long term phase from the affected SG when LOOP is not assumed, case 3. The affected SG depressurisation by water transfer into the neighbour unaffected SG is sufficient to depressurise both the RCP [RCS] and the affected SG to the LHSI/RHR connection pressure range. There is no need to open the steam side of the affected SG to complete the RCP [RCS] depressurisation. LHSI operation is necessary to maintain the pressure required for Reactor Coolant Pump operation.

Reactor Coolant Pump off:

When LOOP is assumed, case 4, steam release from the affected SG is needed to complete the RCP [RCS] depressurisation. Depressurisation by water transfer to the adjacent unaffected SG is not permitting below 20 bar. As the depressurisation by opening the VDA [MSRT] or VIV [MSIV] by-pass line begins when the affected SG level is below the nominal level, there is no water release to the atmosphere. In order to limit as far as possible any risk of SGTR reverse flow, the operator completes the RCP [RCS] depressurisation to the LHSI pump injection pressure of 20 bar before performing VDA [MSRT] isolation or VIV [MSIV] by-pass line isolation. This last step leads to a steam release of 53 tons from the affected SG. The total ASG [EFWS] consumption is 800 tons by the end of the calculation.

In both cases, the safe shutdown state is reached using only F1 systems, the VDA [MSRT], RBS [EBS], MHSI, LHSI, ASG [EFWS], VIV [MSIV], VIV [MSIV] bypass and SGBS SG-connection. This is even with the most onerous single failure and preventative maintenance assumption. In addition, there is no demand on the MSSV.

2.18.7. Conclusion

The present analysis of the "SGTR-2 tubes" accident shows that despite of the most onerous single failure and preventative maintenance assumptions:

- the controlled state, where the SI and SGTR flow match, is reached using only F1A means:
  - RT for nuclear power shutdown,
  - ASG [EFWS] and VDA [MSRT] (incl. partial cooldown) for RCP [RCS] heat removal,
  - MHSI for RCP [RCS] water injection.
- the safe shutdown state (LHSI/RHR connection conditions) is reached using only F1A and F1B means :
  - VIV [MSIV] closure, ARE [MFWS]/AAD [SSS] /ASG [EFWS] isolation, VDA [MSRT] setpoint increase for affected-SG isolation,
  - MHSI and LHSI for RCP [RCS] water injection,
  - ASG [EFWS] and VDA [MSRT] for RCP [RCS] cooling,
  - RBS [EBS] for boration,
  - VIV [MSIV] bypass, SGBS SG-connection for RCP [RCS] and affected-SG depressurisation,
  - LHSI/RHR for long term heat removal.

For the assessment of the radioactivity release, the maximum amount of contaminated fluid released to the atmosphere from the affected SG is calculated with the most conservative single failure and preventative maintenance to be:

- with Reactor Coolant Pumps on (no LOOP) :
  - 79 tons [(1)] of steam released before the affected-SG isolation, with no further steam release until and after LHSI/RHR connection,
  - no liquid release[(2)].
- with Reactor Coolant Pumps off (LOOP):

[(1)] Corresponding to the single failure of MSRCV to close in the affected SG

[(2)] Therefore, for activity release calculation performed in Sub-chapter 14.6, the liquid release will be limited to the liquid carry-over within steam flow. A maximum steam moisture level of 0.25% is assumed.

- less than 79 tons of steam released before the affected-SG isolation, plus 53 tons of steam released during the RCP [RCS] and affected-SG depressurisation but only prior to LHSI/RHR connection,
- no liquid release (2).

During the course of the "SGTR-2 tubes" accident, there is no demand on any MSSV, and no overfilling of the affected SG. This is despite of the most conservative single failure and preventative maintenance assumptions. The affected-SG pressure has been maintained below or equal to the RCP [RCS] pressure throughout the transient. This ensures that the SGTR reverse flow has been limited as far as possible. The core remains covered throughout the transient. Thus core cooling is never impaired and no clad heat-up is experienced.

Therefore, all the decoupling criteria and targets are met.

APPENDIX 14B.2.18 – TABLE 1

Initial conditions - Cases 1 and 3
(SGTR 2 tubes Short term - Without LOOP)

Parameters	**Limiting values used**
Reactor coolant system	
◆ Initial reactor power (% of nominal power)	100 + 2 = 102
◆ Initial average RCP [RCS] temperature (°C)	311.25 - 2.5 = 308.75
◆ Initial reactor coolant pressure (bar)	155 + 2.5 = 157.5
◆ Reactor cooling flow (kg/s)	22240 (Thermal-hydraulic)
◆ Pressuriser water volume / level (m^3 / m)	43.4 / 7.41 (nominal + 5% MR)
◆ Initial boron concentration	10 ppm
Steam generators	
◆ Initial steam pressure (bar)	70.1
◆ Initial SG level (m)	16.54 (nominal + 5%NR)
Feedwater	
◆ Main feedwater flow (% of nominal flow)	100 + 2 = 102
◆ Initial ARE [MFWS] temperature (°C)	232

APPENDIX 14B.2.18 – TABLE 2

Initial conditions - Cases 2 and 4
(SGTR 2 tubes Short & long term - With LOOP)

Parameters	**Limiting values used**
Reactor coolant system	
◆ Initial reactor power (% of nominal power)	2
◆ Initial average RCP [RCS] temperature (°C)	301.2 - 2.5 = 298.7
◆ Initial reactor coolant pressure (bar)	155 + 2.5 = 157.5
◆ Reactor cooling flow (kg/s)	21715 (Thermal-hydraulic)
◆ Pressuriser water volume / level (m^3 / m)	24.5 / 4.37 (nominal + 5% MR)
◆ Initial boron concentration	10 ppm
Steam generators	
◆ Initial steam pressure (bar)	82.7
◆ Initial SG level (m)	16.54 (nominal + 5%NR)
Feedwater	
◆ Main feedwater flow (% of nominal flow)	2
◆ Initial ARE [MFWS] temperature (°C)	123

APPENDIX 14B.2.18 – TABLE 3

Sequence of events - Case 1
(SGTR 2 tubes Short term - Without LOOP)

0.0s	SGTR occurs, 2 tubes (4A)
	ARE [MFWS] control on,
	102 % FP
425.s	Low pressuriser level (< 2.1 m)
	Isolation of RCV [CVCS] letdown
563.s	RT signal on pressuriser pressure < MIN2 (135.0 - 1.5 bar)
	Turbine trip
	ARE [MFWS] isolation
	=> beginning of short term radiological release to atmosphere
577.s	VDA [MSRT] of affected SG opens and remains stuck open (SF)
725.s	SI signal on pressuriser pressure < MIN3 (115.0 + 1.5 bar)
	Actuation of Partial Cooldown
~ 1000 s (17 min)	SGTR leak flow matched by MHSI and RCV [CVCS] injection
	Controlled state reached
1265.s (21 min)	MIN2 SG level in SG2 (8.0 - 0.4 m)
1315.s (22 min)	MIN2 SG level in SG1 and SG4 (8.0 - 0.4 m)
1395.s (23 min)	SG low pressure < MIN1 (50 + 1.5 bar)
	closure of VIV [MSIV] of all SG
1515.s (25 min)	Pressure in SGa < MIN2 (40.0 - 1.5 bar)
	Isolation of affected SG :
	- VDA [MSRT] isolation in the affected SG (MSRIV closure)
	=> end of short term radiological release to atmosphere
	(79 ton of steam released to atmosphere from SGa during short term)
2360.s (39 min)	RT+30 minutes : beginning of operator action
2365.s (39 min)	RCV [CVCS] isolation by the operator
8000.s (2.2 h)	Leak cancellation
	End of short term calculation

SGa: affected SG
SG2: related to the unaffected SG attached to pressuriser loop
SG1 & SG4: related to the unaffected SG

APPENDIX 14B.2.18 – TABLE 4

Sequence of events - Cases 2 and 4
(SGTR 2 tubes Short & long term - With LOOP)

0 s	SGTR occurs, 2 tubes (4A)
	2 % FP,
	Pressuriser level control on
	Full heaters power on (2500 kW).
179 s	SG level MAX1 reached in affected SG (17.2m + 2% NR)
181 s	Turbine Trip
	ARE [MFWS] isolation
	Loss of offsite Power (LOOP)
	ASG [EFWS] actuated in one unaffected SG
	=> beginning of short term radiological release to atmosphere
223 s	Low pressuriser level (< 2.1 m)
	Isolation of RCV [CVCS] letdown
475 s	SG level MAX2 reached in affected SG (18.1 m + 2% NR)
	Automatic initiation of Partial Cooldown
570 s	Pressuriser is empty
	Heaters shut off
730 s	SI signal on pressuriser pressure < MIN3 (115.0 + 1.5 bar)
~ 1120 s (19 min)	SGTR leak flow compensated by MHSI and RCV [CVCS] injection
	Controlled state reached
1580 s (26 min)	End of Partial Cooldown
	SG level MAX2 reached in affected SG -2nd time - (18.1 m + 2% NR)
1590 s (26 min)	Isolation of the affected SG :
	- MSRV setpoint of SGa increased (98 bar)
	- VIV [MSIV] closure of SGa
	RCV [CVCS] isolation
	=> end of short term radiological release to atmosphere
2660 s (44 min)	**Leak termination**
	=> end of short term calculation
	(SGa physical level = 19.3 m)
3780 (1.1 hour)	Re-alignment of ASG [EFWS] associated with the affected SG to an unaffected SG (SG1)

APPENDIX 14B.2.18 – TABLE 4

	Beginning of cooldown by unaffected SG at 25°C/h
	Beginning of boration by 1 RBS [EBS] train
8790 s (2.4 h)	Core boron concentration : 280 ppm
14790 (4.1 h)	End of boration by 1 RBS [EBS] train
	Core boron concentration : 1100 ppm
26930 s (7.5 h)	End of cooldown by unaffected SG: SG pressure = 5 bar abs.
	MHSI shutdown (LHSI remain in operation)
	Isolation of SG4 (VIV [MSIV] closure and lift-up of VDA [MSRT] to 98 bar)
	Opening of the blowdown-line between affected SG and SG4
27450 s (7.6 h)	Affected SG level (WR) < nominal level
	Beginning of depressurisation of RCP [RCS] and affected SG by VDA [MSRT] of affected SG (with VIV [MSIV] bypass capacity)
30410 s (8.5 h)	End of depressurisation of RCP [RCS] and affected SG : closure of VDA [MSRT] of affected SG
	⇒ End of long term radiological release to atmosphere
	RCP [RCS] temperatures < 180°C, RCP [RCS] pressure = 20 bar < 30 bar
	→ LHSI/RHR connecting conditions are reached
	Safe shutdown state reached
39000 s (10.8 h)	End of long term calculation

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

APPENDIX 14B.2.18 – TABLE 5

Sequence of events - Case 3
(SGTR 2 tubes Long term - Without LOOP)

0.s	SGTR occurs, 2 tubes (4A) ARE [MFWS] control on, 102 % FP
425.s	Low pressuriser level (< 2.1 m) RCV [CVCS] letdown isolation
565.s	RT signal on pressuriser pressure < MIN2 (135.0 - 1.5 bar) Turbine trip ARE [MFWS] isolation => beginning of short term radiological release to atmosphere
820.s	SI signal on pressuriser pressure < MIN3 (115.0 + 1.5 bar) Automatic initiation of Partial Cooldown
~ 1300 s (22 min)	SGTR leak flow matched by MHSI and RCV [CVCS] injection **Controlled state reached**
1600 s (27 min)	MIN2 SG level in SG2 (8.0 - 0.4 m) ASG [EFWS] actuation in SG2
1645 s (27 min)	MIN2 SG level in SG1 (8.0 - 0.4 m) ASG [EFWS] actuation in SG1
2365 s (40 min)	Time of operator action : isolation of the affected SG - MSRV setpoint of SGa increased (95 bar) - VIV [MSIV] closure of SGa RCV [CVCS] charging line isolation
7765 s (2.2 h)	Beginning of cooldown by unaffected SG at 25°C/h Beginning of boration by 1 RBS [EBS] train Core boron concentration : 606 ppm
13765 s (3.8 h)	End of boration by 1 RBS [EBS] train Core boron concentration : 1292 ppm
30700 s (8.5 h)	End of cooldown by unaffected SG : SG pressure = 5 bar 3 out of 4 Reactor Coolant Pumps shutdown MHSI shutdown (LHSI remain in operation) Isolation of SG2 (VIV [MSIV] closure and increase of VDA [MSRT] setpoint to 40 bar) Depressurisation of the affected SG by opening of the dedicated transfer line to SG2
~31800 s (8.8 h)	RCP [RCS] Temperatures < 180°C, RCP [RCS] pressure < 30 bar => LHSI/RHR connecting conditions are reached **Safe shutdown state reached**
~31850.s (8.8 h)	RCP [RCS] < 20 bar LHSI injection
36000.s (10 h)	End of long term calculation : SGTR leak termination

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

APPENDIX 14B.2.18 – TABLE 6

SGTR 2 tubes: Definition of studied cases

	SGTR 2 TUBES (4A) - SECTION 2.18			
	Initiator	Controlled State	Safe shutdown	**Comments**
Case 1 (short term)	102 % FP Without LOOP Calculation of maximum steam release	SGa isolation ↓		Calculation until reaching SGa isolation in order to evaluate max short term steam release (activity) Conservative assumption : max power
Case 2 (short term)	2% FP With LOOP Verification of no overfilling of affected SG	Leak termination ↓		Calculation until reaching SGTR leak termination in order to demonstrate no overfilling Conservative assumptions : - max SG water content at t = 0 - min power for min release
Case 3 (long term)	102% FP Without LOOP Depressurisation of SG a by transfer to unaffected SG Capability of F1 systems to reach safe shutdown (adequacy of ASG [EFWS] tank capacity)			Calculation until reaching LHSI/RHR connection in order to validate EFW-tanks capacity Conservative assumptions : - max power - min cooldown gradients
Case 4 (long term)		Restart of case 2 (with LOOP) Depressurisation of SGa ↓ Max steam release Verification of no liquid release to atmosphere		Restart of case 2 Calculation until reaching LHSI/RHR connection conditions to evaluate max long term release Conservative assumptions: - max SG water content - LOOP for SGa depressurisation (need for affected SG draining before VDA [MSRT]a opening)

APPENDIX 14B.2.18 – TABLE 7

Single Failure and Preventative Maintenance Assumptions - Case 1
(SGTR 2 tubes Short term - Without LOOP)

MSRCV	AVAILABLE	AVAILABLE	**SF STUCK OPEN**	AVAILABLE
MSRIV	AVAILABLE	AVAILABLE	AVAILABLE	AVAILABLE
	SG1	**SG2**	**SG3 SGTR**	**SG4**
ASG [EFWS]	AVAILABLE	AVAILABLE	**PM**	AVAILABLE
	LOOP 1	**LOOP 2**	**LOOP 3**	**LOOP 4**
MHSI	AVAILABLE	AVAILABLE	AVAILABLE	AVAILABLE

OFF SITE POWER	AVAILABLE

APPENDIX 14B.2.18 – TABLE 8

Single Failure and Preventative Maintenance Assumptions - Cases 2 and 4
(SGTR 2 tubes Short & Long term - With LOOP)

MSRCV	AVAILABLE	AVAILABLE	**AVAILABLE setpoint > MHSI delivery pressure**	AVAILABLE
MSRIV	AVAILABLE	AVAILABLE	AVAILABLE	**SF to open** (1)

ASG [EFWS]	**UNAVAILABLE loss of diesel**	**UNAVAILABLE PM**	AVAILABLE	AVAILABLE

	LOOP 1	LOOP 2	LOOP 3	LOOP 4
MHSI	**UNAVAILABLE loss of diesel**	**UNAVAILABLE PM**	AVAILABLE	AVAILABLE
LHSI	**UNAVAILABLE loss of diesel**	**UNAVAILABLE PM**	AVAILABLE (2)	AVAILABLE (2)

RBS [EBS]	**UNAVAILABLE PM**	AVAILABLE

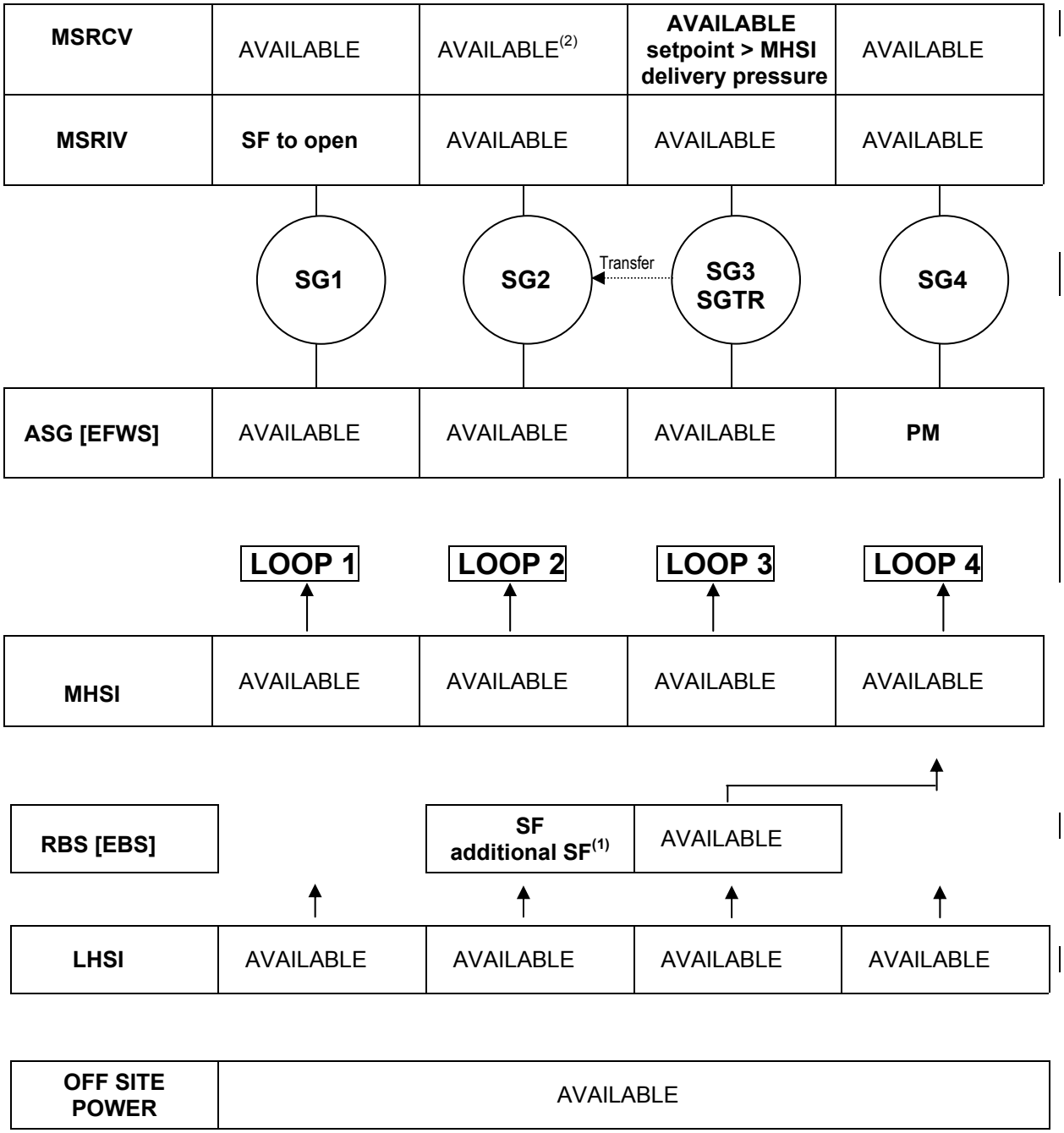
DIESEL	**SF**	**PM**	AVAILABLE	AVAILABLE

OFF SITE POWER	**UNAVAILABLE**

(1) additional SF to limit the number of case presented
(2) the safe shutdown state is achieved by 1 LHSI train in SI mode and the other train in RHR mode

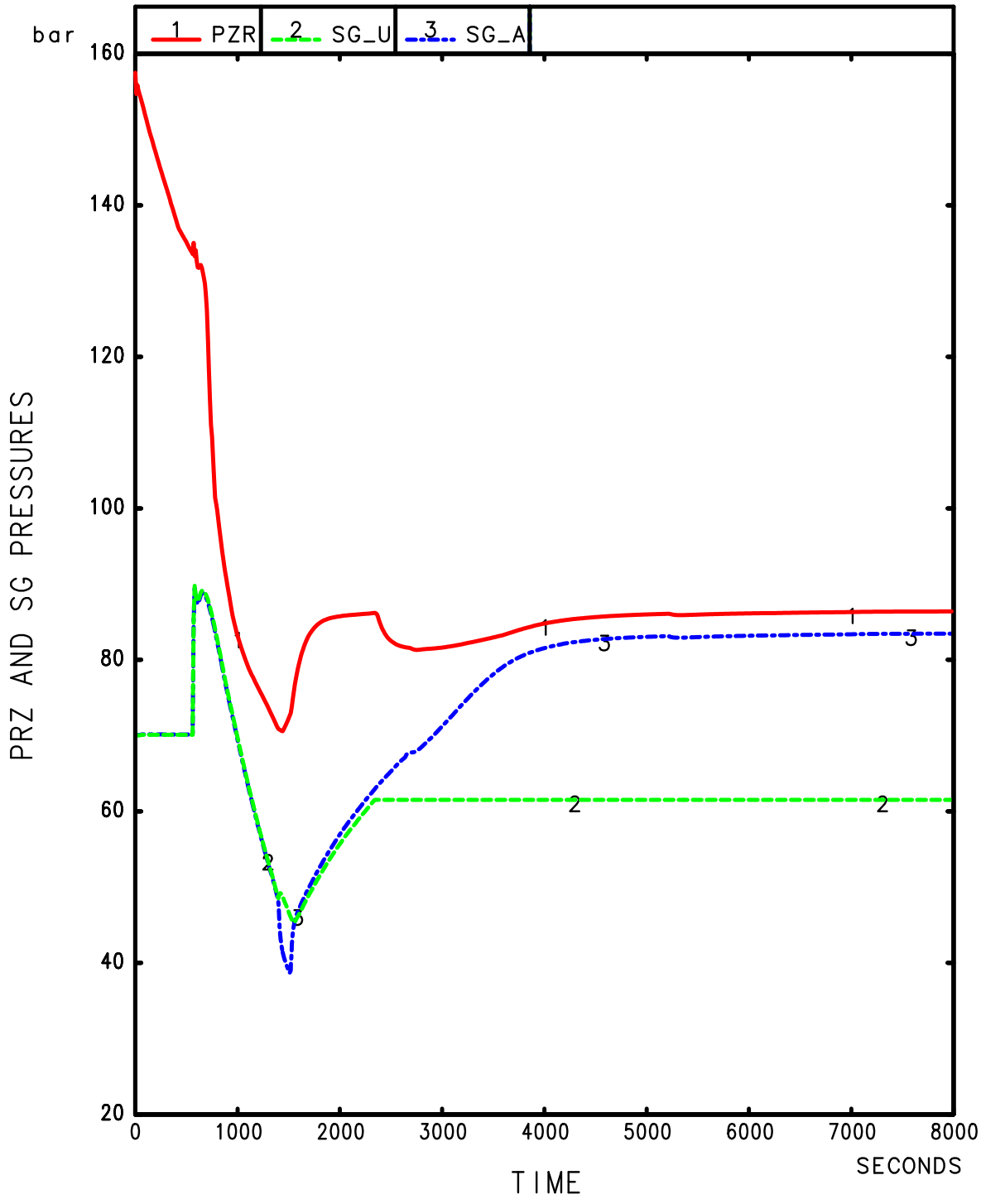
APPENDIX 14B.2.18 – TABLE 9

Single Failure and Preventative Maintenance Assumptions - Case 3
(SGTR 2 tubes Long term - Without LOOP)



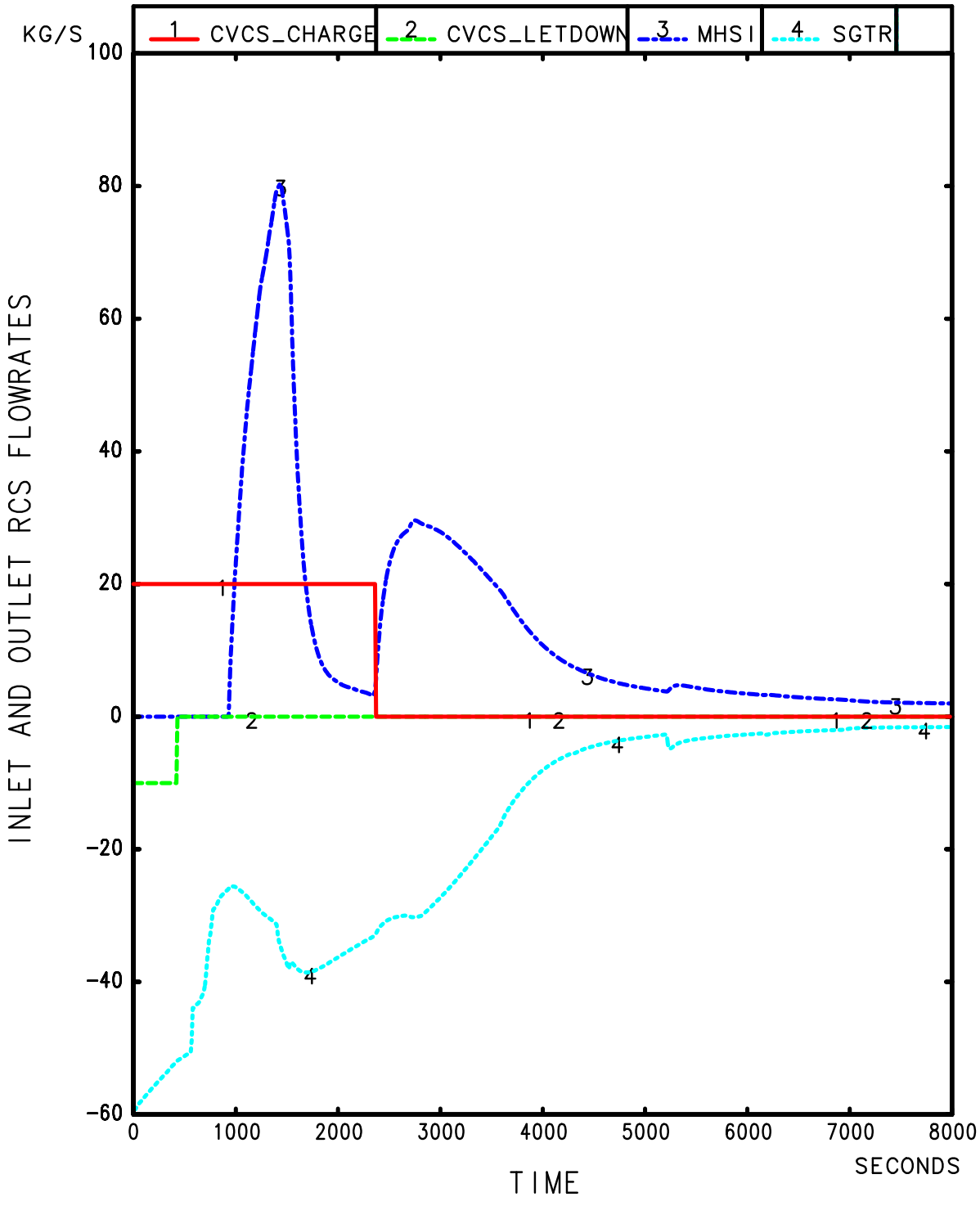
(1) additional SF to limit the number of case presented
(2) setpoint > MHSI delivery pressure at the beginning of the transfer

APPENDIX 14B.2.18 – FIGURE 1



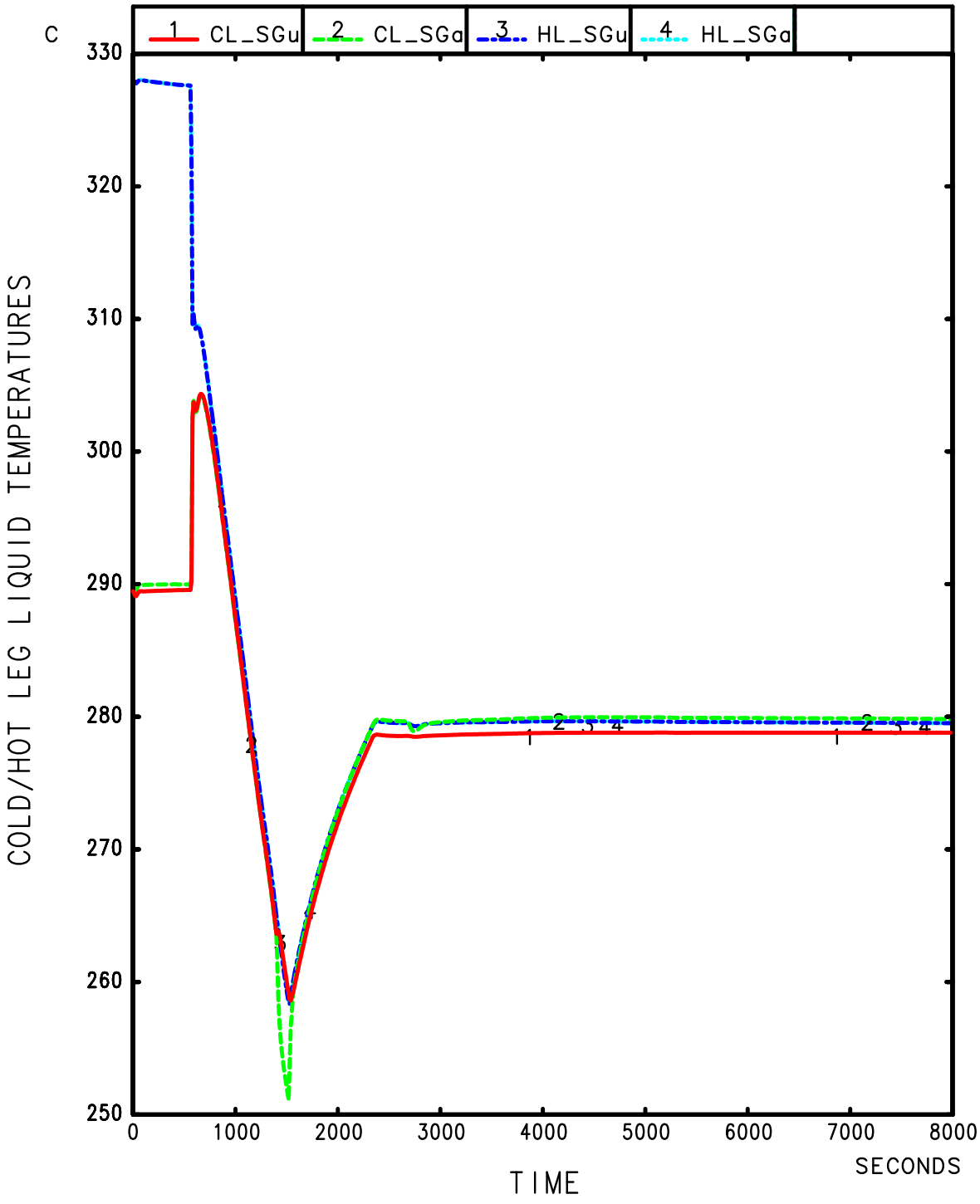
Pressuriser and SG pressures
SGTR 2 tubes / Case 1: Short term - Without LOOP

APPENDIX 14B.2.18 – FIGURE 2



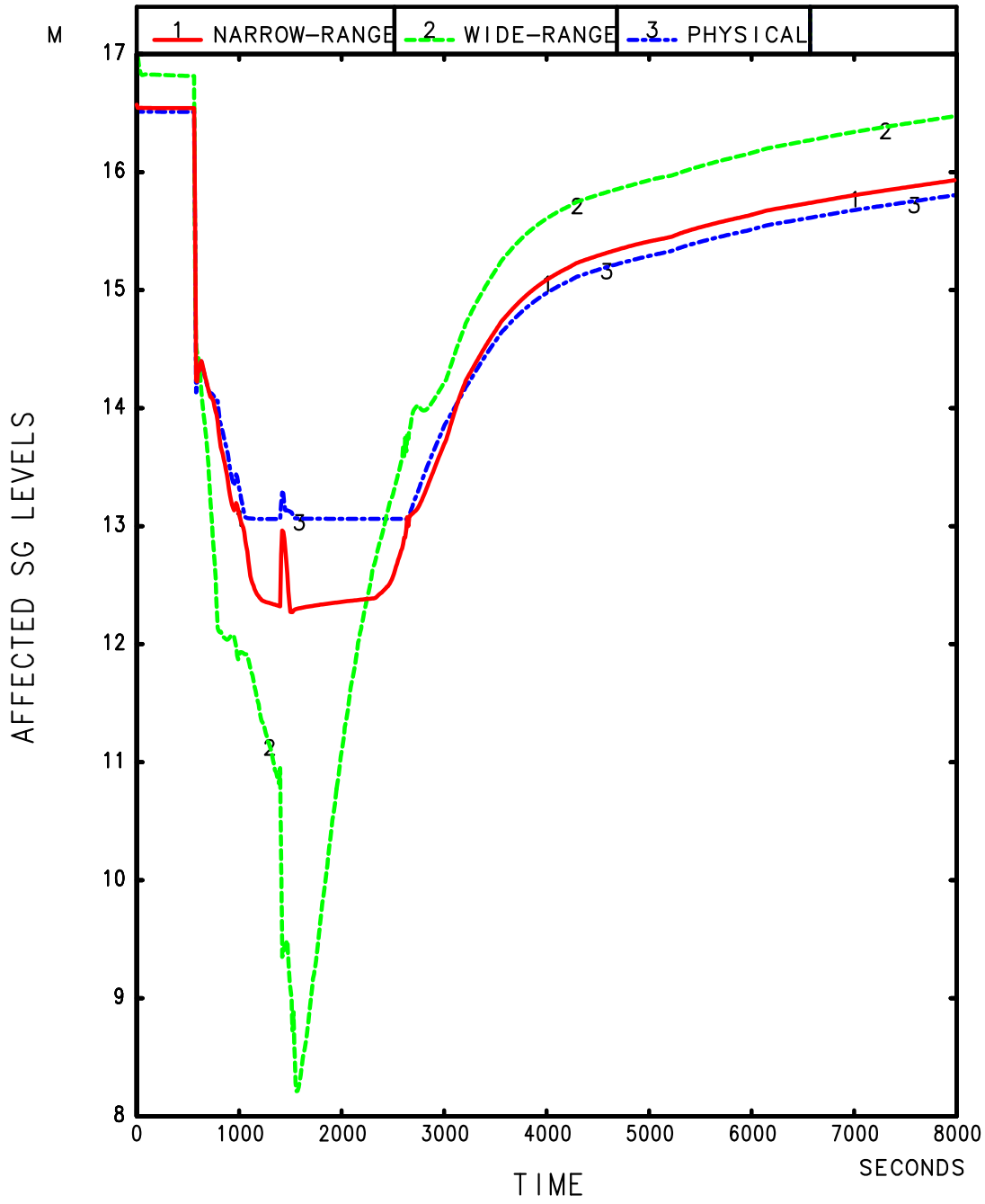
RCP [RCS] inlet and outlet flow rates
SGTR 2 tubes / Case 1: Short term - Without LOOP

APPENDIX 14B.2.18 – FIGURE 3



Cold/hot leg temperatures
SGTR 2 tubes / Case 1: Short term - Without LOOP

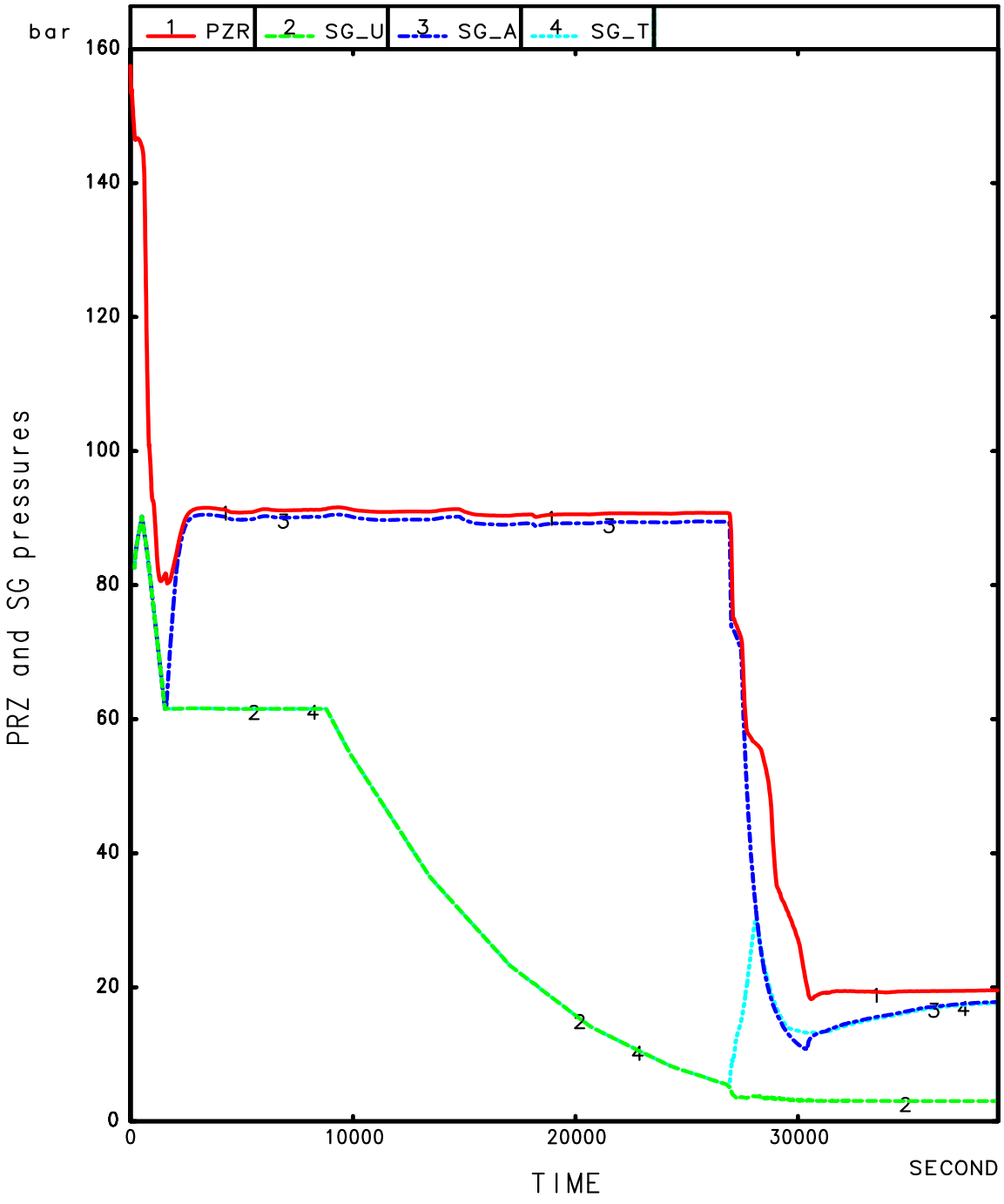
APPENDIX 14B.2.18 – FIGURE 4



Note : Physical level is valid only between 13 and 21.2 m (top of SG)

Affected SG level
SGTR 2 tubes / Case 1: Short term - Without LOOP

APPENDIX 14B.2.18 – FIGURE 5

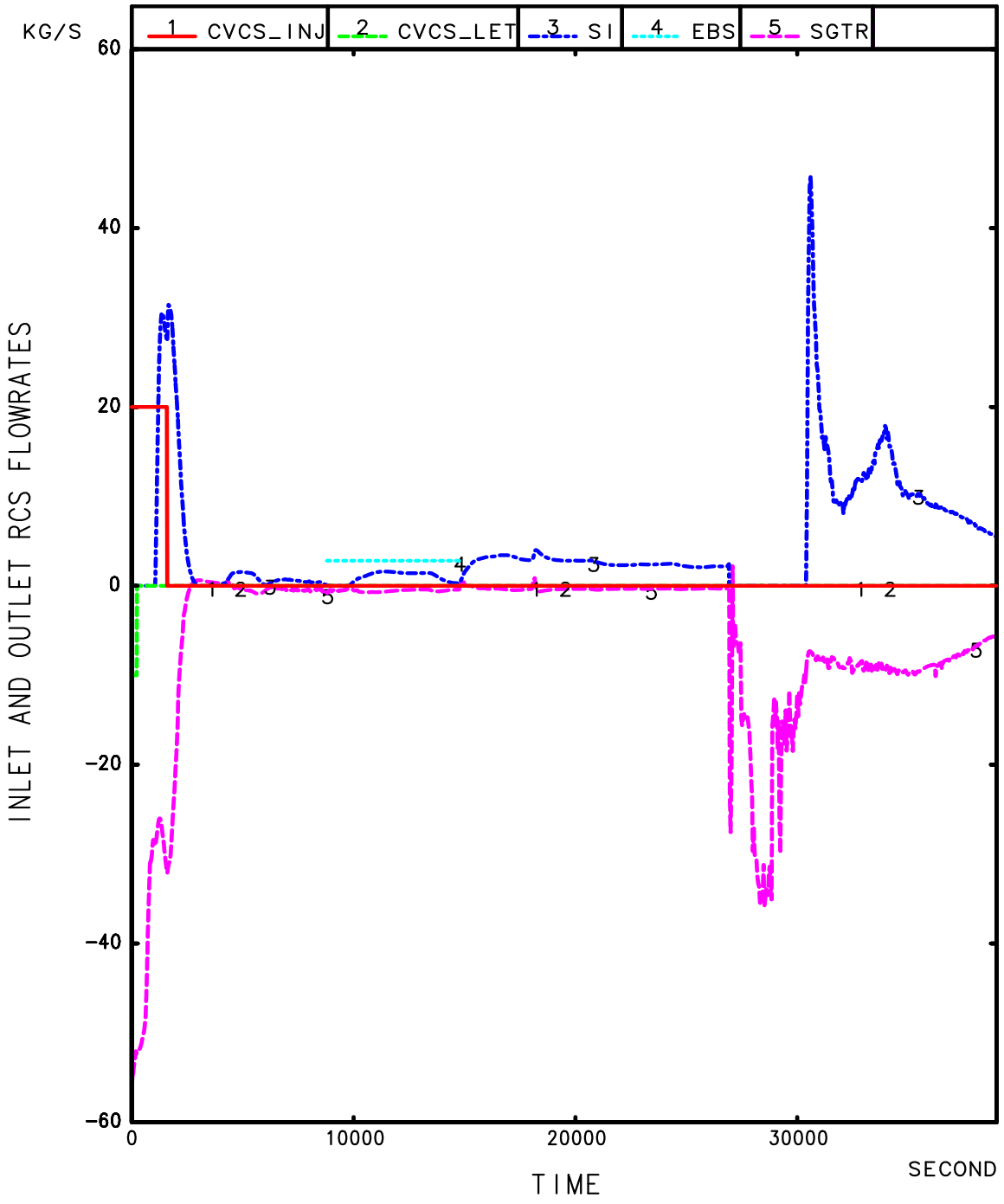


SG_U: Unaffected SG
SG_A: Affected SG

SG_T: Unaffected SG used for water transfer from SG_A

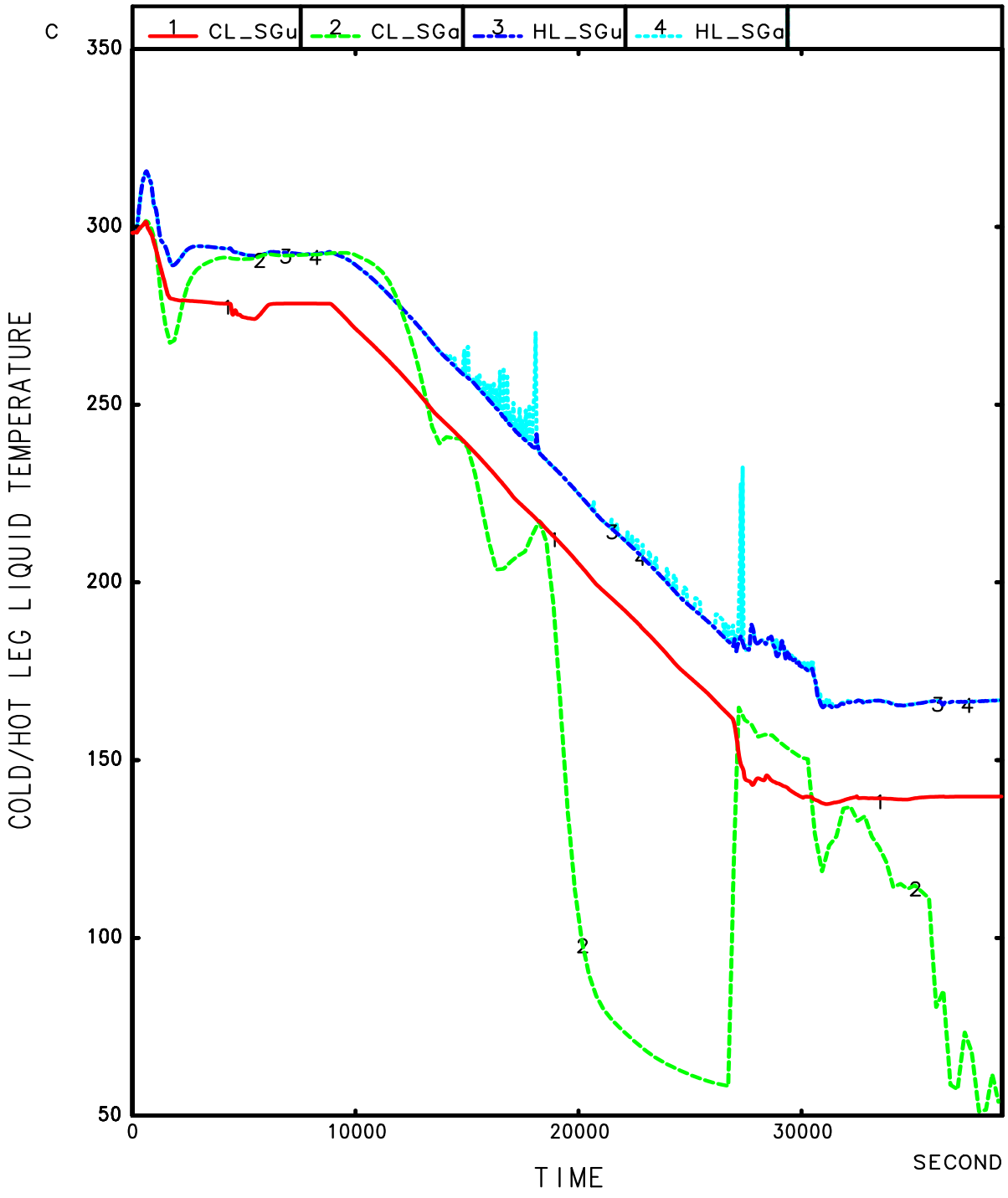
Pressuriser and SG pressures
SGTR 2 tubes / Cases 2 and 4: Short and long term - With LOOP

APPENDIX 14B.2.18 – FIGURE 6



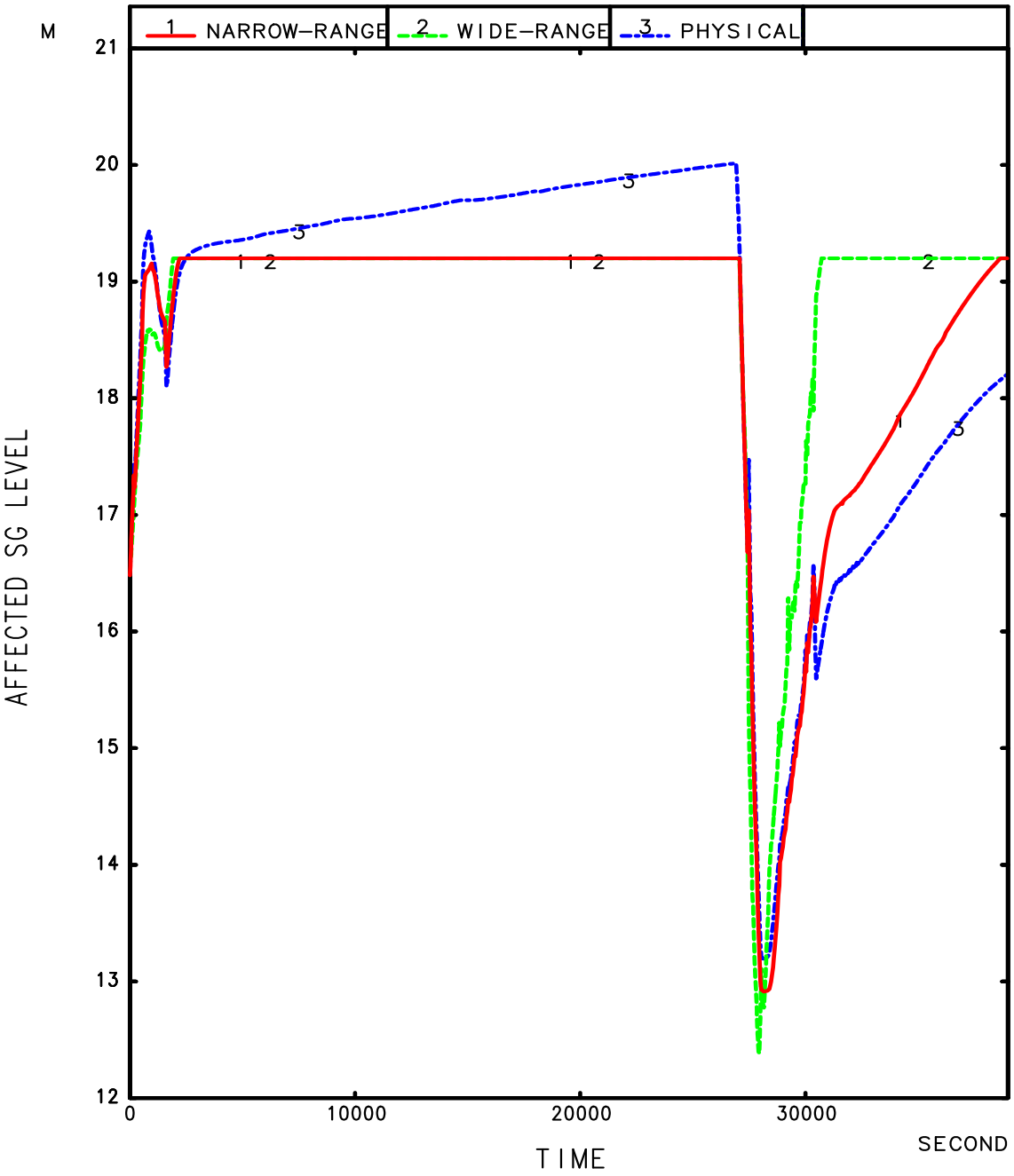
RCP [RCS] inlet and outlet flow rates
SGTR 2 tubes / Cases 2 and 4: Short and long term - With LOOP

APPENDIX 14B.2.18 – FIGURE 7



Cold/hot leg temperatures
SGTR 2 tubes / Cases 2 and 4: Short and long term - With LOOP

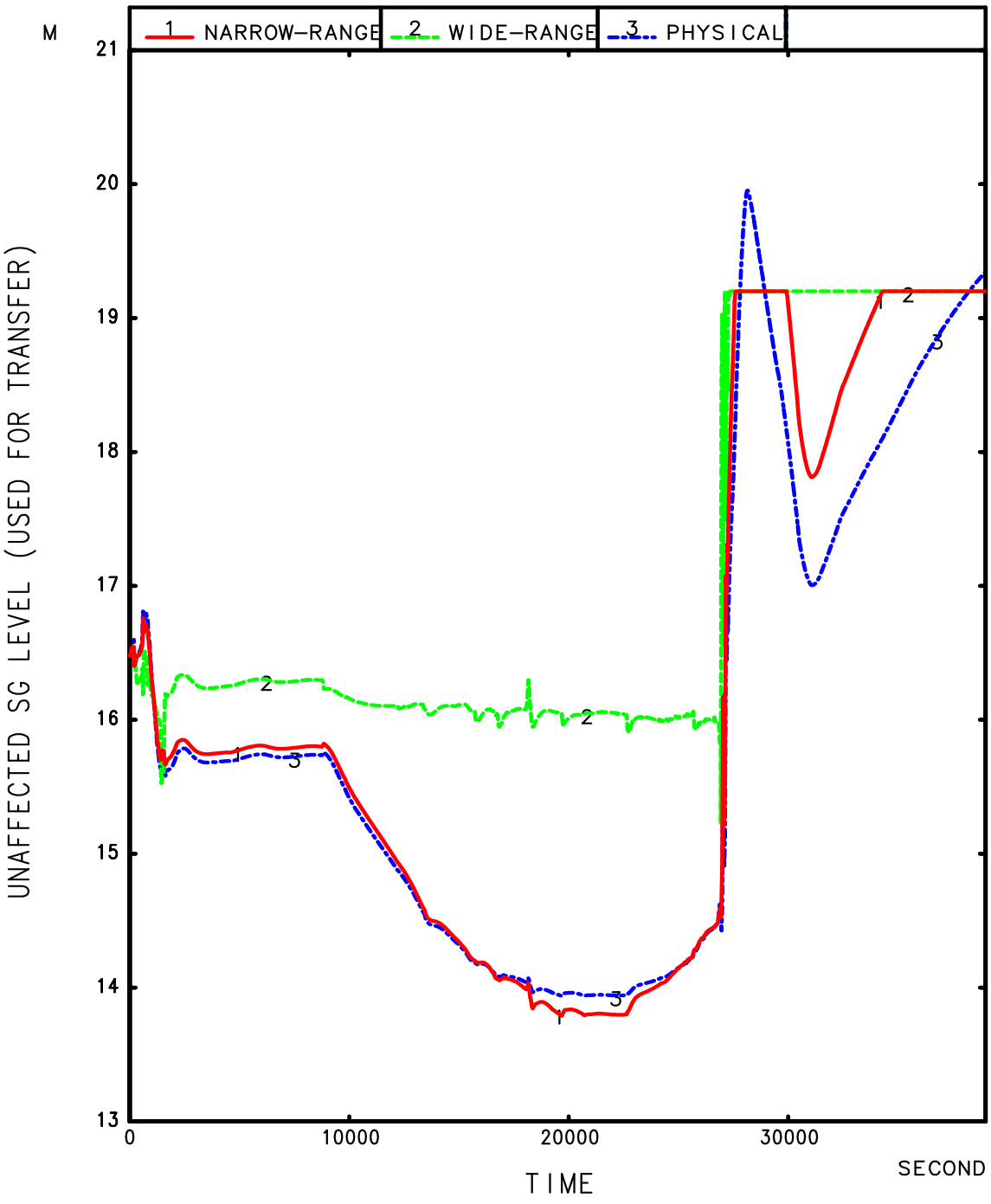
APPENDIX 14B.2.18 – FIGURE 8



Affected SG level
SGTR 2 tubes / Cases 2 and 4: Short and long term - With LOOP

Note : Top of SG corresponds to 21.2 m
Top of wide & narrow range corresponds to 19.2 m
Physical level is valid between 13 and 21.2 m

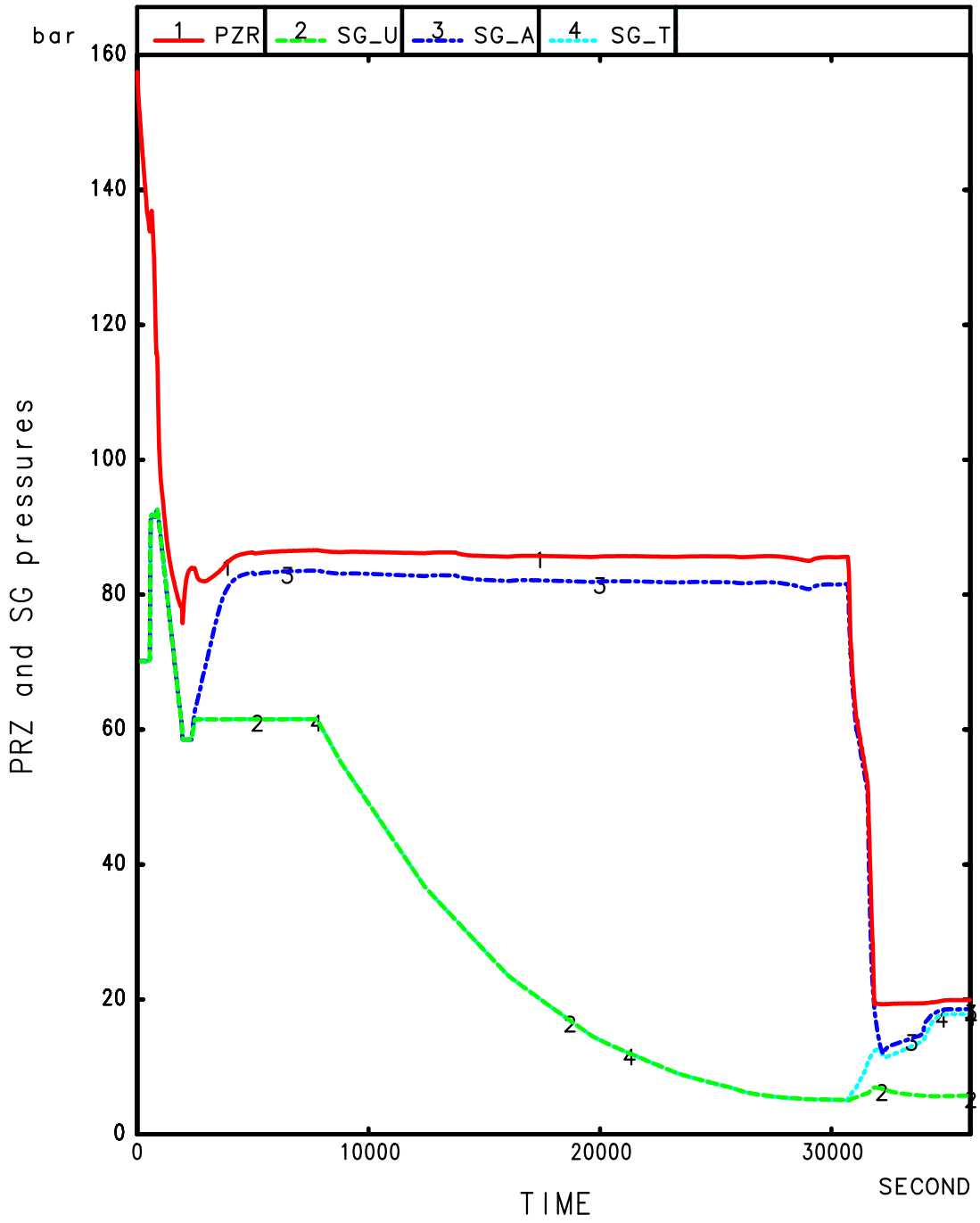
APPENDIX 14B.2.18 – FIGURE 9



Unaffected SG level (used for transfer)
SGTR 2 tubes / Cases 2 and 4: Short and long term - With LOOP

Note : Top of SG corresponds to 21.2 m
Top of wide & narrow range corresponds to 19.2 m
Physical level is valid between 13 and 21.2 m

APPENDIX 14B.2.18 – FIGURE 10

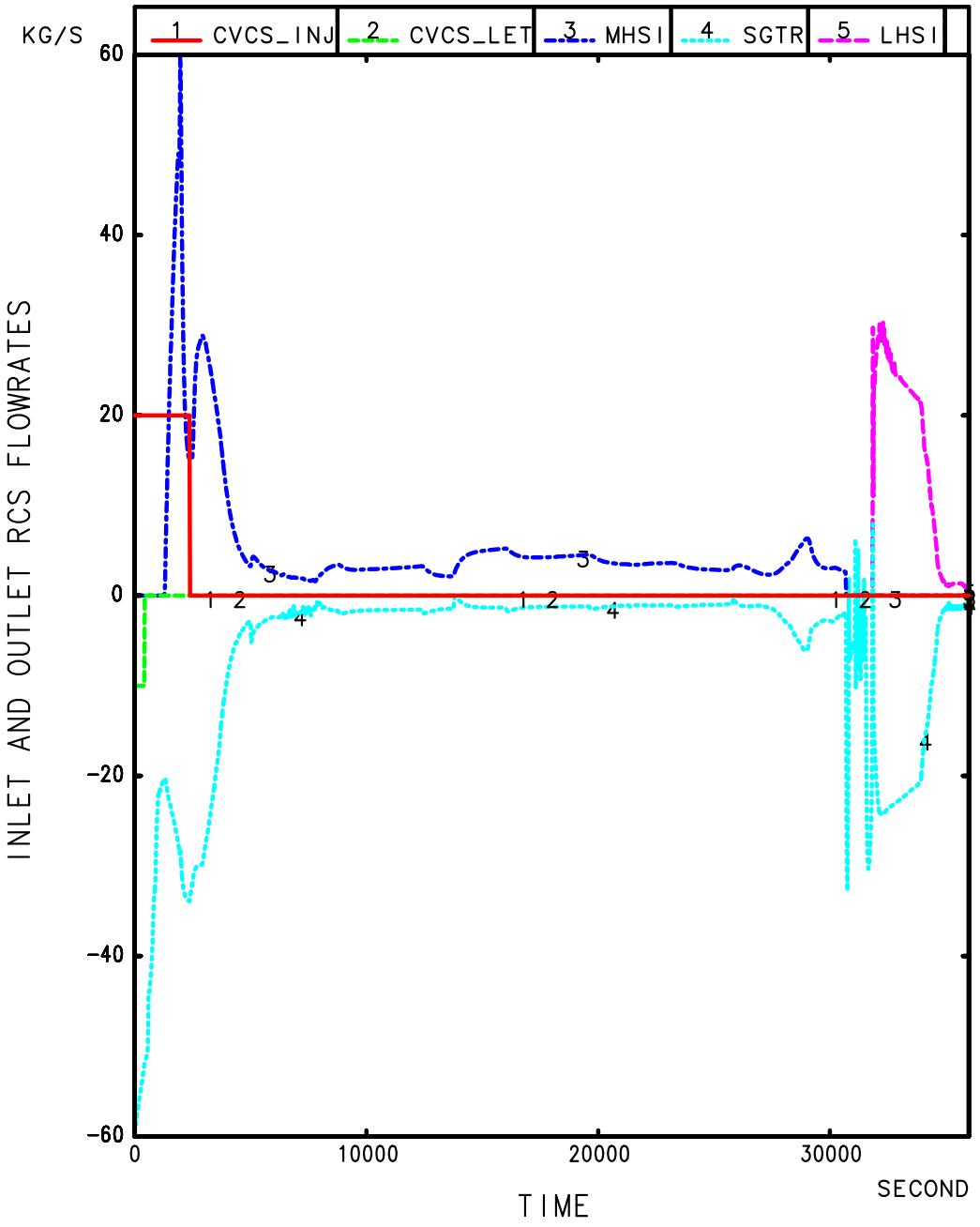


SG_U: Unaffected SG
SG_A: Affected SG

SG_T:Unaffected SG used for water transfer from SG_A

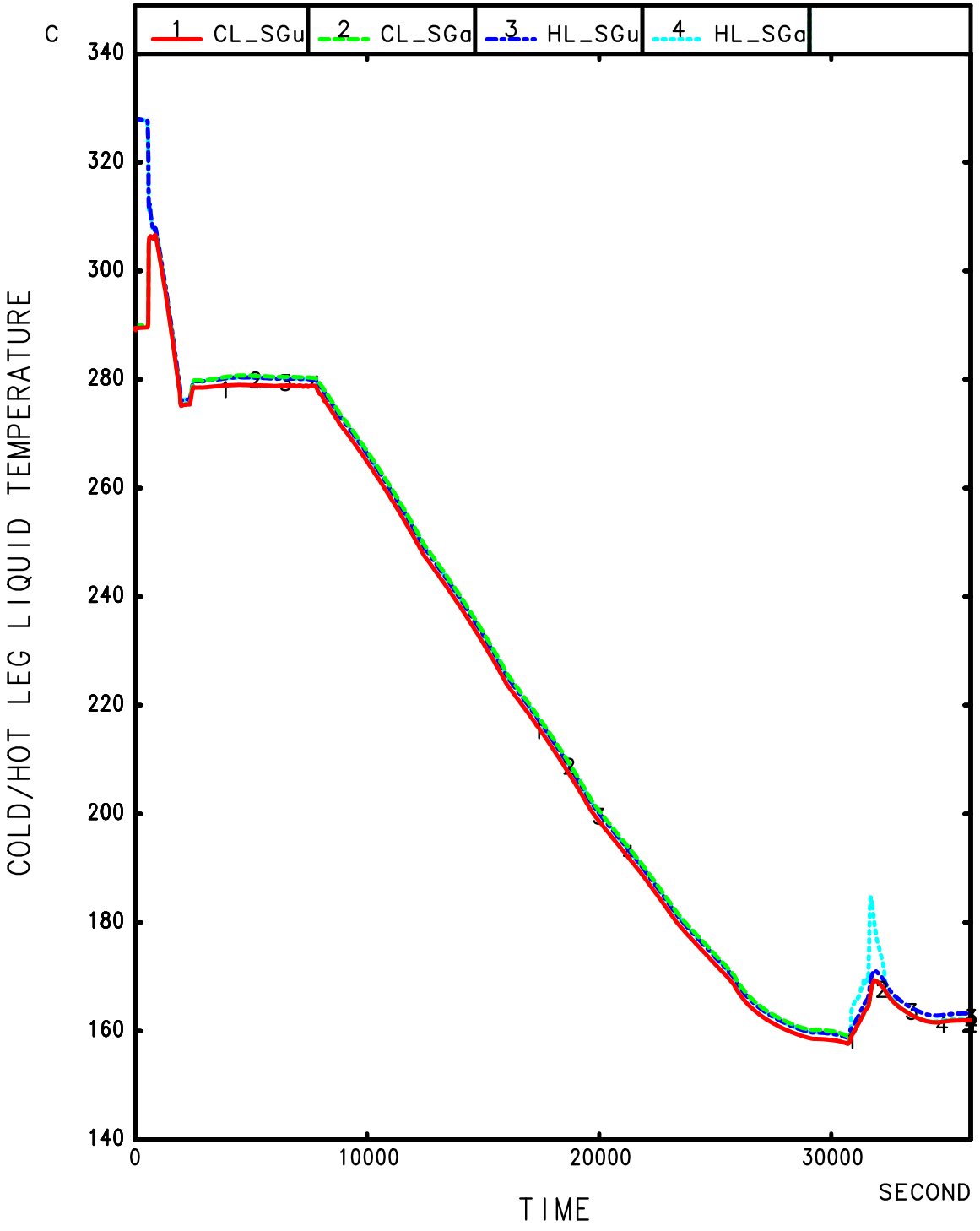
Pressuriser and SG pressures
SGTR 2 tubes / Case 3: Long term - Without LOOP

APPENDIX 14B.2.18 – FIGURE 11



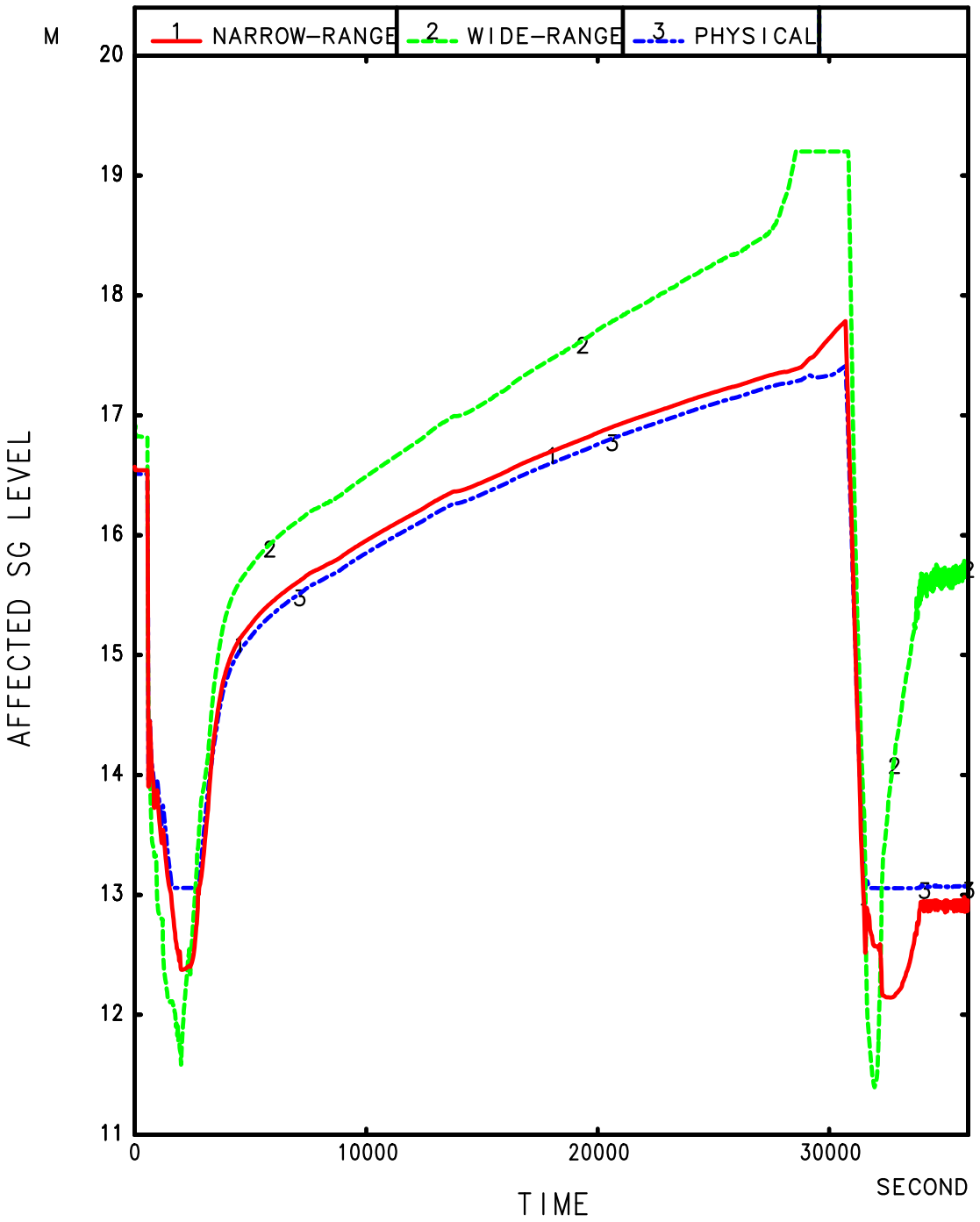
RCP [RCS] inlet and outlet flow rates
SGTR 2 tubes / Case 3: Long term - Without LOOP

APPENDIX 14B.2.18 – FIGURE 12



Cold/hot leg temperatures
SGTR 2 tubes / Case 3: Long term - Without LOOP

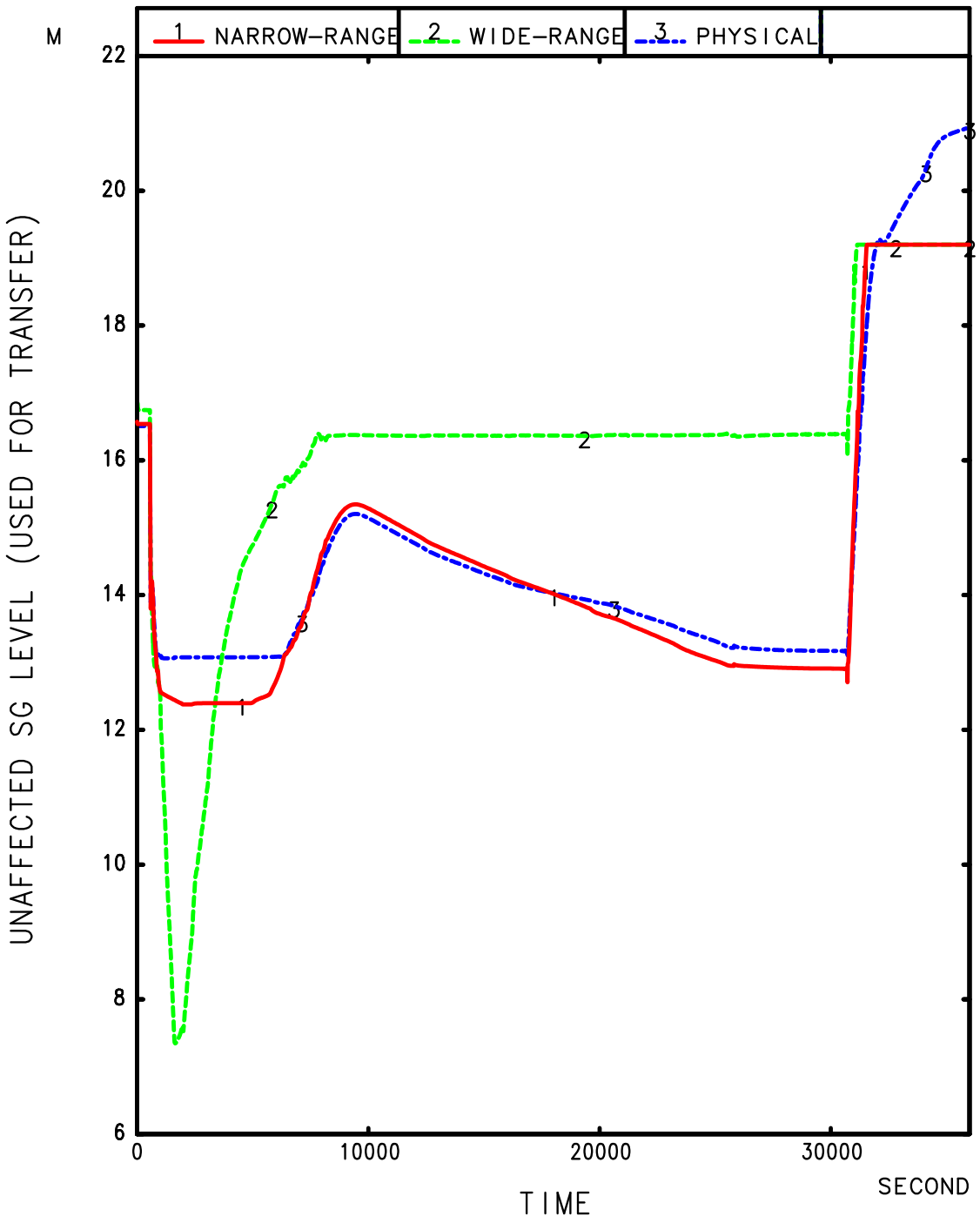
APPENDIX 14B.2.18 – FIGURE 13



Affected SG level
SGTR 2 tubes / Case 3: Long term - Without LOOP

Note : Top of SG corresponds to 21.2 m
Top of wide & narrow range corresponds to 19.2 m
Physical level is valid between 13 and 21.2 m

APPENDIX 14B.2.18 – FIGURE 14



Unaffected SG level (used for transfer)
SGTR 2 tubes / Case 3: Long term - Without LOOP

Note: Top of SG corresponds to 21.2 m
Top of wide & narrow range corresponds to 19.2 m
Physical level is valid between 13 and 21.2 m

2.2. SMALL BREAK LOCA

2.2.1. Small Break LOCA in state A (PCC-3)

The SB(LOCA) is defined as a break of equivalent diameter less than or equal to 50 mm (equivalent area less than or equal to 20 cm^2).

2.2.1.1. Identification of causes and accident description

2.2.1.1.1. General concern

The following small breaks are not analysed in this section:

- Steam Generator Tube Rupture which involves different physical phenomena. These transients are dealt with in the sections 2.17 and 2.18 of this appendix,
- Leaks on the RCP [RCS] which are compensated by RCV [CVCS].

A LOCA induces the following:

- Loss of reactor coolant inventory and possibly core heat-up,
- Containment loads by overpressurisation due to mass and energy release (see section 1 of Sub-chapter 6.2),
- Mechanical loads on RCP [RCS] components and their associated supports and structures,
- Mechanical loads on RPV internals.

A LOCA is classified as PCC-3 and PCC-4 event according to the break size and the plant initial state. This section only deals with the small break LOCA in state A, classified as a PCC-3 event (see section 0.1 in this appendix: scope of events).

2.2.1.1.2. Typical sequence of events

a) From initiation to controlled state

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

A reactor trip occurs on low pressuriser (PZR) pressure (< MIN2). The RT signal automatically trips the turbine and closes the ARE [MFWS] full load lines.

As the secondary side pressure increases, the Main Steam Bypass valves open initiating steam dump to the condenser. In case of unavailability of steam dump to the condenser (e.g. in case of LOOP), the Main Steam Relief Trains open permitting steam dump to atmosphere.

The SGs are fed by the ARE [MFWS] system through the low load lines. In case of unavailability of the ARE [MFWS] system, the start-up and shutdown pump starts and feeds the SGs through the low load lines. In case of unavailability of the AAD [SSS] (e.g. in case of LOOP), the Emergency Feed Water System is actuated on low SG level (< MIN2).

The Safety Injection signal is actuated on very low pressuriser pressure (< MIN3). The SI signal automatically starts the MHSI and LHSI pumps and initiates a partial cooldown of the secondary system. The partial cooldown cools the primary system and lowers the RCP [RCS] pressure.

During the partial cooldown, the RCP [RCS] pressure decreases sufficiently to allow MHSI injection into the cold legs. The partial cooldown is performed by all SGs via steam dump to condenser or to the atmosphere, by automatically decreasing the respective relief valve setpoints at the constant cooling rate of -100°C/h down to a fixed pressure value, low enough to permit the required MHSI injection and high enough to prevent core re-criticality.

After the partial cooldown the RCP [RCS] pressure remains constant, roughly at the same level than the secondary side, until the break energy flow rate becomes sufficient to remove the decay heat. As long as the MHSI flow rate is insufficient to compensate for the break flow rate, the RCP [RCS] inventory continues to decrease. During this phase the break flow is subcooled, eventually reaching saturation conditions.

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

Prior to the time when MHSI flow is able to compensate for the break flow, the core may become uncovered. As a consequence, the fuel clad temperature increases above the saturation level in the uncovered part of the core. The larger the core uncovery depth and duration, the higher the resulting clad temperature increase. Specific criteria (see section 2.2.1.2 of this appendix) must be met in order to avoid unacceptable core damage and to limit radiological consequences on the environment.

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

- reactivity under control,
- core power being removed,
- reactor coolant inventory stable or increasing.

b) From controlled state to safe shutdown

At the controlled state, core cooling is ensured by the RCP [RCS] water supply via the SIS and the RCP [RCS] heat removal via the break and the SGs if needed. This state cannot last indefinitely for the following reasons:

- ASG [EFWS] tank depletion,
- Continued increase of containment pressure and temperature.

A transfer to the safe shutdown is therefore required. In case of SB(LOCA), the safe shutdown is achieved when the following conditions are attained:

- the core is subcritical allowing for xenon depletion,

- the break flow is compensated by RIS [SIS] flow,
- The decay heat is removed by the cooling chain LHSI/RRI [CCWS]/SEC [ESWS] 1 (LHSI used in RHR mode) and partially by the break flow,
- The activity release is within the limits of PCC-3.

The transfer to LHSI/RHR connection conditions is thus necessary to reach the safe shutdown state. The sequences of actions to be performed are as follow:

- Boration phase: the RCP [RCS] must be borated sufficiently to keep the core subcritical throughout the transient up to the safe shutdown state.
  - For smaller break sizes[2] where MHSI boration is not sufficient, because of insufficient injection flow, RCP [RCS] boration is performed during RCP [RCS] cooldown by the RCV [CVCS] (via the charging line) or by the F1-A classified Extra Boration System (RBS [EBS]), with injection of boric acid at 7000 ppm enriched boron. The RCP [RCS] cooling rate is either -25°K/h (if 1 RBS [EBS] pump is in operation) or -50°K/h (if 2 RBS [EBS] pumps are in operation). The RBS [EBS] is designed so that RBS [EBS] boration matches the reactivity insertion resulting from the RCP [RCS] cooling rate.
  - For larger break sizes[3], MHSI boration is sufficient.
- RCP [RCS] cooldown: RCP [RCS] cooldown is manually initiated via the secondary side by decreasing the GCT [MSB] setpoint, if available, or by the VDA [MSRT] (F1-B classified). The operator manually controls the RCP [RCS] temperature. The minimum cooling rate assumed in the EPR-design is -25 °K/h provided this is not limited by the VDA [MSRT] or GCT [MSB] capacity. The Main Steam Header (MSH) is generally open: if necessary the operator is instructed to inhibit the MSH isolation on SG low pressure. In case of MSH isolation and failure of the VDA [MSRT], the redundant SG depressurisation system (VIV [MSIV] bypass, F1-B classified) would be used to depressurise the isolated SG.
- RCP [RCS] depressurisation: this is performed by shutting off the MHSI pumps at an RCP [RCS] hot leg temperature below 200°C (ensuring that LHSI injection is operational), provided that the LHSI pumps are already operating and the RCP [RCS] water inventory is sufficient. The Reactor Pressure Vessel Level (RPVL) measurements and ΔT_{sat}[1] measurements (F1-B classified) provide the required information on RCP [RCS] water inventory, in conjunction with emergency procedures.

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

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

[1] 1 out of 4 LHSI/RHR trains is sufficient to ensure the heat removal at a 180°C RCP [RCS] hot leg temperature.

[2] break sizes typically lower than 5 cm^2 (∅ 25 mm), with the RCP [RCS] in subcooled conditions.

[3] break sizes typically higher than 5 cm^2 (∅ 25 mm), with the RCP [RCS] in saturated or super-heated conditions.

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

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

Safe shutdown is thus ensured by control of RCP [RCS] water inventory via LHSI operating in SI-mode, and by control of RCP [RCS] temperature by the LHSI operating in RHR-mode. One LHSI/RHR train is sufficient to remove the decay-heat at 180°C. If necessary, with regard to the RCP [RCS] water inventory, LHSI injection can be supplemented by the MHSI system, operating in its appropriate configuration (large mini flow line open before LHSI/RHR cut-in, so as to limit the MHSI delivery pressure to around 40 bar). IRWST cooling is performed by the LHSI mini-flow. The switch-over of the LHSI injection from the cold legs to the hot legs is not required.

2.2.1.2. Safety criteria

The safety criteria to be met are the dose equivalent limits for PCC-3 events.

For LOCA analysis, the following decoupling criteria must be met (see section 1.1 in this appendix):

- The peak cladding temperature must remain below 1204°C,
- The maximum cladding oxidation must remain lower than 17% of the total cladding thickness,
- The maximum hydrogen generation must remain lower than 1% of the amount that would be generated if all the active part of the cladding were to react,
- The core geometry must remain coolable: calculated changes in core geometry must be such that the core remains capable of being cooled,
- Long term cooling must be ensured: the calculated core temperature must be maintained at an acceptably low value and decay heat must be removed.

Demonstration must be provided that the two following safe states are reached with application of the safety analysis rules defined in section 1:

- The controlled state, assuming availability of F1-A equipment only,
- The safe shutdown state, assuming availability of F1-A and F1-B equipment only.

2.2.1.3. Methods and assumptions

2.2.1.3.1. Method

The CATHARE computer code is used.

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

- the separate effect test (SET) and component test matrix contains about 300 tests from various experiments,

- the integral effect tests (IET) matrix contains a selection of 27 tests coming from the BETHSY, LOBI, LOFT, LSTF, PACTEL, PMK and SPES facilities.

Technical Report 97/53 'CATHARE, Small and IB(LOCA)-Methodology for PCC-3 and PCC-4 events' provides a description and justification of the deterministic methodology used for PCC - LOCA analyses performed with CATHARE.

The deterministic methodology is characterised as follows:

- key code models for SB and IB(LOCA) dominant phenomena are realistic though conservatively oriented, bounding experimental results without excessive conservatism;
- initial and boundary conditions are conservatively selected.

The basic steps in the deterministic methodology consist of:

- a phenomenological analysis of the LOCA scenario, and the identification of key phenomena;
- the judgement on adequacy of the code to calculate the LOCA scenario, based on physical understanding, use of the experimental database, code assessment examination, supplemented when necessary by sensitivity studies;
- an evaluation of calculation uncertainty with emphasis on dominant parameters (via sensitivity studies), checking on the conservative approach to key phenomena modelled by the code, making use of the code assessment matrix;
- the introduction, where necessary, of conservative biases matched as closely as possible to the uncertainty in the key phenomena; these are introduced either in a code model, or in the nodalisation scheme, or via a boundary conditions;
- the use of conservative assumptions for initial and boundary conditions.

The realistic deterministic methodology provides a conservative result which can be directly compared to the decoupling criteria.

The CATHARE code provides a detailed representation of the primary and secondary systems. For the PCC-LOCA analysis, the broken loop is explicitly modelled, two intact loops are lumped into a single one with a weighting factor of two, the fourth loop is modelled with the pressuriser connected to the hot leg.

The break is conservatively assumed to be located in the lower part of a cold leg.

The CATHARE point kinetics model is not activated; the residual fission power (A term) is introduced as an input to the CATHARE code. The A term is derived from a decoupled conservative RT-simulation (see Appendix 14B.2.2 – Table 1).

The transient analyses are performed according to the conservative PCC-analysis rules defined in section 1.

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

A second CATHARE calculation is then performed, the hot assembly calculation, which models only the hot assembly and the hot rod within that assembly, using core boundary conditions from the system calculation.

2.2.1.3.2. Main assumptions

a) Accident definition
The break size analysed is 20 cm^2 (∅ 50 mm).

The analysis concentrates on core cooling effectiveness. The aim is to demonstrate that the relevant decoupling criteria are met.

b) Protection and mitigation actions
In case of a LOCA event, automatic protection actions, using F1-A classified systems, carry out tripping of the reactor, removal of residual heat and compensation for the inventory losses, in order to achieve a controlled state.

In accordance with the rules defined for safety analyses in section 1, the controlled state must be achievable only relying on F1-A classified systems. The safe shutdown must be achievable relying only on F1-A and F1-B classified systems.

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

- Reactor Trip, on pressuriser pressure < MIN2,
- Turbine Trip, on RT signal,
- Safety Injection, on pressuriser pressure < MIN3,
- RCP [RCS] trip, on RCP [RCS] pressure drop < MIN1,
- VDA [MSRT] opening and pressure control, on SG pressure > MAX1,
- Partial cooldown, on SI signal,
- ASG [EFWS] injection on SG level < MIN2,
- Isolation of APG [SGBS], on an ASG [EFWS] actuation signal.

Operator actions

No operator action is considered before 30 minutes after the reactor trip. After 30 minutes, some operator actions are required to reach the safe shutdown state.

In accordance with procedures (see section 0.3 in this appendix), the operator diagnoses the RCP [RCS] state using information based on two measurements: Reactor Pressure Vessel Level (RPVL) and ΔT_{sat} . He has then to perform the following actions:

- Control of criticality: The sequence of operator actions to perform RCP [RCS] boration depends on the RCP [RCS] state described by RPVL and ΔT_{sat} measurements. In standard LOCA transients, the RCP [RCS] boration is performed in parallel with the RCP [RCS] cooldown, by means of the RCV [CVCS] (non-F1 classified) or/and the RBS [EBS] (F1-A classified) actuated manually by the operator. In extreme case where high flux is detected on the intermediate range channels (F1-B classified), or where the MHSI pumps are unavailable, the operator are requested to immediately actuate the RBS [EBS], and RCV [CVCS] if available.

- Control of water inventory: the MHSI pump(s) can be stopped when, and provided:

  - the core outlet temperature is below 200°C,

  - the LHSI/RHR pumps (number still to be defined) are available and in operation,

  - the core outlet temperature is not excessive, and

  - the RPVL measurement indicates the core remains covered.

The accumulators are isolated when the MHSI shut-off criterion is reached. If there is a severe reduction in the RCP [RCS] coolant inventory, other subsequent operator actions are performed (see 'Large break LOCA' section 2.3 in this appendix).

- Control of RCP [RCS] pressure and temperature: in the RCP [RCS] state described previously, the operator is required to initiate RCP [RCS] cooldown either at -50°K/h or at -25°K/h, according to the parameters describing the reactor state (containment pressure, ΔT_{sat}, RPVL).

The list of F1B operator actions, with indication of the main F1-B information needed are:

- Manual MHSI shut-off:

  - ΔT_{sat}, core outlet temperature (T_{co}), RPV level (RPVL),

- Manual accumulator isolation:

  - ΔT_{sat}, core outlet temperature (T_{co}), RPV level (RPVL),

- Manual RBS [EBS] actuation:

  - high neutron flux from ex-core detectors, MHSI cold leg flow rate measurements,

- Manual VDA [MSRT] opening/closing:

  - SG pressure, RCP [RCS] temperature,

- Manual ASG [EFWS] control (if no F1-B automatic control of SG level):
    - SG level, ASG [EFWS] flow rate,
- Manual opening of ASG [EFWS] headers at pump suction/discharge:
    - ASG [EFWS] tank level, ASG [EFWS] flow rate,
- Manual VIV [MSIV] bypass opening:
    - SG pressure,
- Manual connection of LHSI/RHR trains:
    - RCP [RCS] hot leg pressure, RCP [RCS] hot leg temperature, ΔT_{sat}, RPVL.

2.2.1.4. Definition of cases studied

The cases studied in this section correspond to a 20 cm^2 break (∅ 50 mm) (the largest SB(LOCA) size considered) located in a cold leg pump discharge piping. The cold leg break location produces the most adverse consequences for the core, compared to other RCP [RCS] break locations. The transient analysis cases include:

- the most adverse single failure,
- the most adverse preventative maintenance state,
- LOOP coincident with the LOCA event.

Two CATHARE calculations have been performed:

- a calculation ending when the controlled state is reached. The objective is to demonstrate that decoupling criteria related to the controlled state (prevention of excessive fuel clad heat-up) are met, taking credit only for use of F1-A systems (see section 2.2.1.5 of this appendix).
- a second calculation ending when the safe shutdown is reached. The objective is to demonstrate that the decoupling criteria related to the safe shutdown state (long term core cooling) are met, taking credit only for use of F1-A and F1-B systems, and to demonstrate the transfer to the safe shutdown state. For this calculation CATHARE is coupled with the CONPATE containment code (see section 2.2.1.6).

2.2.1.5. Description of cases studied: from initiation to controlled state

The safety analysis is performed using conservative assumptions.

2.2.1.5.1. Choice of single failure and preventative maintenance

The most adverse single failure is the loss of 1 emergency diesel at the time of LOOP. 1 RIS [SIS]/RRA [RHRS] train (1 MHSI pump + 1 LHSI pump) and 1 ASG [EFWS] pump are thus unavailable.

Preventative maintenance of 1 diesel is the most pessimistic configuration because 1 RIS [SIS]/RRA [RHRS] train (1 MHSI pump + 1 LHSI pump) and 1 ASG [EFWS] pump are made unavailable.

The assumption of LOOP is pessimistic for SB(LOCA) analysis because of application of the SFC and preventative maintenance principle.

2.2.1.5.2. Initial state

The initial state conditions, given in Appendix 14B.2.2 - Table 2 are used to maximise the time to generation of a RT and SI signal, which is pessimistic for SB(LOCA) mitigation. The initial water temperature within the RPV dome is maximised (taken as the hot leg temperature), which is pessimistic for the SB(LOCA) transient with respect to core cooling.

The axial power shape assumed for the average rod in the average assembly is identical to the one used in the PCC-4 analyses, given in Appendix 14B.2.3 - Figure 1. This power shape, which is skewed at the top of the core, has the following characteristics:

- average linear power = 182.2 W/cm at 102% NP,
- enthalpy rise factor of 1.00,
- peaking factor of 1.46,
- axial offset of 18% (15% + 3% uncertainty).

The axial power shape for the hot rod in the hot assembly is identical to that used in the PCC-4 analyses, given in Appendix 14B.2.3 - Figure 2. This power shape, which is also skewed at the top of the core, has the following characteristics:

- maximum linear power =470 W/cm at 102% NP,
- enthalpy rise factor of 1.80,
- hot spot peaking factor of 2.58 occurring 3.5 m above the bottom of the active core,
- axial offset of 21% (18% + 3% uncertainty).

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 power versus core height distribution. The power is maximised in the upper part of the core which is limiting for SB(LOCA) analysis because of core uncovering process: as the core uncovers, the cladding in the upper part of the core heats up and the peak temperature is sensitive to the linear power at that elevation: the cladding temperature in the lower part of the core remains close to the saturation temperature.

The initial fuel temperature is identical to the one used in the PCC-4 analyses, shown in Appendix 14B.2.3 - Figure 3. This temperature is averaged out over a section of fuel and is given as a boundary condition limit. The initial temperature assumption has a negligible impact on the maximum cladding temperature reached, because the core heats long after RT in a typical SB(LOCA) sequence.

2.2.1.5.3. Specific assumptions

a) Neutronic data and decay heat

Core power is set at 102% of full power until reactor trip. After RT, the residual fission power (term A) is defined by a highly conservative curve (see Appendix 14B.2.2 – Table 1). The decay heat (terms B and C) used is defined in section 0.2.4 within this appendix.

b) Assumptions related to non-F1 systems

Non-classified systems (control systems, etc) are not taken credit for when they perform a mitigating function.

The pressuriser pressure control system (pressuriser heaters) is modelled because it delays the RT signal. A total heating power of 2500 kW is applied until the pressuriser is totally empty.

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

The ARE [MFWS] flow is assumed to be constant until reactor trip.

The GCT [MSB] and AAD [SSS] are disregarded.

The RCV [CVCS] is not modelled before it is isolated. The letdown line is normally isolated on low pressuriser level (not F1A) and on an SI signal (F1A). This modelling is conservative with respect to the RCP [RCS] water inventory balance.

c) Assumptions related to F1 systems

Reactor Trip (F1A): the RT signal is actuated on low pressuriser pressure (< MIN2: 135 bar - 3 bar). The uncertainty in the setpoint allows for degraded containment conditions.

The specific assumptions related to the resulting actions, considering adverse delays, are listed below:

- Beginning of rod insertion 1.2 seconds after RT signal,
- 5 second for complete rod insertion (seismic conditions),
- Total main feed water isolation 1.2 seconds after RT signal,
- Turbine Trip 1.5 seconds after RT signal.

Safety Injection (F1A): the SI signal is actuated on very low pressuriser pressure (< MIN3: 115 bar - 3 bar). The uncertainty in the setpoint allows for the effect of degraded containment conditions in delaying the start-up of SIS pumps and the commencement of partial cooldown.

Specific assumptions related to the resulting actions, including pessimistic delays, are listed below:

- 2.9 seconds delay to actuate the VDA [MSRT] -100 °K/h setpoint decrease,
- 40.9 seconds delay in RIS [SIS] pump start-up, which allows for the delay in the diesel unloading and reloading sequence and the RIS [SIS] pump starting time,
- A minimum characteristic for RIS [SIS] pumps is assumed, as defined in section 0.2.

- A 50°C initial IRWST temperature and injection flow temperature is assumed; this temperature is assumed constant for the duration of the transient (because of the short time duration of the transient, the temperature increase of IRWST is neglected.

The following specific assumptions are applied to the accumulators aimed at minimising the accumulator injection:

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

The water temperature in each accumulator is assumed to be 50°C corresponding to the maximum containment building temperature.

The RIS [SIS]/RRA [RHRS] train connected to the RCP [RCS] loop containing by the break is assumed to spill directly into the containment, without contributing to RCP [RCS] injection[1]. As a result, and accounting for the single failure and preventative maintenance hypotheses, the only RIS [SIS]/RRA [RHRS] components performing cold leg injection are:

- 1 MHSI pump,
- 3 accumulators,
- 1 LHSI pump.

In accordance with the reactor modelling in section 2.2.1.3.1, the following RIS [SIS] modelling is adopted:

- 1 RIS [SIS] train injects into one intact loop,
- 3 accumulators inject into the intact loops (one double-weighted accumulator and one single accumulator),

VIV [MSIV] (F1A): There is no isolation of the main steam header.

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

- 93.0 + 1.5 bar at the beginning of partial cooldown,
- 60.0 + 1.5 bar at the end of partial cooldown.

All the VDA [MSRT] trains (one per SG) are assumed to be available.

[1] As no core heat-up is expected during the SB(LOCA) transient, this assumption has no significant impact with respect to the decoupling criteria (see section 2.2.1.2).

ASG [EFWS] (F1A): the ASG [EFWS] is actuated train by train on very low SG level (< MIN2 (8 m) - 5% uncertainty on the wide range).

The uncertainty in the setpoint allows for the degradation of containment conditions which could delay the start-up of the ASG [EFWS] pumps.

Two ASG [EFWS] trains are available. They are simulated as injecting into SG associated with the double-weighted loop.

Specific assumptions related to the resulting actions, including pessimistic delays, are listed below:

- A 51.5 second delay is applied to ASG [EFWS] pump start-up: this delay includes allowances for the diesel unloading and reloading sequence and the ASG [EFWS] pump start time,
- Minimum characteristic are assumed for ASG [EFWS] pumps: a constant injection flow rate of 93.5 t/h is assumed.
- A 50°C injection flow temperature is assumed.

RCP [RCS] TRIP (F1A): In the EPR design, it is intended to provide an automatic RCP [RCS] trip in case of LOCA, based on a "ΔP over RCP [RCS]" measurement (80% $\pm$ 5% of nominal ΔP, Appendix 14B.0.2 – Table 9).

In order to be conservative in the present analysis, it is postulated that the RCP [RCS] are tripped at the reactor trip occurrence, which is the most pessimistic RCP [RCS] trip time within the time period from reactor trip to hereupon RCP [RCS] trip signal occurrence. Indeed, for the smaller break areas (below 20 cm^2), the early RCP [RCS] trip contributes to an additional loss of RCP [RCS] water inventory by maintaining the primary pressure high at the beginning of transient while the RCP [RCS] is still under liquid phase.

d) Other assumptions

The Loss Of Off-site Power is superposed to the event and occurs at time of the turbine trip.

2.2.1.5.4. Results

The sequence of events is given in Appendix 14B.2.2 - Table 3.

Key parameters are presented in the following figures:

- Appendix 14B.2.2 – Figure 2: RCP [RCS] and secondary side water inventories,
  RCP [RCS] and secondary side pressures.
- Appendix 14B.2.2 – Figure 3: Total and steam break flow rates,
  Total break and RIS [SIS] flow rates.
- Appendix 14B.2.2 – Figure 4: Hot spot cladding temperature,
  Core swell level.

About 3500 seconds after reactor trip, the flow rate injected by 1 MHSI matches the break flow rate. The RCP [RCS] water inventory stops decreasing, with a minimum water content of 85 tons. The decay heat is fully removed, partly by the break, mostly by the SG via ASG [EFWS] and VDA [MSRT]. The controlled state is reached.

The following conclusions can be drawn with respect to the decoupling criteria:

- There is no core uncovery and thus no core heat-up; the peak cladding temperature remains below the acceptance criterion limit (1204°C),
- There is no cladding oxidation,
- There is no cladding rupture,
- Integrity of the core geometry is maintained,
- Long term cooling is addressed in section 2.2.1.6 within this appendix.

In conclusion, it has been shown that the operation of F1A systems (RIS [SIS]/RRA [RHRS] in conjunction with VDA [MSRT] and ASG [EFWS]) is sufficient for reaching the controlled state.

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

The safety analysis is performed with the same level of conservatism as the analysis up to the controlled state, presented in section 2.2.1.5.

In particular, the assumptions of maximum decay heat and the minimum capacity of F1 systems used in section 2.2.1.5 are also used in 2.2.1.6.

2.2.1.6.1. Choice of single failure and Preventative maintenance

The most adverse single failure is the loss of one emergency diesel at the time of LOOP (considered to occur at the time of reactor trip):

- 1 RIS [SIS] train (1 MHSI pump and 1 LHSI pump), and 1 ASG [EFWS] pump are thus unavailable,
- 2 hours after the LOOP, the VDA [MSRT] assigned to this division is also assumed lost because of the battery depletion (assuming that the power supply from the neighbouring division via the dedicated DC cross-connection is unavailable, because of preventative maintenance on the associated diesel).

Preventative maintenance on one emergency diesel is the most adverse configuration since:

- 1 RIS [SIS] train (1 MHSI pump and 1 LHSI pump), and 1 ASG [EFWS] pump are unavailable,
- 2 hours after the LOOP, the VDA [MSRT] assigned to this division is lost because of the battery depletion (assuming that the power supply from the neighbouring division via the dedicated DC cross-connection is unavailable, because of single failure on the associated diesel).

The battery depletion has in fact no consequence on the availability of VDA [MSRT], since the operability of VDA [MSRT] is possible by local action. The consequence of single failure and preventative maintenance is then limited to the loss of two (MHSI + LHSI + ASG [EFWS]) pumps.

The assumption of LOOP is pessimistic for SB(LOCA) analysis because of the application of the SFC and preventative maintenance principle.

2.2.1.6.2. Specific assumptions

a) RCP [RCS] cooldown phase

According to the CATHARE calculation, a stable controlled state, corresponding to a hot shutdown state at end of the partial cooldown (RCP [RCS] and SG pressure in the range of 60 bar), is maintained (without operator action) until a time of RT + 2 hours.

The 4 Reactor Coolant Pumps are tripped at the time of RT.

At RT + 2 hours, the RPVL is stabilised between the Top of the Hot Leg (THL) and the Bottom of the Hot Leg (BHL). ΔT_{sat} indicates saturated condition at the core outlet.

At that time, the operator is assumed to start the RCP [RCS] cooldown to safe shutdown, by using three out of four VDA [MSRT]. 1 VDA [MSRT] is considered unavailable to allow for single failure in the VDA [MSRT] system (failure of one MSRIV to open is assumed to result from the single failure on 1 diesel).

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

b) Assumptions applied to the secondary side

Because of the application of the Single Failure and the Preventative Maintenance principles, only two ASG [EFWS] are assumed available to deliver into two SGs. The flow rate per pump is the same as in the analysis up to the controlled state (section 2.2.1.5). The SG water level is assumed to be controlled in order to prevent the fed SG from overfilling.

The setpoint of the available VDA [MSRT] is decreased to 5 bar (P_{sat} (150°C)) for the RCP [RCS] cooldown phase described in (a) above.

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

c) Assumptions applied to the RIS [SIS]/RRA [RHRS]

The MHSI and LHSI water injection temperatures are maximised to allow for the IRWST temperature increase due to degraded containment conditions (break mass and energy releases). The temperature is obtained from calculations with the coupled CATHARE-CONPATE codes.

The IRWST temperature is calculated in a conservative manner:

- the initial IRWST temperature is assumed to be the maximum containment operating temperature of 50°C,
- the IRWST water volume is assumed to be the minimum vale of 1300 m^3,

- the LHSI heat removal capacity is minimised (minimum capacity assumed for the LHSI/RRI [CCWS] heat exchanger, see section 1.3 in Sub-chapter 6.2.).

The only MHSI assumed available in the analysis is conservatively shut-off as soon as the core outlet temperature decreases below 200°C. The available LHSI contuse to deliver into the RCP [RCS] until RHR connecting conditions are reached.

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

d) Assumptions applied to LHSI/RHR connection conditions.

LHSI/RHR connection conditions are reached when the following conditions are met:

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

In the CATHARE calculation, the RPVL is above the top of hot leg and the subcooling ΔT_{sat} is greater than 10°C, when LHSI/RHR connection conditions are reached.

2.2.1.6.3. Results

The sequence of events is given in Appendix 14B.2.2 - Table 4.

The most representative parameters are presented in the following figures:

- Appendix 14B.2.2 - Figure 5: RCP [RCS] and secondary side water inventories, RCP [RCS] and secondary side pressures.
- Appendix 14B.2.2 - Figure 6: RCP [RCS] cold leg and hot leg temperatures, Integrated ASG [EFWS] water flow rates
- Appendix 14B.2.2 - Figure 7: Total and steam break flow rates, Total break and RIS [SIS] flow rates.
- Appendix 14B.2.2 - Figure 8: Reactor Pressure Vessel Level Subcooling Margin ΔT_{sat}.

LHSI/RHR connecting conditions are reached at 7 hours after reactor trip (including 2 hours without any manual action, and 5 hours of RCP [RCS] cooldown at -25°K/h), at the following RCP [RCS] state:

- hot leg temperature is about 160°C,
- hot leg pressure is about 10 bar,
- the subcooling margin ΔT_{sat} is positive and above 10°C,
- the RPVL is above the top of the hot leg.

The RCP [RCS] loop level rapidly increases when the RCP [RCS] pressure decreases below 10 bar.

At time of LHSI/RHR connection, the MHSI pump is shut off, 1 LHSI train operates in SI-mode (injection into 1 cold leg), and the remaining LHSI train operates in RHR-mode. The flow rate injected by the one available LHSI pump operating in SI-mode is about twice the break flow rate:

- should the break be located in the LHSI pump injection line, the flow rate injected by this pump would largely compensate the break flow rate, even under pessimistic subcooled conditions of injection (the lower the break flow temperature, the higher the break flow rate);

- even if the water temperature in the IRWST were taken as 10°C, the liquid flow rate (at 10°C) through the 20 cm^2 break area would not exceed 60% of the flow rate delivered by a single LHSI pump at 5 bar. Thus, 40% of the LHSI pump flow rate would contribute to water inventory restoration, helping to maintain a sufficient RCP [RCS] loop level in the hot legs to ensure adequate operating conditions for the LHSI pump operating in RHR-mode.

No further RCP [RCS] water inventory degradation is expected after the transfer to safe shutdown conditions.

The MHSI pump which has previously been shut-off according to a criterion defined in emergency operating procedures, remains available to supplement the LHSI pump operating in injection mode. If it were started up again, its delivery head would be limited to around 40 bar as described in section 2.2.1.1.2 b).

With respect to ASG [EFWS] feedwater availability, the transfer to the safe shutdown state occurs without emptying of the ASG [EFWS] tanks. About 600 tons of ASG [EFWS] water is used from the 1500 tons initial inventory of ASG [EFWS] tanks.

With regard to criticality concerns, core subcriticality is ensured throughout the transient by the boron initially injected by the MHSI pump, and later (after LHSI/RHR connection) by the LHSI pump operating in SI-mode. No additional boration is needed. For very small breaks where the SI injection flow would not be sufficient to perform the required boration, safety boration would be ensured by the F1-A classified RBS [EBS].

2.2.1.6.4. Conclusion

This analysis of an SB(LOCA) shows that, even assuming the most adverse single failure and the most adverse preventative maintenance assumptions:

- The controlled state can be reached fulfilling all safety criteria (section 2.2.1.5),

- The safe shutdown can be reached fulfilling all safety criteria, using the following F1 means:

    - during the transfer from the controlled state to the safe shutdown state,

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

        - The LHSI/RHR heat exchanger capacity for IRWST heat removal,

- The MHSI injection capacity for maintaining the RCP [RCS] water inventory,

- The MHSI and RBS [EBS] (needed only for smaller breaks) boron injection capacities for maintaining core subcriticality.

o at LHSI/RHR connection corresponding to the safe shutdown state,

- The LHSI/RHR heat exchange capacity for core heat removal,

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

- The LHSI/RHR injection capacity for maintaining the RCP [RCS] water inventory,

- The LHSI/RHR boron injection capacity for maintaining core subcriticality.

APPENDIX 14B.2.2 – TABLE 1

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

- the point-kinetic neutronics model is used with a conservative set of neutronics data, enveloping all UO2 and MOX fuel management schemes:
  - maximum moderator temperature coefficient of 0.49 ΔK/K per g/cm^3 (constant value)
  - minimum Doppler temperature coefficient of – 4.1 pcm/°C (constant value)
  - minimum Doppler power coefficient, according to Appendix 14B.0.2 - Table 4
  - minimum rods worth of - 4500 pcm (N-1 rods, UO2 fuel)
  - minimum rods worth insertion versus rods dropping time of Appendix 14B.0.2 - Table 8
  - maximum rods dropping time of 5 seconds (with earthquake)
- the RCP [RCS] and SG thermal-hydraulic calculation is based on conservative boundary conditions, which are intended to bound any future ARE [MFWS] and GCT [MSB] characteristics (not yet defined precisely):
  - minimum decrease of ARE [MFWS] flow after RT
    - from RT to RT+60s: 100% to 20% (linear decrease)
    - from RT+60s to 0% SG-level setpoint: 20% (constant value)
    - afterwards: flow controlled to keep SG-level
  - maximum decrease of ARE [MFWS] temperature after RT
    - at RT: from 230°C down to 120°C in one step
  - minimum GCT [MSB] opening-time after RT
    - GCT [MSB]-valve opening time: 0 seconds
    - GCT [MSB] I&C-processing delay: 0 seconds

Note: At this stage of EPR design, pessimistic data have been used even if the resulting shutdown margin does not meet the –2000 pcm design criteria after RT, which is highly pessimistic.

(continued…)

TIME (s)

from beginning of rod drop	FISSION POWER % of initial core power
0	92.4
1	89.0
2	85.3
3	69.2
4	20.4
5	13.3
10	11.9
50	10.5
100	6.4
200	1.3
300	0.0

FISSION POWER (TERM A)

APPENDIX 14B.2.2 – TABLE 2

Initial conditions: 20 cm^2 (∅ 50 mm) Cold Leg Break (state A, PCC-3)

Parameters	Limiting values used
Reactor coolant system	
Initial reactor power (% of nominal power)	100 + 2 = 102.
Initial average RCP [RCS] temperature (°C)	311 + 2.5 = 313.5
Initial pressuriser pressure (bar)	155 + 2.5 = 157.5
Reactor cooling flow (kg/s)	22240 (thermal-hydraulic flow rate)
Pressuriser water volume / level (m^3 / m)	43.4/7.4(nominal + 5%MR)
Dome liquid temperature (°C)	332
Steam generators	
Initial steam pressure (bar)	76.9
Initial SG level (m)	16.2 (nominal)
Feedwater	
Main feedwater flow (% of nominal flow)	100 + 2 = 102
Initial ARE [MFWS] temperature (°C)	232

APPENDIX 14B.2.2 – TABLE 3

Sequence of events up to controlled state: 20 cm^2 (∅ 50 mm) Cold Leg Break (state A. PCC-3)

TIME	EVENT
0.0 s	Break opening
95.1s	RT signal on pressuriser pressure < MIN2 (132.0 bar) Turbine Trip. LOOP. RCP [RCS] trip
230 s	Pressuriser emptying
300 s	Safety Injection & Partial Cooldown signal (112 bar)
304 s	Beginning of Partial Cooldown (SG pressure MSRV setpoint)
342 s	Starting of MHSI & LHSI pumps (Train 2)
887 s	Beginning of MHSI (RCP [RCS] pressure < 80 bar)
1136 s (1) 1153 s (2)	Beginning of ASG [EFWS] (SG level < 7.1 m)
5000.s	End of calculation

(1) SG2: related to 1 unaffected SG
(2) SGR: related to the affected SG

APPENDIX 14B.2.2 – TABLE 4

Sequence of events up to safe shutdown state: 20 cm^2 (∅ 50 mm) Cold Leg Break (state A. PCC-3)

TIME	EVENT
0.0 s	Break opening
7300 s (2.0 h)	Beginning of RCP [RCS] cooldown (reactor trip + 2 hours)
7300 s (2.0 h)	Accumulators isolation
19075 s (5.3 h)	MHSI pump cut-off on criterion $T_{Hot\ Leg}$ < 200°C
25420 s (7.1 h)	RPVL greater than Top of Hot Leg
25940 s (7.2h)	$\Delta T_{sat\ 1}$ > 10°C (subcooling > 10°C)
25940 s (7.2 h)	LHSI/RHR connecting conditions are met (P_p about 10 bar. $T_{Hot\ Leg}$ about 160°C)
27000 s (7.5 h)	End of calculation

[1] $T_{sat} - T_{core\ outlet}$

APPENDIX 14B.2.2 – TABLE 5

Initial conditions: 20 cm² (∅ 50 mm) Cold Leg Break (state C. PCC-4)

Parameters	Limiting values used	
Reactor coolant system		
– Initial reactor power (% of full power)	0.92% / 46 MW	(max)
– Initial average RCP [RCS] temperature (°C)	120	(max)
– Initial reactor coolant pressure (bar)	30	(max)
– 2 LHSI/RHR trains in operation		
– Pressuriser water volume / level (m^3/m)	24.4 / 4.4 (21+5% MR)	(max)
Steam generators		
– Initial water temperature (°C)	120	(max)
– Initial steam pressure (bar)	2	(P_{sat} 120°C)
– Initial SG level(m)	16.2	

APPENDIX 14B.2.2 – TABLE 6

Sequence of Events: 20 cm² (∅ 50 mm) Cold Leg Break (state C. PCC-4)

TIME	EVENT
0.0 s	Break opening
0.0 s	RCP [RCS] trip
175 s	Pressuriser emptying
232 s	Safety injection signal on ΔP_{sat} < 5 bars
157 s	Leak compensation by MHSI pump
425 s	Minimum RCP [RCS] subcooling margin
2500 s	End of calculation

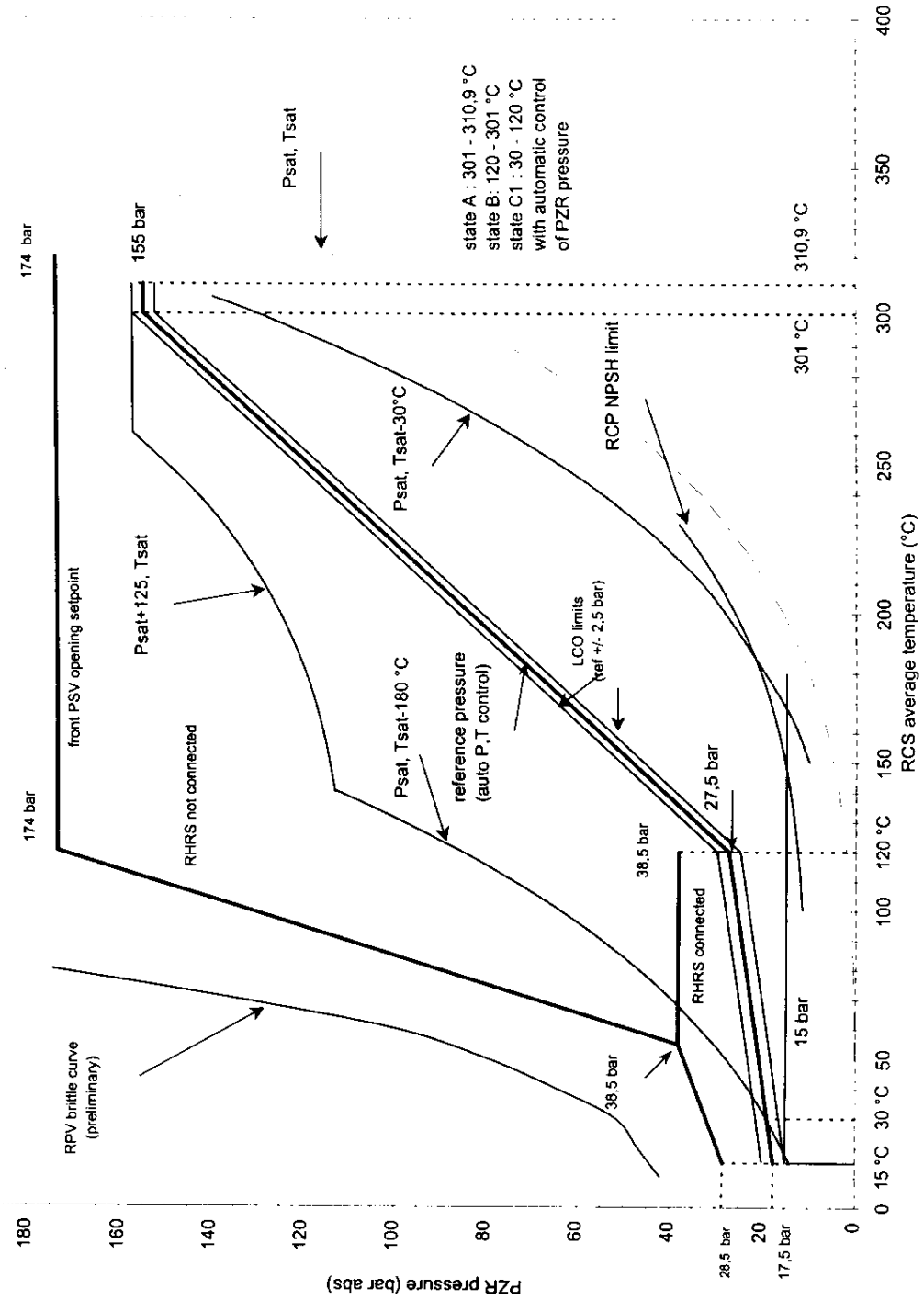
APPENDIX 14B.2.2 – TABLE 7

RCP [RCS] PLANT STATUS in STATE C

	RCP [RCS] thermalhydraulic STATUS			EQUIPMENT in OPERATION		EQUIPMENT at STANDBY		
						SAFETY INJECTION (RIS [SIS])		
							RCP [RCS] HEAT REMOVAL	
	RCP [RCS] Pres.	RCP [RCS] Temp.	RCP [RCS] water inventory	Reactor Coolant Pump on	LHSI/RHR on	Automatic actuation on SI-signal	Manual actuation	Automatic actuation on SG-pressure
STATE C1	30 bar	120°C ↓ 100°C	PZR level	2 RCP on	2 LHSI/RHR on	4 MHSI (1)	2 LHSI (1)	2 SG
STATE C2	30 bar	100°C ↓ 55°C	PZR level	2 RCP on (T>≅70°C) 1 RCP on (T>≅55°C)	4 LHSI/RHR on (1)	4 MHSI (1)		2 SG
STATE C3	30 bar ↓ 1bar	55°C	PZR level ↓ mid-loop	0 RCP on	4 LHSI/RHR on (1)	4 MHSI (1)		2 SG

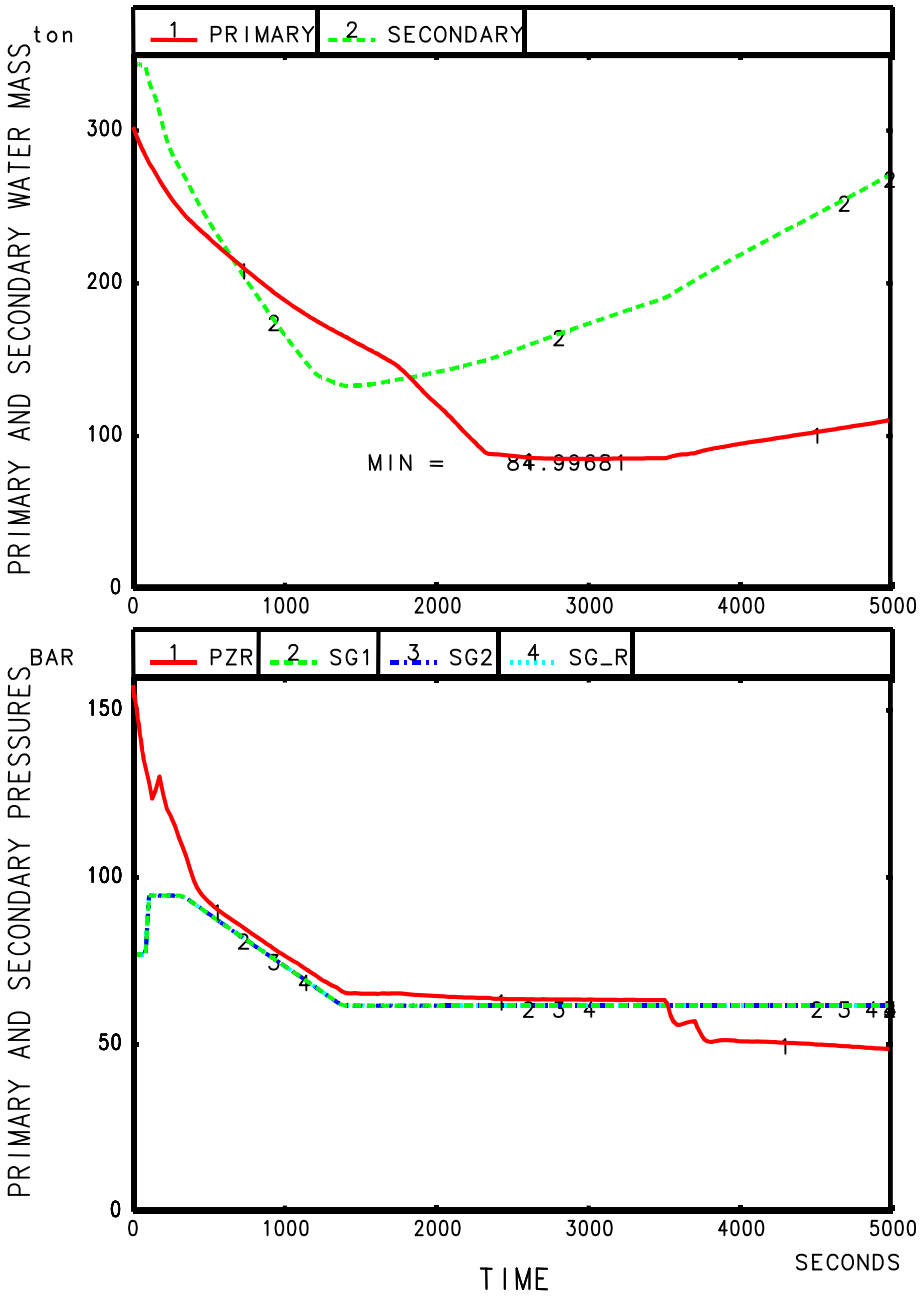
(1) Possible Preventative Maintenance on 1 safety division. With consequential unavailability of 1 LHSI + 1 MHSI system

APPENDIX 14B.2.2 – FIGURE 1



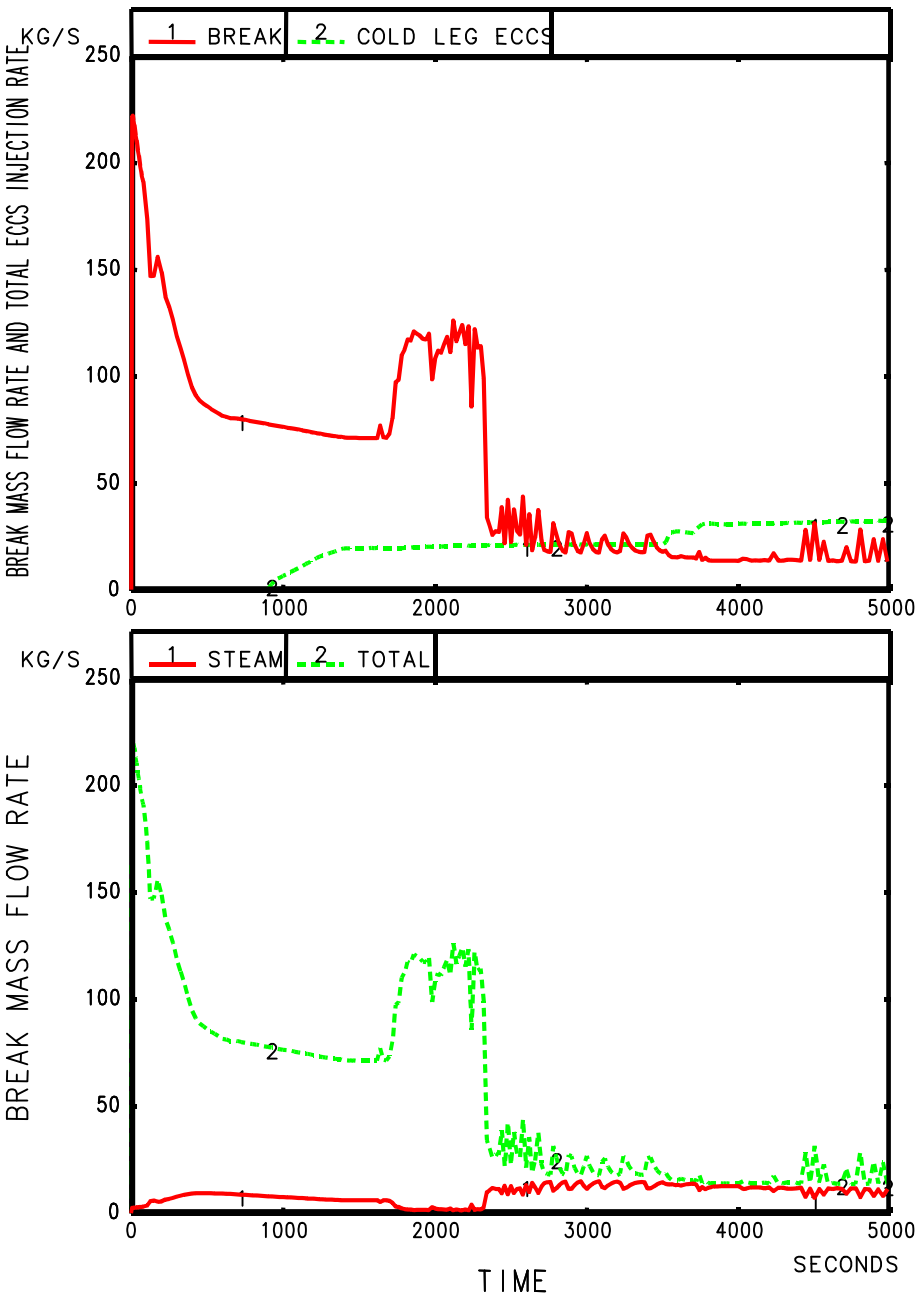
Typical (P,T) reactor operating range

APPENDIX 14B.2.2 – FIGURE 2



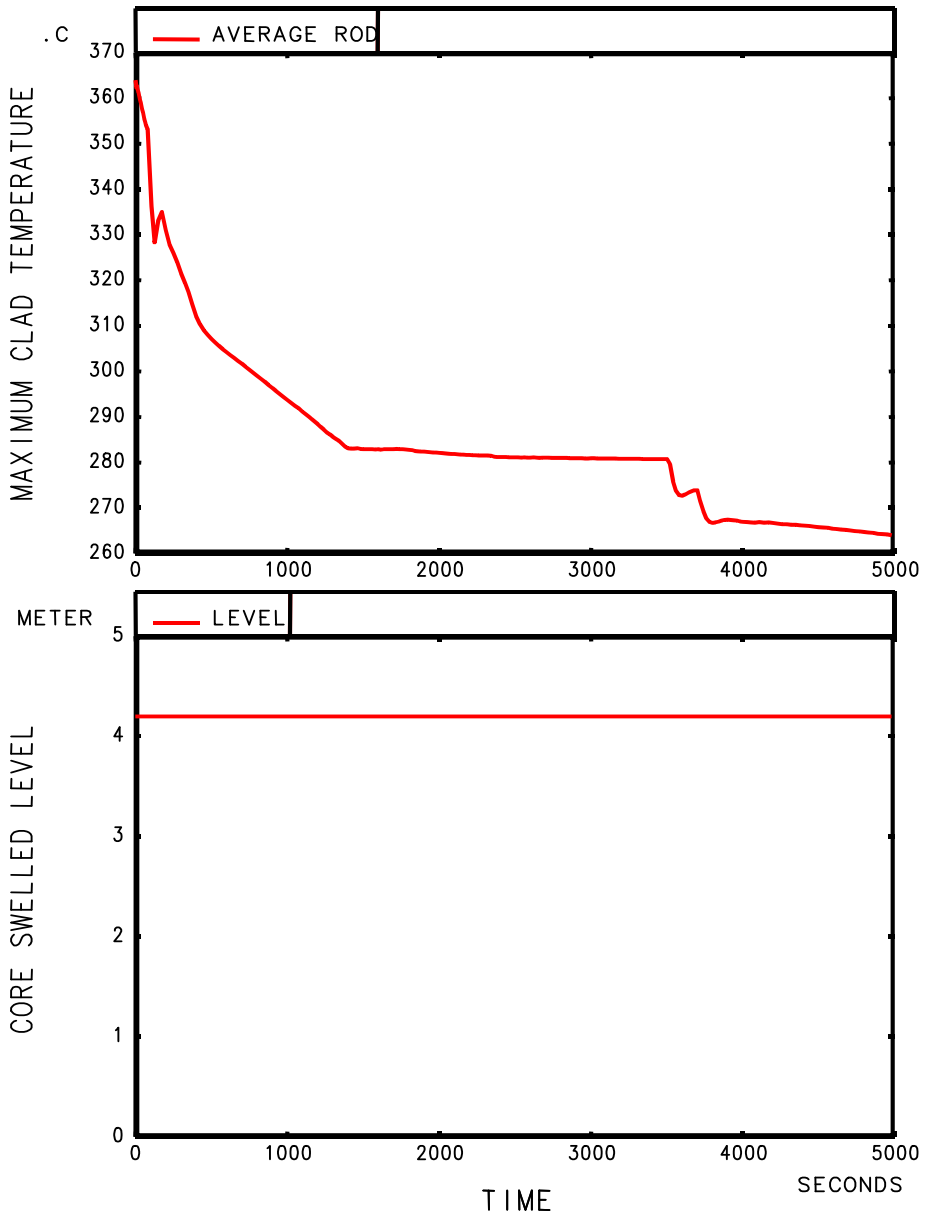
RCP [RCS] and secondary side water inventories
RCP [RCS] and secondary side pressures
20 cm^2 (∅ 50 mm) cold leg break (state A. PCC-3)

APPENDIX 14B.2.2 – FIGURE 3



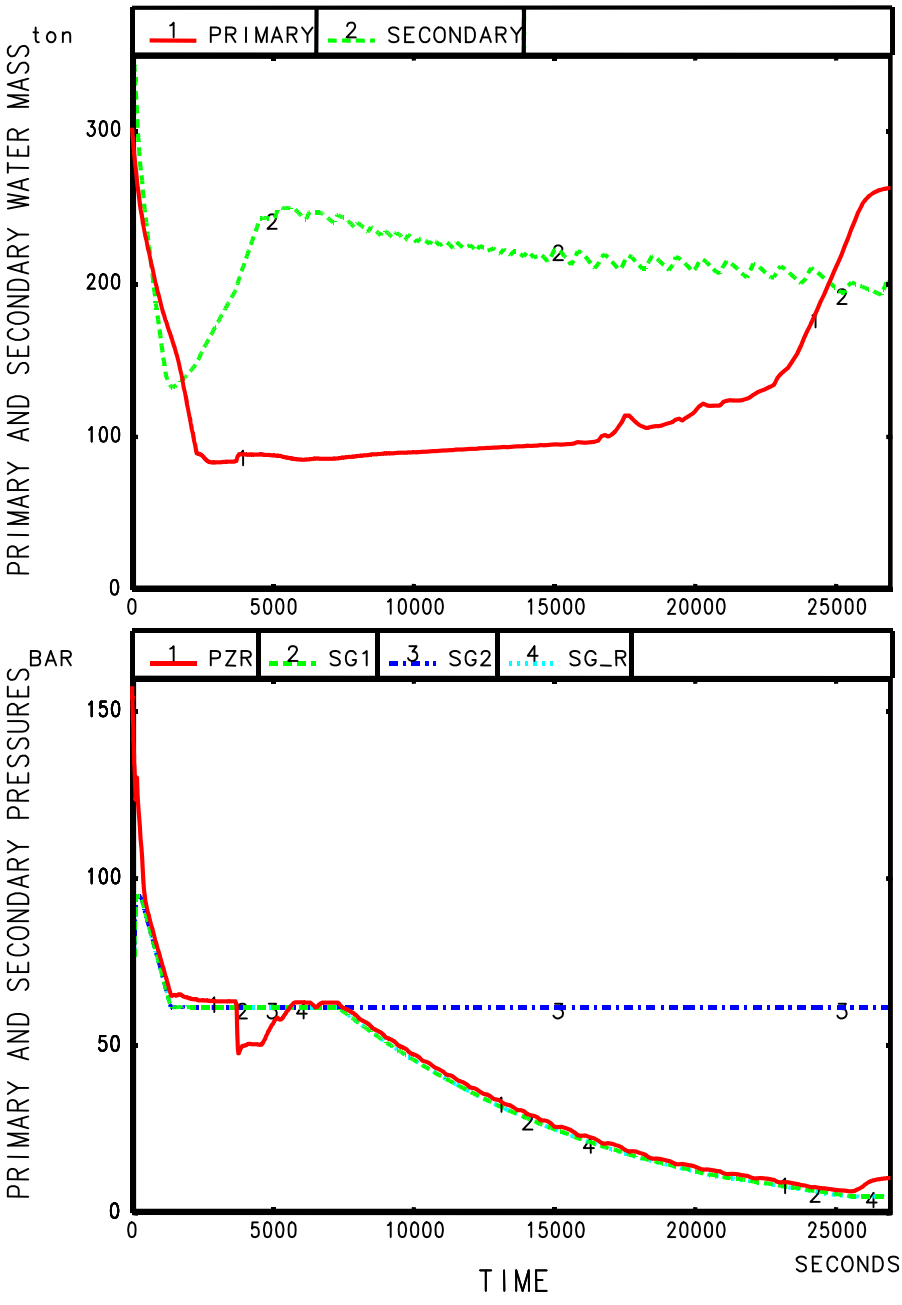
Total break and RIS [SIS] flow rates
Total break and steam flow rates
20 cm^2 (∅ 50 mm) cold leg break (state A. PCC-3)

APPENDIX 14B.2.2 – FIGURE 4



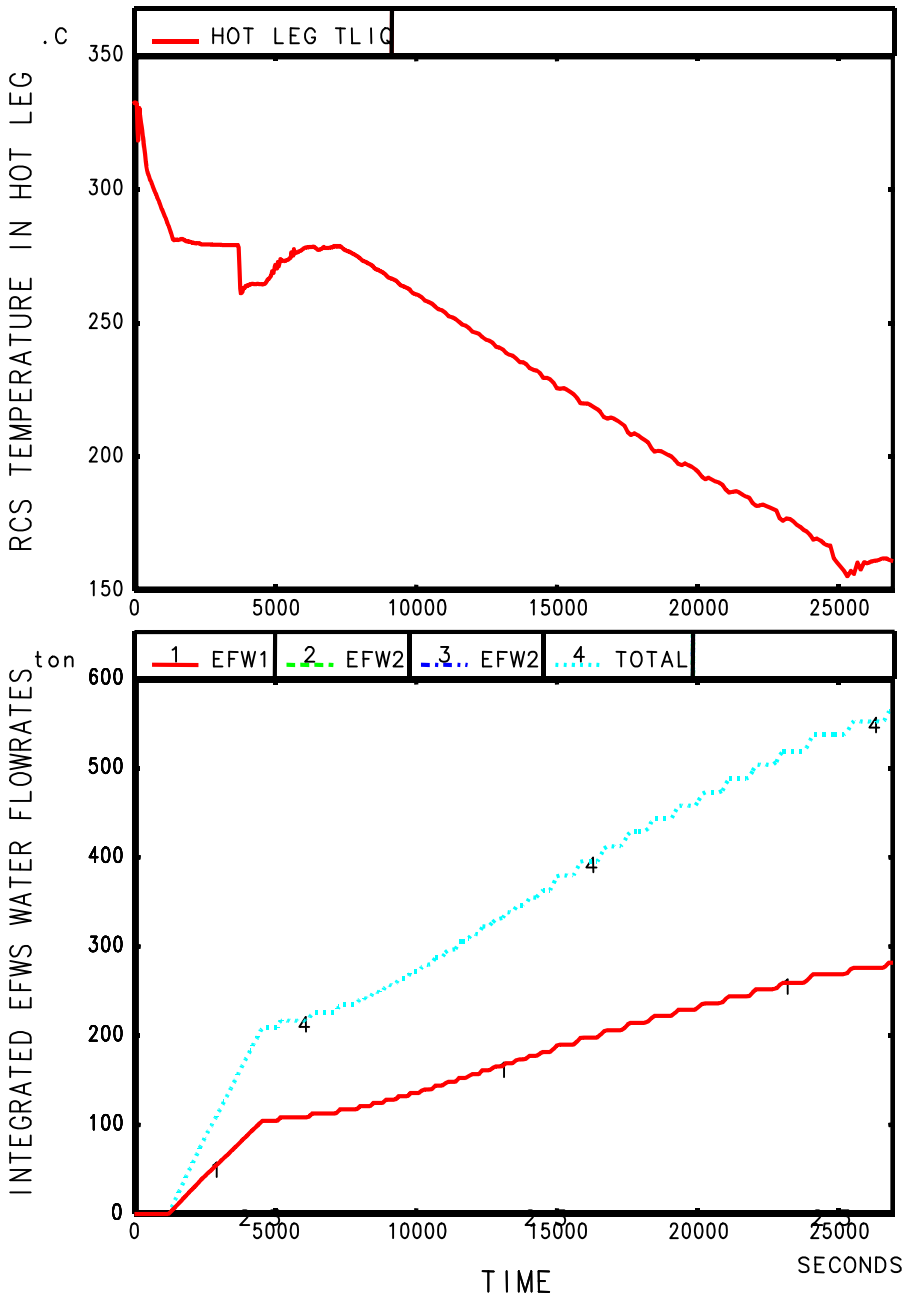
Maximal cladding temperature
Core swell level
20 cm^2 (∅ 50 mm) cold leg break (state A. PCC-3)

APPENDIX 14B.2.2 – FIGURE 5



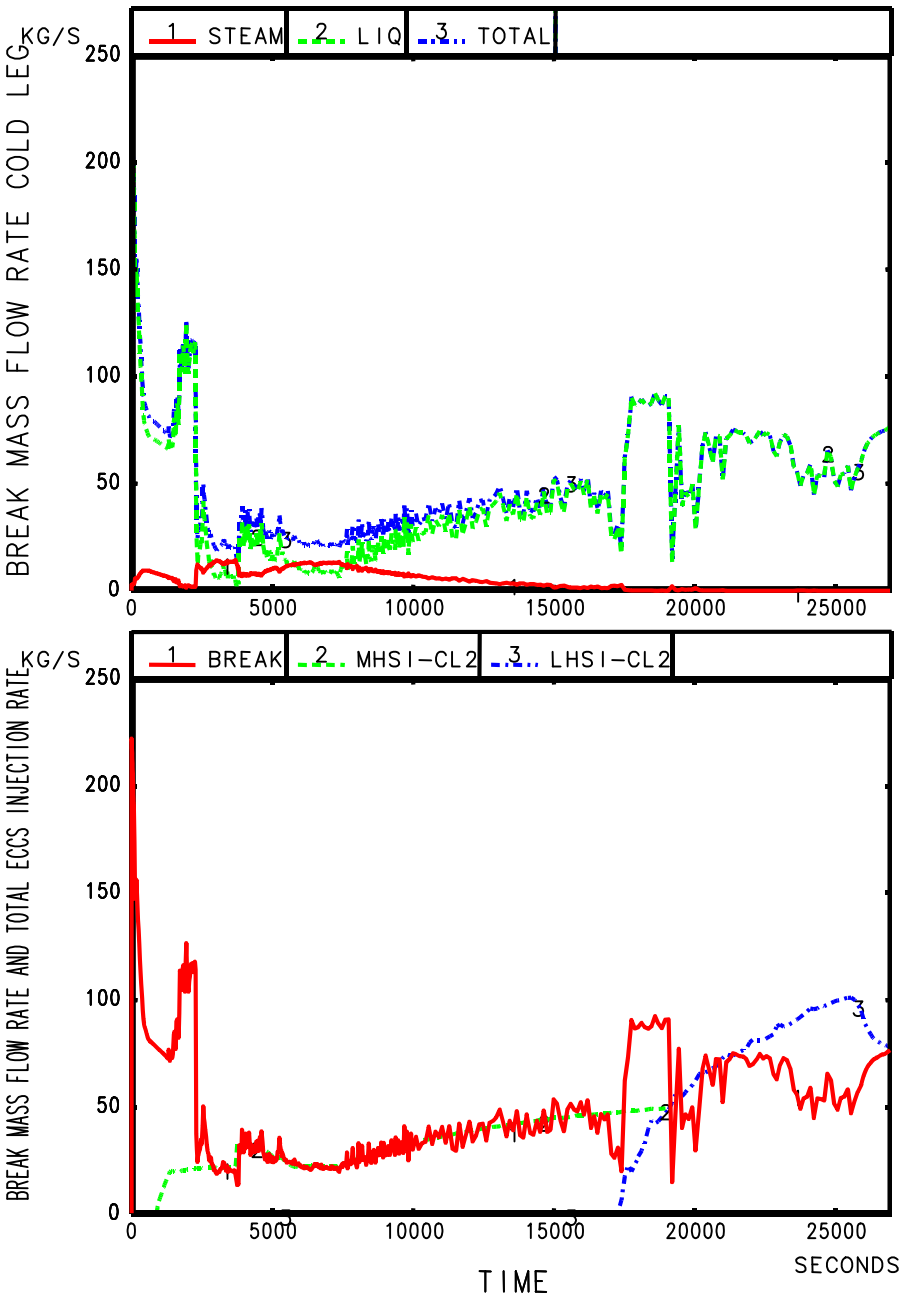
RCP [RCS] and secondary side water inventories
RCP [RCS] and secondary side pressures
20 cm^2 (∅ 50 mm) cold leg break (state A. PCC-3)

APPENDIX 14B.2.2 – FIGURE 6



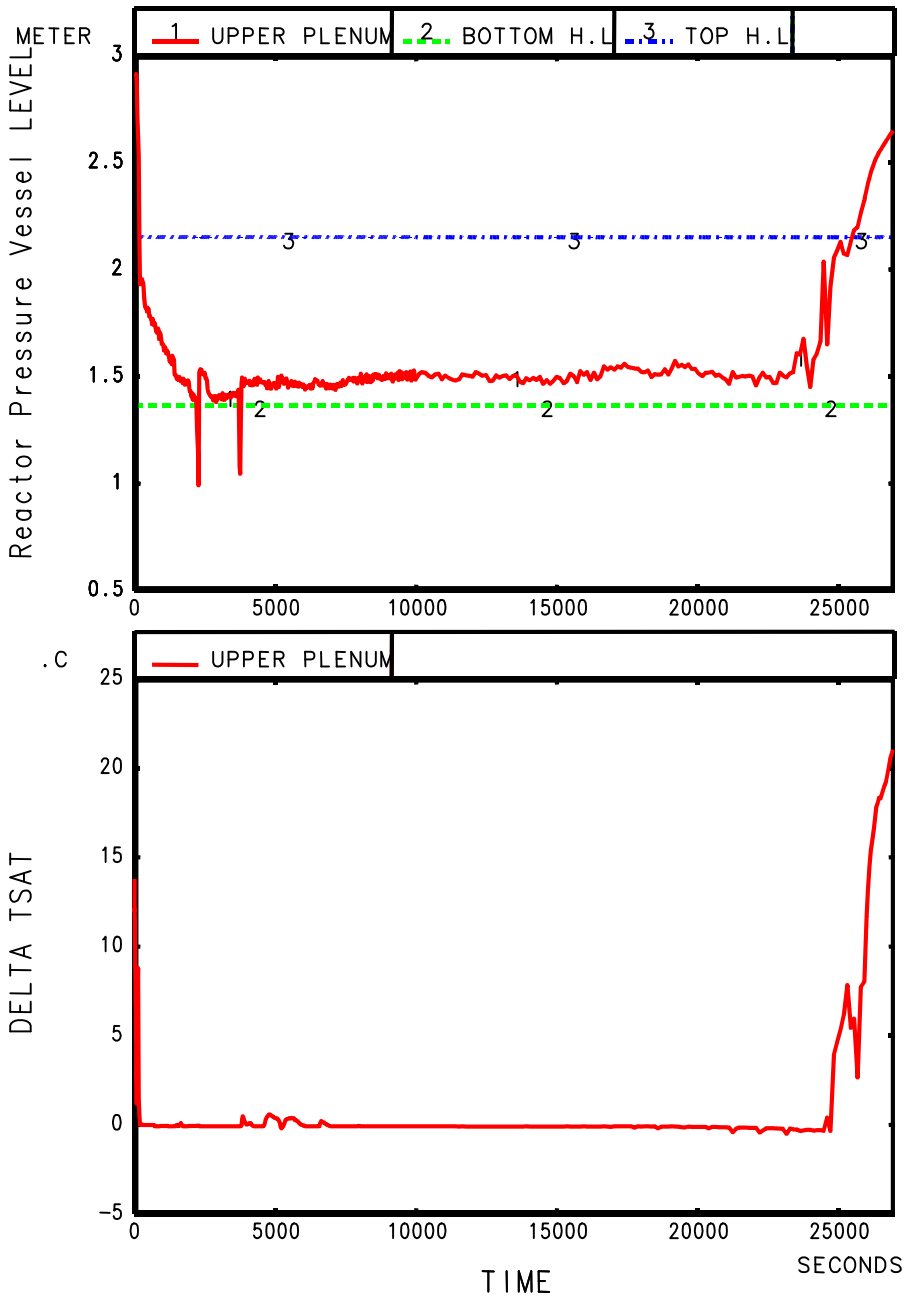
RCP [RCS] hot leg temperature
Integrated ASG [EFWS] water flow rates
20 cm^2 (∅ 50 mm) cold leg break (state A. PCC-3)

APPENDIX 14B.2.2 – FIGURE 7



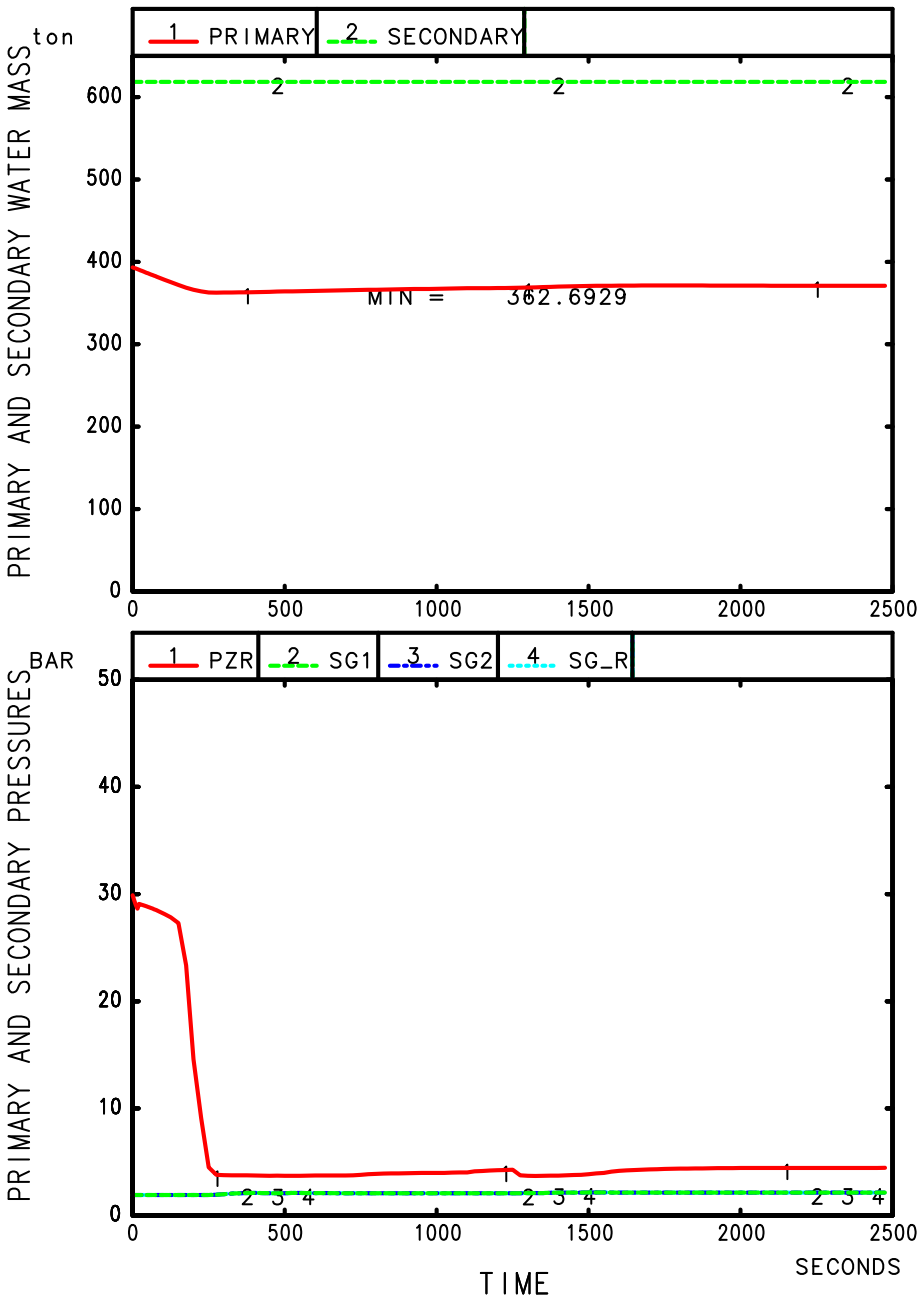
Total and steam break flow rates
Total break and RIS [SIS] flow rates
20 cm^2 (∅ 50 mm) cold leg break (state A. PCC-3)

APPENDIX 14B.2.2 – FIGURE 8



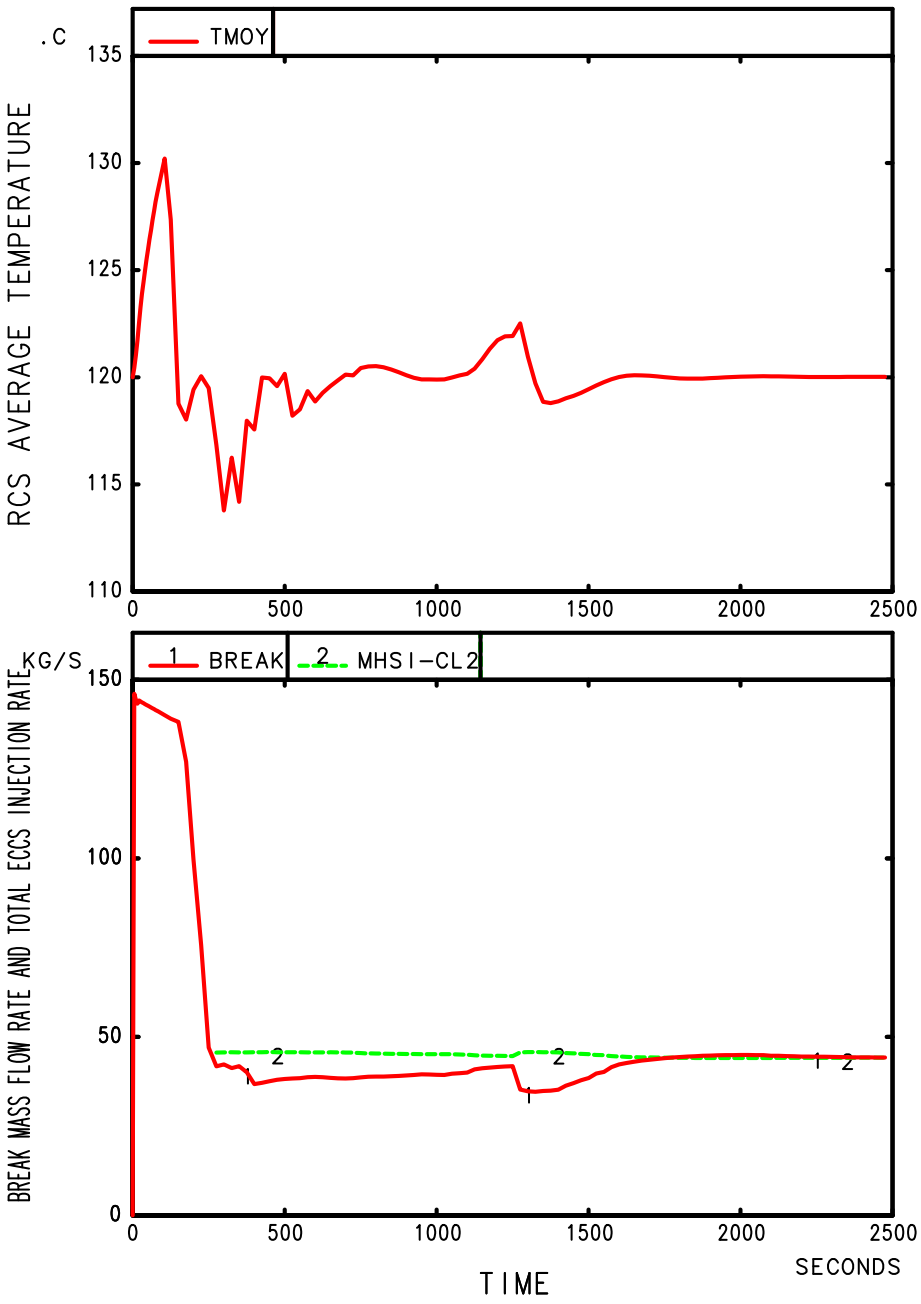
Reactor Pressure Vessel level
Saturation Margin ΔT_{sat}
20 cm^2 (∅ 50 mm) cold leg break (state A. PCC-3)

APPENDIX 14B.2.2 – FIGURE 9



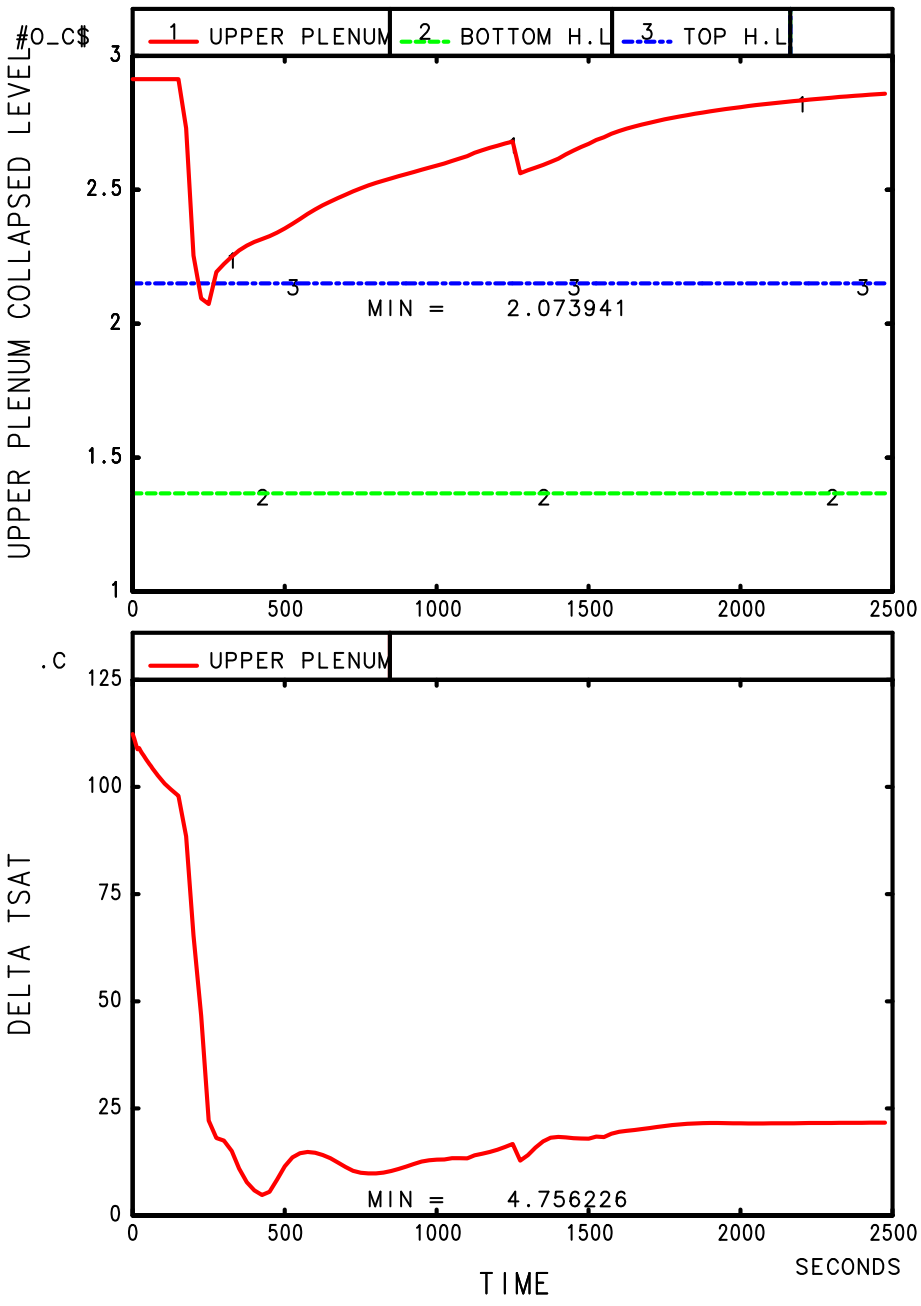
20 cm^2 (∅ 50 mm) cold leg break (state C. PCC-4)
RCP [RCS] and secondary side water inventories
RCP [RCS] and secondary side pressures

APPENDIX 14B.2.2 – FIGURE 10



20 cm^2 (∅ 50 mm) cold leg break (state C. PCC-4)
RCP [RCS] average temperature
Total break and RIS [SIS] flow rates

APPENDIX 14B.2.2 – FIGURE 11



20 cm^2 (∅ 50 mm) cold leg break (state C. PCC-4)
Upper plenum level
RCP [RCS] subcooling margin ΔT_{sat}

2.3.1. Intermediate and Large Break LOCA in state A (PCC-4)

The intermediate break LOCA (IB(LOCA)) is defined as a break of equivalent diameter greater than 50 mm (equivalent area greater than 20 cm²), and less than the LB(LOCA) size.

The large break LOCA (LB(LOCA)) is defined as a guillotine break of the largest pipe connected to the RCP [RCS]-loop, i.e. a surge line break with respect to the hot leg side and the RIS [SIS] line break with respect to the cold leg side.

2.3.1.1. Identification of causes and accident description

2.3.1.1.1. General issues

A LOCA induces:

- A loss of reactor coolant inventory and possibly a core heat-up,
- Containment overpressure loads due to the mass and energy release (see section 1 in Sub-chapter 6.2),
- Mechanical loads on RCP [RCS] components and associated supports and structures,
- Mechanical loads on RPV internals.

A LOCA is classified as a PCC-3 or PCC-4 event according to the break size and the plant initial state. This section deals only with the intermediate and large break LOCAs in state A, which are classified as PCC-4 events (see section 0.1: scope of events).

2.3.1.1.2. Typical sequence of events

2.3.1.1.2.1. From event initiation to attainment of controlled state

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

A reactor trip occurs on low pressuriser (PZR) pressure (< MIN2). The RT signal automatically trips the turbine and closes the MFW full load control lines.

As the secondary side pressure increases, the Main Steam Bypass valves open allowing steam dump to the condenser. In case of unavailability of the steam dump to condenser (typically in case of LOOP), the Main Steam Relief Trains open allowing steam dump to atmosphere.

The SGs are fed by the ARE [MFWS] via the low load control lines. In case of unavailability of the ARE [MFWS], the start-up and shutdown pump starts and feeds the SGs through the low load control lines. In case of unavailability of the AAD [SSS] (typically in case of LOOP), the ASG [EFWS] is actuated on low SG level (< MIN2).

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

During the partial cooldown, the RCP [RCS] pressure decreases sufficiently to allow MHSI injection into the cold legs. The partial cooldown is carried out using all SGs, by dumping steam to the condenser or to the atmosphere. The cooldown is performed by automatically decreasing the relief valve setpoints to achieve a cooling rate of –100°K/h, to reach a target pressure value that is low enough to permit MHSI injection and high enough to prevent core recriticality.

For the smallest intermediate breaks, RCP [RCS] leakage does not remove sufficient volumetric flow to exceed the steam production in the core due to decay heating, and RCP [RCS] depressurisation terminates at the end of the partial cooldown.

As long as the MHSI flow rate is insufficient to compensate for the break flow rate, RCP [RCS] inventory continues to decrease. The break flow rate decreases as the void fraction in the cold legs increases.

When the break flow switches to single phase steam, the volumetric balance in the RCP [RCS] between steam production due to decay heating and steam venting through the break is modified, and the break size becomes the dominant parameter in the subsequent depressurisation sequence:

- In case of the smallest intermediate breaks, some condensation in the SG tubes may be still necessary in conjunction with the direct steam venting at the break to remove all steam produced in the core. The RCP [RCS] pressure (saturation pressure) remains slightly above the SG pressure to achieve an energy balance,

- In case of larger breaks, the break size is large enough to vent sufficient steam so that further RCP [RCS] depressurisation can occur without the requirement for steam condensation in the SG tubes (eventually the heat transfer reverses between primary and secondary side), driving down the RCP [RCS] pressure, independently of the SG temperature, to the accumulator pressure and possibly the LHSI pressure level.

The subsequent evolution of the RCP [RCS] water inventory depends on the balance between SI flow rate (MHSI, accumulators, LHSI) and the break flow rate.

Prior to the time when SI flow is able to compensate for the break flow, the core may become uncovered. The fuel clad temperature could increase above the saturation level in the uncovered part of the core. The greater the core uncovery depth and duration, the higher the resulting clad temperature increase. In the case of core uncovery, specific criteria must be met in order to prevent any unacceptable core damage and to limit radiological consequences to the environment (see section 2.3.1.2).

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

- The reactivity is under control,

- The reactor coolant inventory is stable or increasing,

- Core cooling is ensured by water supplied to the RCP [RCS] by the RIS/RRA [SIS/RHRS], and RCP [RCS] heat removal via the break and the SGs if needed.

2.3.1.1.2.2. From the controlled state to the safe shutdown state

Transfer to LHSI/RHR conditions is generally not possible because the break flow cannot be compensated by water injection (except for the break sizes close to ∅50 mm considered in section 2.2.1), and thus the RCP [RCS] loops can not be reflooded to permit LHSI/RHR operation (the necessary conditions for LHSI/RHR suction from the hot legs are based on ΔT_{sat} and RPVL information).

In this case the safe shutdown state corresponds to the following conditions:

- The core is subcritical even after xenon depletion,
- The break flow is compensated by the RIS/RRA [SIS/RHRS] flow,
- The decay heat is being removed from the core,
- The break flow rate is at a temperature below the containment temperature limit defined by the equipment qualification envelope (see section 1 in Sub-chapter 6.2),
- Heat is being removed from the containment by the LHSI/RRI [CCWS]/SEC [ESWS] cooling chain within the design conditions for the containment, IRWST and RIS [SIS],
- The activity release is within the allowable limits for PCC-4 events.

The actions to be performed (initiated by the operator) to reach the safe shutdown are as follows:

- Switchover from LHSI cold leg injection to hot leg injection:

  The containment pressure increase is limited in the long term by switching LHSI/RHR injection into the hot legs, the consequence of which is to terminate the flow of steam out of the break (see section 1 in Sub-chapter 6.2) whatever the break location (cold leg or hot leg). The switchover of the LHSI from the cold legs to the hot legs also prevents the core boron concentration from continuing to increase in the case of cold leg breaks, preventing boron from precipitating inside the core. This operator action consists of opening valves in the LHSI hot leg injection lines, and closing valves located in the LHSI cold leg injection lines.

Note: cooldown by the secondary side is not required to reach the safe shutdown (after the partial cooldown), except for break sizes close to Ø50 mm, which are dealt with as a SB(LOCA). However secondary side cooling can be helpful to limit the SG energy release into the containment in the case of cold leg breaks. Secondary side cooldown is initiated by the operator by decreasing the VDA [MSRT] or GCT [MSB] setpoints on criteria linked to RPVL, ΔT_{sat} (subcooling margin) or high containment pressure, in accordance with emergency operating instructions.

The F1 systems involved to achieve the safe shutdown state are:

- The RIS/RRA [SIS/RHRS] (F1A),
- The RRI/SEC [CCWS/ESWS] (F1A part) as support systems to F1A systems.

2.3.1.2. Safety criteria

The safety criteria to be met are the dose equivalent limits as addressed in section 1.

For LOCA analysis the following decoupling criteria must be met (see section 1.1):

- The peak cladding temperature must remain lower than 1204°C,
- The maximum cladding oxidation must remain lower than 17% of the total cladding thickness,
- The maximum hydrogen generation must remain lower than 1% of the amount that would be generated if all the active part of the cladding were to react,
- The core geometry must remain coolable: calculated changes in core geometry must be such that the core remains amenable to cooling,
- Long term cooling must be ensured: the calculated core temperature must be maintained at an acceptably low value with decay heat being removed.
  It must be demonstrated that the two following safe states are reached with the application of the safety analysis rules defined in section 1:
- the controlled state, relying only on F1A means,
- the safe shutdown state, relying only on F1A and F1B means.

2.3.1.3. Methods and assumptions

2.3.1.3.1. Method

The CATHARE computer code is used.

The Technical Report 97/53 'CATHARE, small and IB(LOCA) methodology for PCC-3 and PCC-4 events' provides the description and the justification of the deterministic methodology used for the PCC LOCA analyses performed with the CATHARE code.

The basis of the qualification of CATHARE, and the basic principles of CATHARE deterministic methodology for LOCA, are described in the section 2.2.1.3.1.

The deterministic methodology provides a conservative result which can be compared directly with the decoupling criteria.

The CATHARE code provides a detailed representation of the primary and secondary systems. The broken loop is explicitly modelled; two intact loops are lumped into a single loop of weight 2; the fourth loop is modelled with the pressuriser connected to its hot leg.

The break is conservatively assumed to be located at the lower part of the cold leg for all postulated pipe ruptures in the cold leg.

The CATHARE point-kinetics model is not used; the residual fission power (term A) is introduced as an input to CATHARE. Term A is derived from a decoupled strongly conservative standalone RT-simulation (see Appendix 14B.2.2 – Table 1).

The transient analyses are performed using the conservative PCC-analysis rules defined in section 1.

With respect to the achievement of the controlled state (including the PCT calculation):

- A first CATHARE calculation is performed which models the entire RCP [RCS] and the relevant boundary conditions. In this calculation, the system calculation, only the characteristics of the average core assembly are modelled.
- A second CATHARE calculation is then performed, 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.

With respect to the achievement of the safe shutdown state (long term core cooling):

- A CATHARE/CONPATE coupled calculation is performed, with simultaneous modelling of the RCP [RCS] (system calculation) and the containment).

2.3.1.3.2. Main assumptions

2.3.1.3.2.1. Accident definition

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 (see section 0.1). The surge line is the largest line connected to the RCP [RCS] being an RCP [RCS] hot leg connection; the RIS [SIS] injection line is the largest line connected to the RCP [RCS] cold leg.

Intermediate breaks (IB(LOCA)) refer to the following break size spectrum:

- break size > 20 cm² (∅> 50 mm),
- break size < the break of the RIS [SIS] line.

The large breaks (LB(LOCA)) refer to the following breaks:

- double-ended break of the pressuriser surge line for the hot leg and
- break of the RIS [SIS] line for the cold leg side.

The IB and LB(LOCA) analyses presented in this section concentrate on core cooling effectiveness. The aim is to demonstrate that relevant decoupling criteria are met.

2.3.1.3.2.2. Protection and mitigation actions

In case of LOCA, automatic protection actions, by means of F1-A classified systems, are provided for tripping the reactor, removing residual heat and making up for inventory losses in order to allow the controlled state to be reached.

In accordance with the rules defined for safety analyses in section 1, the controlled state must be reached relying only on F1-A classified systems. The safe shutdown must be reached relying only on F1-A and F1-B classified systems.

The F1-A I&C signals relevant for this accident are:

- Reactor Trip, on pressuriser pressure < MIN2,
- Turbine Trip, on a RT signal,
- Safety Injection, on pressuriser pressure < MIN3,
- Reactor Coolant Pump trip on RCP [RCS] pressure drop < MIN1,
- VDA [MSRT] opening and pressure control, on SG pressure > MAX1,
- Partial cooldown, on SI signal,
- ASG [EFWS] injection, on SG level < MIN2,
- Isolation of APG [SGBS], on ASG [EFWS] actuation signal.

2.3.1.3.2.3. Operator actions

No operator actions are assumed before 30 minutes after reactor trip. After 30 minutes, operator actions are required to reach the safe shutdown.

In the case of smaller intermediate break sizes, for which transfer to LHSI/RHR operating conditions is possible, the operator actions are those mentioned in section 2.2.1.3.2.
For the larger break sizes where the RCP [RCS] reaches saturation conditions, the operator must perform the following actions:

- Switchover from LHSI cold leg injection to hot leg injection, by opening of LHSI hot leg injection valves and closing LHSI cold leg injection valves. The operator is assumed to perform this action at 1.5h after RT.
- Maintaining the MHSI injection into the cold legs.

The F1-B operator action, with indication of the main F1-B information needed, is:

- manual switchover of LHSI from CL to HL: containment pressure, RCP [RCS] pressure, ΔT_{sat} and RPVL.

2.3.1.4. Definition of cases studied

The analysis covers a spectrum of breaks up to the largest connecting line (surge line). For a given break size, a cold leg location is more severe in terms of core cooling than a hot leg location, or a pressuriser location including breaks in the pressuriser steam region. The cases studied in this section are listed below:

- 45 cm^2 (∅ 75 mm) break in the Reactor Coolant Pump discharge piping,
- 80 cm^2 (∅ 100 mm) break in the Reactor Coolant Pump discharge piping,
- 125 cm^2 (∅ 125 mm) break in the Reactor Coolant Pump discharge piping,
- 180 cm^2 (∅ 150 mm) break in the Reactor Coolant Pump discharge piping,

- RIS [SIS] line break (390 cm^2 - ∅ 225 mm),
- Surge line break (2 x 830 cm^2 - ∅ 2 x 325 mm).

The RIS [SIS] line corresponds to the largest nozzle in the Reactor Coolant Pump discharge piping, the surge line being the largest pipe connected to the RCP [RCS].

The surge line break is considered as a double ended guillotine break of the surge line, conservatively assumed to be located close to the hot leg.
These cases are analysed with the following aggravating assumptions:

- Most adverse additional single failure,
- Most adverse preventative maintenance state,
- Coincident LOOP.

Two CATHARE calculations are performed:

- the first CATHARE calculation ends when the controlled state is reached. Its object is to demonstrate that the decoupling criteria related to the controlled state (prevention of excessive fuel clad heat-up) are met, with the use of F1-A systems only. The calculation is described in section 2.3.1.5.
- the second CATHARE calculation ends when the safe shutdown state is reached. Its object is to demonstrate that the decoupling criteria related to the safe shutdown state (long term cooling after LHSI switchover to hot leg injection) are met, with the use of F1A and F1B systems only. In this calculation CATHARE is coupled with the CONPATE containment code. The calculation is described in section 2.3.1.6.

2.3.1.5. Description of calculations for accident phase up to the controlled state

The safety analyses are performed using conservative assumptions, as discussed below.

2.3.1.5.1. Choice of single failure and preventative maintenance

The most adverse single failure is the loss of 1 emergency diesel at the time of LOOP, which makes 1 RIS/RRA [SIS/RHRS] train (1 MHSI pump + 1 LHSI pump) and 1 ASG [EFWS] pump unavailable.

The preventative maintenance of 1 emergency diesel is the most adverse maintenance configuration because 1 RIS/RRA [SIS/RHRS] train (1 MHSI pump + 1 LHSI pump) and 1 ASG [EFWS] pump are made unavailable.

The assumption of LOOP is pessimistic in IB and LB(LOCA) analyses because of application of the SFC and preventative maintenance principles.

2.3.1.5.2. Initial state

Initial conditions are shown in given in Appendix 14B.2.3 - Table 1. These conditions are chosen to maximise the delay in the RT and SI signals, which is pessimistic for IB and LB(LOCA) mitigation. The initial water temperature in the RPV dome is maximised (assumed to be the hot leg temperature), which is pessimistic for the IB and LB(LOCA) transients with respect to core cooling.

The axial power shape for the average rod in the average assembly is shown in Appendix 14B.2.3 - Figure 1. This power shape, which is skewed at the top of the core, is characterised as follows:

- Average linear power of 182.2 W/cm 102% NP,
- Enthalpy rise factor of 1,
- Peaking factor of 1.42 at 3.5 m above the active core base,
- Axial offset of 18% (15% + 3% uncertainty).

The axial power shape for the hot rod in the hot assembly is shown in Appendix 14B.2.3 - Figure 2. This power shape, which is skewed at the top of the core, is characterised as follows:

- Maximum linear power of 470 W/cm at 102% NP,
- Enthalpy rise factor of 1.80,
- Hot spot peaking factor of 2.58 at 3.5 m above the active core base
- Axial offset of 21% (18% + 3% uncertainty).

The ratio between the hot rod power and the hot assembly power is 1.08.

These power shapes are chosen because they are conservative and maximise the power in the upper part of the core, which is limiting for IB and LB(LOCA) analysis because of core uncovering process. As the core uncovers the cladding in the upper part of the core heats up making the peak temperature sensitive to the linear power in the upper core: the cladding temperature in the lower part of the core remains close to the saturation temperature.

The initial fuel pellet temperature is shown in Appendix 14B.2.3 - Figure 3. This temperature, which is an average for a section of fuel, is prescribed as a boundary condition. The initial temperature does not have a significant impact on the maximum cladding temperature reached, because the core heats up well after RT in the typical LOCA sequences considered in this section.

2.3.1.5.3. Specific assumptions

2.3.1.5.3.1. Neutronic data and decay heat

The core power is assumed to be at 102% of full power until reactor trip. After RT, the maximum decay heat curve as described in section 0.2.4 is used for terms B + C (fission products). The residual fission heat (term A) is introduced as an input to CATHARE (the CATHARE point-kinetics model is not used). The A term is obtained from a separate conservative RT-simulation (see Appendix 14B.2.2 – Table 1).

2.3.1.5.3.2. Assumptions regarding non-F1 systems

Non classified systems (control systems etc.) are disregarded if they perform a beneficial function.

The pressuriser pressure control system (pressuriser heaters) is modelled because it delays the RT signal. A total heater power of 2500 kW is applied until the pressuriser is totally empty.

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

The MFW flow is assumed to be constant until reactor trip.

The GCT [MSB] and AAD [SSS] are disregarded.

The RCV [CVCS] is not modelled before its isolation. The letdown line is normally isolated on low pressuriser level (not F1A) and on the SI signal (F1A). The modelling is conservative with respect to the RCP [RCS] water inventory balance.

2.3.1.5.3.3. Assumptions related to F1 systems

The assumptions are the same as those applied in the analysis of SB(LOCA) - see section 2.2.1.5.3. For analysis of IB/LB(LOCA), ignoring the SI flow rate into the broken leg has the most significant impact.

In case of hot leg breaks (surge line break), it is not appropriate to assume that the emergency cooling water from one RIS/RRA [SIS/RHRS] train is lost directly to the break. Therefore in accordance with the modelling principles described in section 2.3.1.3.1, the RIS [SIS] modelling in those cases is as follows:

- two RIS/RRA [SIS/RHRS] trains are available to inject into the double-weighted intact loop,
- four accumulators are available to inject into the intact and broken loops.

2.3.1.5.3.4. Other assumptions

The Loss of Off-site Power is assumed to occur at the time of turbine trip. Reactor Coolant Pump trip is assumed to occur at time of the reactor trip, which is the most pessimistic time within the time period from reactor trip to Reactor Coolant Pump trip signal occurrence (Reactor Coolant Pump trip signal "ΔP over Reactor Coolant Pump ", with setpoint 80 $\pm$ 5% nominal ΔP).

2.3.1.5.4. Results

The analysis results for a range of break sizes and the locations are presented below as event sequences and time histories for the most representative parameters.

The event sequences given in the following tables:

- Appendix 14B.2.3 - Table 2: 45 cm^2 ($\varnothing$ 75 mm) cold leg break,
- Appendix 14B.2.3 - Table 3: 80 cm^2 ($\varnothing$ 100 mm) cold leg break,
- Appendix 14B.2.3 - Table 4: 125 cm^2 ($\varnothing$ 125 mm) cold leg break,

- Appendix 14B.2.3 - Table 5: 180 cm^2 (∅ 150 mm) cold leg break,
- Appendix 14B.2.3 - Table 6: RIS [SIS] line break (390 cm^2 - ∅ 225 mm),
- Appendix 14B.2.3 - Table 7: Surge line break (2 x 830 cm^2 - 2 x ∅ 325 mm).

Time histories of representative parameters are presented as follows:

- Appendix 14B.2.3 - Figure 4: 45 cm^2 (∅ 75 mm) cold leg break
  RCP [RCS] and secondary side water inventories
  RCP [RCS] and secondary side pressures
- Appendix 14B.2.3 - Figure 5: 45 cm^2 (∅ 75 mm) cold leg break
  Total and steam break flow rates
  Total break and RIS [SIS] flow rates
- Appendix 14B.2.3 - Figure 6: 45 cm^2 (∅ 75 mm) cold leg break
  Hot spot peak cladding temperature
  Core swell level
- Appendix 14B.2.3 - Figure 7: 80 cm^2 (∅ 100 mm) cold leg break
  RCP [RCS] and secondary side water inventories
  RCP [RCS] and secondary side pressures
- Appendix 14B.2.3 - Figure 8: 80 cm^2 (∅ 100 mm) cold leg break
  Total flow rate and steam flow rate through break
  Total break and RIS [SIS] flow rates
- Appendix 14B.2.3 - Figure 9: 80 cm^2 (∅ 100 mm) cold leg break
  Hot spot peak cladding temperature
  Core swell level
- Appendix 14B.2.3 - Figure 10: 125 cm^2 (∅125 mm) cold legs break
  RCP [RCS] and secondary side water inventories
  RCP [RCS] and secondary side pressures
- Appendix 14B.2.3 - Figure 11: 125 cm^2 (∅ 125 mm) cold leg break
  Total flow rate and steam flow rate through break
  Total break and RIS [SIS] flow rates
- Appendix 14B.2.3 - Figure 12: 125 cm^2 (∅ 125 mm) cold leg break
  Hot spot cladding temperature
  Core swell level
- Appendix 14B.2.3 - Figure 13: 180 cm^2 (∅150 mm) cold legs break
  RCP [RCS] and secondary side water inventories
  RCP [RCS] and secondary side pressures
- Appendix 14B.2.3 - Figure 14: 180 cm^2 (∅ 150 mm) cold leg break
  Total flow rate and steam flow rate through break
  Total break and RIS [SIS] flow rates

- Appendix 14B.2.3 - Figure 15: 180 cm^2 (∅ 150 mm) cold leg break
  Hot spot cladding temperature
  Core swell level
- Appendix 14B.2.3 - Figure 16: RIS [SIS] line break (390 cm^2 - ∅ 225 mm)
  RCP [RCS] and secondary side water inventories
  RCP [RCS] and secondary side pressures
- Appendix 14B.2.3 - Figure 17: RIS [SIS] line break (390 cm^2 - ∅ 225 mm)
  Total flow rate and steam flow rate through break
  Total break and RIS [SIS] flow rates
- Appendix 14B.2.3 - Figure 18: RIS [SIS] line break (390 cm^2 - ∅ 225 mm)
  Hot spot cladding temperature
  Core swell level
- Appendix 14B.2.3 - Figure 19: Surge line break (2 x 830 cm^2 - 2 x ∅ 325 mm)
  RCP [RCS] and secondary side water inventories
  RCP [RCS] and secondary side pressures
- Appendix 14B.2.3 - Figure 20: Surge line break (2 x 830 cm^2 - 2 x ∅ 325 mm)
  Total flow rate and steam flow rate through break
  Total break and RIS [SIS] flow rates
- Appendix 14B.2.3 - Figure 21: Surge line break (2 x 830 cm^2 - 2 x ∅ 325 mm)
  Void fraction in the hot assembly
  Hot spot peak cladding temperature

It should be noted that residual heat term A (residual fissions) which is input to the CATHARE calculation, is derived using very conservative neutronic and thermal-hydraulic data (see Appendix 2.2 – Table 1), which do not account for the reactivity reduction due to core voidage. These data assumptions are highly conservative for break sizes above 180 cm^2 (∅ 150mm), and result in a very pessimistic calculation of core heat-up during the first core uncovery.

Significant results are given in Appendix 14B.2.3 - Table 8; the maximum hot spot cladding temperature, hot spot clad oxidation percentage, and clad rupture rate are presented.

The worst case break is the 80 cm^2 (∅ 100 mm) cold leg break. The peak clad temperature reaches 900°C, and the hot spot clad oxidation is 1.5%.

The following conclusions can be drawn with regard to IB and LB(LOCA):

- The maximum cladding temperature is below the decoupling criterion (1204°C),
- The maximum percentage of total cladding thickness oxidised at the hot spot is less than the limit value (17%),
- There is no cladding rupture[(1)],
- Integrity of the core geometry is maintained,

[(1)] Note: Based on an assumed initial internal rod pressure of 90 bar. A higher pressure could be assumed without resulting in a significant change to the maximum cladding temperature.

- Note that long term core cooling is addressed in section 2.3.1.

The overall conclusion is that operation of the F1A systems (RIS/RRA [SIS/RHRS] in conjunction with the VDA [MSRT] and ASG [EFWS] systems) is sufficient to allow the controlled state to be reached.

2.3.1.6. Description of cases studied: controlled state to safe shutdown

The transfer to the safe shutdown state is achieved either by reaching LHSI/RHR connection conditions, or by switching over from LHSI cold leg injection to hot leg injection, depending on the plant state.

a) Safe shutdown state: LHSI into the CL + LHSI system operating in RHR-mode

For smaller break sizes within the PCC-4 break spectrum, transfer to long term cooling using the LHSI/RHR system may be possible, as for the PCC-3 SB(LOCA) events described in section 2.2.1.6. Connection of LHSI/RHR is possible when the following RCP [RCS] conditions are achieved:

- 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 these smaller PCC-4 breaks, the consumption of ASG [EFWS] feed-water before connection of the LHSI/RHR, is lower than for the PCC-3 SB(LOCA), since the larger break size results in more residual heat being removed through the break (decreasing the heat transferred to the SG secondary side) and a reduced time to reach the LHSI/RHR connection conditions (due to the faster RCP [RCS] depressurisation). The water inventory in the ASG [EFWS] tanks is thus generally sufficient to allow LHSI/RHR connection conditions to be reached.

The safe shutdown state is reached at time of connection of 1 LHSI in RHR-mode (1 LHSI/RHR is sufficient for heat removal), with the remaining LHSI operating in SI-mode (possibly enhanced by MHSI with large mini-flow line open). Attainment of the safe shutdown is state is dealt with in section 2.2.1.6.

b) Safe shutdown: MHSI into the CL + LHSI into the HL

For the larger break sizes of the PCC-4 break spectrum, the degradation RPVL and ΔT_{sat} conditions does not allow connection of the LHSI in RHR mode until at least 1.5 hours after reactor trip. In such cases, switch-over of LHSI pumps into RCP [RCS] hot legs is performed by the operator, while MHSI into the cold legs is maintained.

Switch-over of LHSI from the cold leg to the hot legs is required for the following reasons:

- to stop steaming and prevent occurrence of excessive pressures and temperatures inside the containment. Indeed, in the case of a cold leg break, only part of the RIS [SIS] flow injected into the cold legs enters the core (steaming through the core), the remainder being discharged directly from the break. To terminate core steaming, RIS [SIS] injection into the core outlet is needed. LHSI flow injected into the hot legs must be able to enter the upper plenum.

- to stop the increase in core boron concentration before the boron crystallisation limit is reached. In case of a cold leg break where only steam escapes from the core (in which the boron concentration is negligible), the core boron concentration would continue to increase due to the continued injection of boron in the coolant injected into the cold leg. To arrest the boron concentration increase, it is necessary to begin discharging liquid from the core. Switching LHSI into the hot legs results in liquid coolant being discharged from the core.

The timing of the switch-over must be:

- early enough so that steaming is stopped before pressure or temperature limits in the containment are exceeded,
- early enough to prevent unacceptable boron dilution in the IRWST, which could lead to a return to core criticality on initiation of hot leg injection (insufficiently borated water entering the core after the boron content has been reduced by liquid exiting the core),
- late enough to ensure that the flow injected into the hot leg is able to enter the RPV upper plenum and then the core, without significant impairment of core cooling (no unacceptable core uncovery).

A switch-over time of 1.5 hours after RT, is presently envisaged.

For the PCC-4 breaks considered above, the safe shutdown state is achieved after completion of the switch-over. For smaller break sizes, RCP [RCS] cooldown (via the SGs) is needed to reach the LHSI injection pressure (~20 bar). The RCP [RCS] cooldown capacity must allow LHSI to be initiated, before the core boron crystallisation and IRWST boron dilution limits are reached. The safety analysis presented below is demonstrates that the RIS [SIS] design is capable of ensure that all the safety and decoupling criteria are met before switch-over (from controlled state to safe shutdown state) and after switch-over (stable long term safe shutdown state), with use of only F1A and F1B systems.

The decoupling criterion considered relate to long term core cooling, (see section 2.3.1.2). The following must be demonstrated:

1 - adequacy of core cooling before switch-over at 1.5 hours, with MHSI and LHSI into CL,

2 - adequacy of cooling after switch-over at 1.5 hours, with MHSI into CL and LHSI into HL,

3 - existence of liquid discharge from the core after switch-over at 1.5 hours,

4 - core boron crystallisation limit not exceeded before switch-over at 1.5 hours,

5 - IRWST boron dilution limit not exceeded before switch-over at 1.5 hours.

The consequences of the IB/LB(LOCA) with respect to the containment are addressed in section 2.3.1.

The safety analysis is performed in two parts as follows:

- The first part of the analysis consists of a thermal-hydraulic transient calculation of the most severe IB/LB(LOCA) using the coupled CATHARE/CONPATE codes. CATHARE calculates the RCP [RCS] and SG behaviour; CONPATE calculates the containment behaviour, including the IRWST conditions.

- The second part of the analysis consists of a conservative estimation of the latest switch-over times which could lead to exceedance of the boron crystallisation limit in the core (risk of boron precipitation), or the boron dilution limit in the IRWST (risk of core recriticality). This assessment uses enveloping mass and energy balances and conservative boundary conditions with respect to core boration and IRWST dilution. The objective is to show that the switch-over time of 1.5 hours is well below these limiting values.

All the analyses are performed using conservative assumptions for the achievement of the controlled state. The number of LHSI and MHSI pumps available is minimised, considering the most adverse single failure and preventative maintenance state.

As discussed previously, the limiting RCP [RCS] breaks with respect to the phenomena of concern are cold leg breaks. For hot leg breaks, switch-over is not necessary, cold leg injection being the most efficient injection mode (all the coolant injected into the CL flows into the core).The most severe PCC-4 IB/LB(LOCA) event is the largest cold leg break, i.e. the 390 cm^2 RIS/RRA [SIS/RHRS] line break.

2.3.1.6.1. Decoupling criteria

With regard to boron concentration limits, the following values are used:

The limit in boron solubility in water at 100°C is 27.5% w/o boric acid solution. Taking into account a 4% w/o margin for uncertainties, this becomes 23.5%, which corresponds to a limiting core boron concentration of 41130 ppm.

The minimum boron concentration required at cold shutdown to avoid a return to core criticality is 1920 ppm natural boron. This value covers all fuel cycles and UO2/MOX fuel management schemes, with consideration of uncertainties, and assumes all rods inserted except the one with the highest negative reactivity worth, which is assumed to be stuck in its fully withdrawn position.

The boron concentration limits are thus:

- core boron concentration < 41130 ppm, to prevent crystallisation,

- IRWST boron concentration >1920 ppm natural boron, to prevent core criticality.

2.3.1.6.2. Choice of single failure and preventative maintenance assumption

The most adverse single failure is loss of 1 emergency diesel generator at the time of LOOP. One RIS/RRA [SIS/RHRS] train (1MHSI pump + 1 LHSI pump) and 1 ASG [EFWS] pump are thus made unavailable.

The preventative maintenance on 1 diesel is the most adverse maintenance configuration because one RIS/RRA [SIS/RHRS] train (1 MHSI pump + 1 LHSI pump) and 1 ASG [EFWS] pump are made unavailable.

The assumption of LOOP is pessimistic for IB and LB(LOCA) analyses because of application of the SFC and preventative maintenance principles.

2.3.1.6.3. Specific assumptions

a) Assumptions concerning the secondary side

Because of the application of the single failure and the preventative maintenance principles, only two ASG [EFWS] are available to deliver feed to the SGs. The flow rate per pump is the same as used in the analysis up to the controlled state. The SG level is controlled to prevent SG overfeeding.

The SG pressure is kept unchanged after the end of partial cooldown being assumed to remain at 61.5 bar (60 bar+1.5 bar uncertainty) applicable in the controlled state. This assumption maximises reverse heat transfer from the SG to the RCP [RCS], which is conservative for core cooling (increased steam binding) and for the IRWST water temperature (higher mass and energy releases into the containment).

b) Assumptions concerning the RIS/RRA [SIS/RHRS]

At the controlled state, only one RIS/RRA [SIS/RHRS] train (1 MHSI and 1 LHSI pumps) is assumed to be injecting into 1 RCP [RCS] cold leg. 2 SIS/RHRS trains are assumed unavailable due to the single failure and preventative maintenance assumptions and one further train is assumed to delivering coolant into the containment directly through the break and is unavailable for RCP [RCS] water injection.

1.5 hours after the reactor trip, the operator performs the switch-over of all the LHSI pumps from the cold legs to the hot legs. The LHSI pump which is already effective and the LHSI pump previously assumed to be delivering to the break are now considered to be available for injection into 2 RCP [RCS] hot legs.

Minimum characteristics for the MHSI and LHSI pumps are assumed (see section 0.2).

The accumulators are conservatively assumed to be isolated at the time of first operator action, i.e. 30 minutes after the reactor trip.

The initial IRWST temperature is assumed to be at a maximum value of 50°C. The initial IRWST water volume is assumed to be at a minimum in order to maximise the IRWST temperature increase. The initial volume is set equal to 1300 m^3, (see Appendix 14B.0.2 - Table16)

The MHSI and LHSI water temperature increase is also maximised by making allowance for the IRWST temperature increase due to the degraded containment conditions. This temperature is obtained from a coupled CATHARE/CONPATE calculation.

CATHARE/CONPATE thermal-hydraulic analysis

The initial boron concentration in the IRWST is set to a conservative value of 1800 ppm enriched boron, which is enveloping for MOX and UO2 fuel management schemes. Dilution in the IRWST is not considered in this calculation in contrast with the boron assessment described below. (This is because the CATHARE boron calculation is only used to provide qualitative information on the termination of the boron concentration increase after switch-over).

Boron concentration assessment

With respect to the estimation of the minimum switch-over time at which the 41130 ppm core crystallisation limit could be reached, the following assumptions are made which maximise the core boron concentration:

- Initial IRWST water volume: 1940 m^3 (maximum value)
- Initial IRWST boron concentration: 1800 ppm (maximum value)
- Initial accumulator boron concentration: 1800 ppm (maximum value)
- Initial RCP [RCS] boron concentration: 1500 ppm (maximum value) - (BOL – enveloping value for UO2/MOX fuel management schemes)
- Initial accumulator water volume: 35 m^3 per accumulator (maximum value)
- Initial RCP [RCS] water mass: 300 tons

Note: in this calculation, the boron concentrations refer to enriched boron.

For the estimation of the minimum switch-over time at which the 1920 ppm IRWST dilution limit could be reached, the following assumptions are made so as to minimise the IRWST boron concentration:

- Initial IRWST water volume: 1300 m^3 (minimum value)
- Initial IRWST boron concentration: 2300 ppm for UO2, and 2700 ppm for MOX (minimum value corresponding in both cases to a 1600 ppm enriched boron)
- Initial accumulator boron concentration: same as IRWST
- Initial RCP [RCS] boron concentration: 1530 ppm for UO2, and 2100 ppm for MOX (minimum values at Beginning of Life (BOL))
- Initial accumulator water volume: 30 m^3 per accumulator (minimum value)
- Initial RCP [RCS] water volume: 300 tons

Note: in this calculation, the boron concentrations refer to natural boron.

c) Residual heat

The maximum decay heat curve described in section 0.2.4 is used, similar to that applied in section 2.3.1.5.3.1 for the calculation of the controlled state.

2.3.1.6.3. Results

a) CATHARE-CONPATE thermal-hydraulic analysis

Time histories of representative parameters are presented as follows:

- Appendix 14B.2.3 - Figure 22: RIS [SIS] line break (390 cm^2 - ∅ 225 mm)
  RCP [RCS] and secondary side pressures
  RCP [RCS] core water level
- Appendix 14B.2.3 - Figure 23: RIS [SIS] line break (390 cm^2 - ∅ 225 mm)
  MHSI and LHSI pumps injection
  Break flow rate
- Appendix 14B.2.3 - Figure 24: RIS [SIS] line break (390 cm^2 - ∅ 225 mm)
  IRWST/MHSI water temperature
  LHSI water temperature
- Appendix 14B.2.3 - Figure 25: RIS [SIS] line break (390 cm^2 - ∅ 225 mm)
  RPV inlet integrated liquid flow rates
  RPV outlet integrated liquid flow rates
- Appendix 14B.2.3 - Figure 26: RIS [SIS] line break (390 cm^2 - ∅ 225 mm)
  Liquid flow rate circulation in the RPV before switch-over
  Liquid flow rate circulation in the RPV after switch-over
- Appendix 14B.2.3 - Figure 27: RIS [SIS] line break (390 cm^2 - ∅ 225 mm)
  Boron concentration transient in the core

Appendix 14B.2.3 - Figures 25, 26, and 27 show that, at switch-over at 1.5 hours after the reactor trip, the LHSI flow rates injected into the hot legs are sufficient to establish negative flow in the core, to dilute the core boron and to remove the core residual heat, even for the largest cold leg break. The core always remains covered after the controlled state is reached (Appendix 14B.2.3 - Figure 22), and the IRWST temperature remains well below the design limit for RIS [SIS] pump operation (Appendix 14B.2.3 - Figure 24).

In the case of a hot leg break, assuming the most adverse single failure and preventative maintenance state, two MHSI pumps are available to deliver into the cold legs at the switch-over time of 1.5 hours. At this time, the minimum MHSI flow rate available to be injected is sufficient to remove about 3.5 times the core residual power under saturated steam conditions.

It is thus seen that, the RIS/RRA [SIS/RHRS] injection and heat removal capacities are sufficient to remove core residual heat, without core uncovery, before and after the switch-over at 1.5 hours after RT.

To ensure effective heat removal, it must also be confirmed that the core geometry remains coolable (no boron crystallisation), and that core re-criticality is prevented (unacceptable IRWST dilution avoided). This confirmation is provided by the Boron Assessment described below.

b) Boron concentration assessment

The estimation of the latest switch-over time to avoid boron crystallisation in the core uses a simplified conservative mass and energy balance. The following assumptions are made:

- only steam is assumed to escape from the core (no boron discharge),
- steam flow rate based on the maximum decay heat curve,
- steam flow rate discounts the effect of any subcooling within the RCP [RCS] (steaming from saturated water),
- minimum volume of liquid assumed for boron concentration calculation (only core and lower-plenum volumes taken into account)

On this basis the earliest switch-over time at which the 41130 ppm core crystallisation limit could be reached is calculated as 18 hours after RT (assuming an initial maximum IRWST boron concentration of 1800 ppm enriched boron). This is an enveloping value for both UO2 and MOX fuel management schemes. There is consequently a high confidence that core boron crystallisation will be prevented with a switch-over time of 1.5 h. The core geometry will thus remain coolable.

The estimation of the latest switch-over time to avoid unacceptable boron dilution in the IRWST relies on a simplified conservative mass and energy balance. The following assumptions are made:

- only steam is assumed to escape from the core (no boron discharge),
- steam flow rate based on the maximum decay heat curve,
- steam flow rate discounts the effect of any subcooling within the RCP [RCS] (steaming from 100^{o}C saturated water),
- minimum volume of liquid assumed for boron concentration calculation (only core and lower-plenum volumes taken into account)

On this basis, the earliest switch-over time at which the 1920 ppm IRWST dilution limit could be reached is 3 hours after RT (assuming an initial minimum IRWST boron concentration of 1600 ppm enriched boron). This is an enveloping value for both UO2 and MOX fuel management schemes. There is consequently a high confidence that core re-criticality will be avoided with a switch-over time of 1.5 hours.

It is concluded that with a switch-over time of 1.5 hours after RT there is an insignificant risk of boron crystallisation in the core and unacceptable dilution in the IRWST. In summary

- Switch-over time considered: 1.5 hours after RT,
- Latest switch-over time to avoid risk of core boron crystallisation: 18 hours after RT,
- Latest switch-over time to avoid risk of core re-criticality: 3 hours after RT[(1)].

[(1)] The RCP [RCS] cool-doown capacity is able to reach the LHSI injection conditions (p≈ 20bar) earlier than 3 hours for the smaller break sizes.

Conclusion

The capability of RIS/RRA [SIS/RHRS] design to ensure adequate core heat removal before and after switch-over at 1.5 hours after RT, avoiding boron crystallisation in the core and the risk of re-criticality due to boron dilution in the IRWST, has been confirmed.

The present analysis has shown that despite assuming the most adverse single failure and preventative maintenance state, that:

- the safe shutdown can be reached with fulfilment of all safety criteria, using the following F1 means:
  - for 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 exchanger capacity for IRWST heat removal,
    - The MHSI, accumulators, and LHSI if needed, cold leg injection capacities from IRWST for maintaining the RCP [RCS] water inventory,
    - The MHSI and LHSI boron injection capacities for maintaining core subcriticality.
  - at switch-over at 1.5 hours after RT, corresponding to the attainment of the safe shutdown state,
    - The LHSI/RHR heat exchange capacity for IRWST heat removal,
    - The LHSI hot leg injection capacity to maintain the RCP [RCS] water inventory,
    - The MHSI cold leg injection capacity for maintaining the RCP [RCS] water inventory,
    - The MHSI and LHSI boron injection capacities for maintaining core subcriticality.

2.3.1.7. Conclusions

The present analysis shows that despite assuming the most adverse single failure and preventative maintenance state:

- the controlled state can be reached fulfilling all safety criteria,
- the safe shutdown can be reached fulfilling all safety criteria and without exceeding:
  - RIS/RRA [SIS/RHRS] design conditions (IRWST temperature),
  - the containment design pressure and temperature,
  - the containment equipment qualification pressure and temperature limit.

APPENDIX 14B.2.3 – TABLE 1

Initial Conditions - IB and LB(LOCA) (state A, PCC-4)

Parameters	Limiting values used
Reactor coolant system	
Initial reactor power (% of nominal power)	100 + 2 = 102
Initial average RCP [RCS] temperature (°C)	311 + 2.5 = 313.5
Initial reactor coolant pressure (bar)	155 + 2.5 = 157.5
Reactor cooling flow (kg/s)	22240 (thermal-hydraulic flow rate)
Pressuriser water volume / level (m^3 / m)	43.4 / 7.4 (nominal + 5% MR)
Dome liquid temperature (°C)	332 (maximum = hot leg temp.)
Steam generators	
Initial steam pressure (bar)	76.9
Initial SG level (m)	16.2 (nominal)
Feedwater	
Main feedwater flow (% of nominal flow)	100 + 2 = 102
Initial MFW temperature (°C)	232

APPENDIX 14B.2.3 – TABLE 2

Sequence of events - 45 cm^2 (∅ 75 mm) cold leg break - (State A, PCC-4)

TIME	EVENT
0.0 s	Break opening
40.3 s	Turbine Trip, LOOP, RCP [RCS] trip, RT signal on pressuriser pressure < MIN2 (132.0 bar)
156 s	Safety Injection & Partial Cooldown signal
159 s	Beginning of Partial Cooldown (SG pressure > VDA [MSRT] setpoint)
172 s	Pressuriser empty
197 s	Starting of MHSI & LHSI pumps (Loop 2)
767 s	Beginning of MHSI (RCP [RCS] pressure < 80 bar)
1140 s (1) 1165 s (2)	Beginning of ASG [EFWS] (SG1 level < 7.1 m)
2140 s	Beginning of core heat-up
2635 s	Secondary side no longer needed (RCP [RCS] pressure < SG pressure)
3135 s	Beginning of accumulator injection (RCP [RCS] pressure < 45 bar)
3275 s	End of core heat-up
4500 s	End of calculation

(1): related to SG (SG2) connected to an intact loop
(2): related to SG (SGR) connected to the broken loop

APPENDIX 14B.2.3 – TABLE 3

Sequence of events - 80 cm^2 (∅ 100 mm) cold leg break - (State A, PCC-4)

TIME	EVENT
0.0 s	Break opening
23.1 s	Turbine Trip, LOOP, RCP [RCS] trip, RT signal on pressuriser pressure < MIN2 (132.0 bar)
47.3 s	Safety Injection & Partial Cooldown signal
50.3 s	Beginning of Partial Cooldown (SG pressure > VDA [MSRT] setpoint)
88.2 s	Starting of MHSI & LHSI pumps (Loop 2)
115 s	Pressuriser empty
615 s	Beginning of MHSI (RCP [RCS] pressure < 80 bar)
990 s	Beginning of core heat-up
1125 s	Secondary side no longer needed (RCP [RCS] pressure < SG pressure)
1360 s	Beginning of accumulator injection (RCP [RCS] pressure < 45 bar)
2400 s	End of core heat-up
2500 s	End of calculation

SG1: related to 2 unaffected SG (weight 2)

APPENDIX 14B.2.3 – TABLE 4

Sequence of events - 125 cm^2 (∅ 125 mm) cold leg break - (State A, PCC-4)

TIME	EVENT
0.0 s	Break opening
16 s	Turbine Trip, LOOP, RCP [RCS] trip, RT signal on pressuriser pressure < MIN2 (132.0 bar)
33.7 s	Safety Injection & Partial Cooldown signal
36.7 s	Beginning of Partial Cooldown (SG pressure > VDA [MSRT] setpoint)
58	Pressuriser empty
74.6 s	Starting of MHSI & LHSI pumps (Loop 2)
473 s	Secondary side no longer needed (RCP [RCS] pressure < SG pressure)
486 s	Beginning of MHSI (RCP [RCS] pressure < 80 bar)
500 s	Beginning of core heat-up
805 s	Beginning of accumulator injection (RCP [RCS] pressure < 45 bar)
1400 s	End of core heat-up
2500 s	End of calculation

SG1: related to 2 unaffected SG (weight 2)

APPENDIX 14B.2.3 – TABLE 5

Sequence of events - 180 cm^2 (∅ 150 mm) cold leg break - (State A, PCC-4)

TIME	EVENT
0.0 s	Break opening
13 s	Turbine Trip, LOOP, RCP [RCS] trip, RT signal on pressuriser pressure < MIN2 (132.0 bar)
29.2 s	Safety Injection & Partial Cooldown signal
32.2 s	Beginning of Partial Cooldown (SG pressure > VDA [MSRT] setpoint)
38 s	Pressuriser empty
70.1 s	Starting of MHSI & LHSI pumps (Loop 2)
200 s	First core heat-up under conservative boiling conditions (*)
262 s	End of first core heat-up
311 s	Secondary side no longer needed (RCP [RCS] pressure < SG pressure)
344 s	Beginning of MHSI (RCP [RCS] pressure < 80 bar)
512 s	Beginning of 2nd core heat-up
577 s	Beginning of accumulator injection (RCP [RCS] pressure < 45 bar)
688 s	End of 2nd core heat-up
1418 s	Beginning of LHSI (RCP [RCS] pressure < 20 bar)
2500 s	End of calculation

SG1: related to 2 unaffected SG (weight 2)

APPENDIX 14B.2.3 – TABLE 6

Sequence of events - RIS [SIS] line break (390 cm^2 - ∅ 225 mm) - (State A, PCC-4)

TIME	EVENT
0.0 s	Break opening
9.6 s	Turbine Trip, LOOP, RCP [RCS] trip, RT signal on pressuriser pressure < MIN2 (132.0 bar)
21.2 s	Safety Injection & Partial Cooldown signal
24.2 s	Beginning of Partial Cooldown (SG pressure > VDA [MSRT] setpoint)
30 s	Pressuriser empty
62.1 s	Starting of MHSI & LHSI pumps (Loop 2)
100 s	Beginning of core heat-up
143 s	Secondary side no longer needed (RCP [RCS] pressure < SG pressure)
174 s	Beginning of MHSI (RCP [RCS] pressure < 80 bar)
256 s	Beginning of accumulator injection (RCP [RCS] pressure < 45 bar)
350 s	End of core heat-up
359 s	Beginning of LHSI (RCP [RCS] pressure < 20 bar)
1000 s	End of calculation

SG1: related to 2 unaffected SG (weight 2)

APPENDIX 14B.2.3 – TABLE 7

Sequence of events - Surge Line Break (2 x 830 cm^2 - 2 x ∅ 325 mm) - (State A, PCC-4)

TIME	EVENT
0.0 s	Break opening
5.2 s	Turbine Trip, LOOP, RCP [RCS] trip, RT signal on pressuriser pressure < MIN2 (132.0 bar)
15 s	Pressuriser empty
16 s	Beginning of Partial Cooldown (SG pressure > VDA [MSRT] setpoint)
8.9 s	Safety Injection & Partial Cooldown signal
60 s	Secondary side no longer needed (RCP [RCS] pressure < SG pressure)
49.8 s	Starting of MHSI & LHSI pumps (Loop 1)
65 s	Beginning of core heat-up
108 s	Beginning of MHSI (RCP [RCS] pressure < 80 bar)
139 s	Beginning of accumulator injection (RCP [RCS] pressure < 45 bar)
156 s	Beginning of LHSI (RCP [RCS] pressure < 20 bar)
385 s	End of core cladding heat-up by rewetting (*)
495 s	Accumulator empty (4/4)
1000 s	End of calculation

Loop1: 2 RIS [SIS] trains inject into loop1 (weight 2)

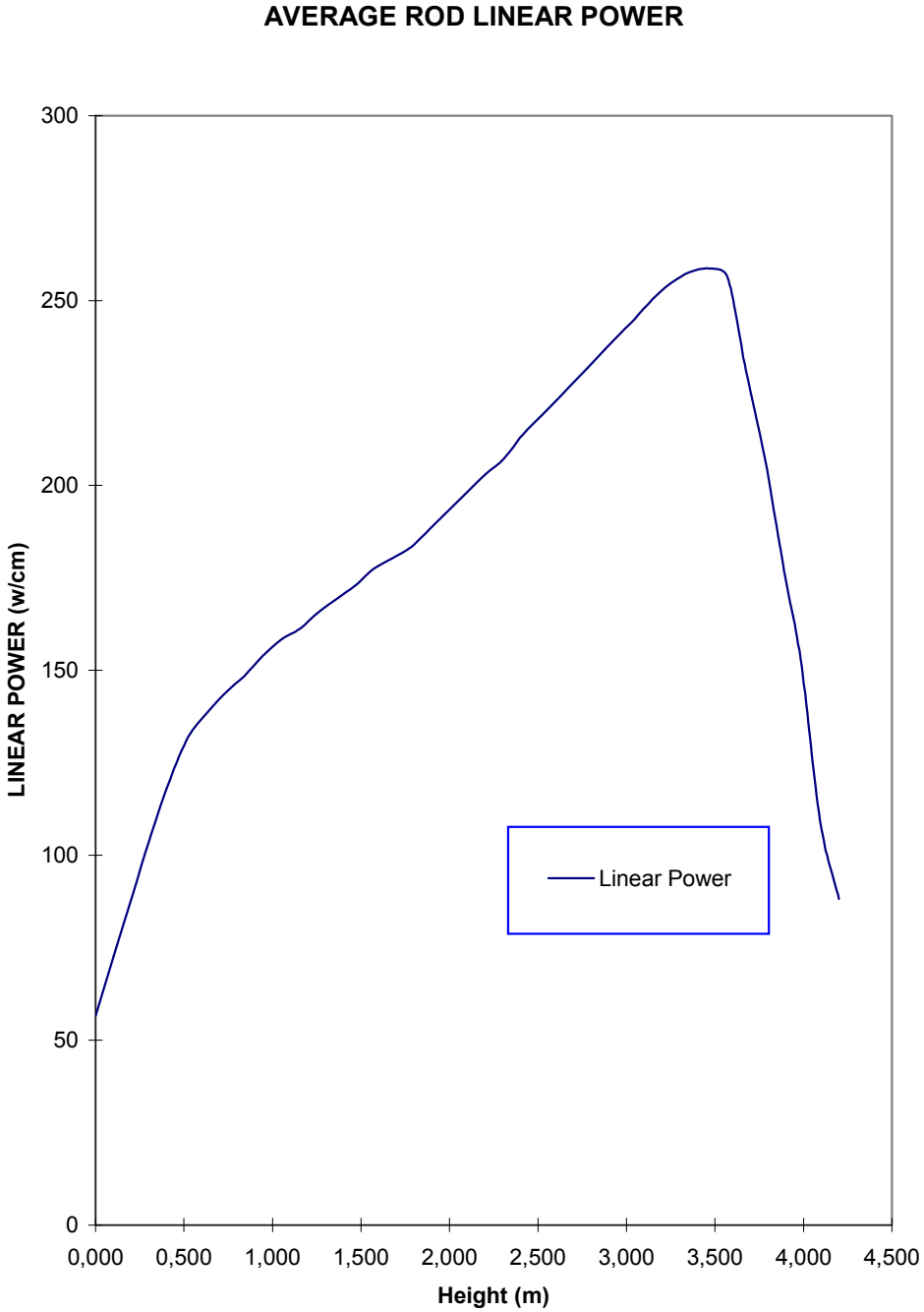
(*) note: after core emptying stopped by accumulator injection, heat transfer between cladding and reactor coolant occurs under film boiling conditions until the temperature decrease allows the rewetting of the hot rod cladding.

APPENDIX 14B.2.3 – TABLE 8

Summary of results - IB and LB(LOCA) - (State A, PCC-4)

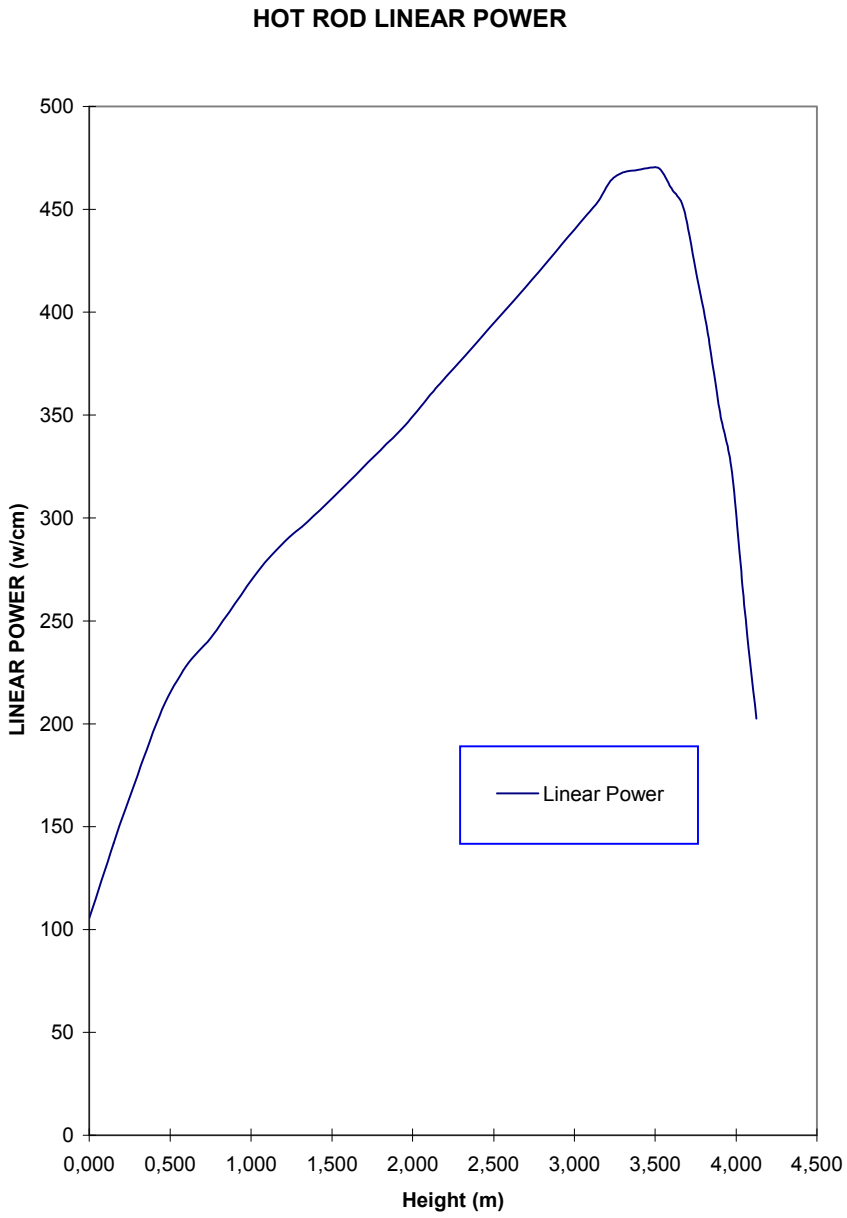
BREAK SIZE & LOCATION	MAXIMUM HOT SPOT CLADDING TEMPERATURE (°C)	HOT SPOT CLAD OXIDATION PERCENTAGE (%)	PERCENTAGE OF CLAD RUPTURE (%)
45 cm^2 (∅ 75 mm) cold leg break	590	Less than 1%	0
80 cm^2 (∅ 100 mm) cold leg break	900	1.5%	0
125 cm^2 (∅ 125 mm) cold leg break	800	Less than 1.5%	0
180 cm^2 (∅ 150 mm) cold leg break	585	Less than 1%	0
RIS [SIS] LINE BREAK (390 cm^2 - ∅ 225 mm)	830	Less than 1.5%	0
SURGE LINE BREAK (2 × 830 cm^2 - ∅ 2 x 325 mm)	810	Less than 1.5%	0

APPENDIX 14B.2.3 – FIGURE 1



PCC LOCA analyses : axial power shape
Average rod in average assembly

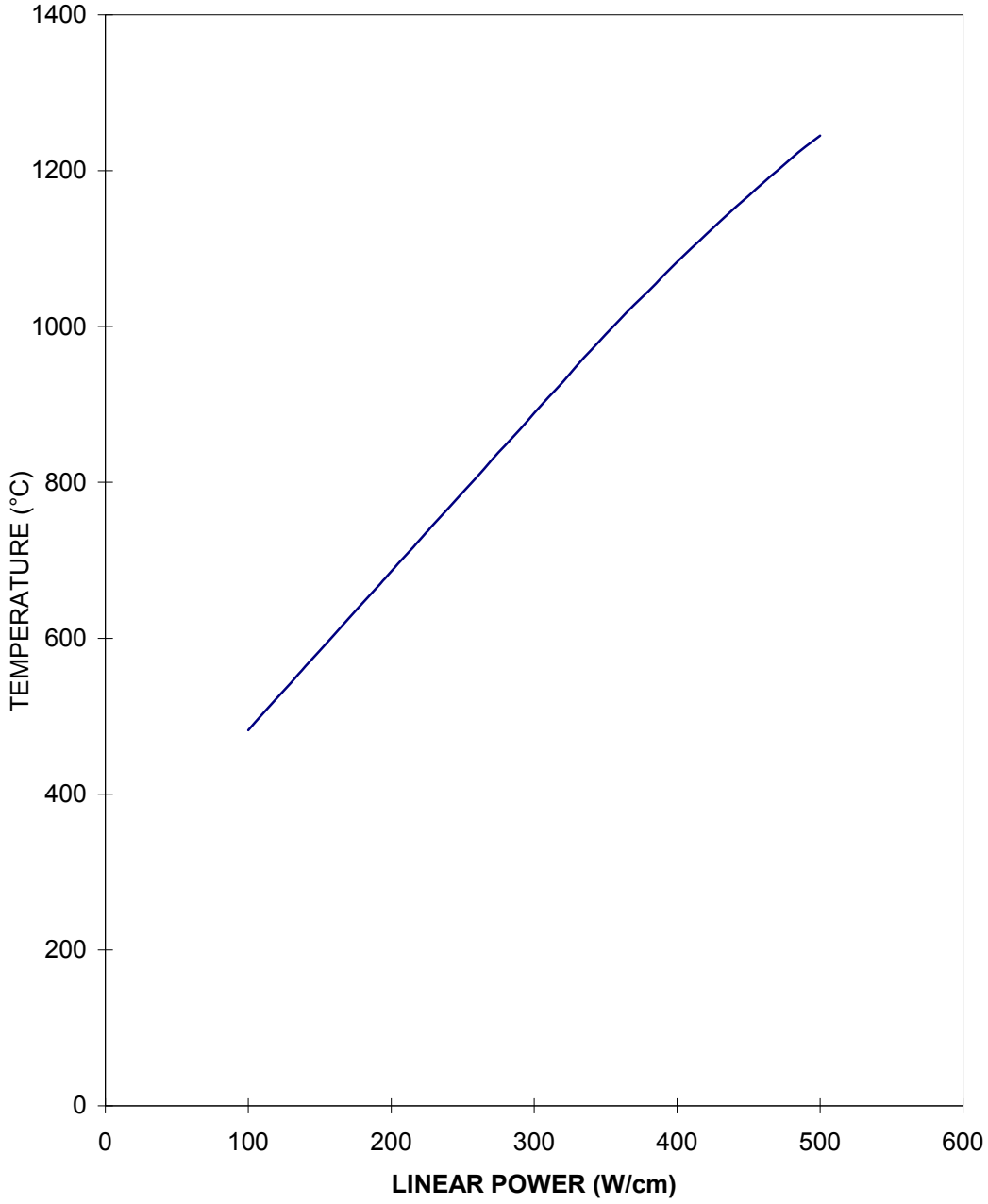
APPENDIX 14B.2.3 – FIGURE 2



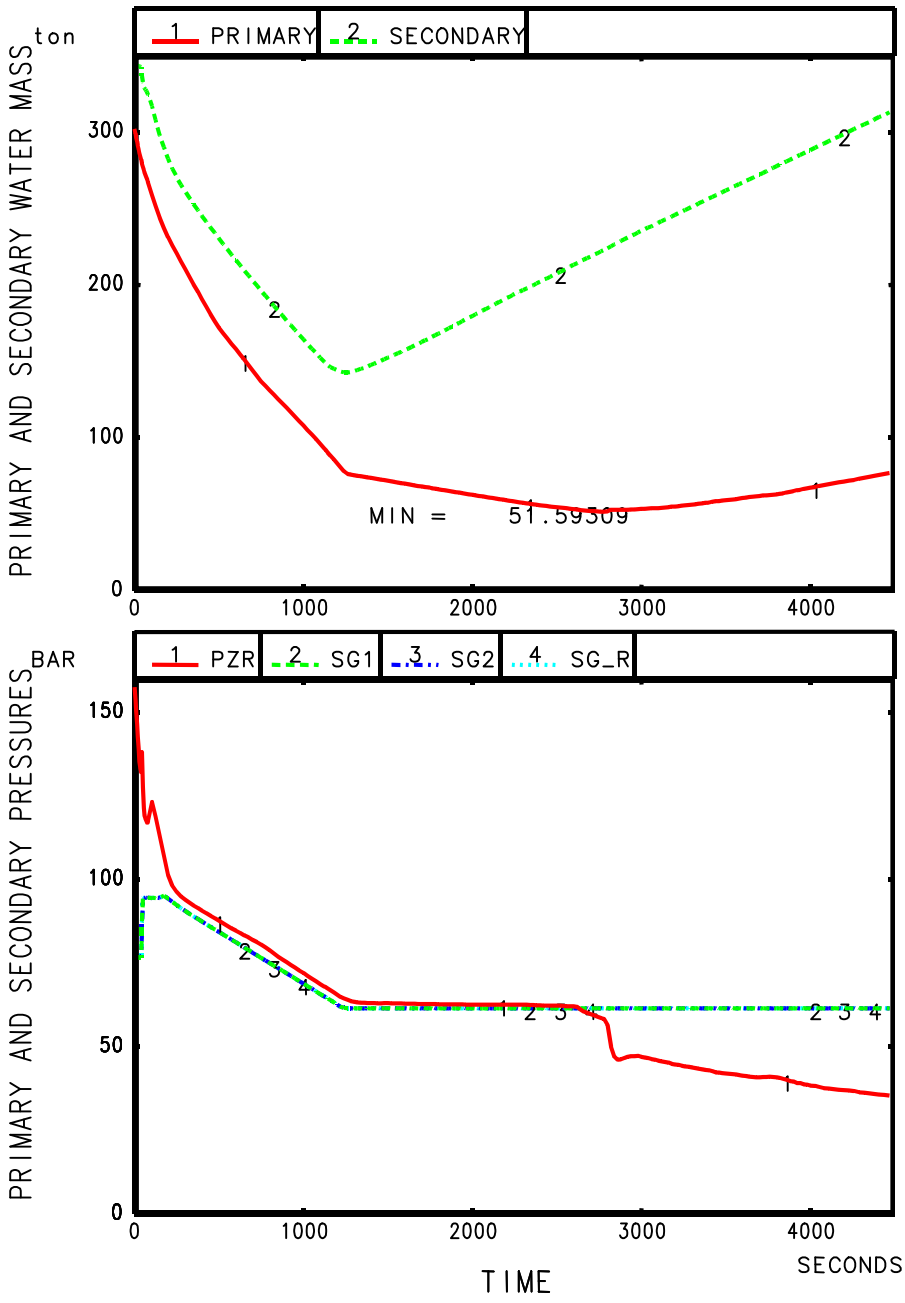
PCC LOCA analyses : axial power shape
Hot rod in hot assembly

APPENDIX 14B.2.3 – FIGURE 3

PCC LOCA analyses: Initial Pellet Temperature

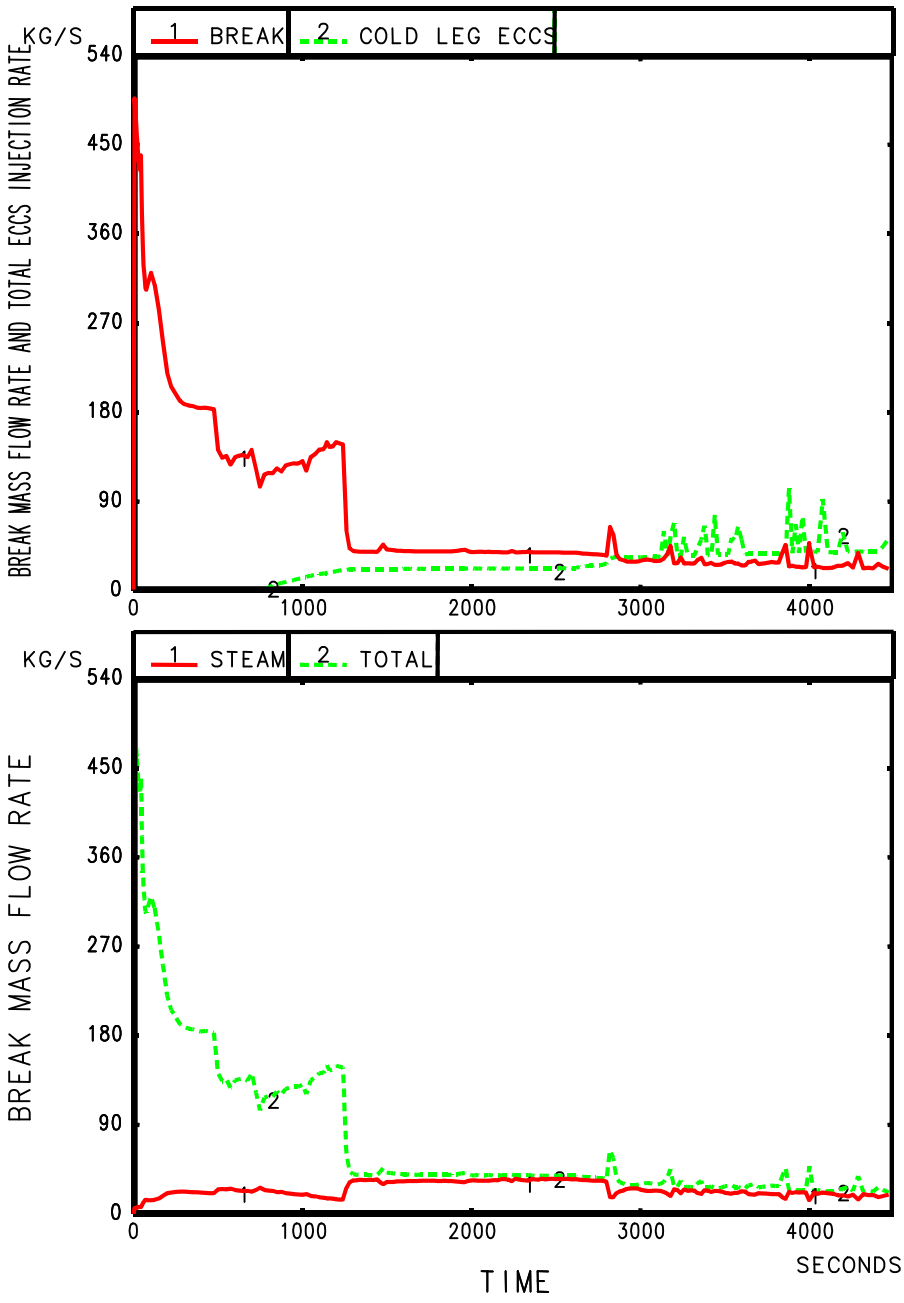


APPENDIX 14B.2.3 – FIGURE 4



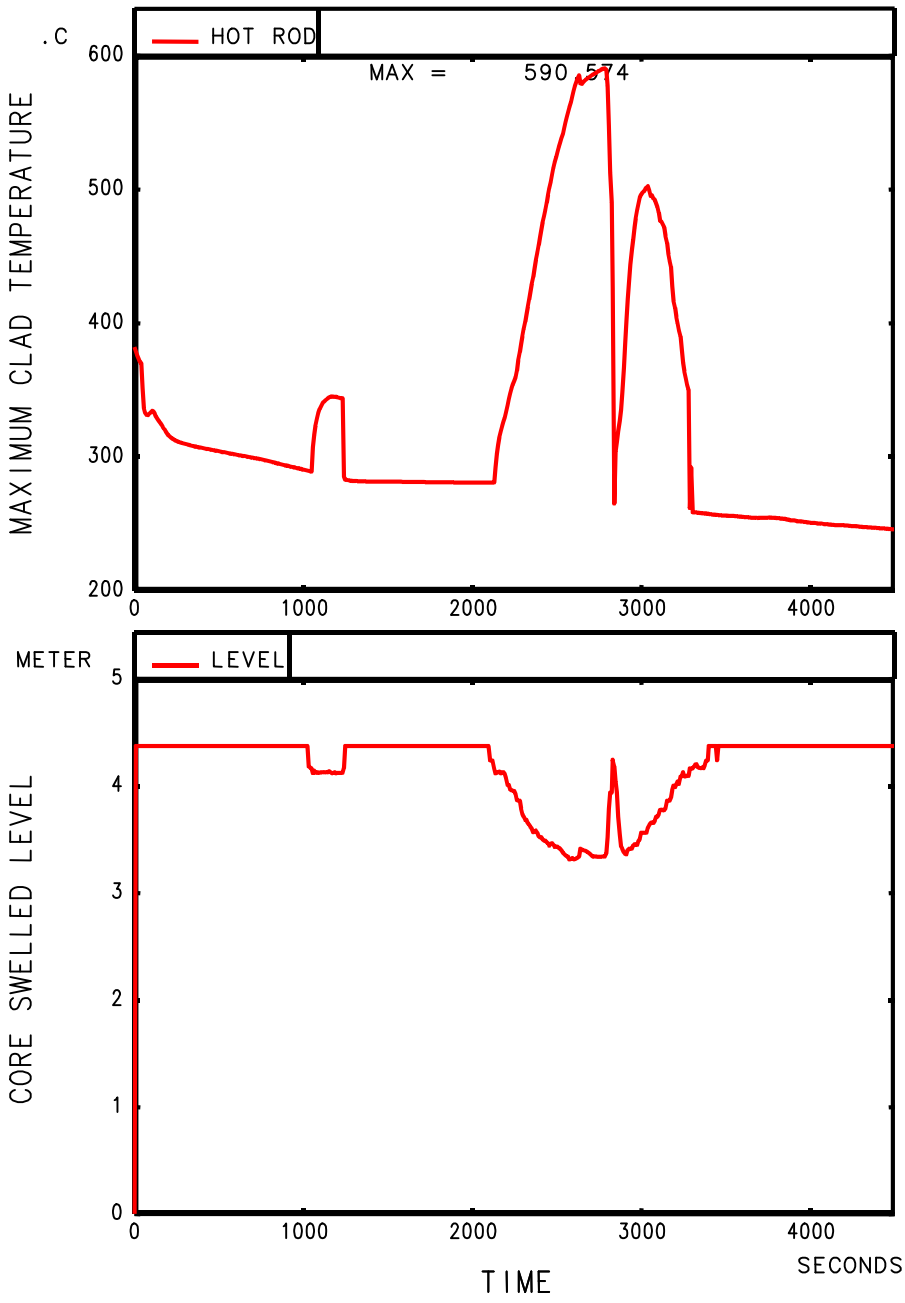
RCP [RCS] and secondary side water inventories
RCP [RCS] and secondary side pressures
45 cm^2 (ϕ 75 mm) cold leg break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 5



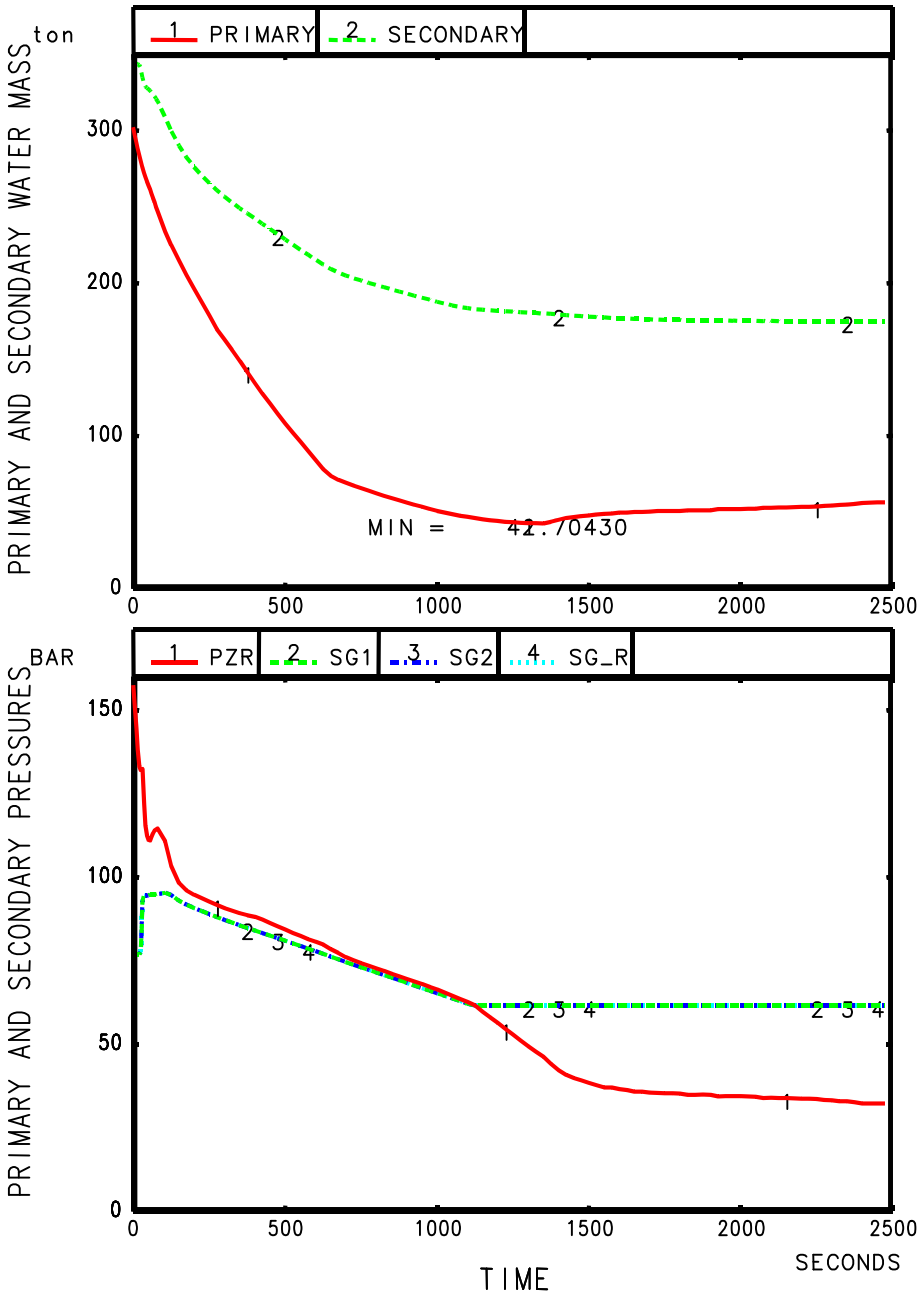
Total break and RIS [SIS] flow rates
Total break and steam flow rates
45 cm^2 (ϕ 75 mm) cold leg break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 6



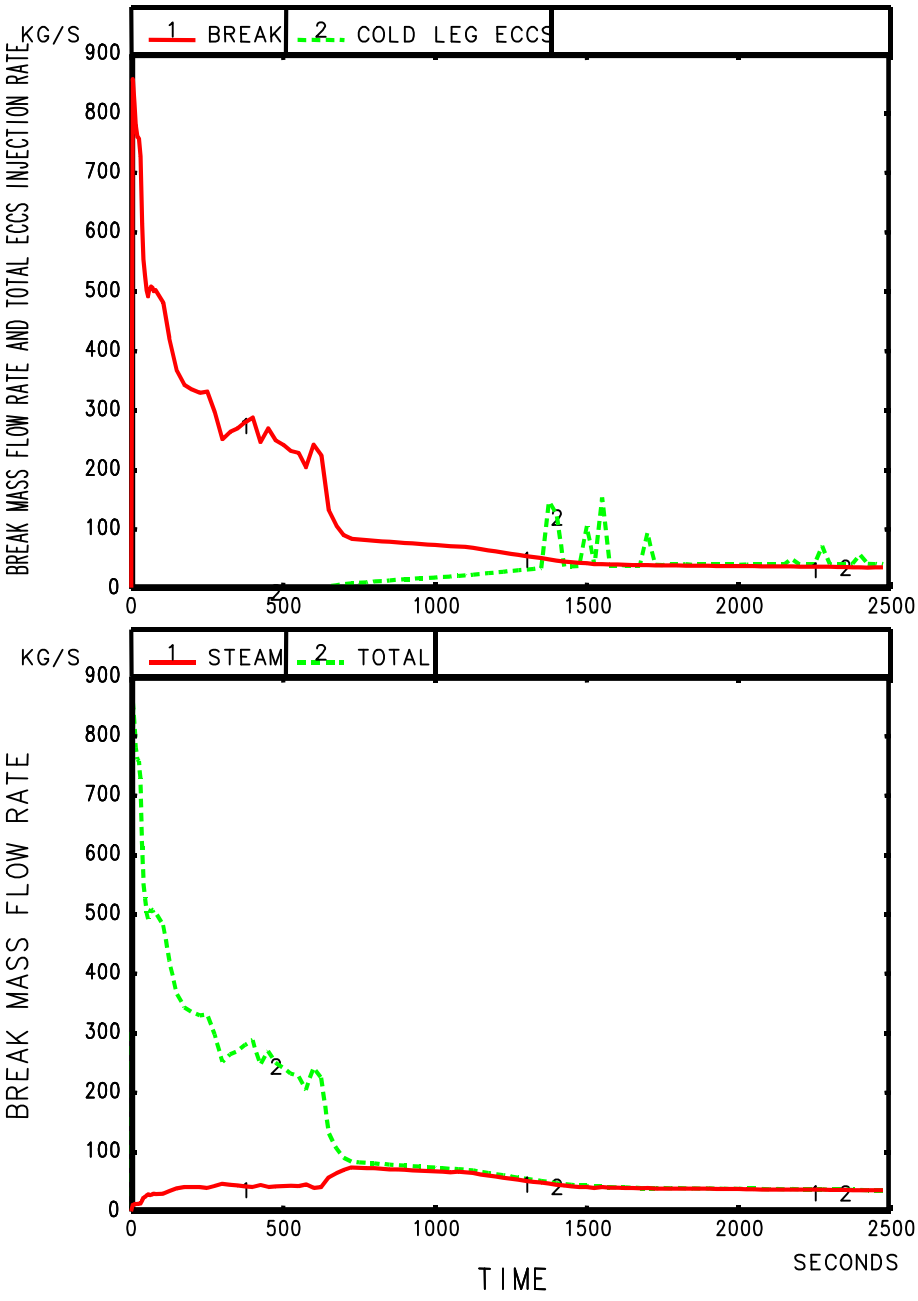
Maximum cladding temperature
Core swell level
45 cm2 (ϕ 75 mm) cold leg break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 7



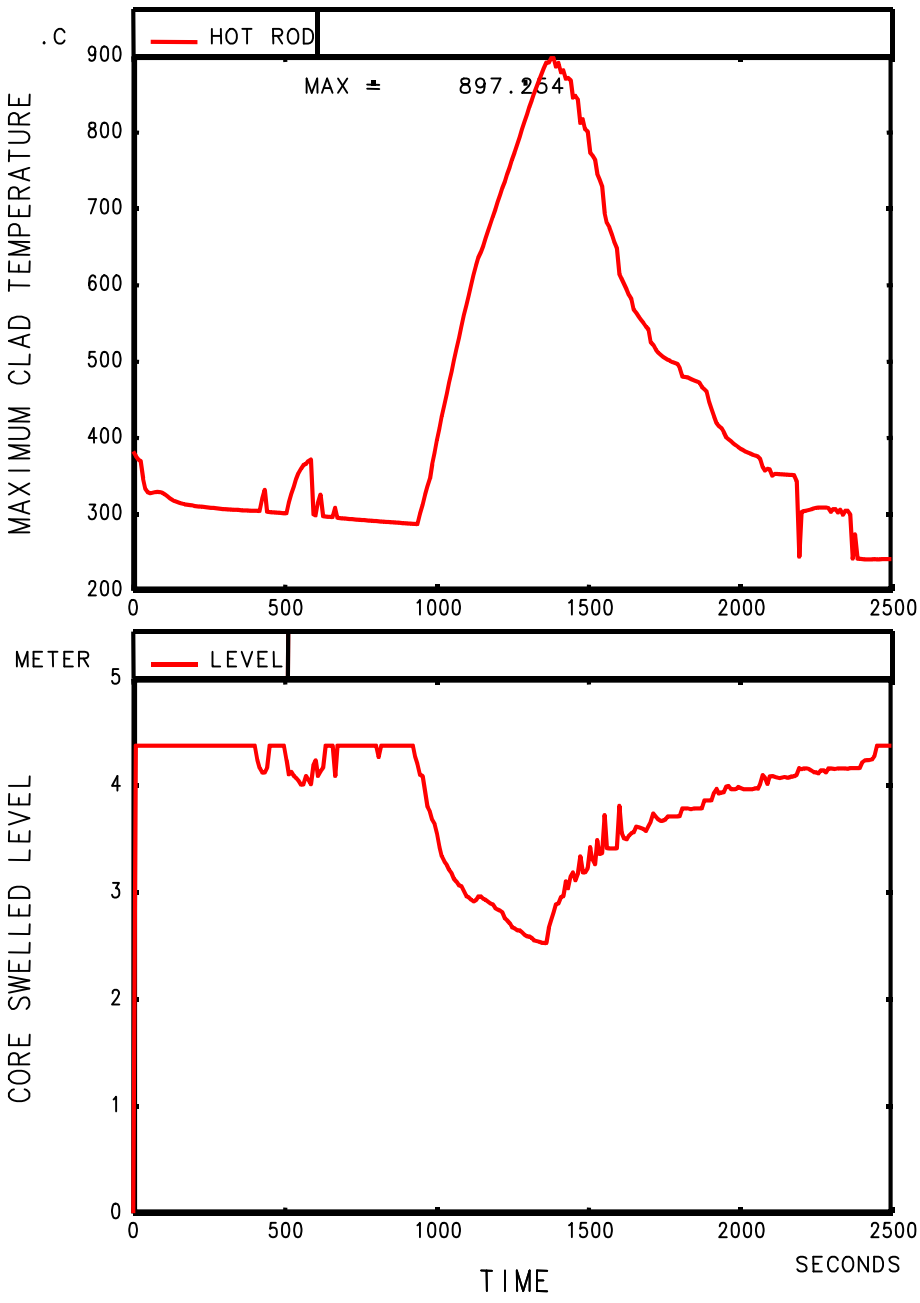
RCP [RCS] and secondary side water inventories
RCP [RCS] and secondary side pressures
80 cm^2 (ϕ 100 mm) cold leg break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 8



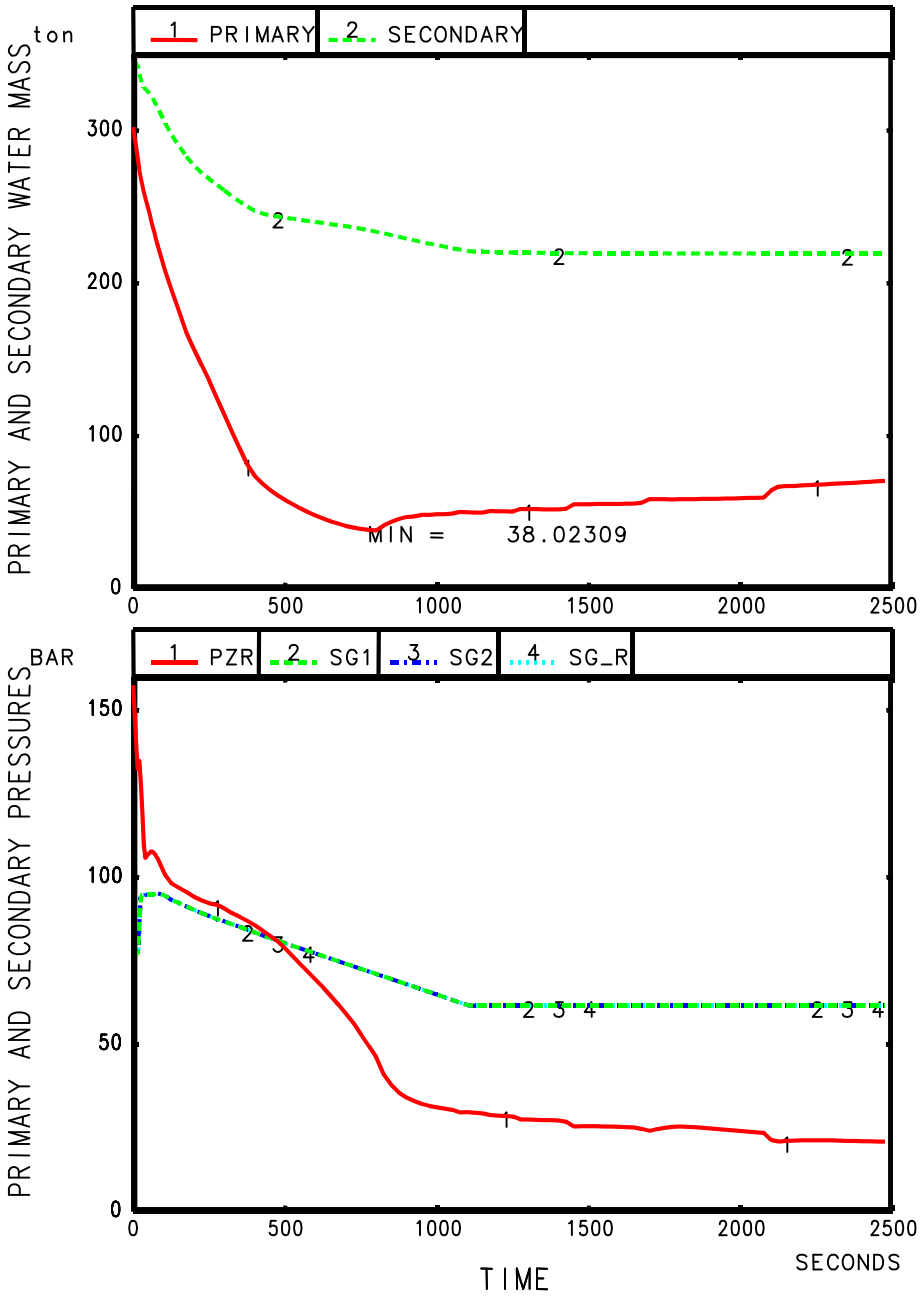
Total break and RIS [SIS] flow rates
Total break and steam flow rates
80 cm2 (ϕ 100 mm) cold leg break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 9



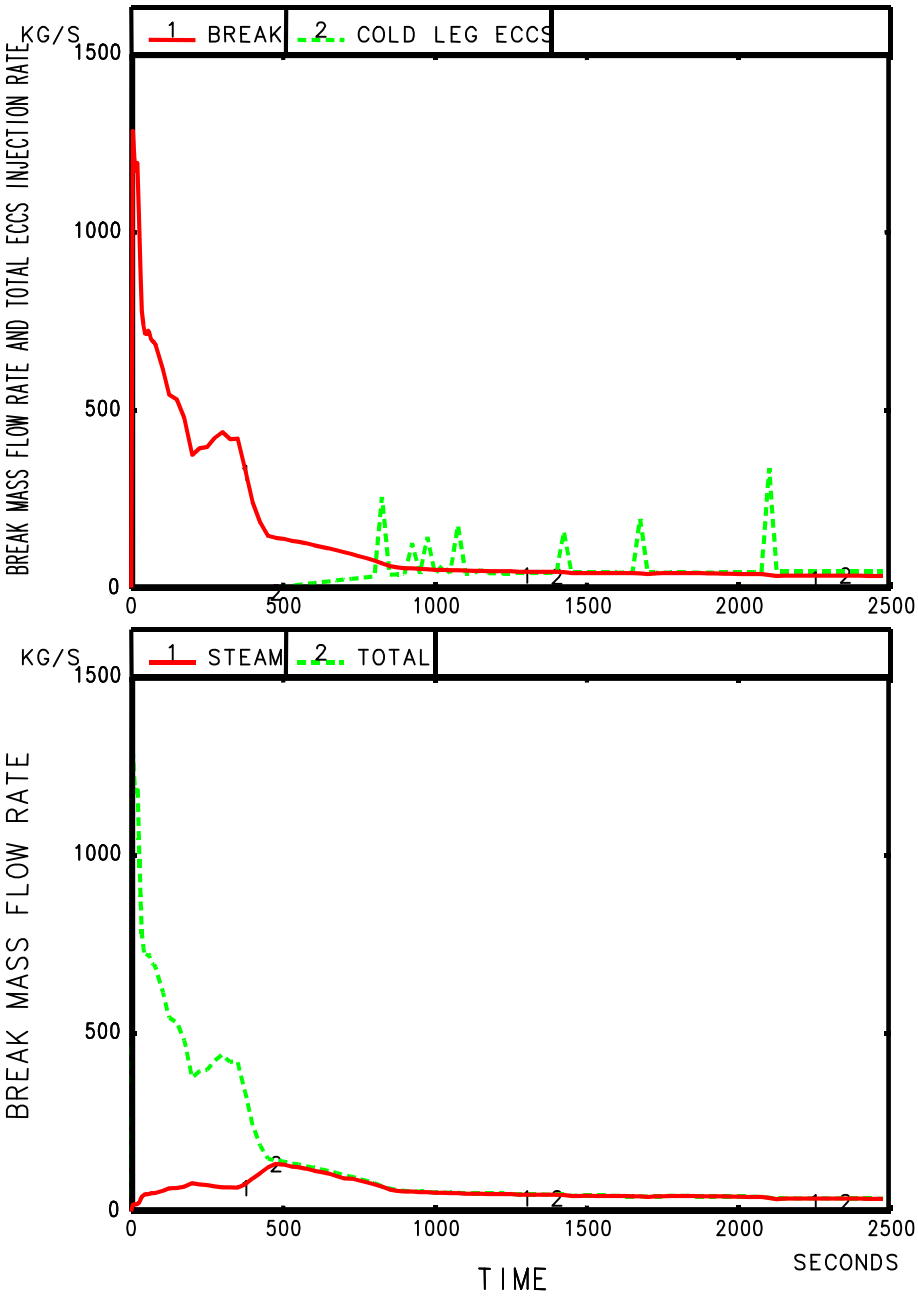
Maximum cladding temperature
Core swell level
80 cm^2 (ϕ 100 mm) cold leg break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 10



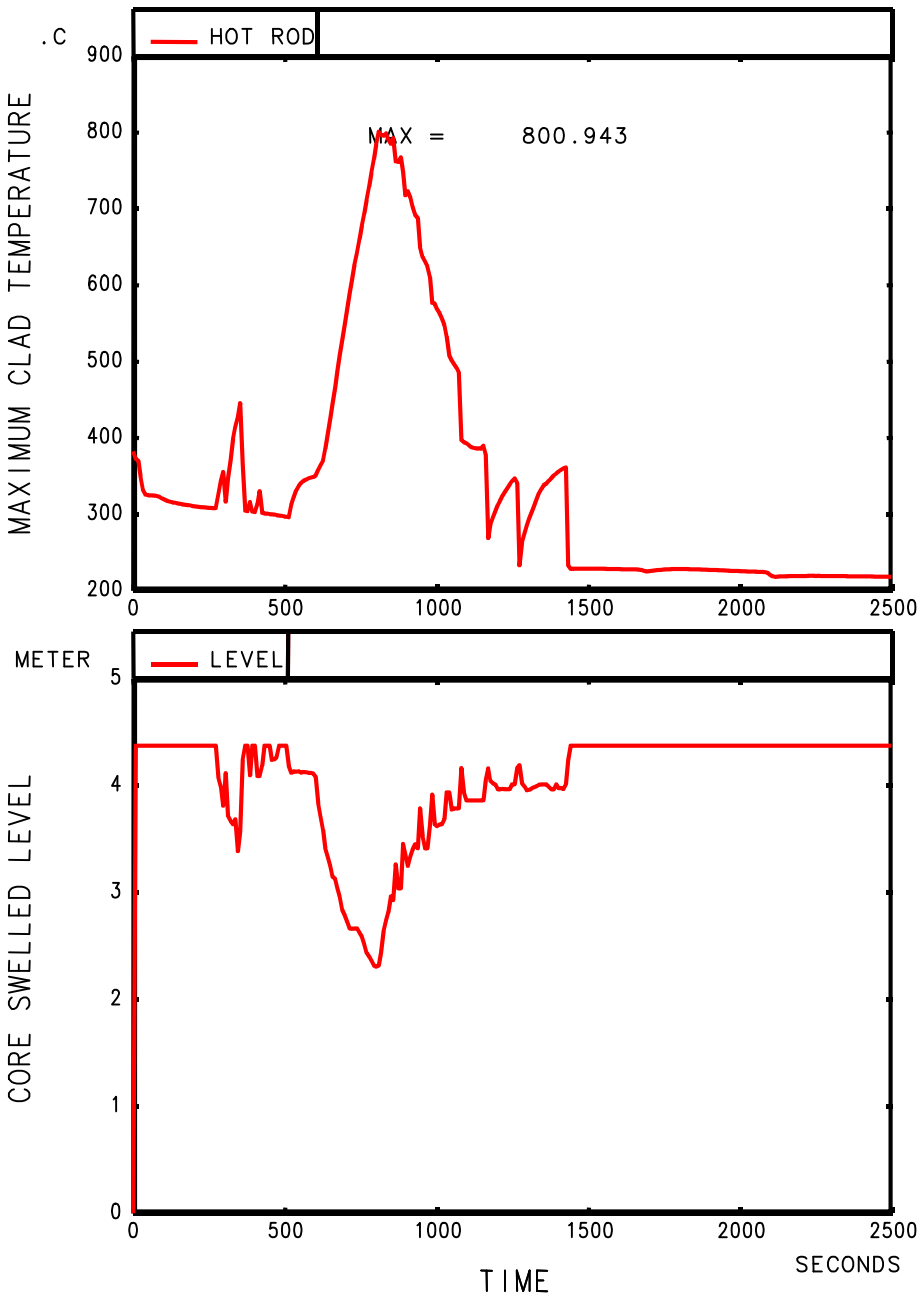
RCP [RCS] and secondary side water inventories
RCP [RCS] and secondary side pressures
125 cm^2 (ϕ 125 mm) cold leg break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 11



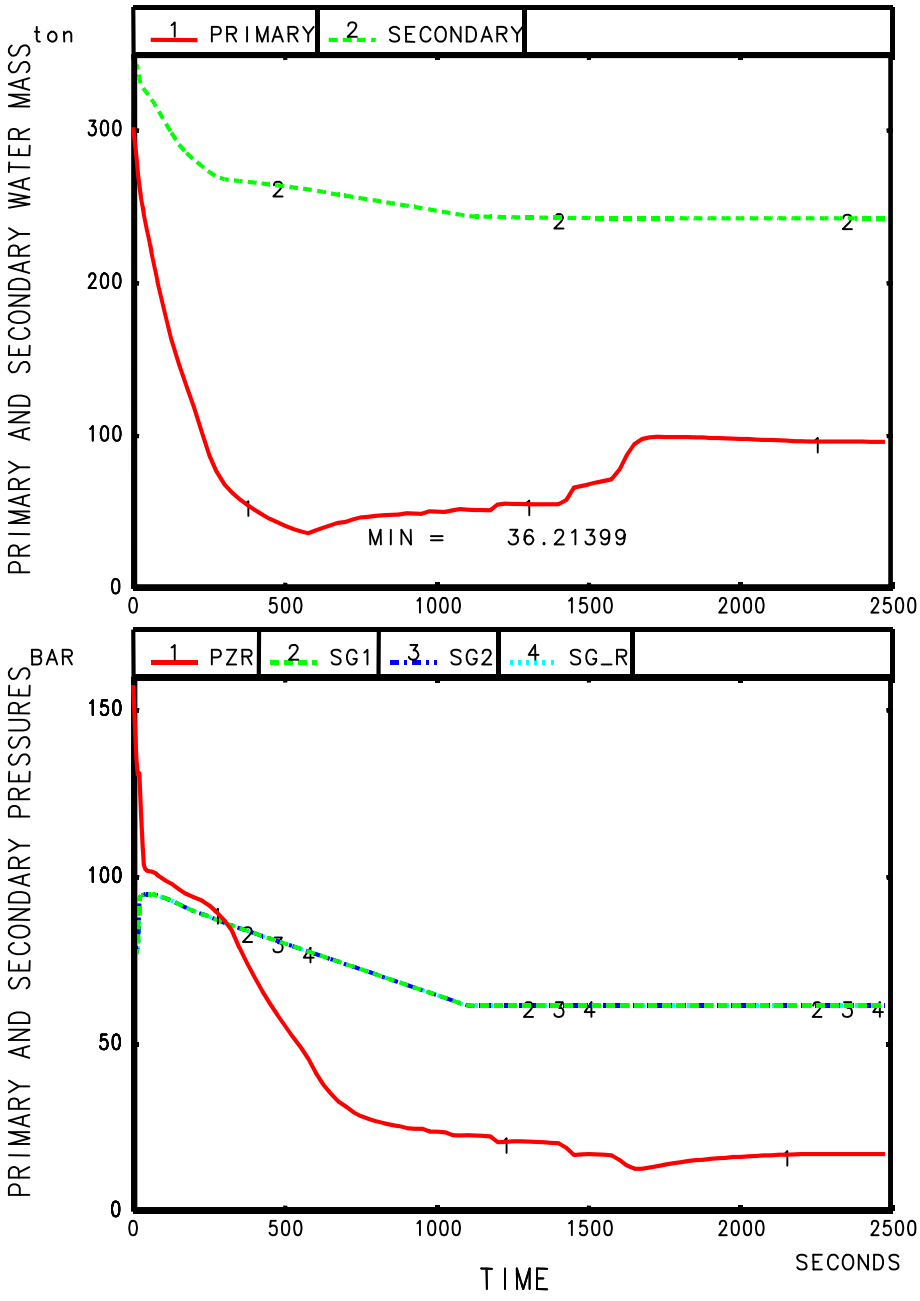
Total break and RIS [SIS] flow rates
Total break and steam flow rates
125 cm^2 (ϕ 125 mm) cold leg break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 12



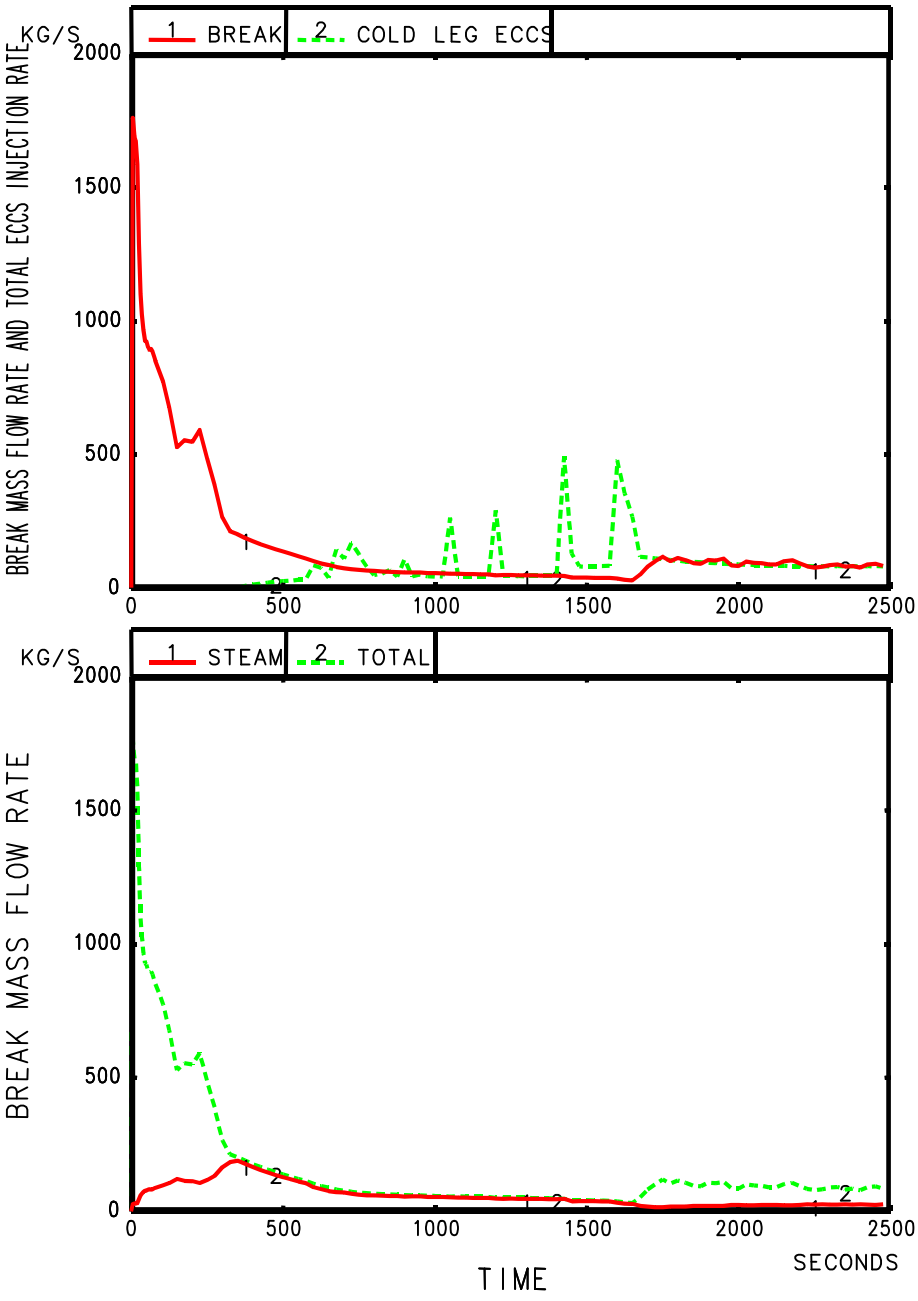
Maximum cladding temperature
Core swell level
125 cm^2 (ϕ 125 mm) cold leg break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 13



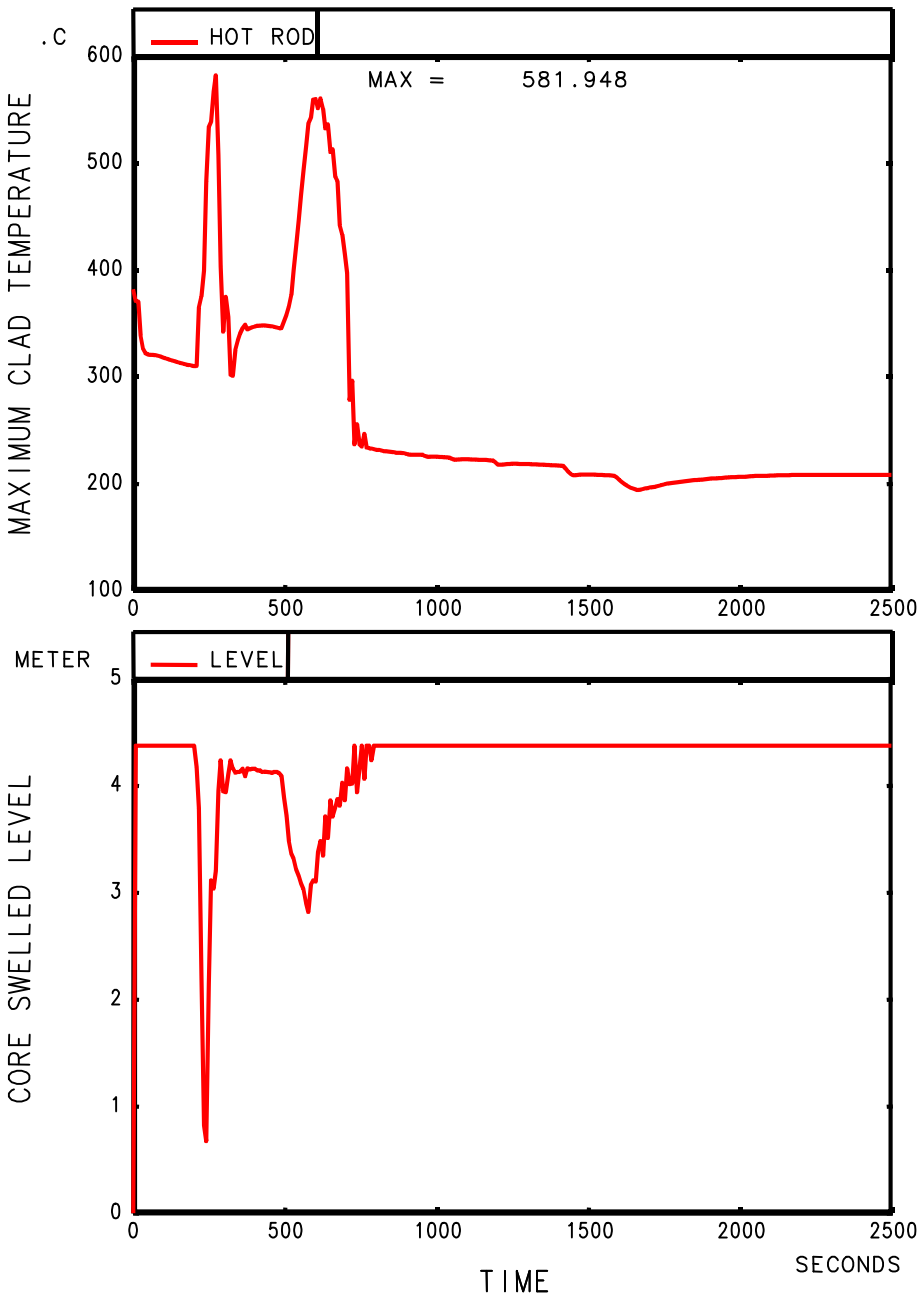
RCP [RCS] and secondary side water inventories
RCP [RCS] and secondary side pressures
180 cm^2 (ϕ 150 mm) cold leg break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 14



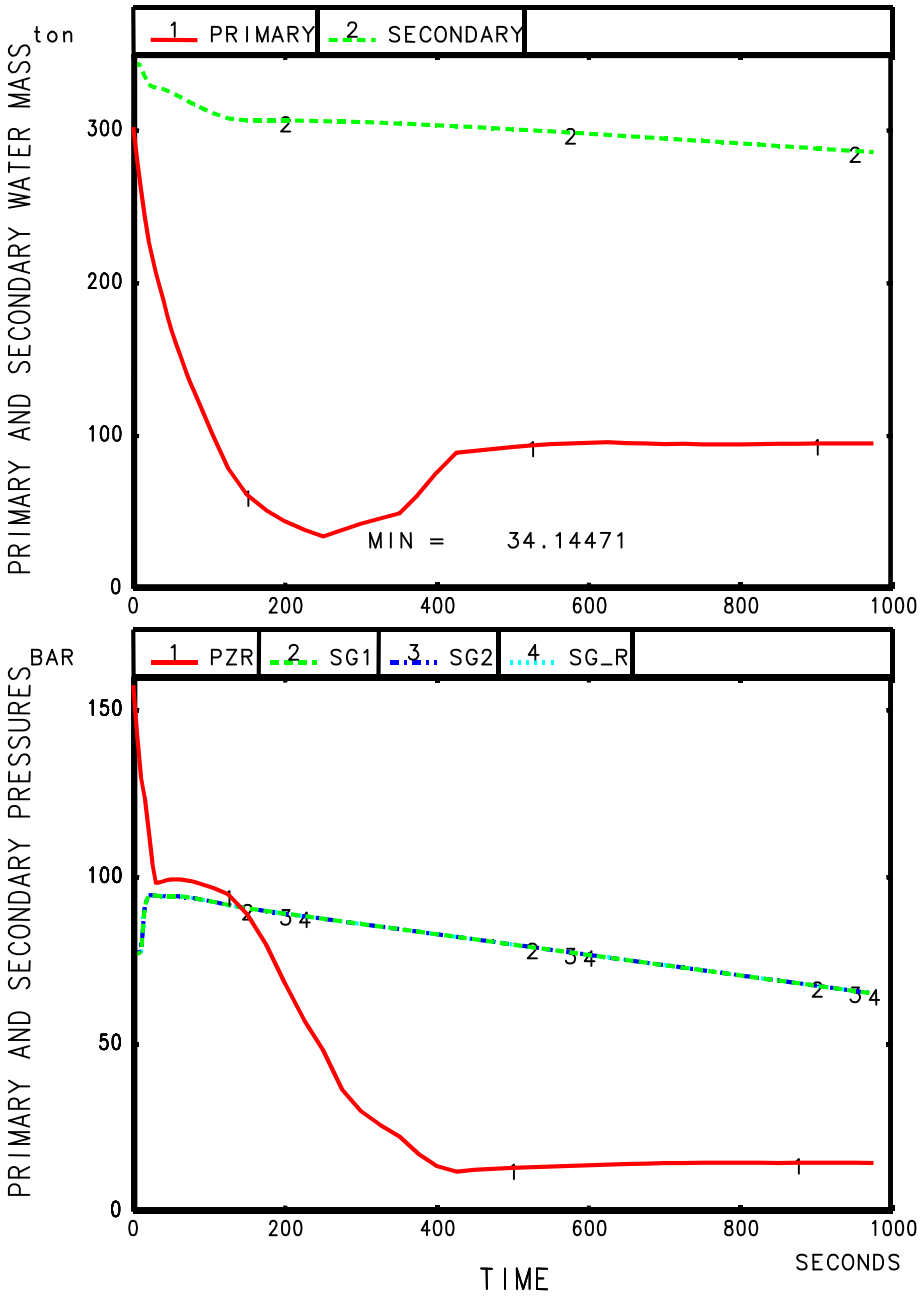
Total break and RIS [SIS] flow rates
Total break and steam flow rates
180 cm^2 (ϕ 150 mm) cold leg break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 15



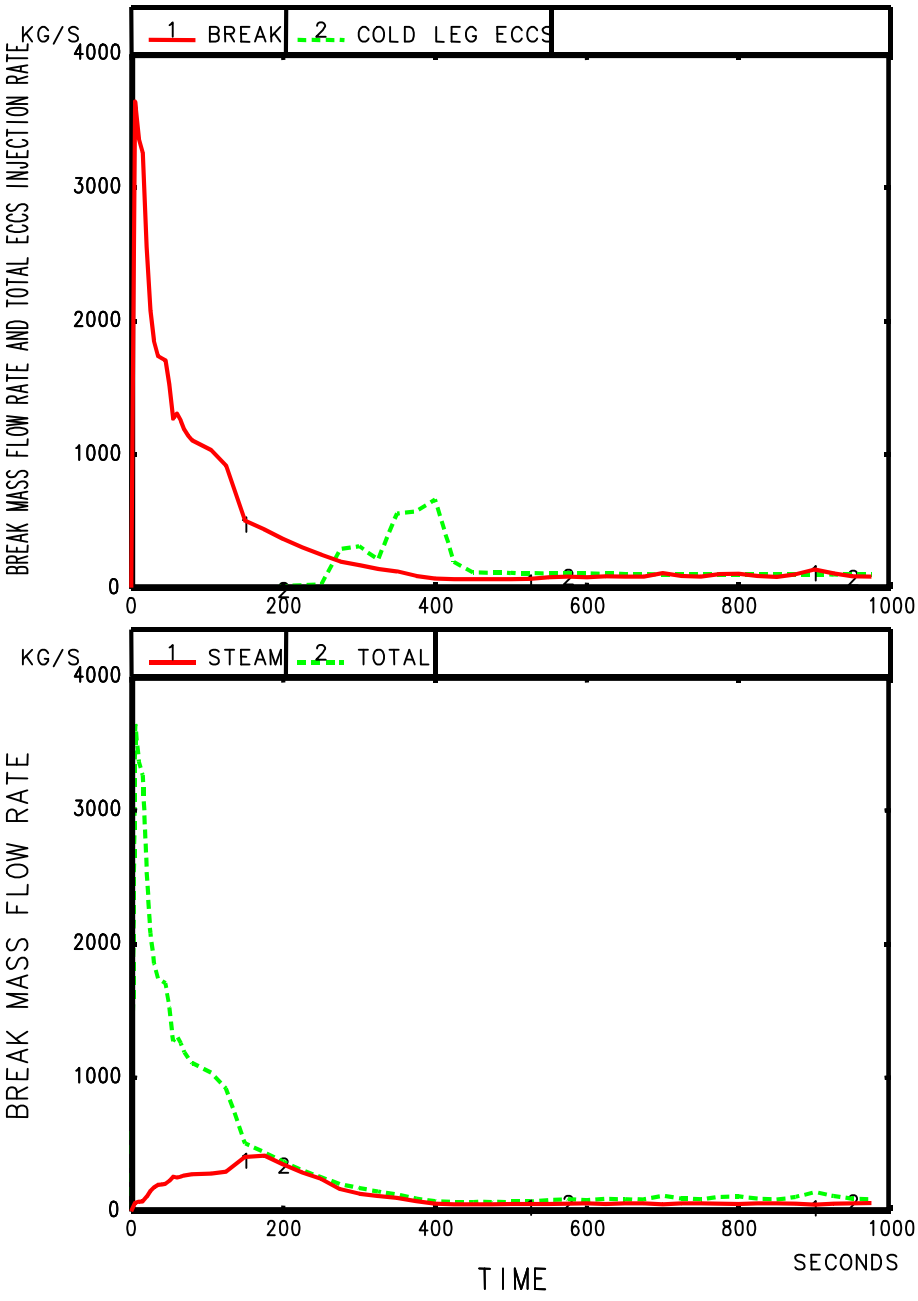
Maximum cladding temperature
Core swell level
180 cm^2 (ϕ 150 mm) cold leg break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 16



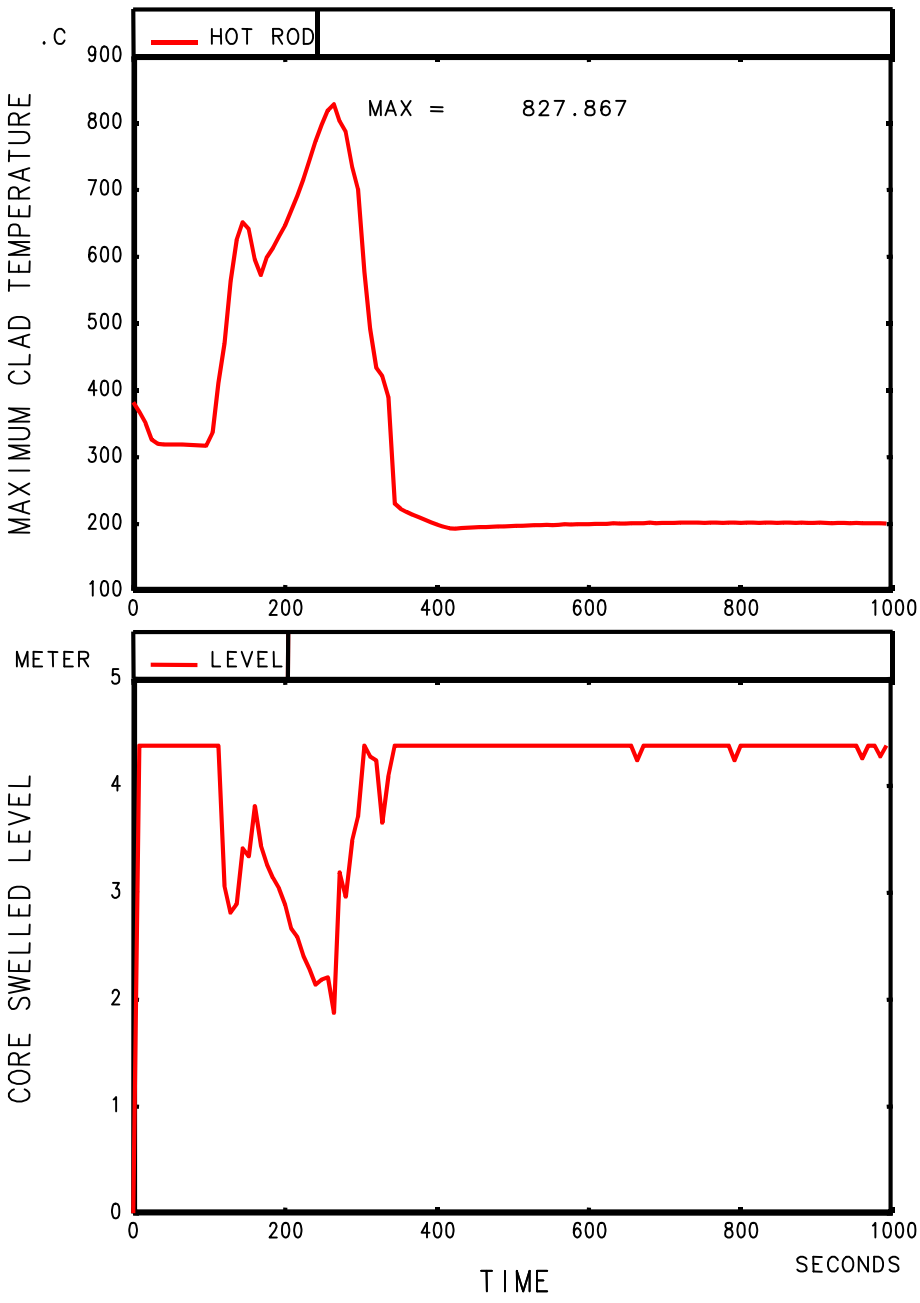
RCP [RCS] and secondary side water inventories
RCP [RCS] and secondary side pressures
390 cm^2 (ϕ 225 mm) RIS [SIS] line break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 17



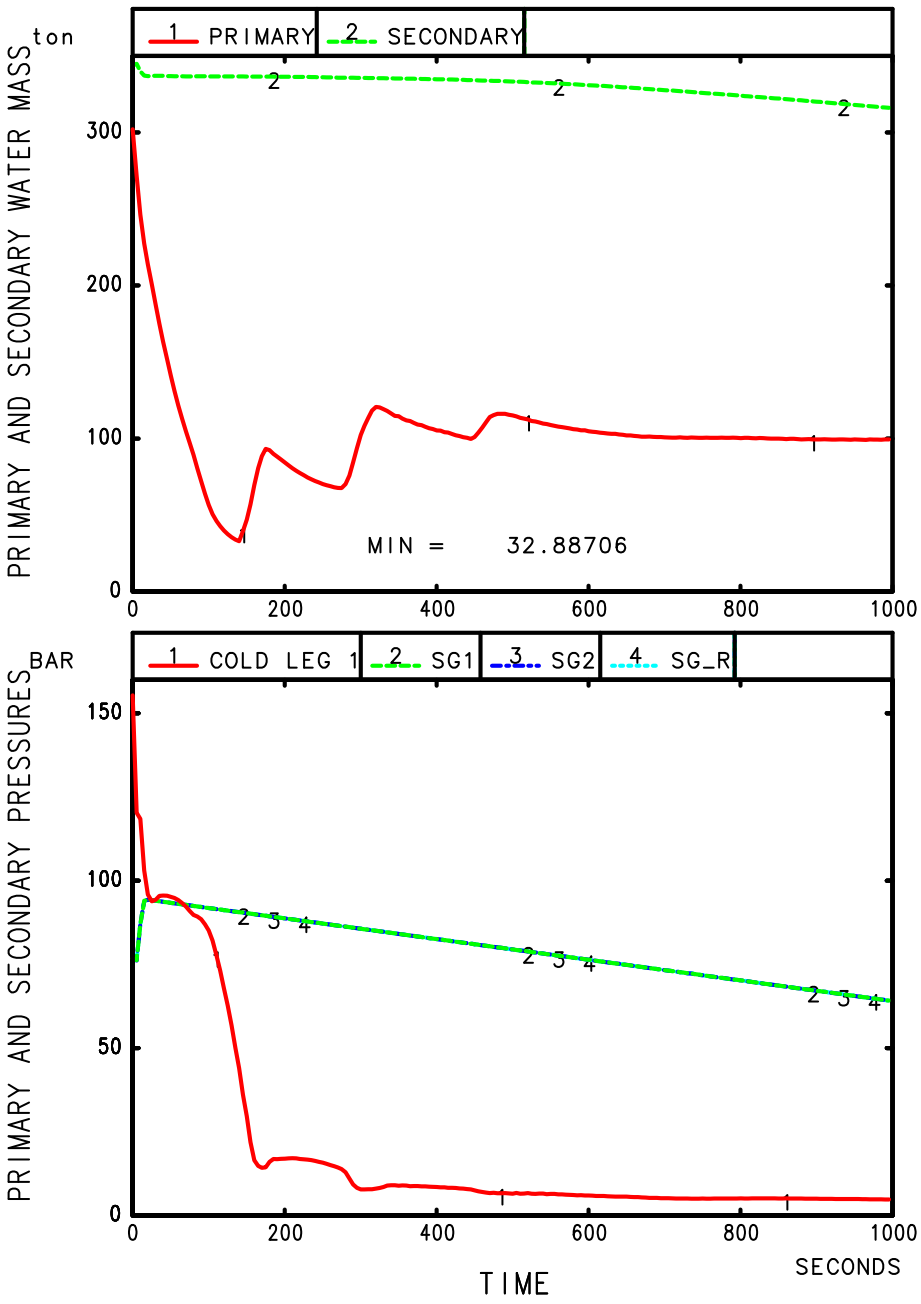
Total break and RIS [SIS] flow rates
Total break and steam flow rates
390 cm^2 (ϕ 225 mm) RIS [SIS] line break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 18



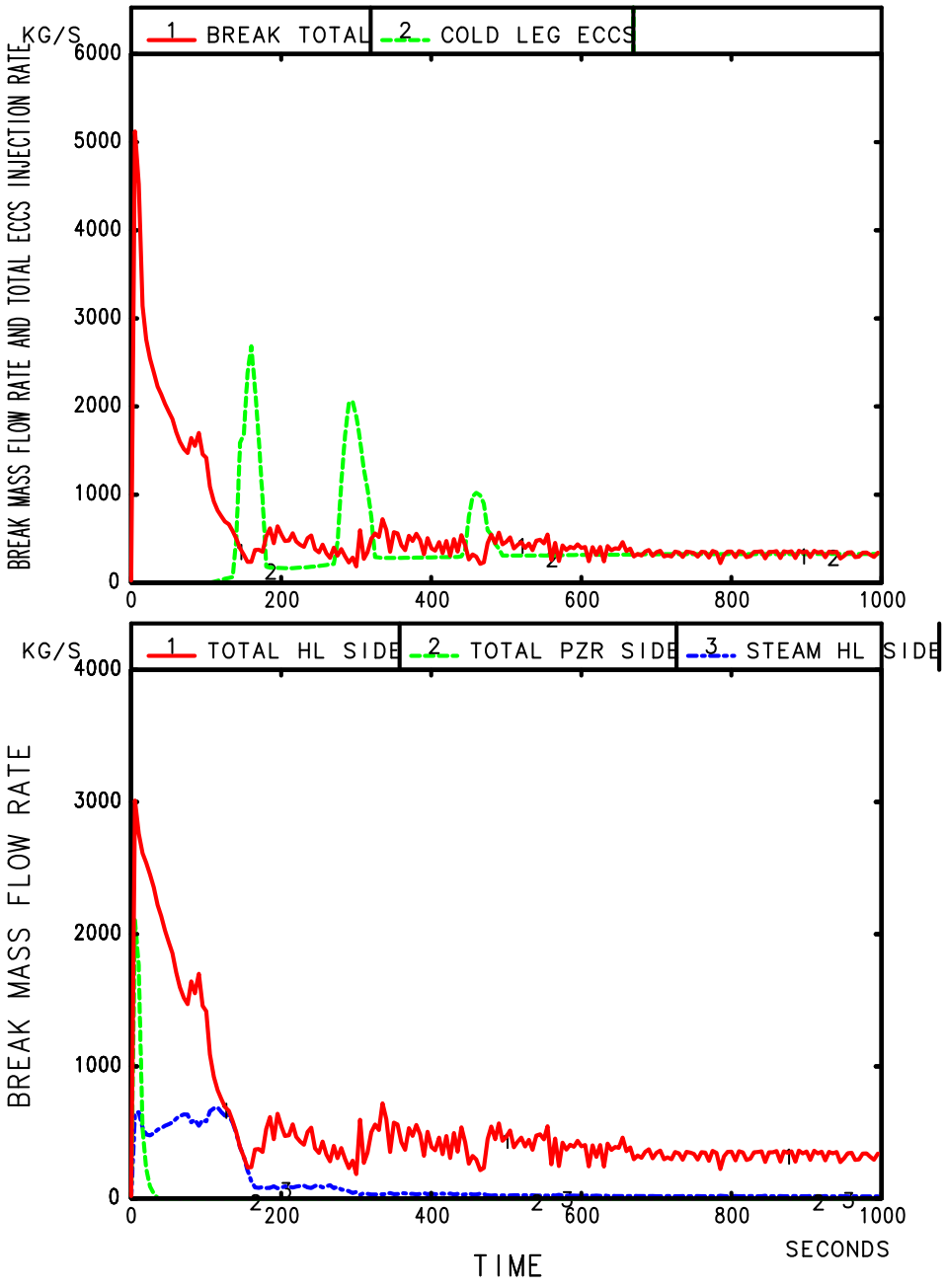
Maximum cladding temperature
Core swell level
390 cm^2 (ϕ 225 mm) RIS [SIS] line break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 19



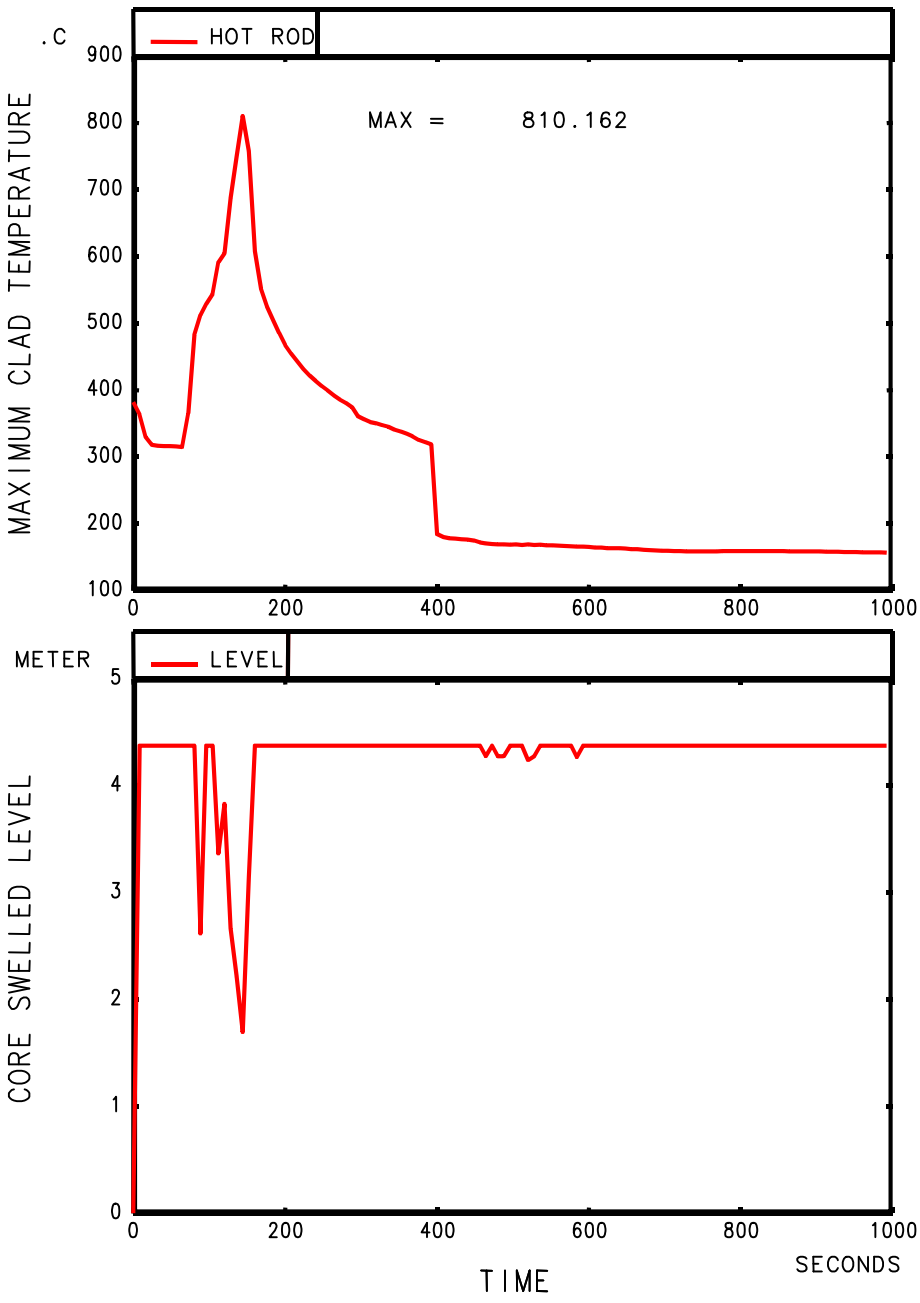
RCP [RCS] and secondary side water inventories
RCP [RCS] and secondary side pressures
2×830 cm^2 ($2 \times \phi$ 325 mm) surge line break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 20



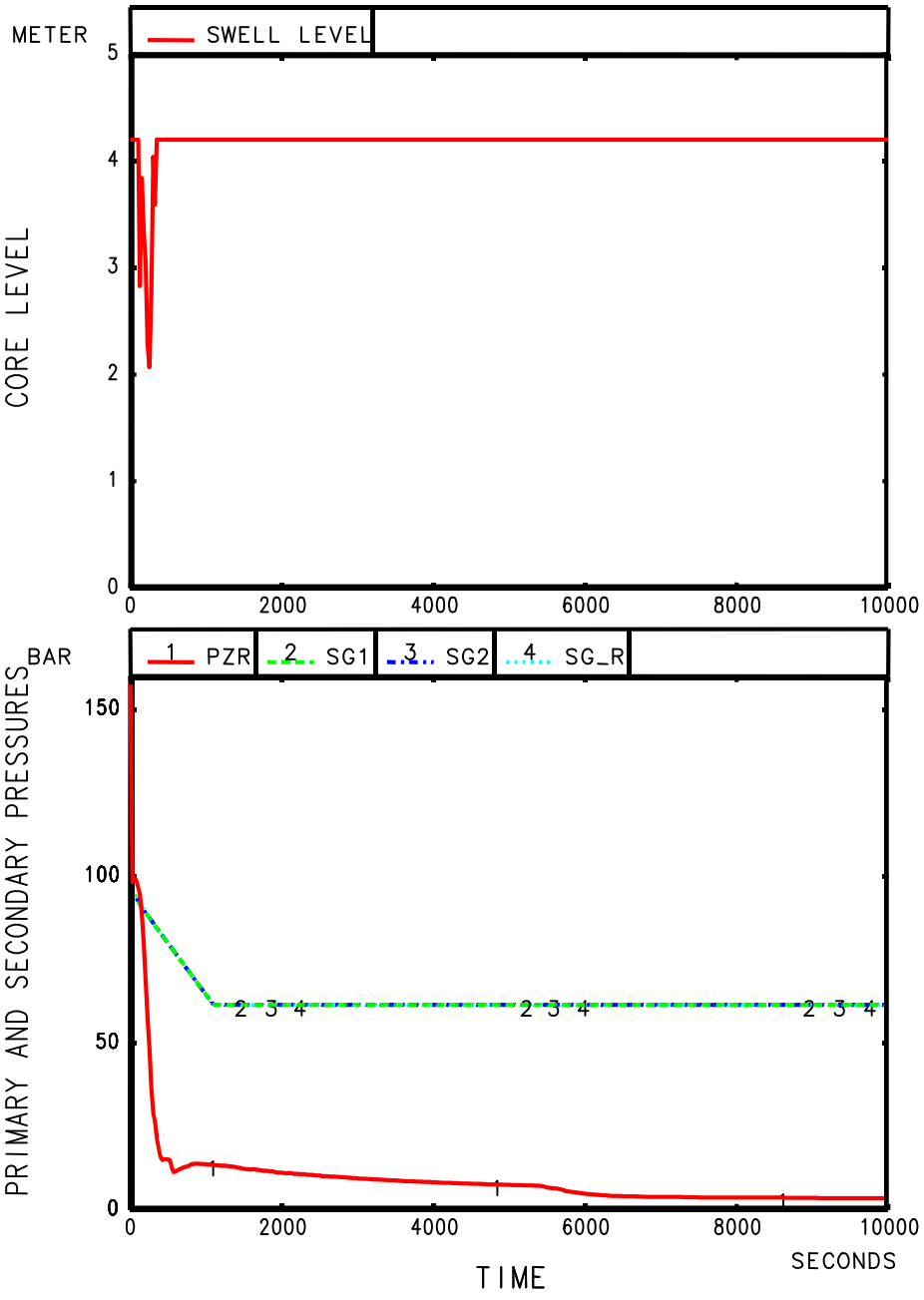
Total break and RIS [SIS] flow rates
Total break and steam flow rates
$2 \times 830\ cm^2$ ($2 \times \phi$ 325 mm) surge line break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 21



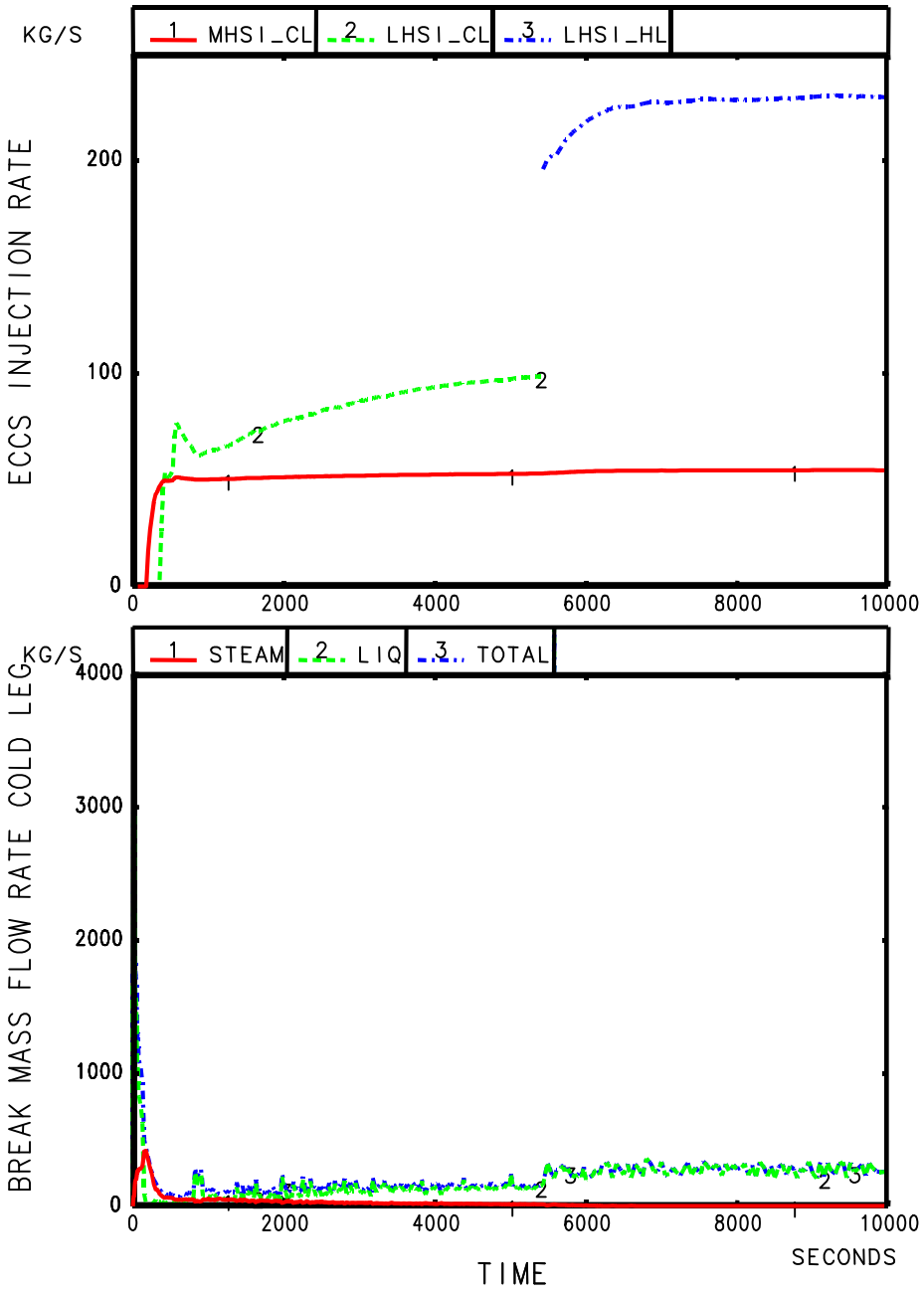
Maximum cladding temperature
Core swell level
$2 \times 830\ cm^2$ ($2 \times \phi$ 325 mm) surge line break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 22



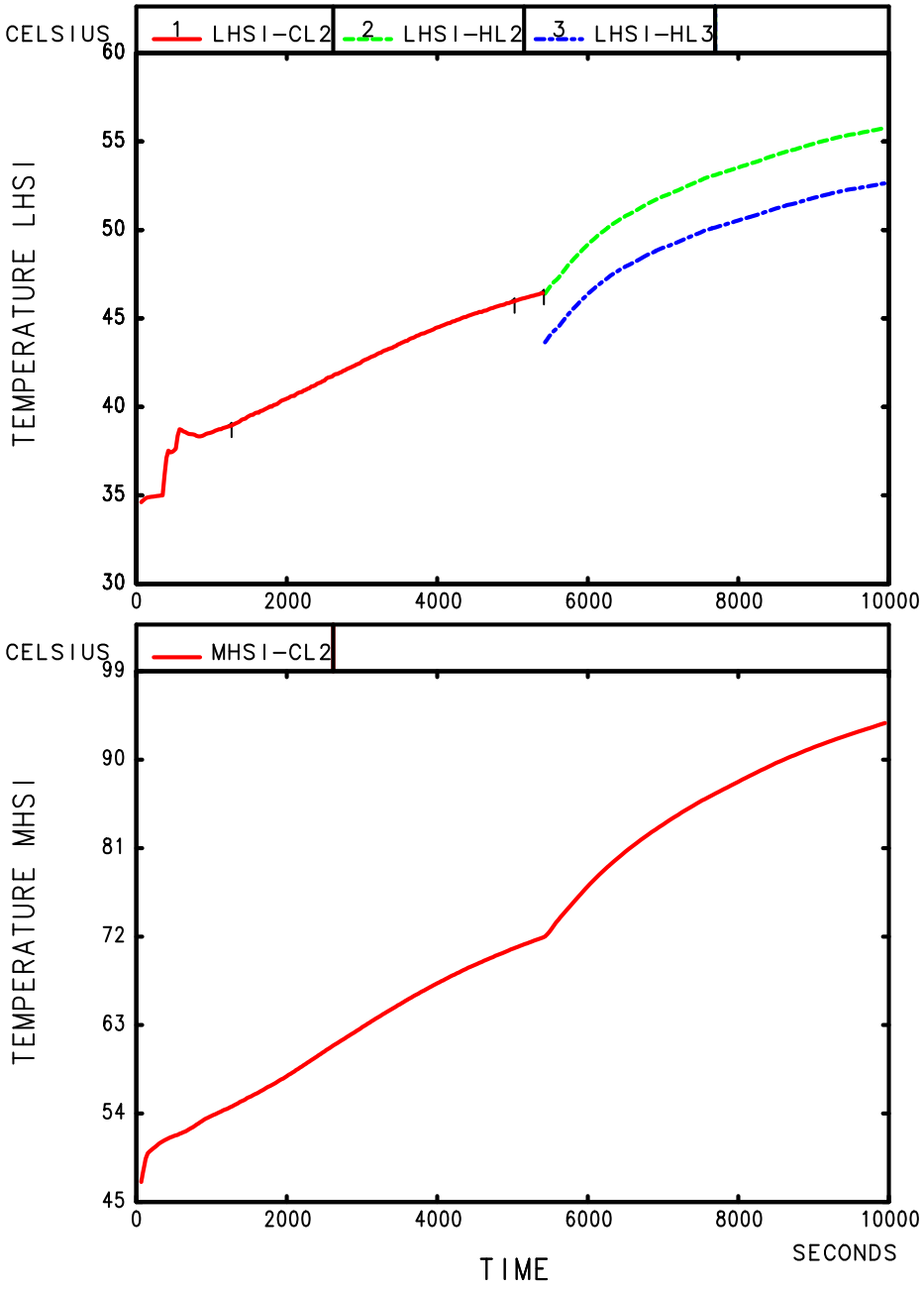
RCP [RCS] core water level
RCP [RCS] and secondary side pressures
390 cm^2 (ϕ 225 mm) RIS [SIS] line break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 23



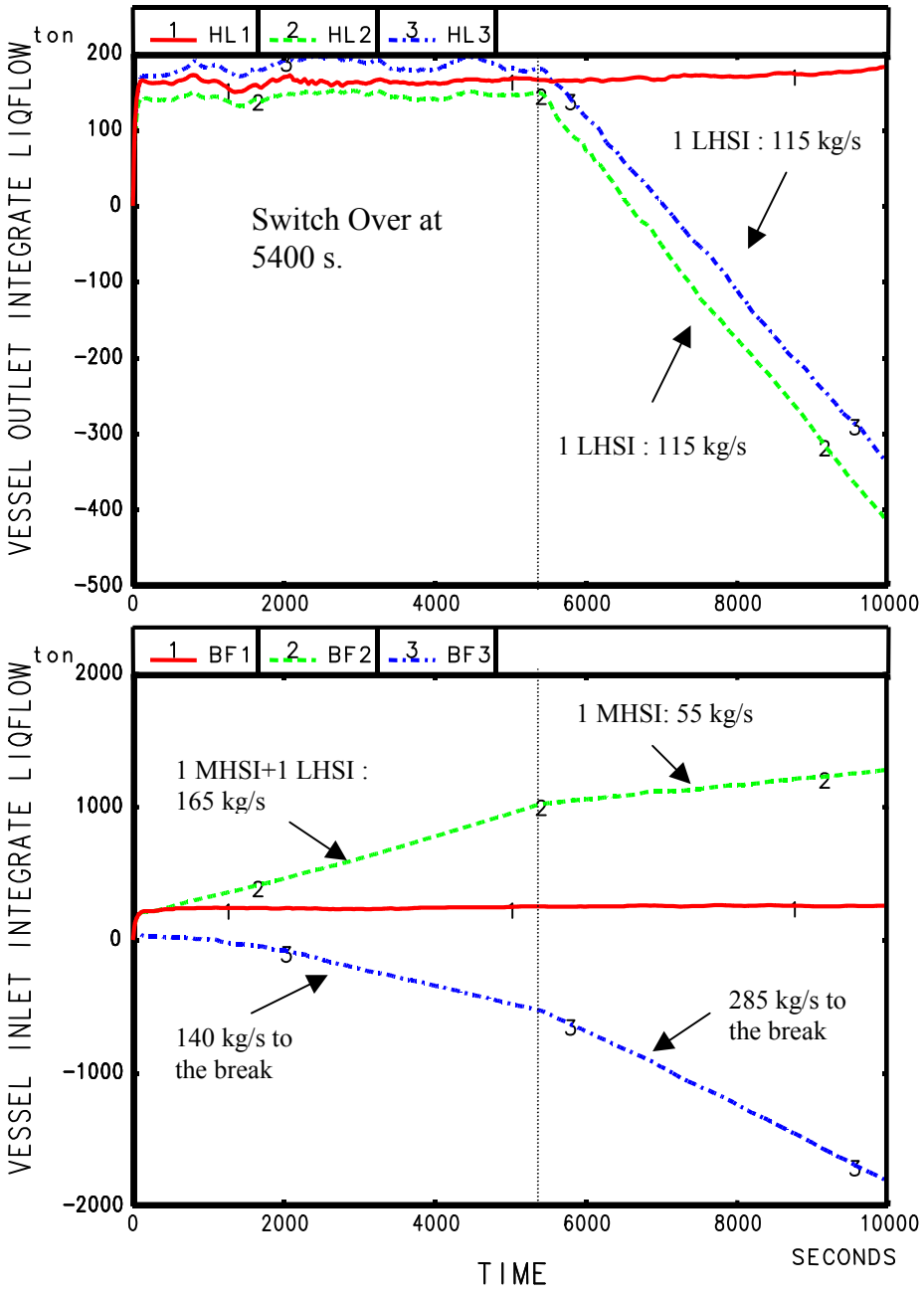
MHSI and LHSI pumps injection
Break flow rate
390 cm^2 (ϕ 225 mm) RIS [SIS] line break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 24



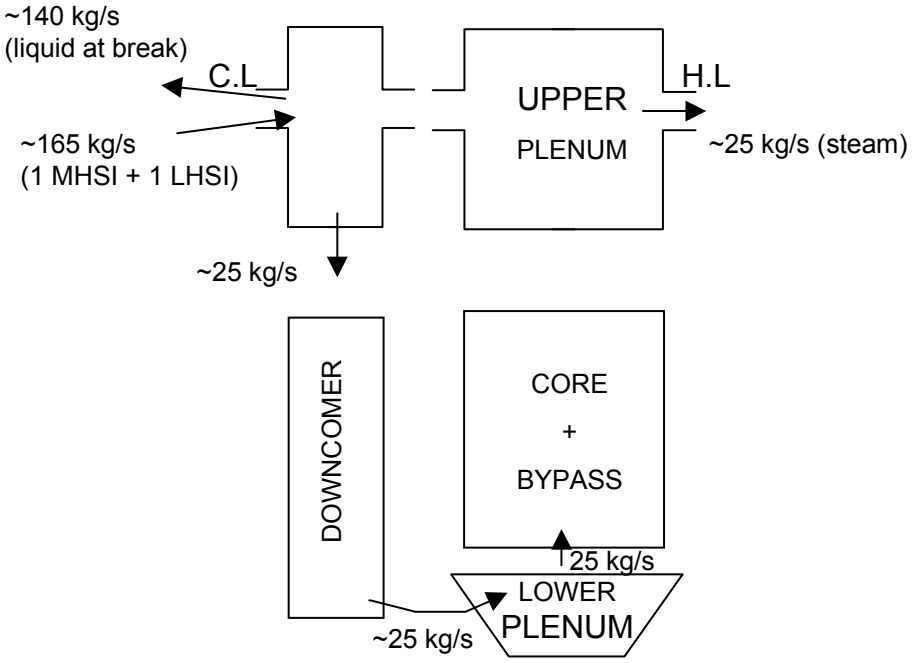
LHSI water temperature
IRWST/MHSI water temperature
390 cm^2 (ϕ 225 mm) RIS [SIS] line break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 25

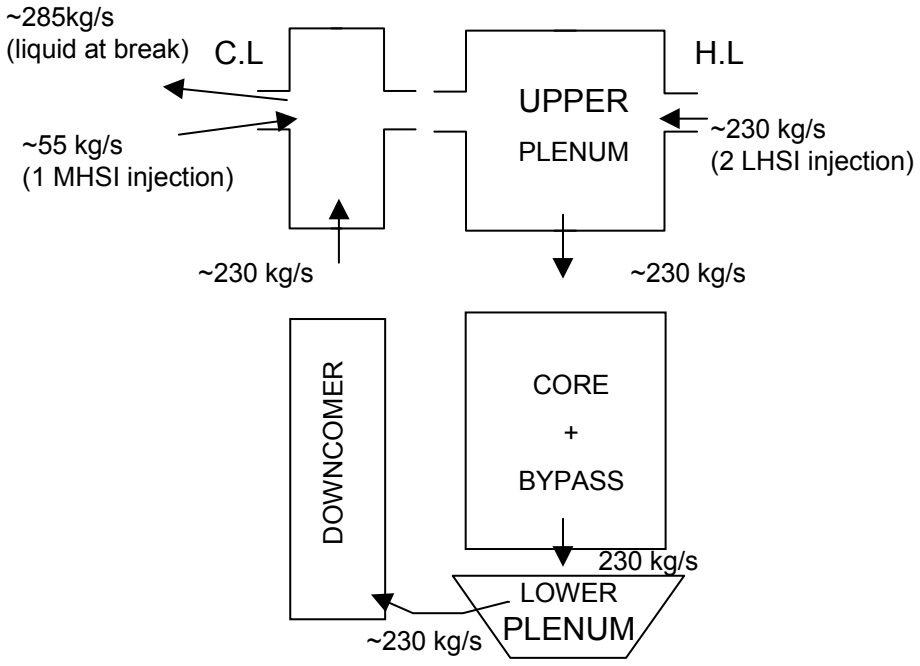


RPV outlet integrated liquid flow rates
RPV inlet integrated liquid flow rates
390 cm^2 (ϕ 225 mm) RIS [SIS] line break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 26

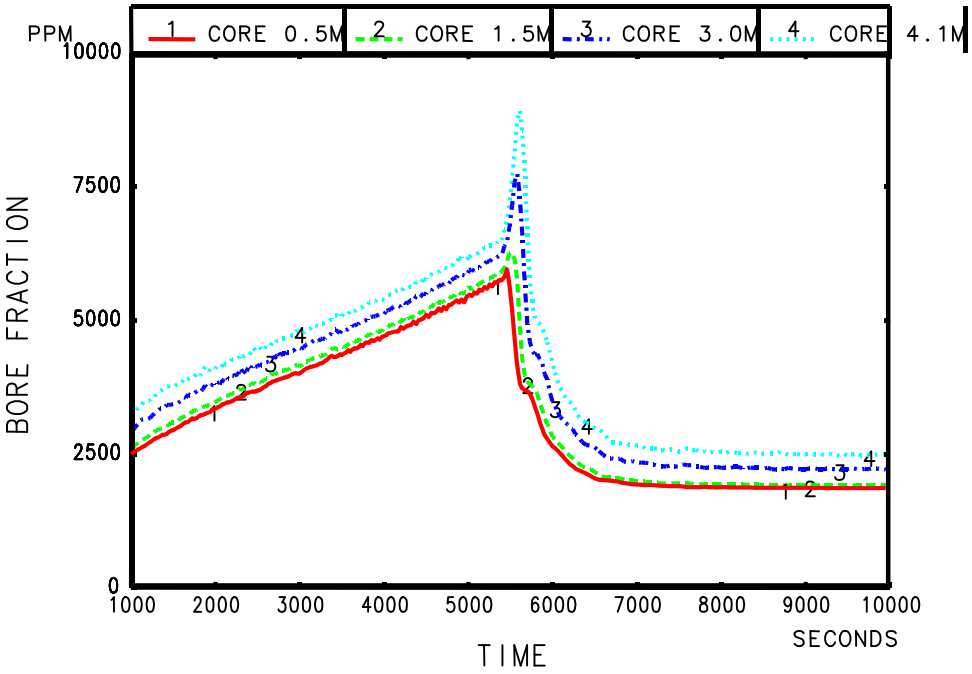


Before switch-over - liquid flow rate circulation in the RPV
390 cm^2 (ϕ 225 mm) RIS [SIS] line break - (State A, PCC-4)



After switch-over liquid flow rate circulation in the RPV
390 cm^2 (ϕ 225 mm) RIS [SIS] line break - (State A, PCC-4)

APPENDIX 14B.2.3 – FIGURE 27



Boron concentration evolution in the core
390 cm^2 (ϕ 225 mm) RIS [SIS] line break - (State A, PCC-4)

2.4. INADVERTENT CLOSURE OF ONE/ALL MAIN STEAM ISOLATION VALVE(S) (PCC-3)

2.4.1 Inadvertent closure of all VIV [MSIV]s (in state A, PCC-3)

2.4.1.1. Identification of causes and accident description

2.4.1.1.1. Issues of concern

Inadvertent closure of all VIV [MSIV]s is an overheating event leading to a risk of DNB and excessive primary and secondary overpressures.

- Identification of causes:

  Inadvertent closure of VIV [MSIV]s might typically be initiated by a spurious closure I&C signal to all 4 VIV [MSIV]s.

- Precautions to avoid accident occurrence:

  The assignment of each of the 4 steam pilots of each VIV [MSIV] to a different electrical division ensures a very low probability of spurious closure of the 4 VIV [MSIV]s due to the spurious opening of one of the pairs of steam pilots in series on each VIV [MSIV].

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

2.4.1.1.2. Typical sequence of events

- From the initiator to the controlled state

  The closure of all VIV [MSIV]s results in termination of the main steam flow. The decrease in heat removal leads to a primary and secondary pressure and temperature increase. A reactor trip and turbine trip occur, with primary and secondary pressures limited by pressuriser and SG pressure relief devices.

  The controlled state is defined in this case as hot shutdown with residual heat being removed via the VDA [MSRT] and ASG [EFWS] (the ARE [MFWS]/AAD [SSS] are not considered since they are not F1 classified).

- From the controlled state to the safe shutdown

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

The sequence of actions performed (initiated by the operator) to reach LHSI/RHR operating conditions are as follows:

- RCP [RCS] boration:

  During the cooldown, RCP [RCS] boration is performed via the RBS [EBS] (the RCV [CVCS] is discounted as it is not F1 classified).

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

- RCP [RCS] cooldown:

  RCP [RCS] cooldown to LHSI/RHR connection conditions is performed via the secondary side by decreasing the VDA [MSRT] setpoints (the GCT [MSB] is unavailable as the VIV [MSIV]s are closed).

  The RCP [RCS] cooling rate is made compatible with the ASG [EFWS] tank capacity, which means that the LHSI/RHR operating conditions are reached before the emptying of the ASG [EFWS] tanks.

  The EPR cooling rate is -50°C/h if 2 RBS [EBS] trains are available, or -25°C/h if only 1 RBS [EBS] train is available, providing it is not limited by the VDA [MSRT] capacity. The RBS [EBS] is designed so that the RBS [EBS] boration matches the reactivity insertion resulting from RCP [RCS] cooling. The ASG [EFWS] tank design allows 2 Reactor Coolant Pumps to be kept in operation during the RCP [RCS] cooling phase (shut down of 2 out of 4 Reactor Coolant Pumps must be performed by the operator).

- RCP [RCS] depressurisation:

  After cooldown to a hot leg temperature of 180°C, if the RCP [RCS] pressure is still greater than the LHSI/RHR connection pressure of 30 bar, the operator is required to momentarily open the PSV in order to depressurise the RCP [RCS]. During the depressurisation phase, the LHSI ensures a minimum RCP [RCS] pressure of about 20 bars so that the RCP [RCS] subcooling margin is not impaired. The LHSI/RHR connection conditions are then met. One LHSI/RHR train is sufficient to remove the decay heat.

2.4.1.2. Safety criteria

The consequences of inadvertent closure of all VIV [MSIV]s are analysed considering the following decoupling criterion:

- the number of rods experiencing DNB must remain below 10%.

2.4.1.3. Definition of studied cases

- Fuel cladding integrity

  An analysis is performed to demonstrate that the minimum DNBR remains above the limit of 1.00 (see section 0.2.8), thus ensuring fuel cladding integrity.

  The very short term phase from the initiating event to the time of minimum DNBR is analysed in detail in the following sections.

  The demonstration of the achievement of the controlled state and the safe shutdown is based on a qualitative evaluation, referring to other analyses in this Appendix.

- Radiological consequences

  The safety criteria to be met are the dose equivalent limits applicable to releases to the atmosphere, as discussed in Section 1.

The bounding transient with regard to radiological release is the loss of condenser vacuum analysed in section 2.5.2. The steam release to the atmosphere is identical in both cases.

2.4.1.4. Methods and assumptions

2.4.1.4.1. Methods of analysis

The analysis of this accident is carried out using the THEMIS code).

The DNBR calculation is performed with the FLICA code.

For modelling the system transient (THEMIS), the analysis methodology involves the following steps:

- identification of dominant phenomena,
- verification of the adequacy of the code to simulate these phenomena,
- application of conservative PCC analysis rules in the transient modelling.

  - The dominant phenomena in the transient are:
    - SG overpressurisation and heat up, due to the loss of steam flow,
    - RCP [RCS] overpressurisation and heat up, resulting from the reduction in SG heat removal,
    - core power changes, resulting from neutronic feedback due to RCP [RCS] coolant density and boron changes.
  - All these phenomena are within the applicability range of THEMIS, which applies a point-kinetics neutronics model accounting for all relevant aspects (moderator density, nuclear power, boron concentration, rod position). Confidence in the thermal-hydraulic qualification of the model is based on:
    - use of recognised and tested correlations, e.g. :
      - Dittus-Boelter correlation for forced convection heat-transfer on the primary side of the SG-tubes; the Jens-Lottes correlation for nucleate boiling heat-transfer on the secondary side of the SG-tubes,
      - Zuber-Wallis, EPRI, and Patricia GV2 drift flux correlations used in the axial SG model,
      - Martinelli-Nelson two-phase pressure drop correlation used in the axial SG model.

- extensive validation of specific models against test results from small-scale simulations, e.g.:
  - validation of the axial SG model (no economiser) against MB2 mock-up tests and the axial SG model (with an N4 SG-type economiser) against the MEGEVE mock-up tests. (The axial SG model refers to the model used for the SG secondary side. The axial SG model is well adapted to the calculation of plant transients involving secondary side overpressure due to loss of steam flow (as in VIV [MSIV] closure), because of its ability to compute the internal recirculation flow and to model the different zones of the SG (subcooled, two-phase, saturated, superheated), and their evolution during transients).
- the overall validation of the code by simulation of PWR plant transients, e.g. :
  - RCP [RCS] overpressurisation and depressurisation transients in the CRUAS 3 x loop plant involving heater and normal/auxiliary spray startup at various power levels and pressuriser levels, used to validate the THEMIS pressuriser model,
  - steady-state operation at different power levels in BUGEY 3 x loop plant and the PALUEL 4 x loop plant, used to validate the SG axial model in various stable operating conditions,
  - load rejection and reactor trip transients in the PALUEL 4 x loop plant which were accurately simulated with THEMIS using the axial SG model. In these tests the calculated loop temperatures compared well with measured values, validating the overall SG/RCP [RCS] heat transfer modelling and the RCP [RCS] hydraulic modelling. The SG secondary side overpressure transient calculated by THEMIS agreed closely with the measured values. These results help demonstrate the applicability of the SG axial model for transient operating conditions.

- The transient analysis uses the conservative PCC analysis rules defined in Sub-chapter 1. These rules require the application of pessimistic boundary conditions appropriate to the decoupling criteria being considered. These pessimisms involve:
  - conservative representation of the initiating event, to maximise the resulting impact,
  - conservative assumptions about plant initial conditions (control dead band limits, maximum measurement uncertainties),
  - assuming minimum effectiveness of protection and mitigation actions (maximum uncertainty on each I&C measurement, signal delay, system response time and system performance capacity),
  - pessimistic quantification of neutronic feedback effects (use of conservative moderator and fuel Doppler coefficients).

The methodology results in conservative analysis which can be used directly for comparison with the decoupling criteria.

2.4.1.4.2. Protection and mitigation actions

The following F1A I&C functions provide protection following inadvertent closure of all VIV [MSIV]s, with regard to DNB protection (for assessments against other safety criteria, see relevant sections in 2.4.1.3 below):

- reactor and turbine trip on “SG pressure > MAX 1”,
- reactor and turbine trip on “PZR pressure > MAX 2”,
- reactor and turbine trip on low DNBR (benefit not claimed in the present accident analysis).

The F1A I&C functions required to reach the controlled state are those involved in residual heat removal via the SG (ASG [EFWS] and VDA [MSRT]) as described in Section 2.5.2 (loss of condenser vacuum). The F1B functions required to transfer the plant from the controlled state to the safe shutdown state, are also described in section 2.5.2.

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

2.4.1.5.1. Choice of single failure and preventative maintenance

The minimum DNBR occurs at the beginning of control rod drop.

It follows that:

- the most onerous single failure is a stuck rod.
- preventative maintenance is not relevant for this event.

2.4.1.5.2. Initial state

The initial conditions correspond to 100% full power operation with maximum uncertainties to minimise the predicted DNBR. The uncertainties applied are given in Appendix 14B.2.4 - Tables 1 to 6.

2.4.1.5.3. Specific assumptions

- Assumptions related to neutronic data and decay heat

  The fission power (term A) is calculated using a point-kinetics model.

  A maximum absolute value for the moderator density coefficient (EOC value) is assumed, in order to maximise the power increase at the beginning of the transient

  A minimum absolute value of the Doppler coefficient is assumed in order to maximise the power increase at the beginning of the transient.

  The maximum decay heat curve (terms B+C) is used (see section 0.2).

- Assumptions related to non F1 systems

  No benefit is claimed for control I&C functions, as they have either no impact or have a beneficial impact on the minimum DNBR value.

Note: pressuriser spray is not modelled. Spray limits the primary pressure increase resulting in a lower power increase, both effects tending to increase the minimum DNBR.

- Assumptions related to F1 systems

  The only F1 systems claimed in the analysis are the reactor trip systems (see section 2.4.1.5.1). Minimum effectiveness of the trip systems are assumed, taking account of all relevant uncertainties (see section 0.2).

- Other assumptions

  The initial DNBR is assumed to be 1.26 (minimum value - see section 2.5.1).

2.4.1.5.4. Results and conclusions

- Results for DNBR

Appendix 14B.2.4 - Table 2 gives the sequence of events.

Appendix 14B.2.4 - Figure 1 to 2 show the evolution of the main parameters during the transient.

At the beginning of the transient, the moderator density increase (due to the RCP [RCS] pressure increase) leads to a power increase and thus DNBR is decreasing.

Then, the combination of a sharp RCP [RCS] pressure increase and a small RCP [RCS] temperature increase results in a stable DNBR.

As soon as RCP [RCS] temperature rise becomes significant, the DNBR decreases again. This decrease is terminated by the thermal power reduction, mainly due to moderator feedback.

The minimum DNBR value of 1.13 is reached immediately before the reactor trip signal (SG pressure > MAX 1). The decoupling criterion of 1.00 is met.

Appendix 14B.2.4 - Table 3 gives the thermal-hydraulic conditions at the time of minimum DNBR.

Demonstration of the attainment of the controlled state (hot shutdown) is the same as for a loss of condenser vacuum event i.e.

- regarding reactivity control: the initial power level and the shutdown rod worths in the two events are identical,
- regarding residual heat removal: the SG heat removal capabilities (VDA [MSRT] and ASG [EFWS]) are the same,
- regarding coolant inventory: neither event impacts on the primary coolant inventory.

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

The safe shutdown state is defined as the state in which LHSI/RHR connection conditions are reached.

The transition from the controlled state to the safe shutdown state is similar to that following a loss of condenser vacuum (see section 2.5.2).

Indeed, both the initial and final states are the same and the capabilities of F1B systems required in both cases are identical (VDA [MSRT], ASG [EFWS], RBS [EBS]).

2.4.2. Inadvertent closure of 1 VIV [MSIV] (in state A, PCC-3)

2.4.2.1. Identification of causes and accident description

2.4.2.1.1. Issues of concern

The inadvertent closure of 1 VIV [MSIV] is an overheating event leading to a risk of DNB and primary and secondary system overpressurisation.

- Identification of causes

  Inadvertent closure of a single VIV [MSIV] could be initiated:

  - either by a spurious I&C signal on one VIV [MSIV],
  - or by spurious opening of 2 steam pilots in series on one VIV [MSIV].

- Precautions to avoid accident occurrence

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

Inadvertent closure of 1 VIV [MSIV] is classified as a PCC-3 event.

2.4.2.1.2. Typical sequence of events

- From the initiating event to the controlled state:

  Closure of 1 VIV [MSIV] results in termination of the steam flow from the affected SG. The heat removal decrease leads to primary and secondary pressures and temperatures increase.

  A reactor trip and a turbine trip occur. The increase in secondary pressure is limited by the SG relief devices.

  The controlled state is defined in this case as hot shutdown with residual heat being removed via VDA [MSRT] and ASG [EFWS] (ARE [MFWS] / AAD [SSS] being not credited since they are not F1 classified).

- Transition from the controlled state to the safe shutdown state

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

The sequence of operator actions required to reach LHSI/RHR operating conditions are as follows:

- RCP [RCS] boration:

  During cooldown, RCP [RCS] boration is assumed to be performed using the RBS [EBS] (the RCV [CVCS] is ignored as it is not F1 classified).

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

- RCP [RCS] cooldown:

  RCP [RCS] cooldown to LHSI/RHR connecting conditions is performed using the secondary system by decreasing the VDA [MSRT] setpoints (the GCT [MSB] on the unaffected SGs is ignored since it is not F1 classified).

  The RCP [RCS] cooling rate is made consistent with the ASG [EFWS] tank capacity, to ensure that LHSI/RHR operating conditions are reached before the ASG [EFWS] tanks are empty.

  The target 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, subject to the available VDA [MSRT] capacity. The RBS [EBS] is designed so that the RBS [EBS] boration compensates for the reactivity insertion due to the RCP [RCS] cooldown. The ASG [EFWS] tank design allows 2 Reactor Coolant Pumps to be kept in operation during the RCP [RCS] cooldown (2 of the 4 Reactor Coolant Pumps must be shut down by the operators).

- RCP [RCS] depressurisation:

  After cooldown to a hot leg temperature of 180°C, if the RCP [RCS] pressure is greater than the LHSI/RHR connecting pressure of 30 bar, the operator will momentarily open the PSV in order to depressurise the RCP [RCS]. During the depressurisation phase, the LHSI ensures a minimum RCP [RCS] pressure of about 20 bar so that the RCP [RCS] subcooling margin is not impaired. After LHSI/RHR connection conditions are met, one LHSI/RHR train is sufficient to remove the decay heat.

2.4.2.2. Safety criteria

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

The consequences of the event assessed against the following decoupling criterion:

- the number of fuel rods experiencing DNB must remain below 10%.

2.4.2.3. Definition of cases studied

- Fuel cladding integrity

  Analysis is performed to demonstrate that the minimum DNBR remains above 1.00 (see section 0.2.8), confirming that fuel cladding integrity is maintained.

  The short term phase from event initiation to the time of minimum DNBR is analysed in detail below.

  Demonstration of the attainment of the controlled state and the safe shutdown state is based on a qualitative evaluation, making reference to other analyses in this Appendix.

- Radiological consequences

  The safety criteria to be met are the dose equivalent limits applicable in the case of a release to the atmosphere, as discussed in section 1.

  The bounding transient with regard to the radiological release is the loss of condenser vacuum transient analysed in section 2.5.2. The quantity of steam released to the atmosphere is identical in both cases.

2.4.2.4. Methods and assumptions

2.4.2.4.1. Method of analysis

Analysis of this accident is carried out using the THEMIS code.

The DNBR calculation is performed with the FLICA code.

The analysis methodology is the same as that used for modelling the 'inadvertent closure of all VIV [MSIV]s' event presented in section 2.4.1.4.1. In contrast to the former event, the 'inadvertent closure of one VIV [MSIV]' event results in asymmetrical RCP [RCS] behavior. The capability of the code to model asymmetrical conditions and the analysis methodology (use of RPV flow mixing coefficients), are presented in section 2.15.1.4.1 which considers the SLB accident.

2.4.2.4.2. Protection and mitigation actions

The following F1A I&C functions provide protection following inadvertent closure of a single VIV [MSIV], with regard to meeting the DNBR criterion (for analyses with regard to other safety criteria, see section 2.4.2.3):

- reactor and turbine trip on SG pressure > MAX 1.

Note: the reactor trip on low DNBR channel could be qualified for such an asymmetrical event. Anyway, this reactor trip signal is not credited in the present accident analysis.

The F1A I&C functions required to reach the controlled state are those involved in residual heat removal via the SGs (ASG [EFWS], VDA [MSRT]) as described in section 2.5.2 (loss of condenser vacuum).

The F1B functions required to transfer the plant from the controlled state to the safe shutdown, are also as described in section 2.5.2 (loss of condenser vacuum).

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

2.4.2.5.1. Choice of single failure and preventative maintenance

The minimum DNBR occurs at the beginning of control rod drop.

It follows that:

- the most onerous single failure is a stuck rod.
- preventative maintenance is not relevant for this event.

2.4.2.5.2. Initial state

The assumed initial conditions correspond to 100% full power, with pessimistic uncertainties with regard to DNBR. The initial conditions are presented in Appendix 14B.2.4 - Table 4.

2.4.2.5.3. Specific assumptions

- Neutronic data and decay heat assumptions

  BOC neutronic data are used as this results in a minimum decrease in the nuclear power at the time the rods begin to drop (this contrasts with the case of inadvertent closure of all VIV [MSIV]s, when use of EOC neutronic data results in the most onerous power increase).

  The maximum decay heat curve is used (see section 0.2).

- Assumptions related to non F1 systems

  I&C control functions are not taken into account, as these have either no impact or have a beneficial impact on the minimum DNBR value.

- Assumptions related to F1 systems

  No F1 system is utilised in this phase of the accident, other than systems involved in the reactor trip (see section 2.4.2.5.1). Minimum effectiveness of the reactor trip function is assumed, allowing for all relevant uncertainties (see section 0.2).

- Other assumptions

  The assumed initial DNBR is 1.26 (minimum value) (see section 2.5.1).

2.4.2.5.4. 2.4.2.5.4 Results and conclusions

- DNBR criterion:

  Appendix 14B.2.4 - Table 5 shows the sequence of events.

Appendix 14B.2.4 - Figure 3 and 4 show the evolution of the main parameters during the transient.

Initially, the small variations of pressure and temperature do not significantly change the DNBR. A slight increase in DNBR then occurs due to the reactor pressure increase.

The increasing coolant temperature ultimately results in a DNBR decrease, which is terminated by the control rod drop.

The minimum DNBR value is calculated as 1.16. The decoupling criterion of 1.00 is met.

Appendix 14B.2.4 - Table 6 gives the thermal-hydraulic conditions at the time of minimum DNBR.

- Controlled state

  Demonstration of the attainment of the controlled state (hot shutdown) is the same as for the loss of condenser vacuum event i.e.

  - regarding reactivity control: the initial power level and the shutdown rod worths in the two events are identical,
  - regarding residual heat removal: the SG heat removal capabilities (VDA [MSRT] and ASG [EFWS]) are the same,
  - regarding coolant inventory: neither event impacts on the primary coolant inventory.

2.4.2.6. Description of cases studied (controlled state to safe shutdown state)

The safe shutdown state is defined as a state in which the LHSI/RHR connection conditions are reached.

The transition from the controlled state to the safe shutdown is the same as in the loss of condenser vacuum event (see section 2.5.2): the initial and final states and the capabilities of the F1A/F1B systems required to reach the safe shutdown state (VDA [MSRT], ASG [EFWS], RBS [EBS]) are identical in the two cases.

APPENDIX 14B.2.4 – TABLE 1

Inadvertent closure of all VIV [MSIV]s
Main assumptions

Parameter	Limiting value used
Reactor Trip "SG pressure > MAX 1" Setpoint Delay	 93 + 1.5 = 94.5 bar 1.2 s
Reactor Trip "PZR pressure > MAX 2" Setpoint Delay	 166.5 + 1.5 = 168 bar 1.2 s
Initial conditions RCP [RCS] flow rate Power Pressure Average temperature PZR level SG level	 thermal-hydraulic 100% + 2% = 102% FP 155 - 2.5 = 152.5 bar 311.25 + 2.5 = 313.75°C 56 + 5% = 61% of measurement range 56% of narrow range

APPENDIX 14B.2.4 – TABLE 2

Inadvertent closure of all VIV [MSIV]
Sequence of events

Event	Time (seconds)
Inadvertent closure of all VIV [MSIV]s	0 s
Minimum DNBR (1.13)	2.1 s
Reactor trip signal (SG pressure > MAX 1)	4.1 s
Start of rod drop	5.3 s

APPENDIX 14B.2.4 – TABLE 3

Inadvertent closure of all VIV [MSIV]
Thermal-hydraulic conditions at time of minimum DNBR

Parameter	Value
Power	107.2%FP
Pressure	156.7 bar
Temperature	295 °C
Flow rate (volumetric)	100% of thermal-hydraulic flow rate

APPENDIX 14B.2.4 – TABLE 4

Inadvertent closure of 1 VIV [MSIV]
Main assumptions

Parameter	Limiting value used
Reactor Trip "SG pressure > MAX 1" Setpoint Delay	 93 + 1.5 = 94.5 bar 1.2 s
Initial conditions RCP [RCS] flow rate Power Pressure Average temperature PZR level SG level	 thermal-hydraulic 100% + 2% = 102% FP 155 - 2.5 = 152.5 bar 311.25 + 2.5 = 313.75 °C 56 - 5% = 51% of measurement range 56 - 5% = 51 % of narrow range

APPENDIX 14B.2.4 – TABLE 5

Inadvertent closure of 1 VIV [MSIV]
Sequence of events

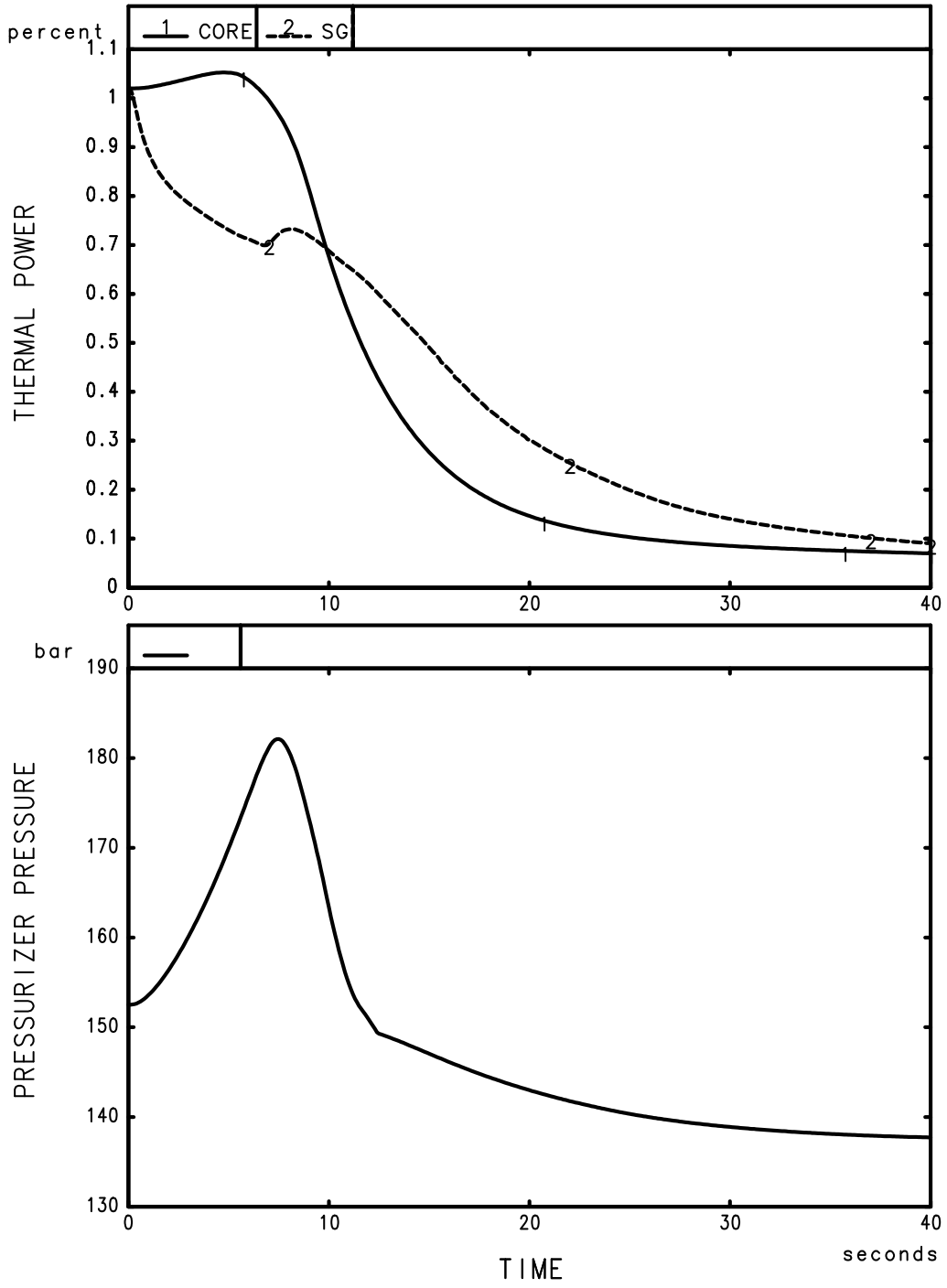
Event	Time (seconds)
Inadvertent closure of 1 VIV [MSIV]	0 s
Reactor trip signal (SG pressure > MAX 1)	4.3 s
Beginning of rods drop	5.5 s
Minimum DNBR (1.16)	5.5 s

APPENDIX 14B.2.4 – TABLE 6

Inadvertent closure of 1 VIV [MSIV]
Thermal-hydraulic conditions at time of minimum DNBR

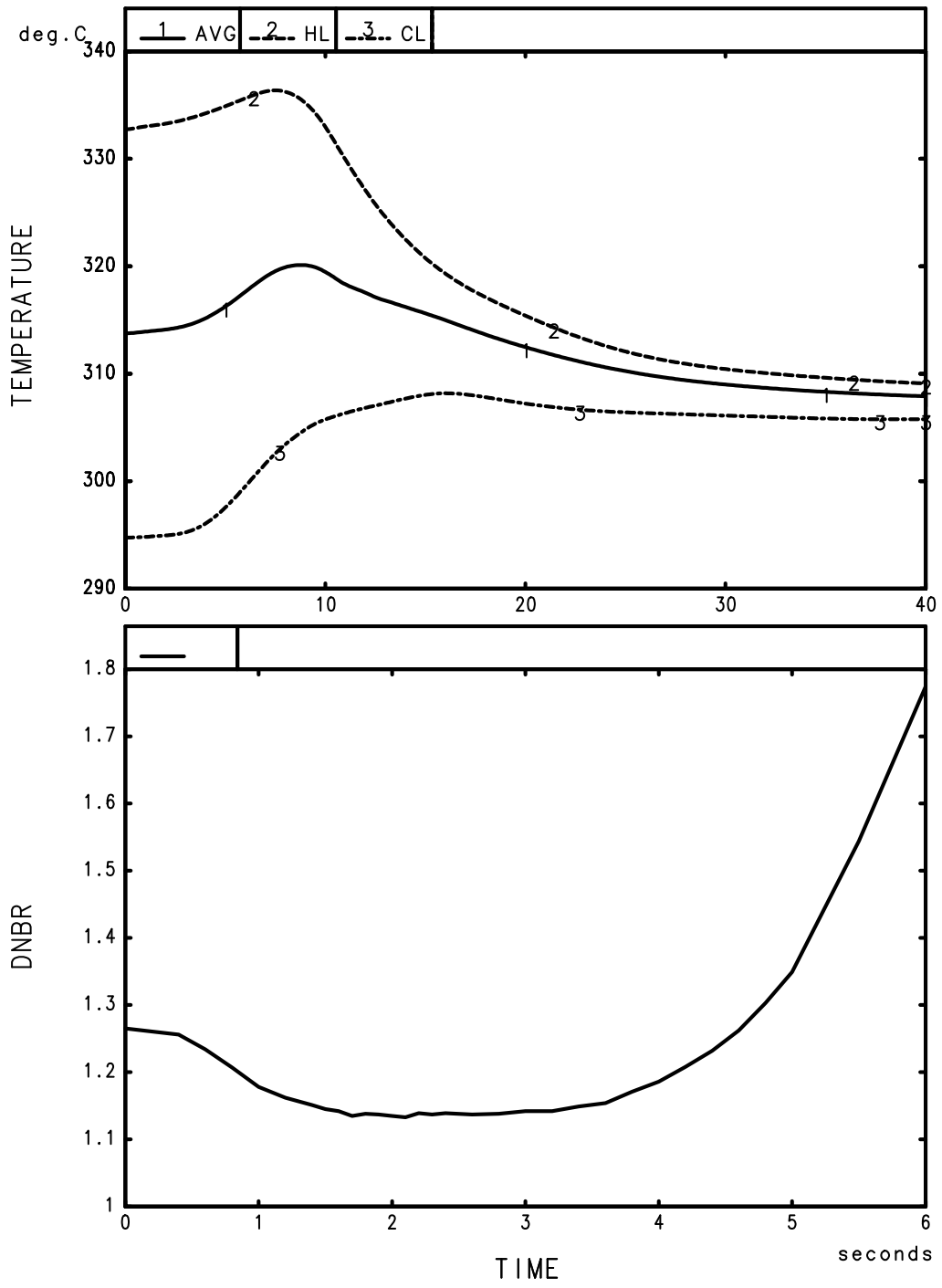
Parameter	Value
Power	101.9%FP
Pressure	154.1 bar
Temperature	297.9°C
Flow rate (volumetric)	100% of thermal-hydraulic flow rate

APPENDIX 14B.2.4 - FIGURE 1



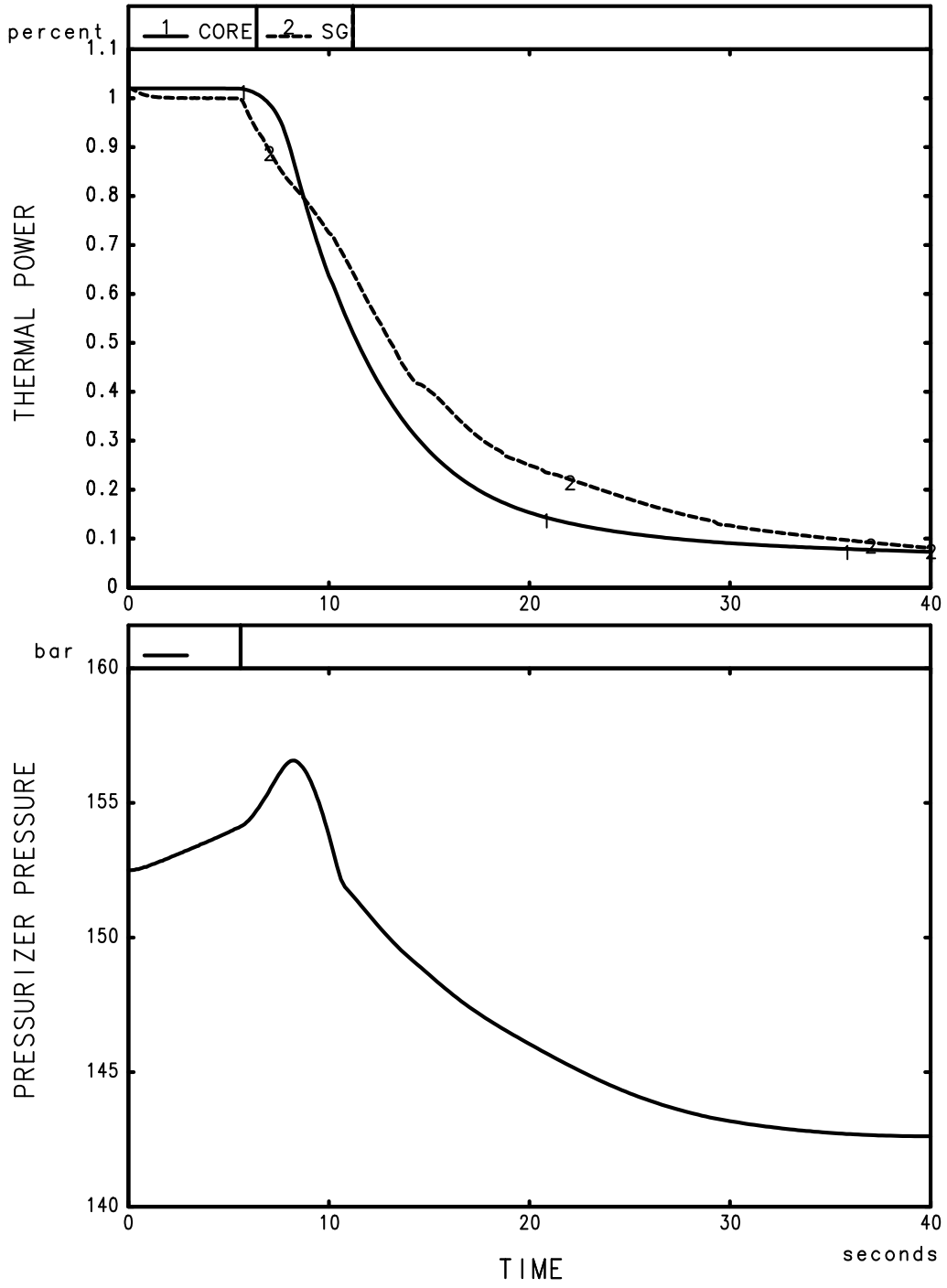
Inadvertent closure of all VIV [MSIV]s

APPENDIX 14B.2.4 - FIGURE 2



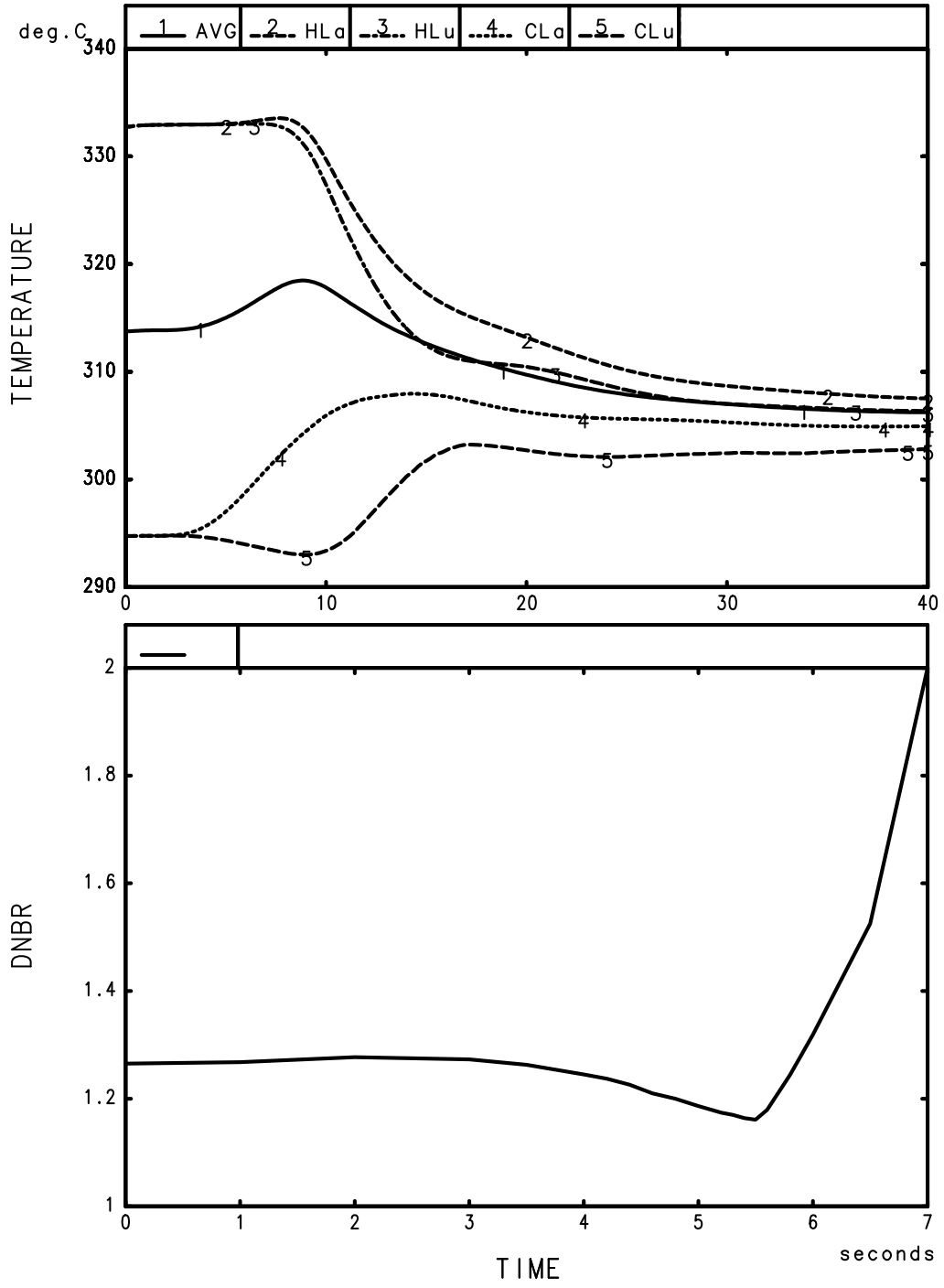
Inadvertent closure of all VIV [MSIV]s

APPENDIX 14B.2.4 - FIGURE 3



Inadvertent closure of 1 VIV [MSIV]

APPENDIX 14B.2.4 - FIGURE 4



Inadvertent closure of 1 VIV [MSIV]

2.5.1. Loss of Non-Emergency AC Power to the Plant Auxiliaries (PCC-2)

This event is analysed in reactor state A at power only, since this state is clearly the most challenging in terms of the demands on safeguard systems required to reach the controlled and safe shutdown states.

2.5.1.1. Identification of Causes and Accident Description

A complete loss of non-emergency AC power results in the loss of all power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, ARE [MFWS] pumps, etc.

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

The event sequence involves a decrease in heat removal by the secondary system, accompanied by a flow coast down which further reduces the capacity of the primary coolant to remove heat from the core. The Main Steam (MS) relief control system maintains a maximum secondary system pressure of about 93 bar.

Further details are given below:

a) Instruments and safety related valve-motors are supplied from non-interruptible battery backed supplies. Other safety related motors are supplied according the load sequence from emergency DC power sources.

   As the steam system pressure rises following reactor and turbine trip, the Main Steam Relief Isolation Valves (MSRIVs) upstream of the Main Steam Relief Control Valves (MSRCVs) are automatically opened. The condenser is assumed not to be available for steam dump because of the loss of AC power to the cooling water pumps, and also because the condenser is not F1 qualified. If the steam flow rate through the relief trains is not available, the steam generator safety valves may lift to remove heat from the primary and secondary systems.

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

c) The standby diesel generators, started on loss of voltage on the plant emergency busses, begin to supply essential plant loads according to the load sequence of the consumers.

d) The emergency feedwater system (ASG [EFWS]) begins operation either as a result of operator action or automatically when the SG level MIN2 setpoint is reached, ensuring long term heat removal from the reactor coolant system.

e) Protection

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

   - The event sequence can be divided into two distinct phases up to attainment of the controlled state (hot standby).
     - a short term phase, characterised by reduced DNB margins, (first few seconds of the event).

- a long term phase, in which it must be shown that acceptable residual core heat removal and activity release to the atmosphere via MS Relief Train (VDA [MSRT]) are achieved up to the time the controlled state is reached.

Controlled and Safe Shutdown State

It must be demonstrated that the controlled state can be reached using only F1A functions, and that the safe shutdown state can be reached using only F1A and F1B functions, without exceeding the following safety and decoupling criteria:

- Safety Criteria:
  - Radiological limits for normal operation.
- Decoupling Criterion:
  - DNB avoided (DNBR limit: 1.0)

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

- Reactor Trip:
  - The reactor trip is initiated from the following signal: “Reactor Coolant Pump speed < 91%”
- Systems:
  - Steam relief trains
  - 4 ASG [EFWS] trains
  - Pressuriser safety valves
- Signals :
  - Actuation of MS relief trains at SG pressure of 93 bar
  - Actuation of ASG [EFWS] at SG water level < 8 m
  - Pressuriser safety valves response setpoints: 174 bar, 178 bar

For the transition from the controlled state to the safe shutdown state the following F1B functions are available (at a minimum):

- 4 MS relief trains for cooldown to RHR connection conditions, (manual action).
- 2 RBS [EBS] trains for boration during cooldown.
- 4 ASG [EFWS] trains for feedwater supply to the SGs (automatic or manual action)

The study of the short term total loss of offsite power also covers slow decay of network frequency (typically up to 2.5 Hz/s). Higher frequency decay rates are considered to be PCC-3 events and are discussed in section 2.6 of this appendix.

2.5.1.2. Methods and Assumptions

Up to attainment of the controlled state the two accident phases discussed above (short term for DNBR, long term for plant behaviour) are studied separately. The long term case is only investigated for maximum primary pressure, because radiological aspects are the same as in the 'Loss of condenser vacuum' event. The codes used for the two accident phases are different because as 3D-kinetic calculations are required for the short term whereas point kinetics calculations are adequate for the long term.

It is assumed that LOOP resulting in turbine trip occurs at t = 10 seconds in the NLOOP (Appendix 14A) calculation of the long term phase, and at t = 0 second in the PANBOX/COBRA (Appendix 14A) calculation of the short term phase.

Choice of Single Failure and Maintenance State

Time Period	Single Failure	Maintenance/ Additional Unavailability
• short term calculation	No impact	No impact
• long term regarding plant behaviour	1 MSRIV sticks in closed position	No impact

2.5.1.2.1. Important phenomena and qualification of the NLOOP and PANBOX/COBRA codes

The accident belongs to the family of transients involving complete or partial loss of forced primary coolant flow.

On occurrence of LOOP, all Reactor Coolant Pumps are lost, together with the secondary side feedwater supply and main condenser.

Phenomena to be modelled

Primary Side

- Loss of forced coolant flow resulting in a reduction of primary to secondary heat transfer.
- Early reactor trip at a relatively high coolant flow followed by a reduction of reactor power to decay heat levels and a transition to cooling by natural circulation.
- Decay heat removal by the MS relief valves at a constant MS pressure.
- Increase in primary coolant temperature and pressure after reactor trip. Possible actuation of pressuriser safety valves to limit the primary system pressure.
- Pressuriser insurge and outsurge with corresponding pressure changes in the Pressuriser and primary circuit.

Secondary side

- MS pressure increase due to closure of the turbine/GCT [MSB] valves.
- Fast opening of the MS relief train occurs followed by re-closure to maintain a constant MS pressure at zero load.
- Heat transfer area of the SG tubes not reduced and separators not overfilled.

Core behaviour with regard to core power and DNBR

- Heat transfer in the average and hot channel until control rods insertion due to reactor trip
- Neutron flux and reactor power (integral and local) depending on the thermal hydraulic parameters.
- Coolant flow behaviour including pressure drop and cross-flow.

Qualification of the NLOOP models for primary and secondary side phenomena

An integral test is available on a PWR power plant in which most of the above mentioned phenomena are addressed. This test was modelled with NLOOP, and good agreement with the plant behavior was obtained. Additionally, further integral tests on actual plants and test facilities were modelled with NLOOP and good agreement also obtained for individual secondary and primary side phenomena. The validation tests are listed below:

Event	Plant/test facility	Phenomena
Emergency power mode	KNU1	see above
Turbine trip without GCT [MSB]	Biblis B	Secondary and primary side pressure increase. Pressuriser water level behaviour, primary coolant heat up with forced RCP [RCS] circulation.
LOOP followed by a total loss of FW supply	PKL3	RCP [RCS] coast down, primary heat up til attainment of natural circulation cooling at core decay heat level.

Qualification of the models in PANBOX/COBRA for DNBR related phenomena

An event occurred on an actual PWR power plant (Reactor Coolant Pump shaft break at full power) in which most of the above mentioned phenomena occurred. This event was modelled with PANBOX, and good agreement with plant behaviour was obtained.

Additionally a further integral test on a PWR plant (reactor trip from full load) was also modelled with PANBOX. The calculation likewise showed a good agreement with the plant behaviour for the reactor trip phenomena.

Validation of COBRA hot channel modelling is performed using a wide range of single heated bundle tests for verification of the CHF tables and for modelling pressure losses at different voidage levels, enthalpy and mass flux distributions, slip flow etc. Calculations are also carried out as part of code benchmark exercises. The validation work against PWR plant data is summarised below:

Event	Plant/test facility	Phenomena
Reactor Coolant Pump shaft break	KKG	fast flow reduction in one loop
Reactor trip	KKU	locally and time depending reactor power due to rod insertion

2.5.1.2.2. Short Term Study (Evaluation of minimum DNBR)

a) Methods of Analysis

Calculations are performed using the following computer codes: PANBOX/COBRA-3-CP

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

According to section 0.2.8 of this appendix, loss of reactor coolant flow due to a loss of off-site power is the limiting PCC-2 event where DNBR is closest to the design limit of 1.0. In order to ensure the design limit is met the initial DNBR prior to the event must exceed a minimum initial value, the DNBR-LCO. The present analysis can thus be regarded as a means of determining a DNBR-LCO level which ensures that the DNBR=1.0 limit is met (no DNB during the event).

The approach applied for determination of the DNBR LCO is as follows:

1) The initial state of the plant is defined taking into account the best-estimate FΔH which is enveloping for possible fuel management schemes, and the available LCOs, including deadbands and uncertainties in important parameters such as RCP [RCS] pressure, temperature, power level, axial offset etc (see Appendix 14B.2.5.1 – Table 1). The resulting DNBR curve, obtained using the coupled neutronic/thermal-hydraulic calculation, is called DNBR1.

2) Starting from this initial state, further arbitrary increases in the local FΔH, (i.e. an arbitrary increase in the power of the assembly containing the hot channel and the surrounding assemblies) is imposed, and further coupled neutronic/thermal-hydraulic analyses performed, until the minimum DNBR becomes equal to the criterion of 1.0. In these analyses the increase in FΔH includes a factor accounting, in an uncoupled manner, for the global uncertainty of the low DNBR surveillance channel (32%).

3) The final calculation in the parametric study gives the variation of DNBR during the transient and defines two limiting initial values of DNBR, corresponding to the low DNBR LCO (without uncertainties due to the low DNBR surveillance channel) called DNBR2, and the absolute minimum DNBR (including uncertainties due to the low DNBR surveillance channel) called DNBR3.

It is noted that this approach is based on increasing the hot channel FΔH, which involves a void fraction increase at the hot channel outlet and thus, inherently, a density reactivity feedback which decreases the hot channel power axial offset.

It is further noted that the FΔH value imposed in the final calculation is much higher than the bounding value expected for fuel management schemes considered in the BDR, i.e. well above the target value of 1.54.

The relevant initial conditions for the final calculation are given in Appendix 14B.2.5.1 – Table 1.

The three initial values and the DNBR curves mentioned above, i.e. DNBR1, DNBR2 and DNBR3, are shown in Appendix 14B.2.5.1 – Figure 1.

b) Initial and Boundary Conditions

Initial conditions were selected to be the most conservative with respect to the DNB limit during steady-state operation.

The major initial and boundary conditions are listed in Appendix 14B.2.5.1 - Table 1.

Reactor Trip

In all cases in which the reactor is assumed to be at power, the latest reactor trip is generated by low reactor coolant pump speed.

The low pump speed setpoint is 91%, including measurement uncertainties. A conservative delay between setpoint actuation and the beginning of rod drop is used (0.6 seconds).

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

- The RCCA drop characteristic, i.e. RCCA position versus t (time), as described in Appendix 14B.0.2 – Table 7
- The RCCA worth as a function of RCCA position calculated by PANBOX on the basis of the actual 3D power distribution

Flow Coast down and Fuel-Cladding Heat Transfer

The core flow coast down curve in Appendix 14B.2.5.1 - Figure 1 is used, which is based on a conservatively reduced Reactor Coolant Pump inertia (15% reduction).

α-gap is taken as 9700 W/m2*K

2.5.1.2.3 Long Term Study of overall plant behaviour up to attainment of controlled state

a) Methods of Analysis

The analysis is performed using the NLOOP code.

b) Initial and Boundary Conditions

The initial and boundary conditions were selected to be the most conservative with respect to high primary/high MS-pressure. The main initial and boundary conditions are listed in Appendix 14B.2.5.1 - Table 1. Availability of the MS bypass in the first 10 seconds after accident is discounted.

The analysis allows for the effects of heat structures in the RCP [RCS] and SGs.

The RCCA rod worth versus time represents the conservative bottom peaked shape shown in Appendix 14B.0.2 – Table 8.

The pump coast down reactor trip setpoint is the same as that used for the short term analysis.

The assumed setpoints of MS/primary safety/relief valves allow for uncertainties (see section 0.2 of this appendix)

The worst case single failure assumed is failure of one MS relief train. Maintenance of safety related components is not considered as it has no impact on the results.

2.5.1.2.4. Long Term Study: Activity Release

Calculation of maximum activity release in the long term accident phase is considered to be covered by the 'Loss of condenser vacuum' case presented in section 2.5.2 of this appendix, based on the following arguments:

- The early automatic phase of the accident involving emptying of the SG with single failure of the MS-relief train is very similar to the 'Loss of condenser vacuum' case in section 2.5.2 of this appendix. The only difference is the coast down of the Reactor Coolant Pumps which have an insignificant impact on drainage of the SG i.e. the SG empties in both cases.

- The steam release to atmosphere in the short and long term phases is lower in the case of loss of AC power because of the absence of heating due to operation of the Reactor Coolant Pumps.

2.5.1.3. Results and Conclusions

2.5.1.3.1. Period from the Initiating Event to the Controlled State

DNBR Evaluation

The average nuclear power, core flow rate, axial offset and the three DNBR curves DNBR1, DNBR2, DNBR3 are plotted in Appendix 14B.2.5.1 - Figure 1.

The pressure and temperature, which influence DNBR, do not change significantly up to the time of minimum DNBR and are therefore maintained constant at their initial values in the calculation.

The final conclusion of the study is that the DNBR LCO must be set to 1.66 (corresponding to DNBR 2 curve) in meet the criterion of 1.0, taking into accounts the 32% uncertainty in the low DNBR surveillance channel. The minimum DNBR occurs at a core elevation of 3.5 m.

Appendix 14B.2.5.1 - Table 2 lists the parameters plotted in Figures within this Appendix.

Overall plant behaviour

Appendix 14B.2.5.1 - Table 3 gives the sequence of events.

Plots of parameter variations with time are shown in Appendix 14B.2.5.1 - Figure 2.

A description of the parameters plotted is given in Appendix 14B.2.5.1 - Table 4.

The RCP [RCS] peak pressure is 173 bar, showing that the heat transfer from the primary to secondary side is sufficient to avoid unacceptable over-pressure.

Heat removal from the secondary side is ensured by the VDA [MSRTs] alone (no demand is made on MSSVs). The ASG [EFWS] is not required in the first 30 minutes: it is actuated automatically when the SG level falls below 8 m.

These results demonstrate that the controlled state, i.e.

- core subcritical (reactivity < 0)
- core power removed by VDA [MSRT] and ASG [EFWS]

is achieved under without exceeding the decoupling criteria. The integrity of the barriers preventing radioactivity release is not impaired.

2.5.1.3.2. Period from the Controlled State to the Safe Shutdown State

The safe shutdown state is defined as a state where the RHR connection conditions are reached.

This phase of the transient is not analysed explicitly since it is covered by analyses of other events (reference cases). The reference cases demonstrating achievement of safe shutdown with compliance with the subcriticality, decay heat removal and activity release/ barrier integrity criteria, are shown below:

Criteria	Reference case	Remark/Reason
Subcriticality	Uncontrolled boron dilution	Rod worth is sufficient (including case of stuck rod).
Activity release	Loss of condenser vacuum	In the reference case one SG is completely emptied.
Heat removal	Feed water line break	In the reference case only one train is available for cooldown.

APPENDIX 14B.2.5.1 - TABLE 1 (1/3)

Event:	Emergency Power mode (< 2 h)
Event-Category (PCC/RRC):	PCC-2
Family of Events:	Decrease in primary and secondary side heat removal
Safety Criteria:	Radiological limits for normal operation covered by 'Loss of Condenser Vacuum'
Decoupling Criteria:	No DNB (DNBR limit : 1.0)
Purpose of Analysis	One analysis to determine minimum DNBR; one analysis to model overall plant behaviour
Safe shutdown conditions: controlled state safe shutdown:	 Hot shutdown with heat removal via MS relief valves See section 2.5.2 of this appendix.

APPENDIX 14B.2.5.1 - TABLE 1 (2/3)

Initial Conditions	Best estimate	Conservative assumption for DNBR calculation	Conservative assumption for overall plant behaviour	Remarks
Reactor/turbine power	100/100%	102/code res.	102/code res.	
Thermal reactor power	4900 MW	4998MW	4998 MW	
Reactor cooling pump power		30.59 MW	30.59 MW	
Thermal steam generator power(per SG)		1256.9 MW	1256.9 MW	
Initial insertion of control rods		ARO	ARO	
• Void fraction at hot channel outlet		0.188		At DNBR LCO = 1.66
		0.303		At DNBR = 1.26
Initial Axial Offset (see initial power		18%		Core avg.
shapes in Appendix 14B.2.5.1 – Figure 1		15%		Hot channel, DNBR = 1.26
FΔH at DNBR LCO		1.745*	not relevant	
FΔH at DNBR = 1.26		1.919*		
Initial local power density		≤506 W/cm	not relevant	hot channel, uncoupled
Reactor cooling flow (total)	22241kg/s	code result	code result	Thermal design
Total core bypass flow	3.4%	5.5%	5.5%	hot RPV head bypass assumed in analysis = 7%
Coolant mixing for temp. distribution at core inlet		not relevant	not relevant	
Average reactor coolant temperature	311.25°C	311.25(+2.5°C)	311.25(+2.5°C)	For DNBR calc. core inlet temp. is 295°C
Pressuriser pressure	155 bar	155(-2.5)bar	155(+2.5)bar	
Pressuriser level	6.97m	not relevant	6.97(+0.55) m	
Feedwater/Main steam flow/SG	694.1 kg/s	not relevant	708 kg/s	
Feedwater temperature	230 °C	not relevant	231 °C	
Steam generator water level	16.2 m	not relevant	16.2 m	uncertainty not relevant
Steam generator water mass	87.6 Mg	not relevant	code result	(RCP [RCS] temp. and flow)
Pressure in steam generator	74.6 bar	not relevant	code result	(RCP [RCS] temp. and flow)

* In this study the FΔH is a sensitivity parameter used to get the LCO value. Considering this, this FΔH cannot be considered as enveloping value for fuel management studies.

APPENDIX 14B.2.5.1 - TABLE 1 (3/3)

Boundary Conditions (kinetics and reactivity)	Best estimate	Conservative assumption for DNBR calculation	Conservative assumption for overall plant behaviour	Remarks
Burn-up state: BOC/EOC		BOC	BOC	
Bank reactivity				
Partial trip		not relevant	not relevant	
Reactor trip		10600 pcm at HSB	4900 pcm	Rod drop characteristic from Appendix 14B.0.2. – Table 7. Rod worth from PANBOX calc.
Control rod drop time	2.5 s	3.5 s	3.5 s	
Moderator R-feed back		-9.6 pcm/°C [1)]	0.0	- 9.6 pcm/°C corresponds to 0.038 * 10^3 pcm/gcm^{-3}
Void reactivity		code result		
Fuel Reactivity feedback		- 3.6.pcm/°C [1)]	-4.1 pcm/°C	
Boron Reactivity coefficient/efficiency		not relevant	not relevant	
Boron concentration		not relevant	not relevant	
Decay power	ORIGEN/S	ORIGEN/S*fact.	ORIGEN/S*fact	fact. is time dependant (between 1.1 and 1.2)
Delayed neutron fraction ($\Sigma\beta$)		730.5 pcm	700 pcm	No influence on results

[1)] at time 0s

APPENDIX 14B.2.5.1 - TABLE 1 (3/3)

Boundary Conditions (failure assumptions)	Best estimate	Conservative assumption for DNBR calculation	Conservative assumption for overall plant behaviour	Remark
Emergency power mode		yes	yes	Initiating event
Single failure F1A or F1B		no	1 MSRIV fails to open	no impact on DNBR transient
Maintenance		not relevant	1 diesel unavailable	no impact
Consequential failure		no	no	
Control and limitation system assumption (event specific):				
Partial trip		no	no	
RCP [RCS] temperature control		not relevant	no	
RCP [RCS] pressure control		not relevant		
• PZR normal spray		not relevant	no	
• PZR heater		not relevant	yes	only diesel supplied rods
PZR level control (RCV [CVCS])		not relevant	no	no change of state till LOOP
Auxiliary spray via RCV [CVCS]		not relevant	no	
Turbine control		not relevant	no	no change of state till turbine trip
Steam dump control		not relevant	no	
MS relief control		not relevant	yes	
SG level control (ARE [MFWS] and AAD [SSS])		not relevant	no	no change of state till turbine trip/LOOP
Start-up/Shutdown system		not relevant	no	

APPENDIX 14B.2.5.1 - TABLE 2

Description of Variables plotted

Plot	Unit	Variable		Description of Variable
Page 1	(-)	MASS FLOW	=	Normalised net core mass flow rate
	(-)	P_BOC	=	Normalised fission and decay power,
Page 2	(-)	OFF_BOC	=	Axial offset core
	(-)	DNBR_BOC1	=	DNBR hot channel, best estimate F∆H, coupled PANBOX-COBRA calculation
	(-)	DNBR_BOC2	=	DNBR hot channel, F∆H increased, coupled PANBOX-COBRA calculation
	(-)	DNBR_BOC3	=	DNBR hot channel as DNBR2 + uncertainties considered in uncoupled COBRA calculation
Page 3	(-)	Core	=	Axial power distribution in the average channel (initial condition)
	(-)	Hot Channel	=	Axial power distribution in the hot channel (initial condition)

APPENDIX 14B.2.5.1 - TABLE 3

Emergency Power Mode

Sequence of Events regarding the overall plant behaviour

Event	Time (seconds)
Turbine trip Loss of off-site power initiated, RCP [RCS] trip	10
Reactor trip from RCP [RCS] speed < 91%,	11.3
1st maximum of RC-pressure - at hot leg (171 bar) - at pressuriser (170 bar)	18
Opening of the MS-relief isolation valves at SG pressure of 94.5 bar, in 3 loops In the fourth loop the MSRIV is assumed to stick in closed position.	25.4

APPENDIX 14B.2.5.1 - TABLE 4

Description of plotted Variables

Plot	Unit	Variable		Description of Variable
A two loop calculation was performed. The variables with the index 1 refer to the loop with closed MSR-train, index 2 denotes to the other loops.				
Page 1	(-)	PGBIST	=	Normalised generator power
	(-)	PBEZ	=	Normalised power (fission + decay + RCP [RCS] power)
	(-)	SQKCB	=	Normalised heat transfer cladding-coolant
	(-)	SQKDEB	=	Normalised total heat transfer in steam generators
Page 2	(-)	WKB(i)	=	Normalised RC flow rate loop(i)
	(-)	PUMPN(i)	=	Normalised RCP speed loop(i)
	(°C)	TK1(i)	=	Coolant temperature RPV outlet loop(i)
	(°C)	TK9(i)	=	Coolant temperature SG outlet loop(i)
Page 3	(bar)	PK	=	RC pressure RPV outlet (hot leg)
	(bar)	PDH	=	Pressuriser pressure
	(bar)	PKRCP(i)	=	RC pressure downstream of RCP (cold leg) loop(i)
	(m)	DHFIST	=	Real pressuriser water level
	(m)	DHFMESS	=	Measured pressuriser water level
	(m)	DHFSOL	=	Pressuriser water level setpoint
Page 4	(kg/s)	WDE(i)	=	Steam flow rate at outlet of SG(i)
	(kg/s)	WSP(i)	=	Main feedwater/AAD [SSS] flow rate of SG(i)
	(kg/s)	WNSP(i)	=	ASG [EFWS] flow rate of SG(i)
	(m)	DEFIST(i)	=	Real steam generator water level of SG(i)
	(m)	DEFS(i)	=	wide range measured SG water level of SG(i)
Page 5	(bar)	PDE(i)	=	Steam pressure at top of SG(i)
	(bar)	PFDSOL(i)	=	Setpoint of MS relief valves of SG(i)
	(bar)	PDAF1(i)	=	Setpoint of “MS pressure drop rate > MAX“ of SG(i)
	(Mg)	RMWDE(i)	=	Liquid water and steam mass in SG(i)
	(Mg)	RMDDE(i)	=	Steam mass in SG(i)

APPENDIX 14B.2.5.1 - FIGURE 1 (1/3)

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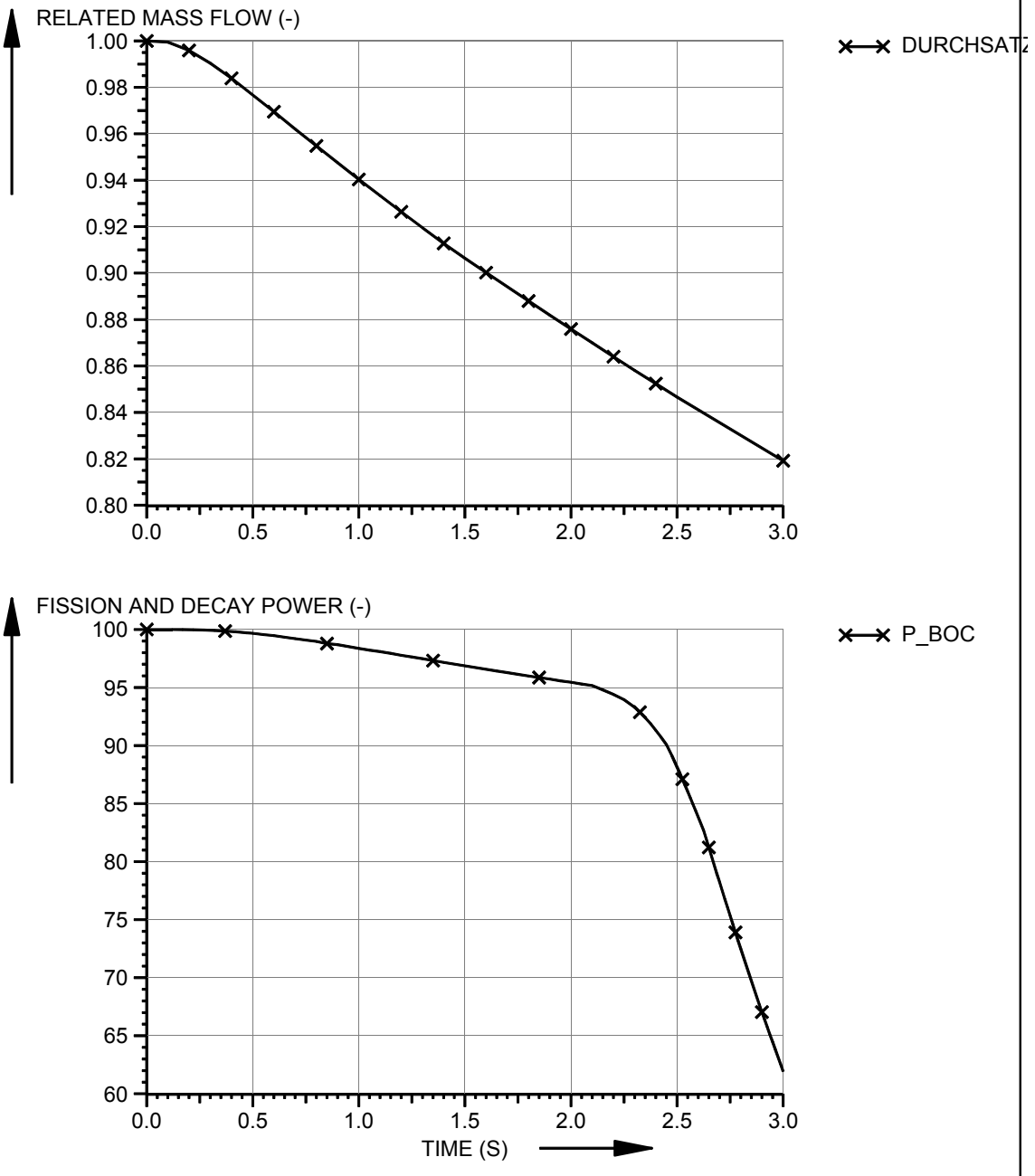
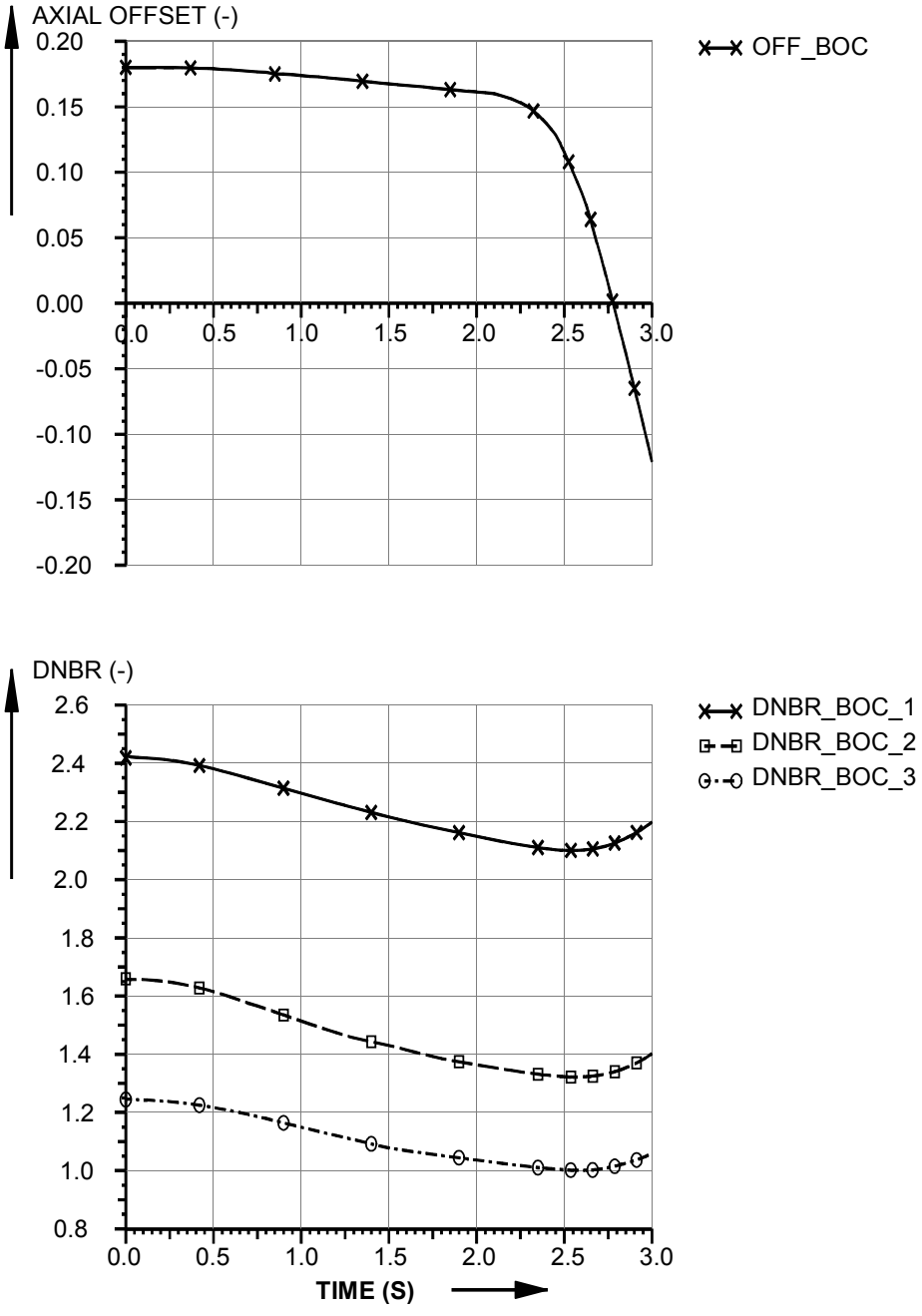


EPR, BASIC DESIGN
BOC CONDITION, EMERGENCY POWER MODE (SHORT TERM REG. MIN. DNBR)
CODE PANBOX/COBRA SIEMENS AG / UB KWU DEP NBTT

APPENDIX 14B.2.5.1 - FIGURE 1 (2/3)



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BOC CONDITION, EMERGENCY POWER MODE (SHORT TERM REG. MIN. DNBR)
CODE PANBOX/COBRA SIEMENS AG / UB KWU DEP NBTT

APPENDIX 14B.2.5.1 - FIGURE 1 (3/3)

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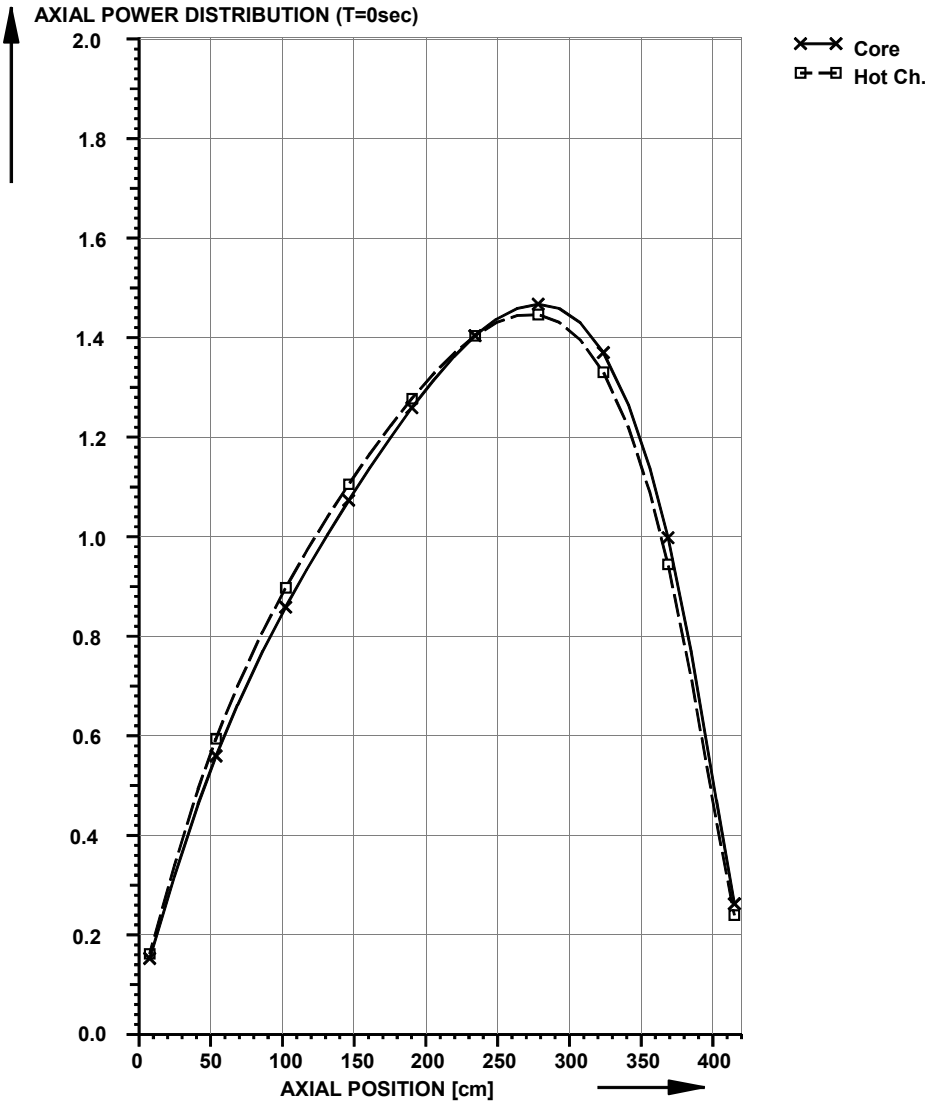
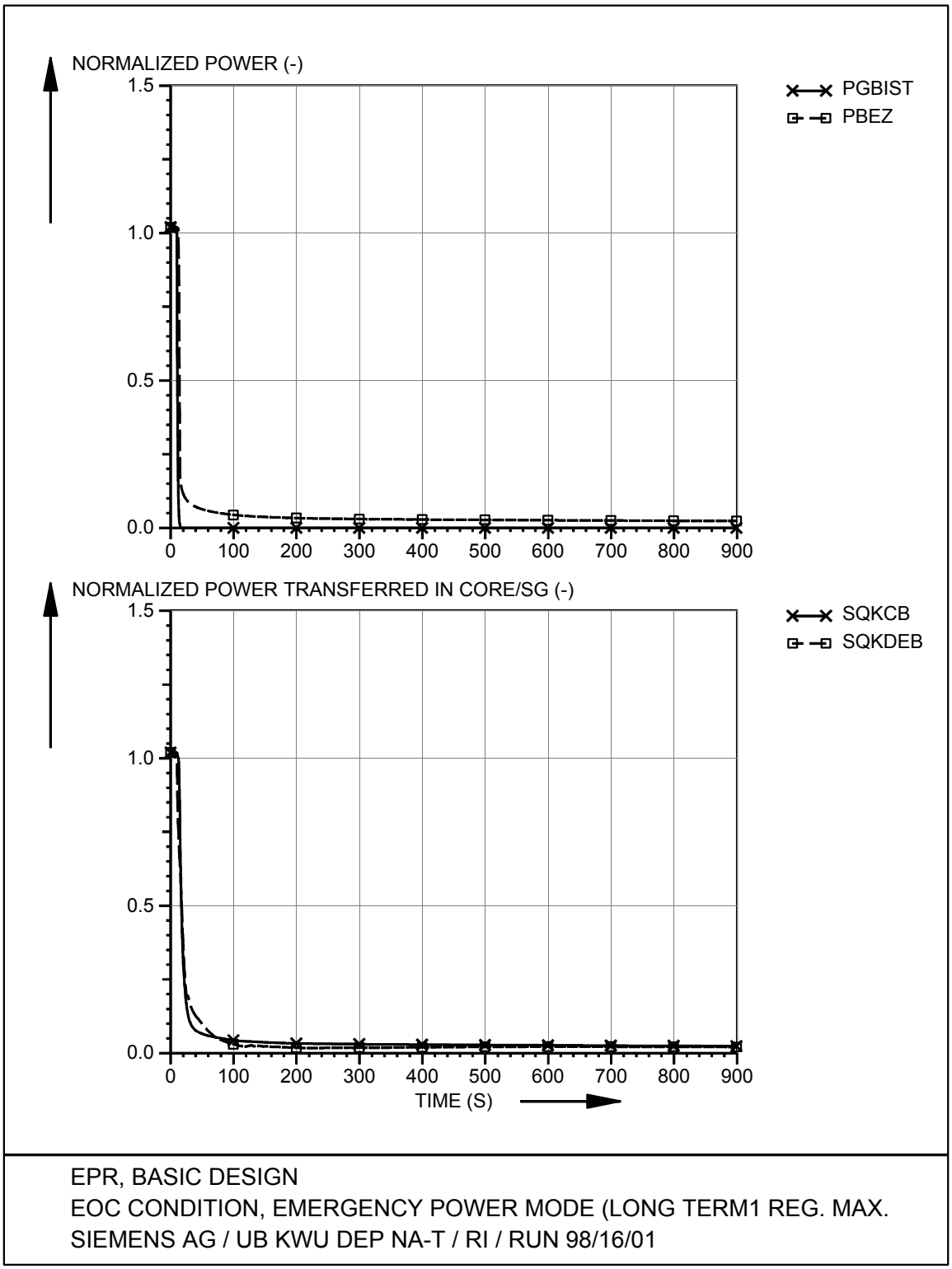


EPR, BASIC DESIGN
BOC CONDITION, EMERGENCY POWER MODE (SHORT TERM REG. MIN. DNBR)
CODE PANBOX/COBRA SIEMENS AG / UB KWU DEP NBTT

APPENDIX 14B.2.5.1 - FIGURE 2 (1/5)

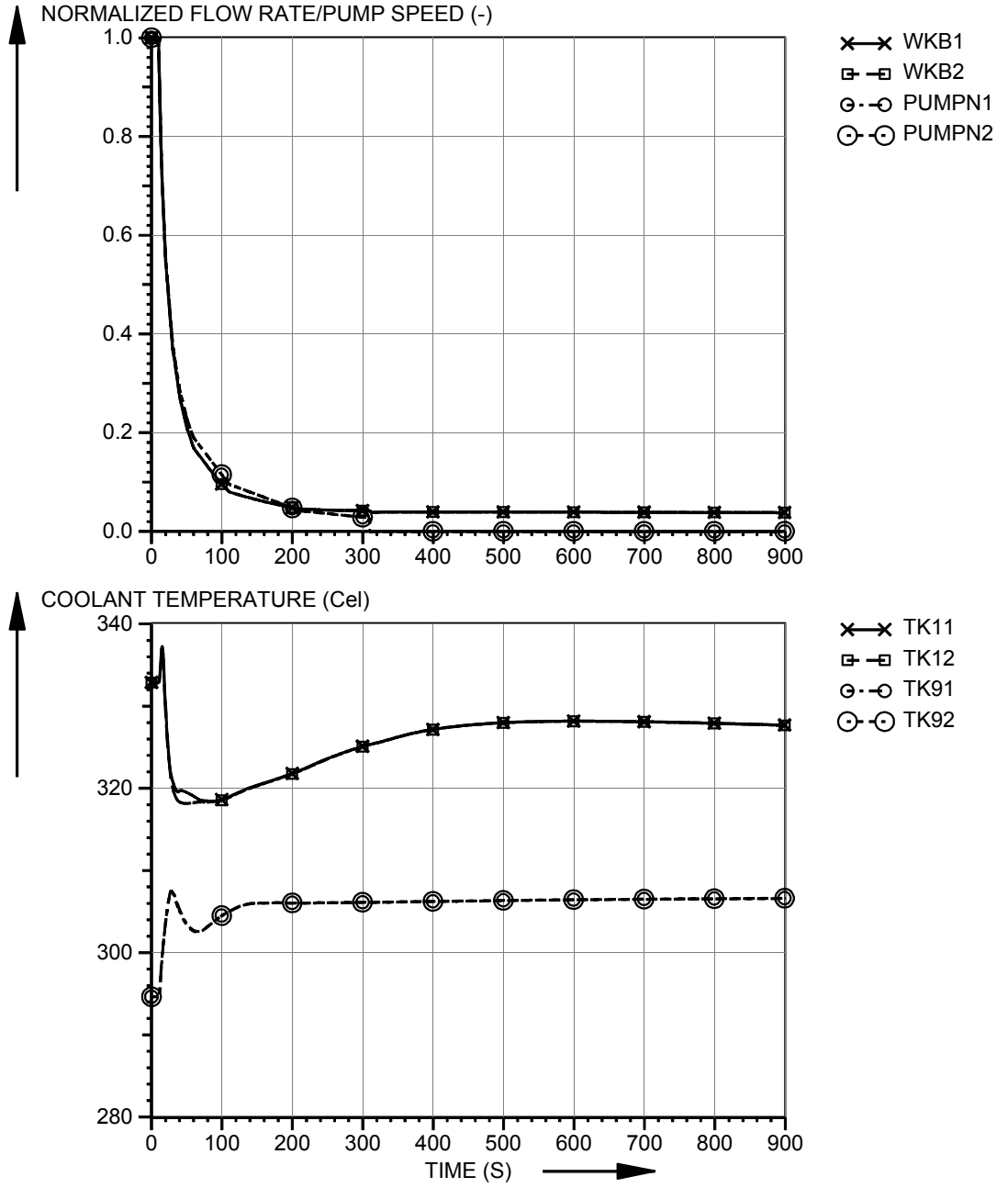
12.11.98 14:22:12 TAPE4_LAUF_101898_171752_01 Page 1



EPR, BASIC DESIGN
EOC CONDITION, EMERGENCY POWER MODE (LONG TERM1 REG. MAX.
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/16/01

APPENDIX 14B.2.5.1 - FIGURE 2 (2/5)

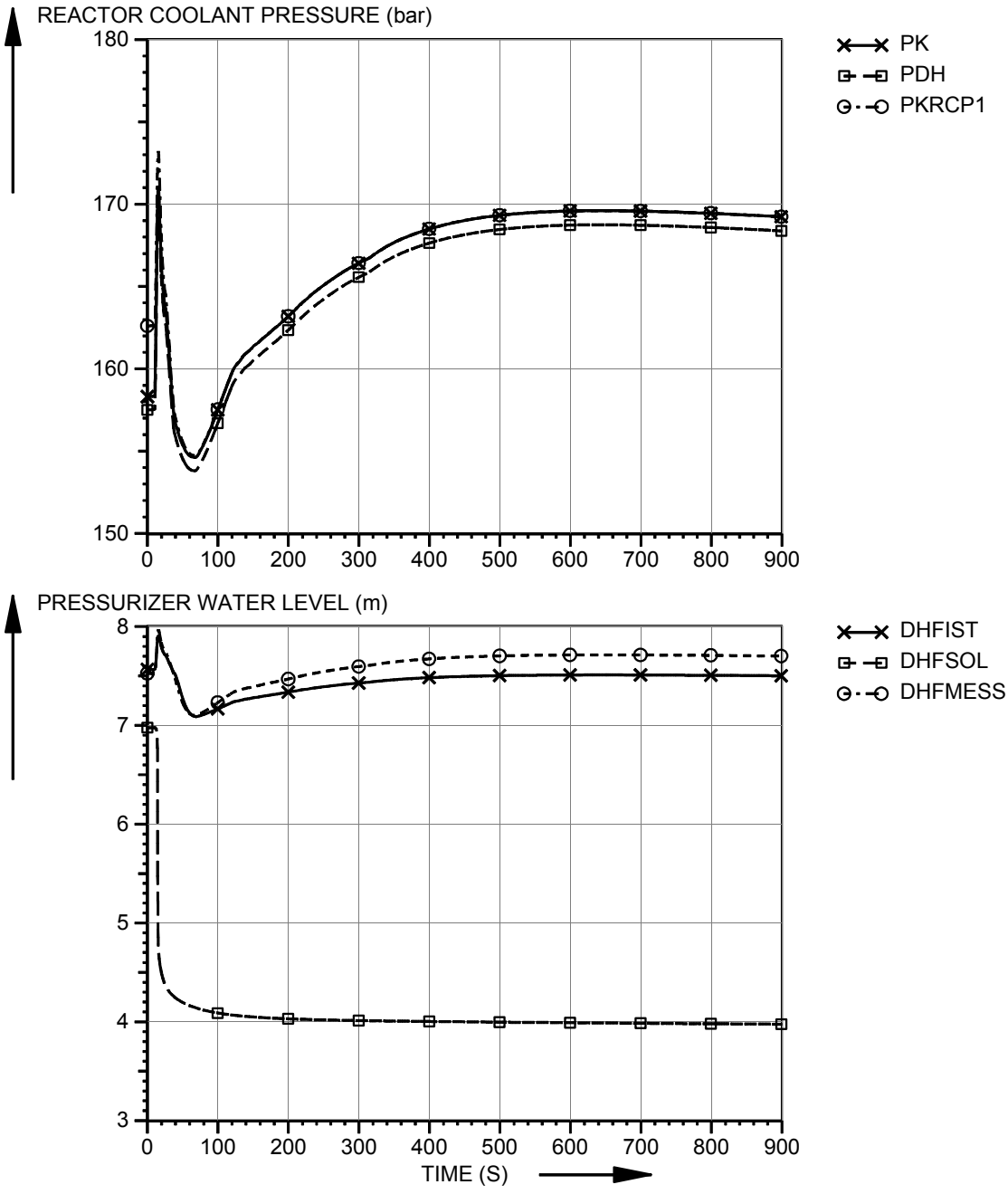
12.11.98 14:22:22 TAPE4_LAUF_101898_171752_01 Page 2



EPR, BASIC DESIGN
EOC CONDITION, EMERGENCY POWER MODE (LONG TERM1 REG. MAX.
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/16/01

APPENDIX 14B.2.5.1 - FIGURE 2 (3/5)

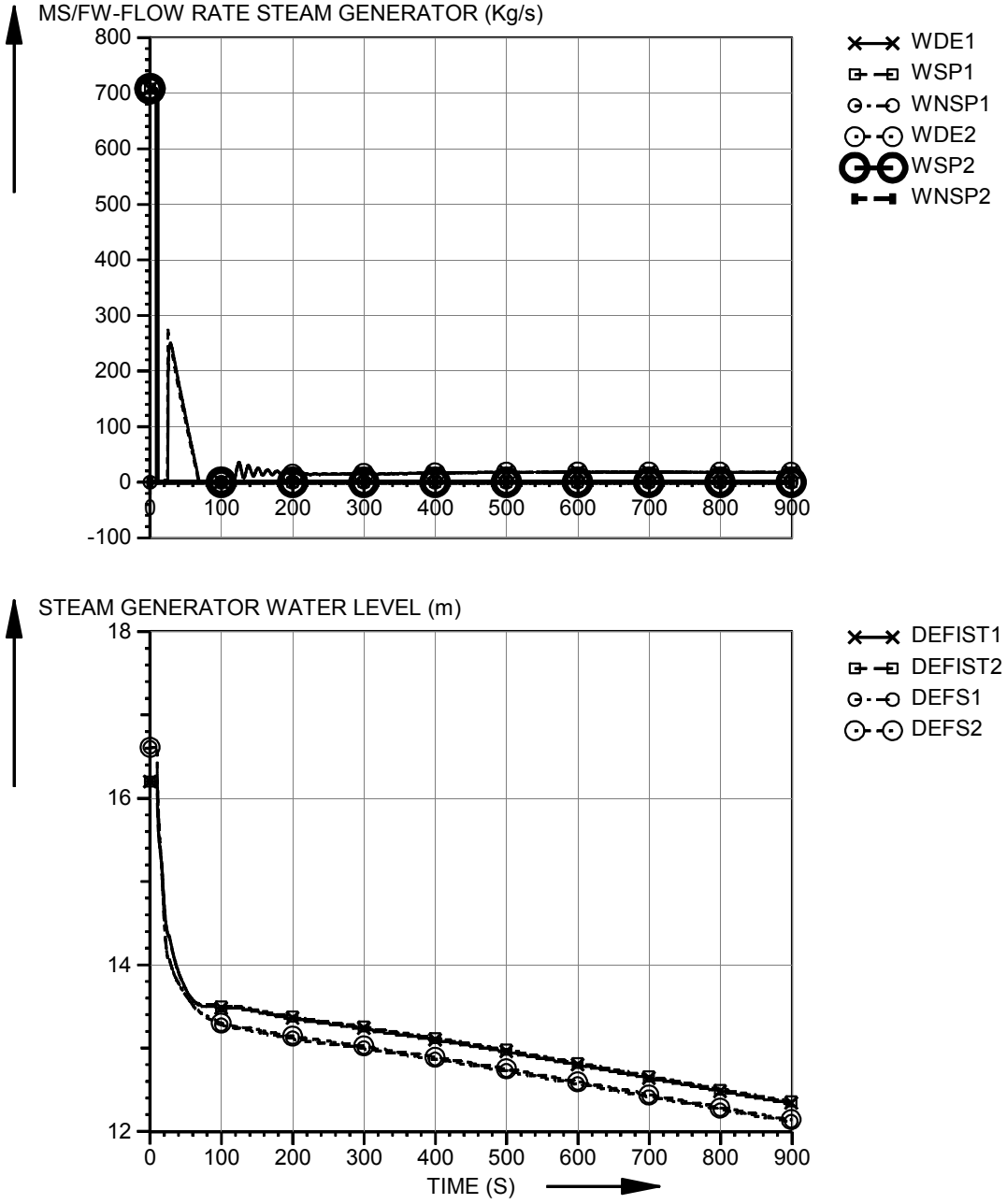
12.11.98 14:22:26 TAPE4_LAUF_101898_171752_01 Page 3



EPR, BASIC DESIGN
EOC CONDITION, EMERGENCY POWER MODE (LONG TERM1 REG. MAX.
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/16/01

APPENDIX 14B.2.5.1 - FIGURE 2 (4/5)

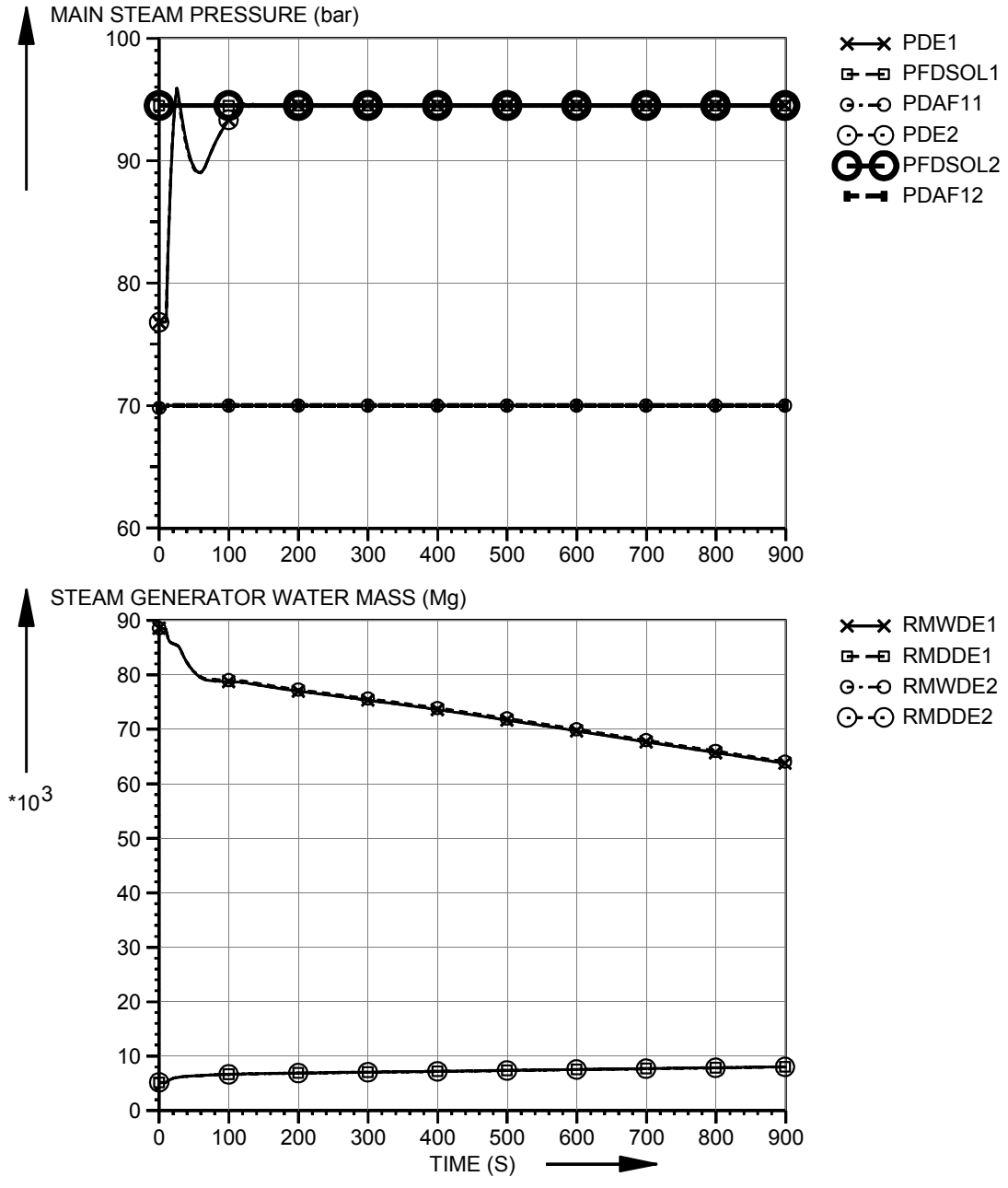
12.11.98 14:22:30 TAPE4_LAUF_101898_171752_01 Page 4



EPR, BASIC DESIGN
EOC CONDITION, EMERGENCY POWER MODE (LONG TERM1 REG. MAX.
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/16/01

APPENDIX 14B.2.5.1 - FIGURE 2 (5/5)

12.11.98 14:22:35 TAPE4_LAUF_101898_171752_01 Page 5



EPR, BASIC DESIGN
EOC CONDITION, EMERGENCY POWER MODE (LONG TERM1 REG. MAX.
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/16/01

2.5.2. Loss of Condenser Vacuum (PCC-2)

The event is analysed in reactor state A only.

2.5.2.1. Identification of Causes and Accident Description

From Initiating Event to Controlled State

A loss of condenser vacuum precludes the use of the MS bypass. The situation is very similar to the other transients involving reactor trip-turbine trip considered in the safety case. After the loss of condenser vacuum turbine trip is actuated: the reactor is tripped on high primary or secondary pressure.

Since the purpose of this transient analysis is to obtain thermal-hydraulic parameters such as

- minimum SG water inventory
- maximum integral MS flow to the atmosphere

for the calculation of activity release and off-site dose, failure of one MSRCV to close after response of the VDA [MSRT] (subsequent to reactor trip) is assumed.

The controlled state is reached after automatic closure of the MSRIVs on signal “MS pressure < 40 bar”.

From Controlled State to the Safe Shutdown State

Boration and plant cooldown are performed by F1 systems as follows:

- Cooldown at 25°C/h down to hot leg temperature of < 180°C is performed via the MS relief trains, with subsequent reduction of primary pressure to < 30 bar (RHR operating conditions).
- Boration during cooldown is carried out with one RBS [EBS] pump

Controlled State and Safe Shutdown State

It must be demonstrated that the controlled state can be reached using only F1A functions and the safe shutdown state can be reached using only F1B and F1A functions, without exceeding the following safety and decoupling criteria:

- Safety Criteria:
  - Radiological limits for normal operation.
- Decoupling Criterion:
  - Avoidance of DNB (DNBR limit= 1.0 in short term emergency power mode)

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

- Reactor trip:

  Reactor trip is initiated from one of the following signals:

  - RCP [RCS] pressure >166.5 bar
  - MS pressure > 93 bar

- Systems:

  - 4 MS relief trains
  - 4 ASG [EFWS] trains

- Signals:

  - Actuation of MS relief trains at a SG pressure of 93.5 bar
  - Closing of MSIVs on ”MS pressure drop >MAX1”
  - Actuation of ASG [EFWS] on “SG water level < 8 m”
  - Actuation of MSRIVs on “SG pressure < 40 bar”
  - Actuation of MHSI on an RIS [SIS] signal (this action is not essential for the mitigation of the transient)

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

- 4 MS relief trains for cooldown to RHR connection conditions, (manual action).
- 2 RBS [EBS] trains for boration during cooldown
- 4 ASG [EFWS] trains for feedwater supply to the SGs (automatic or manual action)
- 1 pressuriser safety valve to reduce the primary pressure to 30 bar
- MHSI (not essential).

2.5.2.2. Methods and Assumptions

a) Methods of analysis

Calculations are performed using the following computer code: NLOOP.

b) Important phenomena to be modelled and qualification of the models used in NLOOP

In this event, two families of transients (loss of secondary heat sink and secondary overcooling) are combined in a single event sequence.

Phenomena

Primary Side

Overcooling during the period of inadvertent opening of the MS relief train (VDA [MSRT]), involving emptying of the pressuriser. Increasing primary temperatures and pressures after isolation of the VDA [MSRT].

Secondary side

Initial increase in MS pressure due to closure of turbine/GCT [MSB] valves. Fast opening of the MS relief trains and their re-closure in the 3 non-faulted loops to maintain constant MS pressure at zero load. In the faulted loop failure to close of the MS relief train, resulting in excessive loss of steam and overcooling until the MS relief train is isolated when the MS pressure falls below 40 bar. Emptying of the faulted SG. Heat transfer at low water level in one SG.

Qualification of the NLOOP models of primary and secondary side phenomena

Some transients on PWR power plants are available in which most of the above mentioned phenomena occur. These events were modelled with NLOOP, and good agreement with the observed plant behaviour was obtained. In addition, a relevant integral test on a test facility was available for validation of NLOOP. Simulation of this test likewise showed good agreement with test results for secondary and primary side phenomena. The plant transients and tests are identified below:

Event	Plant/test facility	Phenomena
Spurious opening of GCT [MSB]	KKP2	Primary and secondary side overcooling followed by heat up
Secondary pressure drop transient	KWO	Primary and secondary side overcooling
LOOP followed by total loss of feedwater supply	PKL3	Circulation depending on the water mass in SG. SG heat transfer at low SG secondary side inventory

c) Choice of Single Failure and Preventative Maintenance

Failure to close of one MSRCV after opening of the MSRIV is assumed as a single failure, in order to maximise the steam outflow to the atmosphere. Additionally, in the phase from the controlled state to the safe shutdown state, failure of one RBS [EBS] pump is assumed in order to delay the cooldown phase and thus also to maximise the steam release to atmosphere. Preventative maintenance is not relevant for the scenario.

The analysis of the transition from the controlled state to the safe shutdown state performed for the 'Loss of Condenser Vacuum' case, is also applicable to other events such as the 'Reactor Coolant Pump Locked Rotor' event.

d) Initial and boundary conditions

Initial conditions are selected to be conservative for maximising the total MS release to the atmosphere and minimising the water mass content in the SGs.

The main initial and boundary conditions such as reactor power, core inlet temperature, primary pressure are chosen to allow for the dead bands, uncertainties, and typical maximum deviations of control systems. The coolant flow assumed corresponds to the thermal design flow.

The MS bypass system is not considered because it is not F1 qualified.

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

The major initial and boundary conditions are listed in Appendix 14B.2.5.2 - Table 1.

2.5.2.3. Results and Conclusions

Appendix 14B.2.5.2 - Table 2 gives the sequence of events for the automatic phase of the transient (i.e. from the initiating event to the controlled state) and operator action phase (i.e. from the controlled state to the safe shutdown state)

The time evolution of parameters is presented in the table, distinguishing between the automatic phase (up to 1800 seconds i.e. from the initiating event up to the controlled state, and the cooldown phase to RHR conditions i.e. from the controlled state to safe shutdown state. The safety functional classification of actions is stated.

Parameter plots are given in Appendix 14B.2.5.2 - Figure 1.

The plotted parameters are described in Appendix 14B.2.5.2 - Table 3.

Results up to attainment of Controlled State:

Reactor trip occurs on RCP [RCS] pressure > 168 bar at 16 seconds. The uncontrolled MS release due to the non-closure of the MSRCV is terminated by the automatic closure of the associated MSRIV.

During the transient the first pressuriser safety valve opens for a short period at the time of reactor trip and before the opening of the MS relief valves at a SG pressure of 92 bar. The primary pressure decreases to 110 bar, at which an RIS [SIS] signal is generated i.e. the MHSI and partial cooldown are actuated.

The controlled state is safely reached with the aid of the following system actions:

- reactor trip and closure of MSRIVs to achieve subcritical conditions
- 3 VDA [MSRTs] and 4 ASG [EFWS] trains to ensure heat removal

Results for period from the Controlled State to Safe Shutdown State

The transition from the controlled state to safe shutdown state confirms that relevant decoupling criteria are not violated:

- subcriticality is ensured by boration using one RBS [EBS] pump
- Heat removal is ensured by 3 VDA [MSRTs] and 4 ASG [EFWS] trains
- Activity release is controlled, since none of the barriers is impaired.

- Reduction of primary pressure to 30 bar is ensured by manual shut-off of the MHSI/RBS [EBS] pumps and a brief opening of one pressuriser safety valve. If the RBS [EBS] pump is shut off at approximately 6100 seconds when the boron concentration required for the safe shutdown state is reached, opening of the pressuriser safety valve to reduce the primary pressure is not required.

The minimum reactor inlet temperature with failure of the MSRCV is 250°C, resulting in no return to criticality.

The minimum SG water mass and maximum integral MS flow via MS relief valves are listed in the following:

Mass content

SG	Time	min. Mass content
	s	t
SG 1(faulted)	1040	15
SG 2-4	960	28

Integral MS flow

SG	Time s	Integral MS flow via MS relief per loop (t)
SG 1(faulted)	16000	249
SG 2-4	16000	295
Sum of all SGs	16000	1134

Note that an assumption of maintenance on the EFW pump supplying the faulted SG, i.e. EFW pump to SG with non-closing MSRCV, results in zero mass content in the faulted SG.

APPENDIX 14B.2.5.2 - TABLE 1 (1/3)

Event:	Loss of Condenser Vacuum (till cooldown to RHR connection)
Event Category (PCC/RRC):	PCC-2
Family of Events:	Decrease in secondary side heat removal
Safety Criteria:	Radiological limits for normal operation
Decoupling Criteria:	No DNB (covered by emergency power mode, short term)
Purpose of Analysis	Analysis for activity release
Safe shutdown conditions:	
controlled state	Hot shutdown with removal via MS relief valves
safe shutdown:	Connection to RHR System

APPENDIX 14B.2.5.2 - TABLE 1 (2/3)

Initial Conditions	Best estimate	Conservative value adopted for DNBR	Conservative value adopted for activity release	Remarks
Reactor/turbine power	100/100 %		102/code res.	
Thermal reactor power	4900 MW		4998 MW	
Reactor cooling pumps power			30.59 MW	
Thermal steam generator power(per SG)			1256.9 MW	
Initial insertion of control rods			not relevant	
Initial power shape			not relevant	
Fxy				
Axial offset				
Initial DNBR			not relevant	
Initial local power density			not relevant	
Reactor cooling flow (total)			code result	
Total core bypass flow	5.5 %		5.5 %	
Coolant mixing for temp. distribution at core inlet			not relevant	
Average reactor coolant temperature	311.25°C		311.25(+2.5) °C	
Pressuriser pressure	155 bar		155(+2.5)bar	
Pressuriser level	6.97m		6.97(+0.55) m	
Feedwater/Main steam flow/SG	694.1kg/s		708 kg/s	
Feedwater temperature	230 °C		231 °C	
Steam generator water level	16.2 m		16.2 (-0.3) m	
Steam generator water mass	87.6 Mg		code result	(RCP [RCS] temp. and flow)
Pressure in steam generator	74.6 bar		code result	(RCP [RCS] temp. and flow)

APPENDIX 14B.2.5.2 - TABLE 1 (3/3)

Boundary Conditions (kinetics and reactivity)	Best estimate	Conservative value adopted for DNBR	Conservative value adopted for activity release	Remarks
Burn-up state BOC/EOC			EOC	
Bank reactivity				
Initial bank reactivity acc. to rod pos.			not relevant	
Partial trip			not relevant	
Reactor trip			5900 pcm	
Control rod dropping time	2.5 s		3.5 s	
Moderator R-feed back			-60 pcm/°C	
Void reactivity			not relevant	
Fuel R-feed back			-4.1 pcm/°C	
Boron R-coefficient/efficiency			min. efficiency	see Appendix 2.5.2 - Figure 2
Boron concentration			10 ppm	
Decay power	ORIGEN/S		ORIGEN/S*fact.	fact is time dependant i.e. between 1.1 and 1.2
Delayed neutrons ($\Sigma\beta$)			70 e-4	

APPENDIX 14B.2.5.2 - TABLE 1 (3/3)

Boundary Conditions (failure assumptions)	Best estimate	Conservative value adopted for DNBR	Conservative value adopted for activity release	Remarks
Emergency power mode			no	
Single failure F1A or F1B			1 MSRCV fails to close	for short term
			1 RBS [EBS] pump	for long term
Maintenance			no	
Consequential failure			no	
Considered control and limitation systems (event specific):				
Partial trip			no	
RCP [RCS] temperature control			no	
RCS pressure control				
Pressuriser normal spray			no	
Pressuriser heater			yes	
Pressuriser level control (RCV [CVCS])			no	
Auxiliary spray via RCV [CVCS]			no	
Turbine control			no	no change of state till turbine trip
Steam dump control			no	
MS relief control			yes	
SG level control (ARE [MFWS] and AAD [SSS])			no	no change of state till turbine trip
Start-up/Shutdown system			no	

APPENDIX 14B.2.5.2 - TABLE 2

Loss of Condenser Vacuum - Sequence of Events

Event	Time (s)
Automatic phase (from initiator to controlled state, all actions F1A)	
• Loss of condenser vacuum with consequential turbine trip	10
Reactor trip from coolant pressure (>168 bar)	16.3
Response of pressuriser safety valve, one short opening	18
• Opening of the MS relief isolation valves at SG pressure of 91.5 bar, assumption of the single failure in one MS relief control valve to close in the regulation mode.	19
• SG1 water level < 7.74 m, start of EFW pump	637
• Pressuriser is empty	695
• SG2 to 4 water level < 7.74 m, start of EFW pumps	707
• Primary pressure <113 bar, ECC signal, start of : MHSI/LHSI pump Partial cooldown with 100K/h	757
• Start of MHSI injection, prim. pressure < 80 bar	920
• SG min absolute pressure of < 50 bar is reached, isolation of MSRIV, full load FW valves are already isolated from RT	961
• SG1 min absolute pressure of < 38.5 bar is reached, isolation of	1017
• Primary pressure has reached 80 bar, no further injection of MHSI MS relief isolation valve minimal SG mass in the SG1 is about 15 t	appr. 1110
Operator action phase to RHR cold shutdown (from controlled state to safe shutdown)	
• Switch on of 1 RBS [EBS] pump	1830
• Initiating of cooldown in the non affected SGs with 25K/h (F1B) (after termination of partial cooldown)	1950
• RCP [RCS] temperature <180 °C ,cooldown finished	15300
• Switch off MHSI pumps (F1B)	
• Switch off RBS [EBS] pump (F1B)	
• Opening of one pressuriser safety valve to reduce the pressure to	
• about 30 bar to reach RHR cooldown conditions i. e. RCP [RCS] pressure about 30 bar Hot leg temperature < = 180°C	15500
• Water mass content in affected SG	86 t
Water mass content in non affected SGs	71 t

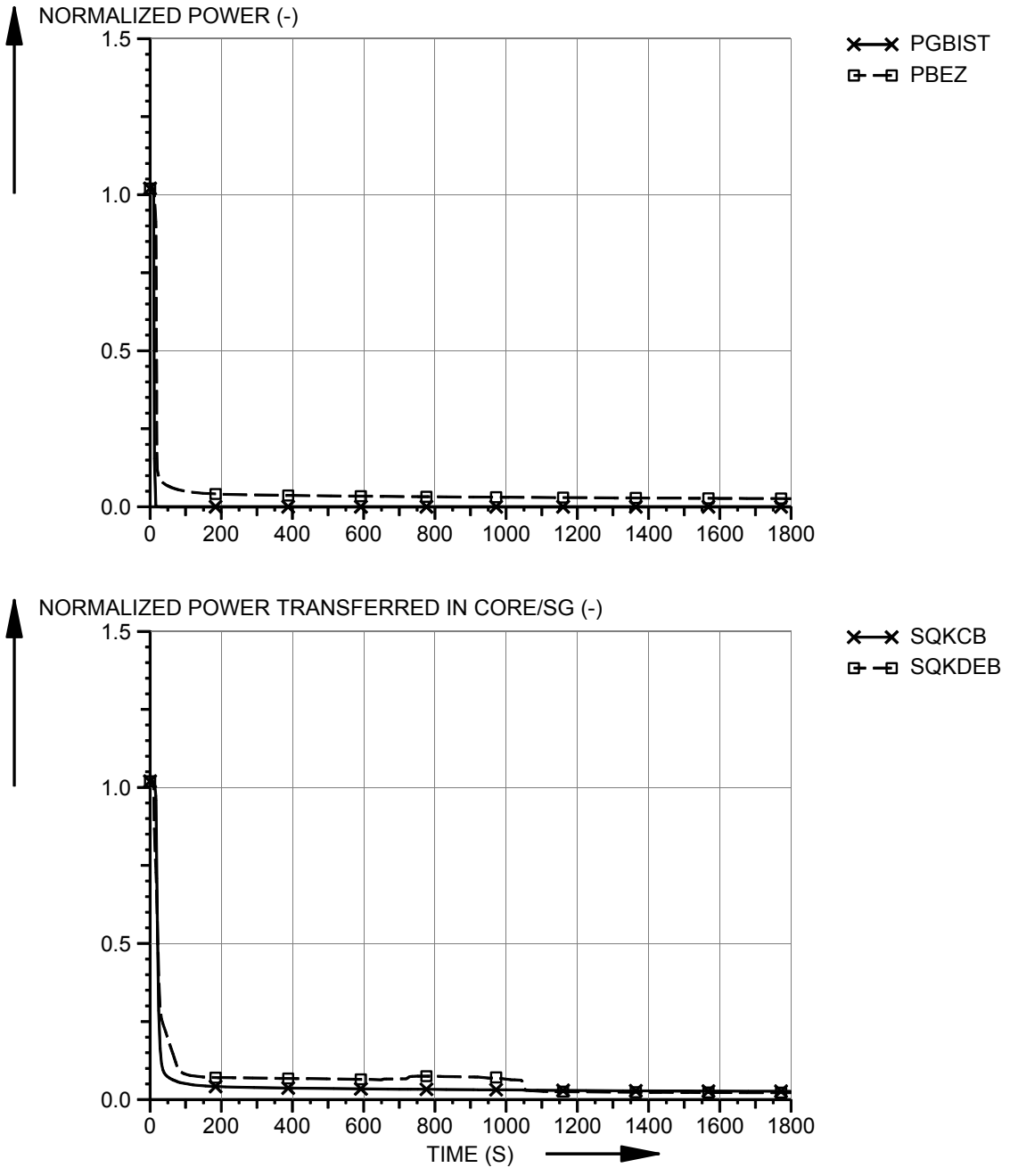
APPENDIX 14B.2.5.2 TABLE 3

Description of Plotted Variables

Plot	Unit	Variable		Description of Variable
Page 1	(-)	PGBIST	=	Normalised generator power
	(-)	PBEZ	=	Normalised power (fission + decay + RCP-power)
	(-)	SQKCB	=	Normalised heat transfer cladding-coolant
	(-)	SQKDEB	=	Normalised total heat transfer in steam generators
Page 2	(-)	WKB(i)	=	Normalised RC flow rate loop(i)
	(-)	PUMPN(i)	=	Normalised RCP speed loop(i)
	(°C)	TK1(i)	=	Coolant temperature RPV outlet loop(i)
	(°C)	TK9(i)	=	Coolant temperature SG outlet loop(i)
Page 3	(bar)	PK	=	RC pressure RPV outlet (hot leg)
	(bar)	PDH	=	Pressuriser pressure
	(bar)	PKRCP(i)	=	RC pressure downstream of RCP (cold leg) loop(i)
	(m)	DHFIST	=	Real pressuriser water level
	(m)	DHFSOL	=	Pressuriser water level setpoint
Page 4	(kg/s)	WDE(i)	=	Steam flow rate at outlet of SG(i)
	(kg/s)	WSP(i)	=	Main feedwater/SSS flow rate of SG(i)
	(kg/s)	WNSP(i)	=	ASG [EFWS] flow rate of SG(i)
	(m)	DEFIST(i)	=	Real steam generator water level of SG(i)
	(m)	DEFS(i)	=	wide range measured SG water level of SG(i)
Page 5	(bar)	PDE(i)	=	Steam pressure at top of SG(i)
	(bar)	PFDSOL(i)	=	Setpoint of MS relief valves of SG(i)
	(bar)	PDAF1(i)	=	Setpoint of “MS pressure drop rate > MAX“ of SG(i)
	(Mg)	RMWDE(i)	=	Liquid water and steam mass in SG(i)
	(Mg)	RMDDE(i)	=	Steam mass in SG(i)
Page 6	(kg/s)	WFDSIV(i)	=	Flow rate MS safety valve of SG(i)
	(kg/s)	WABRV(i)	=	Flow rate MS relief valve of SG(i)
	(kg)	SWVENT(i)	=	Integral flow rate through MS safety and relief valves of SG(i)
Page 7	(pcm)	RHOK1	=	Coolant temperature reactivity per loop
	(pcm)	RHOU1	=	Fuel reactivity per loop
	(ppm)	CBORM(i)	=	Boron concentration in core section (i)
Page 8	(kg/s)	SWSE	=	Total injection flow rate MHSI-pumps
	(kg/s)	WKBA	=	Injection flow rate RCV [CVCS]
	(kg/s)	WES	=	RBS [EBS] flow rate
	(m3)	VVDAMF	=	Steam volume within primary coolant circuit
	(m3)	VVDDEC		Steam volume within RPV head

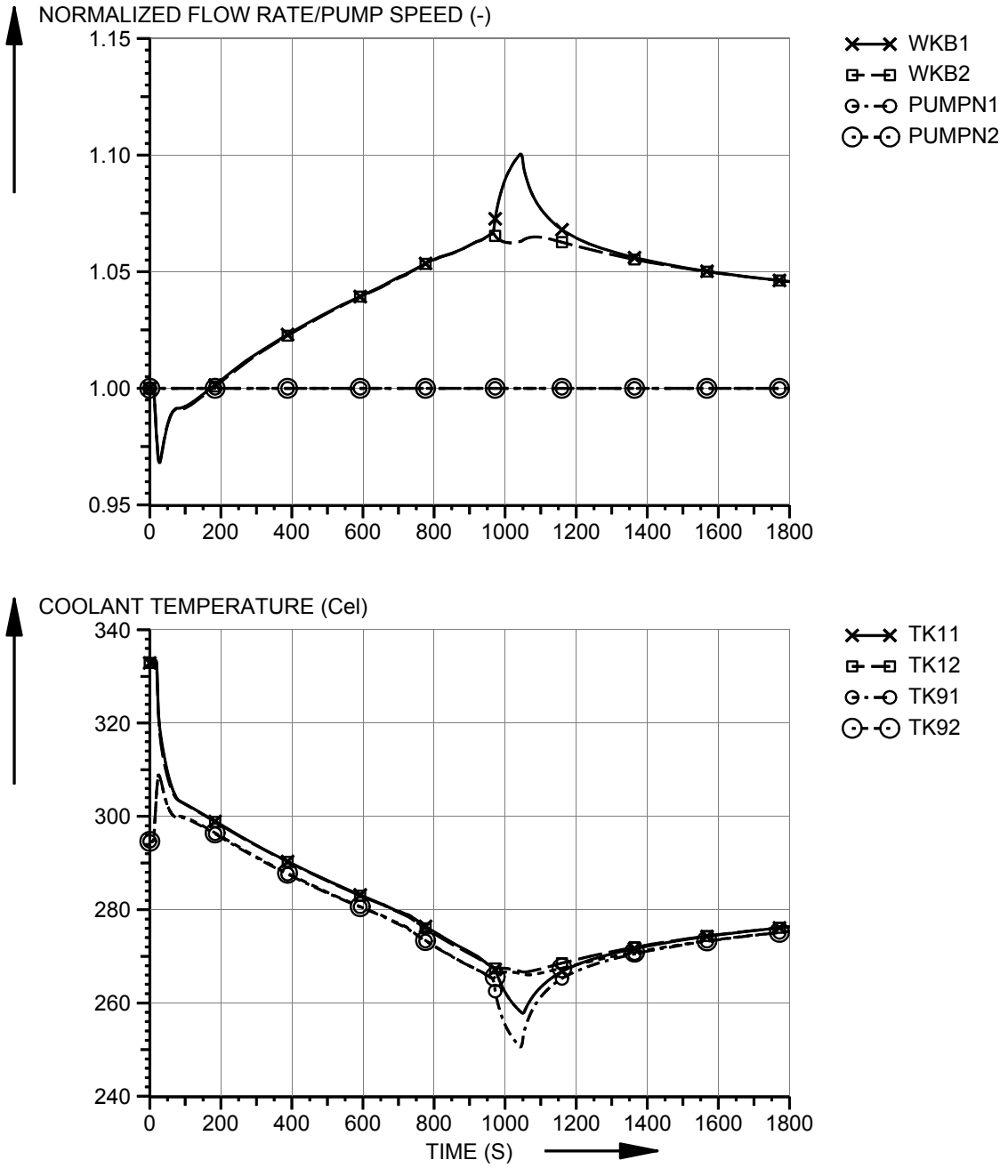
The variables with index 1 refer to the affected loop, index 2 denotes the other loops.

APPENDIX 14B.2.5.2 - FIGURE 1 (1/16)



EPR, BASIC DESIGN ,CODE NLOOP
EOC CONDITION, LOSS OF CONDENSER VACUUM
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (2/16)



EPR, BASIC DESIGN ,CODE NLOOP
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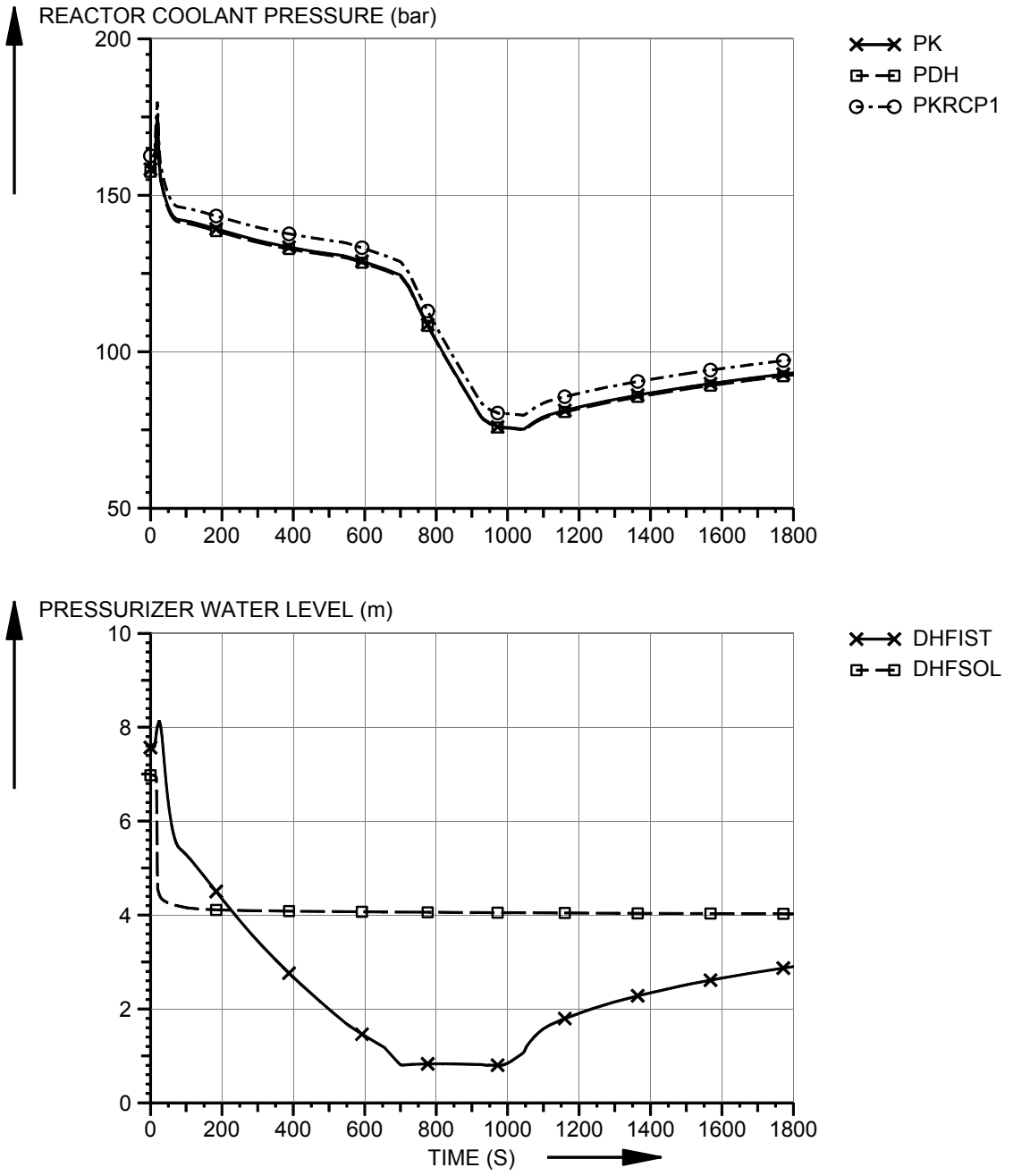
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (3/16)



EPR, BASIC DESIGN ,CODE NLOOP
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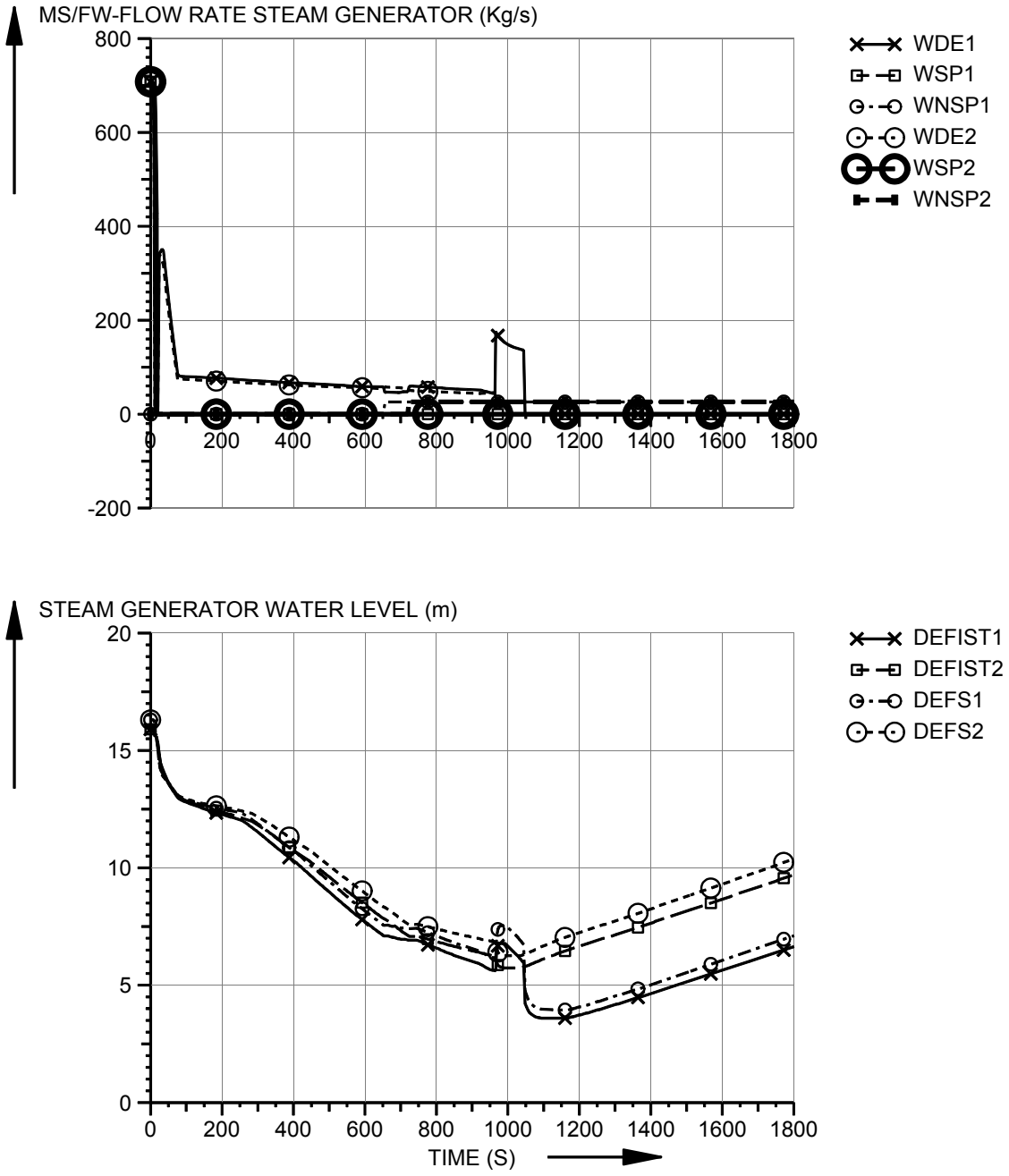
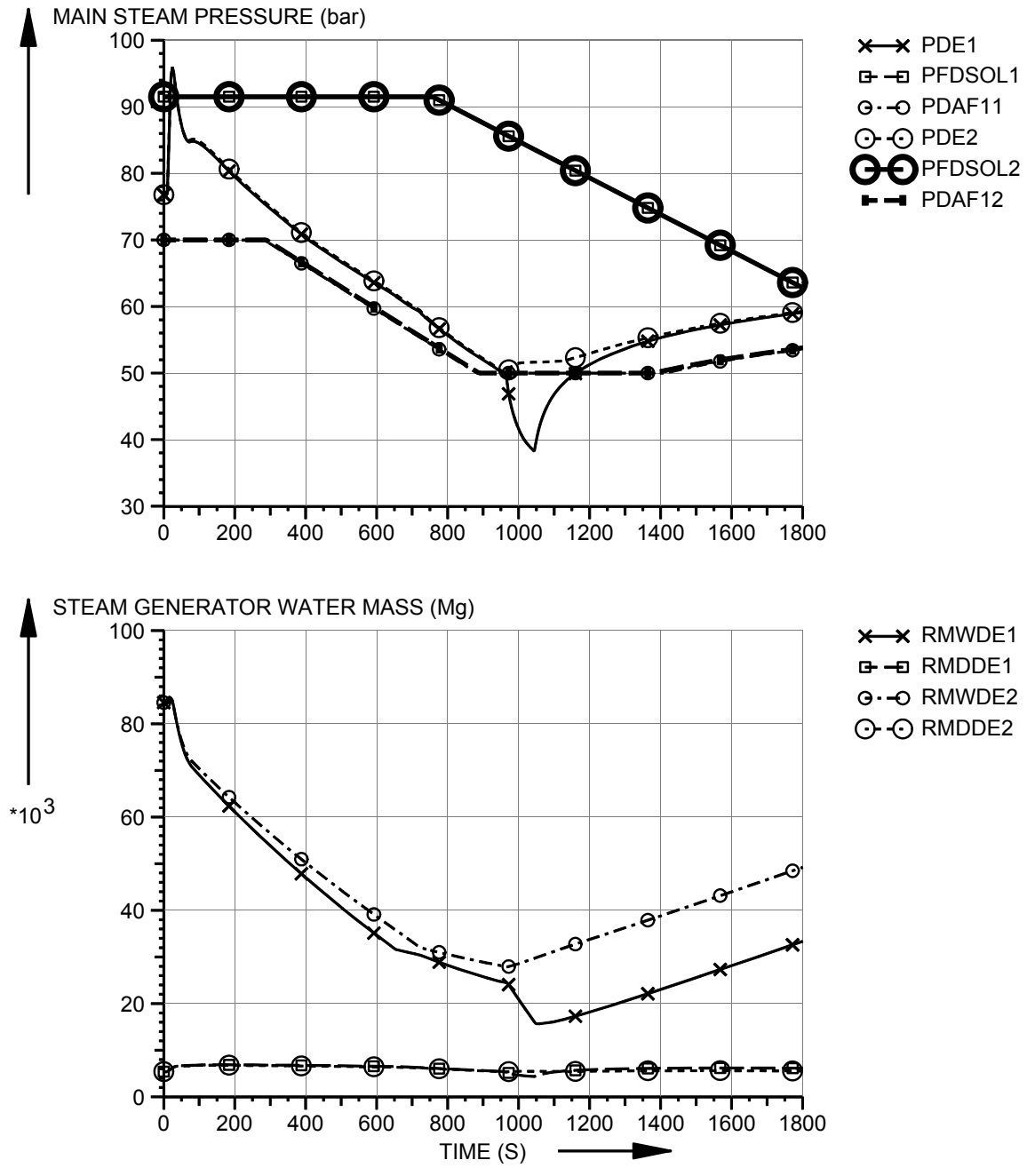
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (4/16)



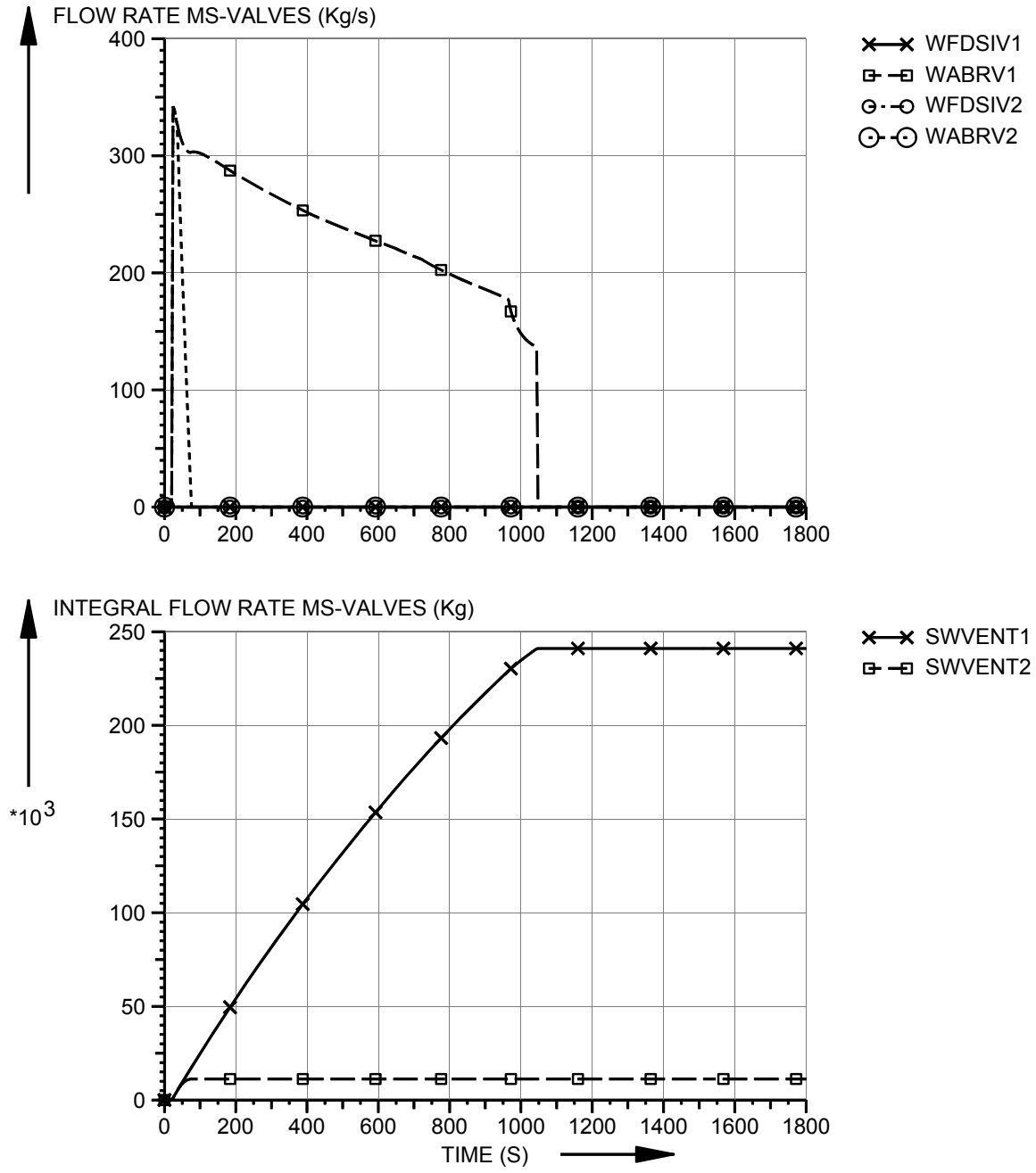
EPR, BASIC DESIGN ,CODE NLOOP
EOC CONDITION, LOSS OF CONDENSER VACUUM
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (5/16)



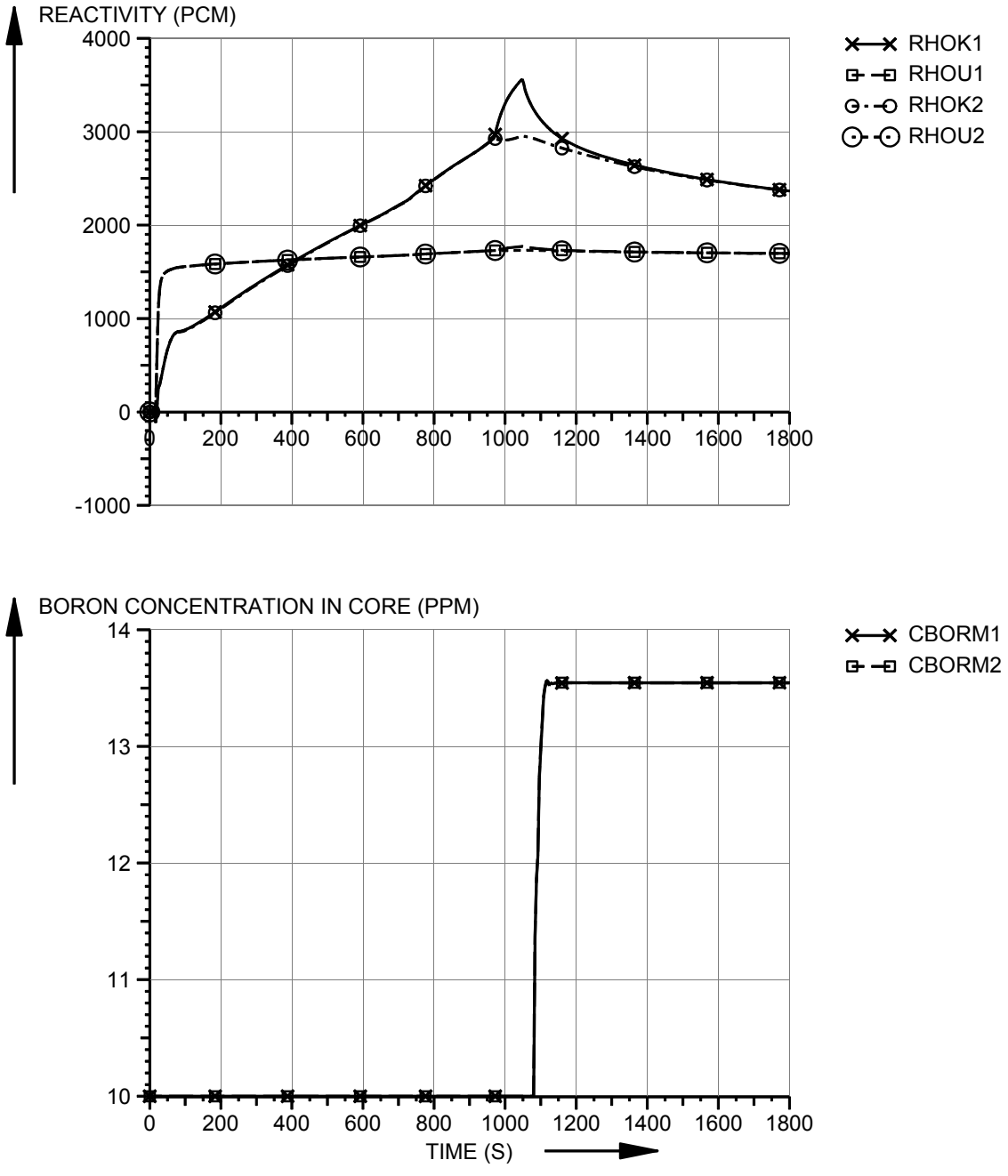
EPR, BASIC DESIGN ,CODE NLOOP
EOC CONDITION, LOSS OF CONDENSER VACUUM
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (6/16)



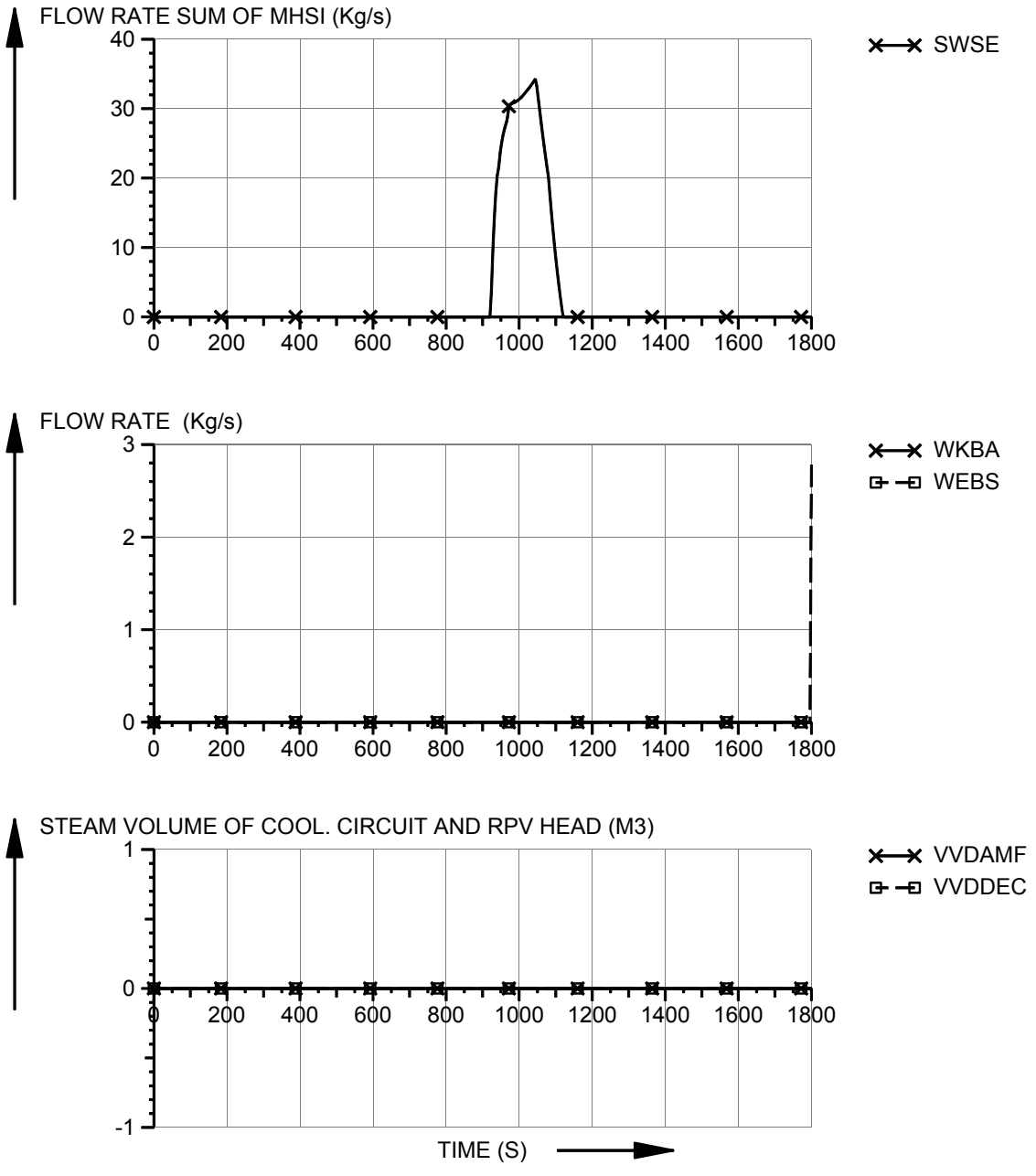
EPR, BASIC DESIGN ,CODE NLOOP
EOC CONDITION, LOSS OF CONDENSER VACUUM
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (7/16)



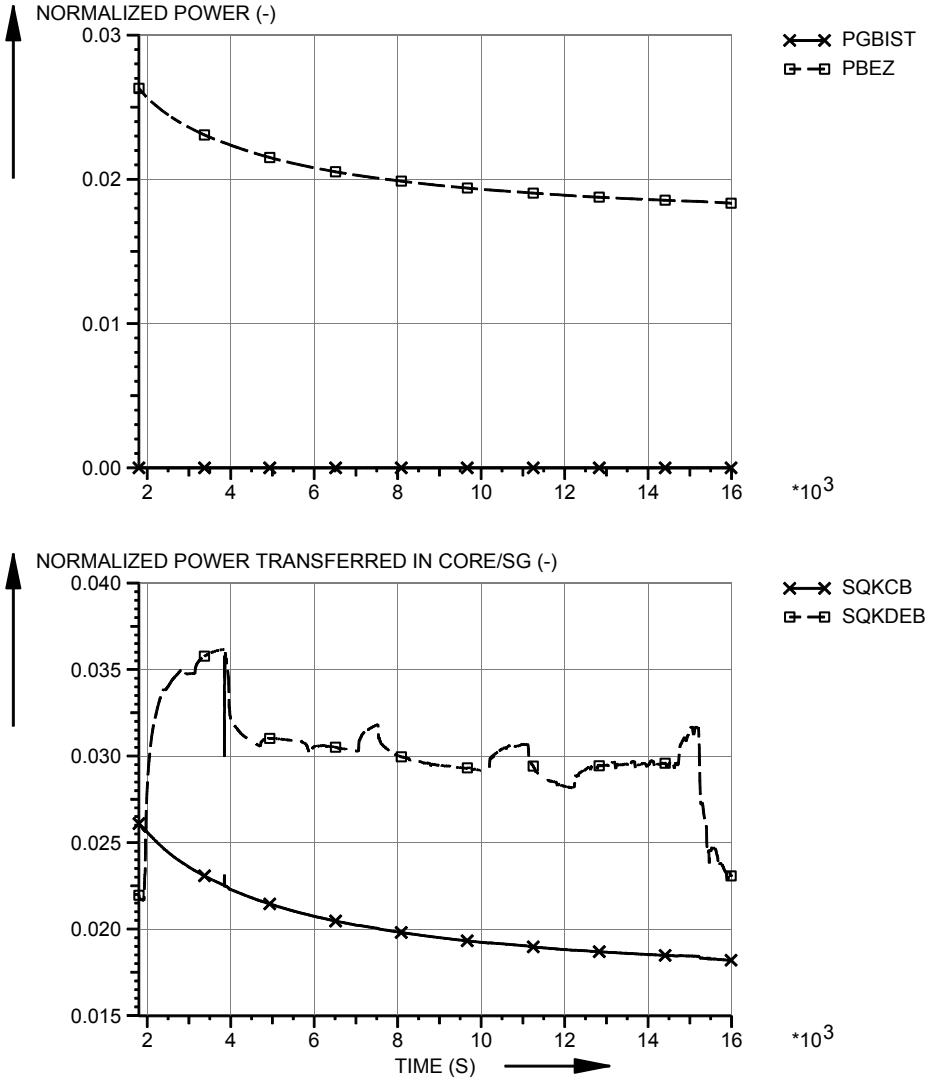
EPR, BASIC DESIGN ,CODE NLOOP
EOC CONDITION, LOSS OF CONDENSER VACUUM
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (8/16)



EPR, BASIC DESIGN ,CODE NLOOP
EOC CONDITION, LOSS OF CONDENSER VACUUM
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

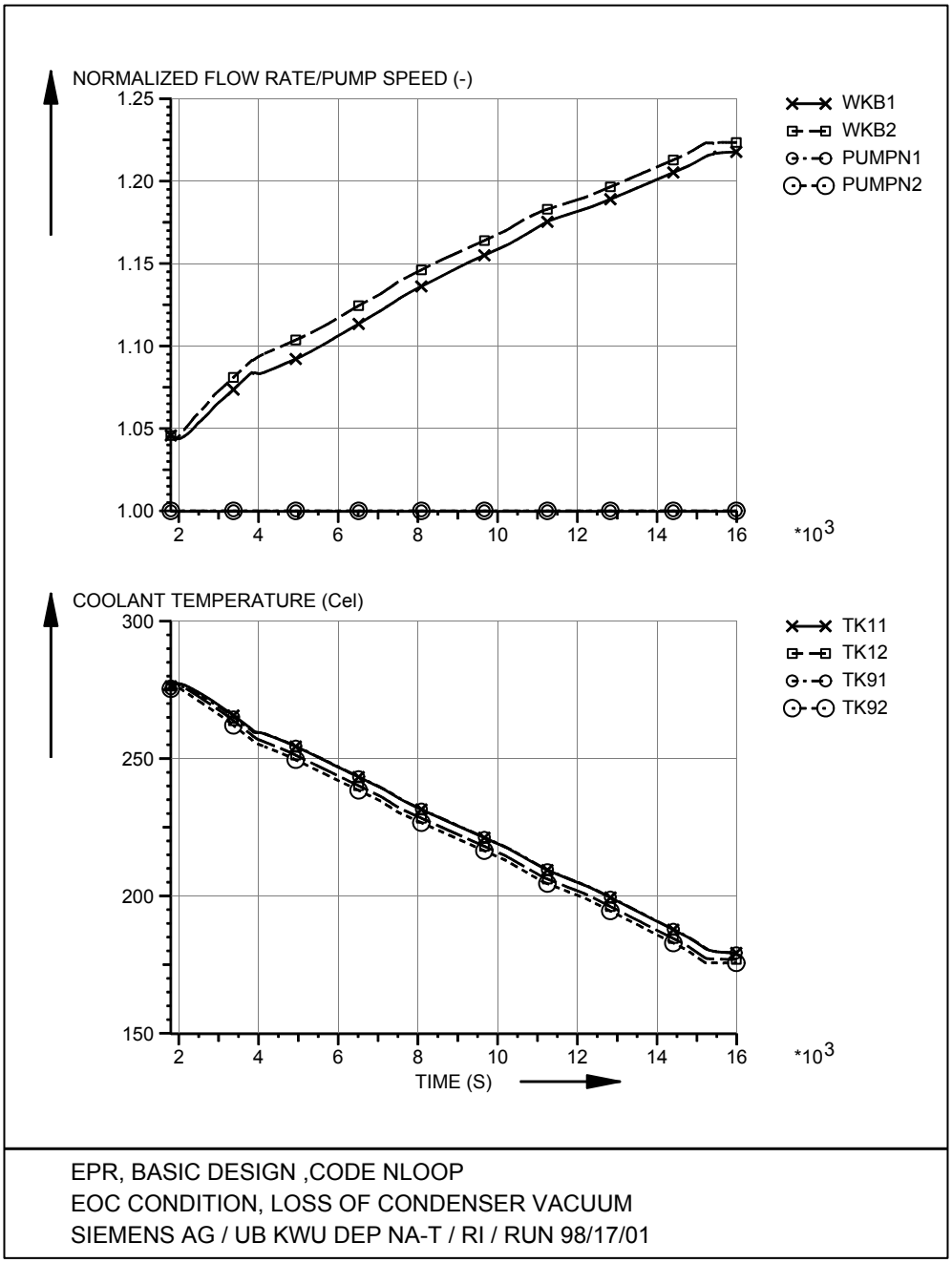
APPENDIX 14B.2.5.2 - FIGURE 1 (9/16)



EPR, BASIC DESIGN ,CODE NLOOP
EOC CONDITION, LOSS OF CONDENSER VACUUM
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (10/16)

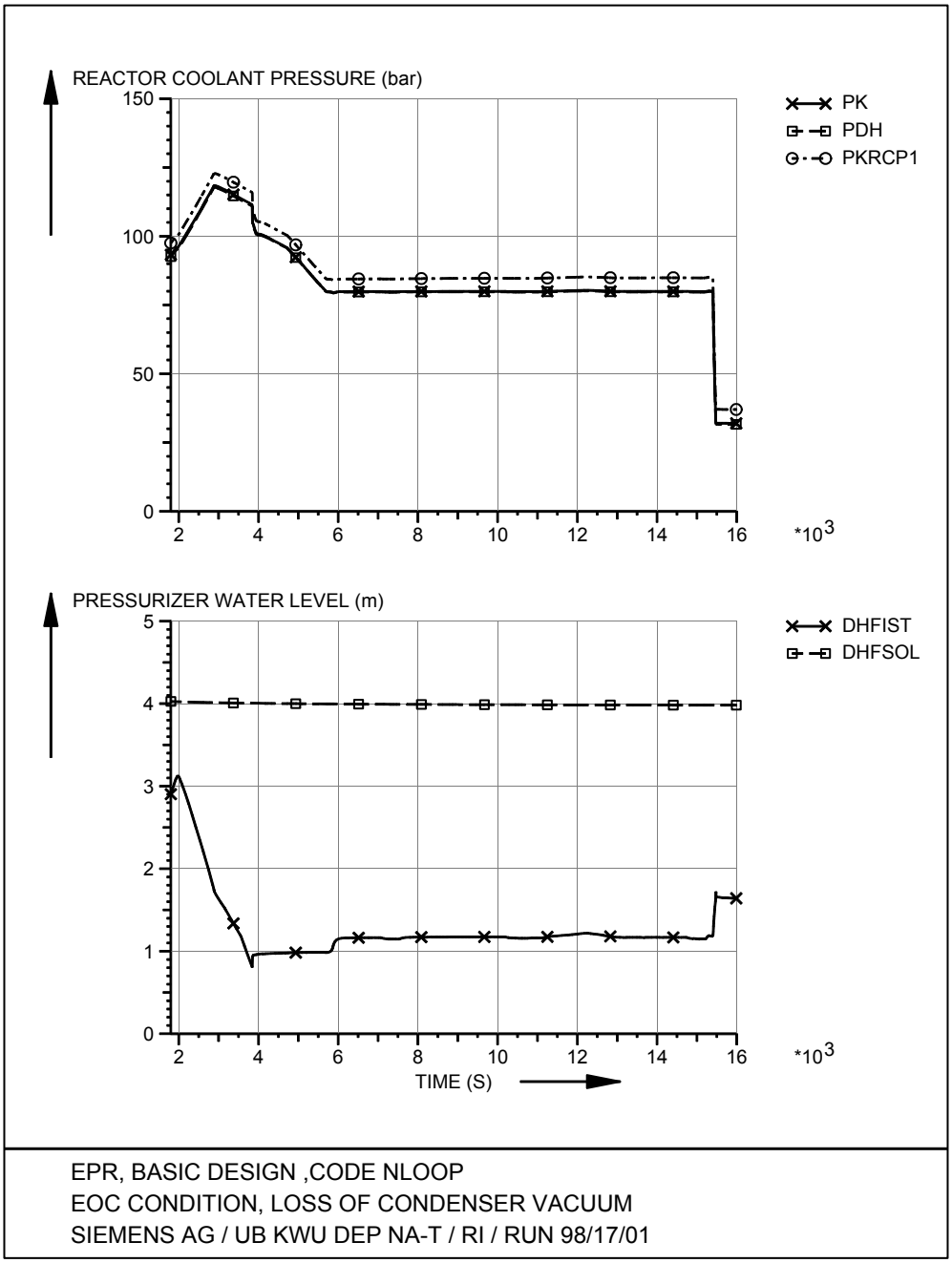
11.11.98 14:40:47 TAPE4_LAUF_102198_103421_01 Page 2



EPR, BASIC DESIGN ,CODE NLOOP
EOC CONDITION, LOSS OF CONDENSER VACUUM
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (11/16)

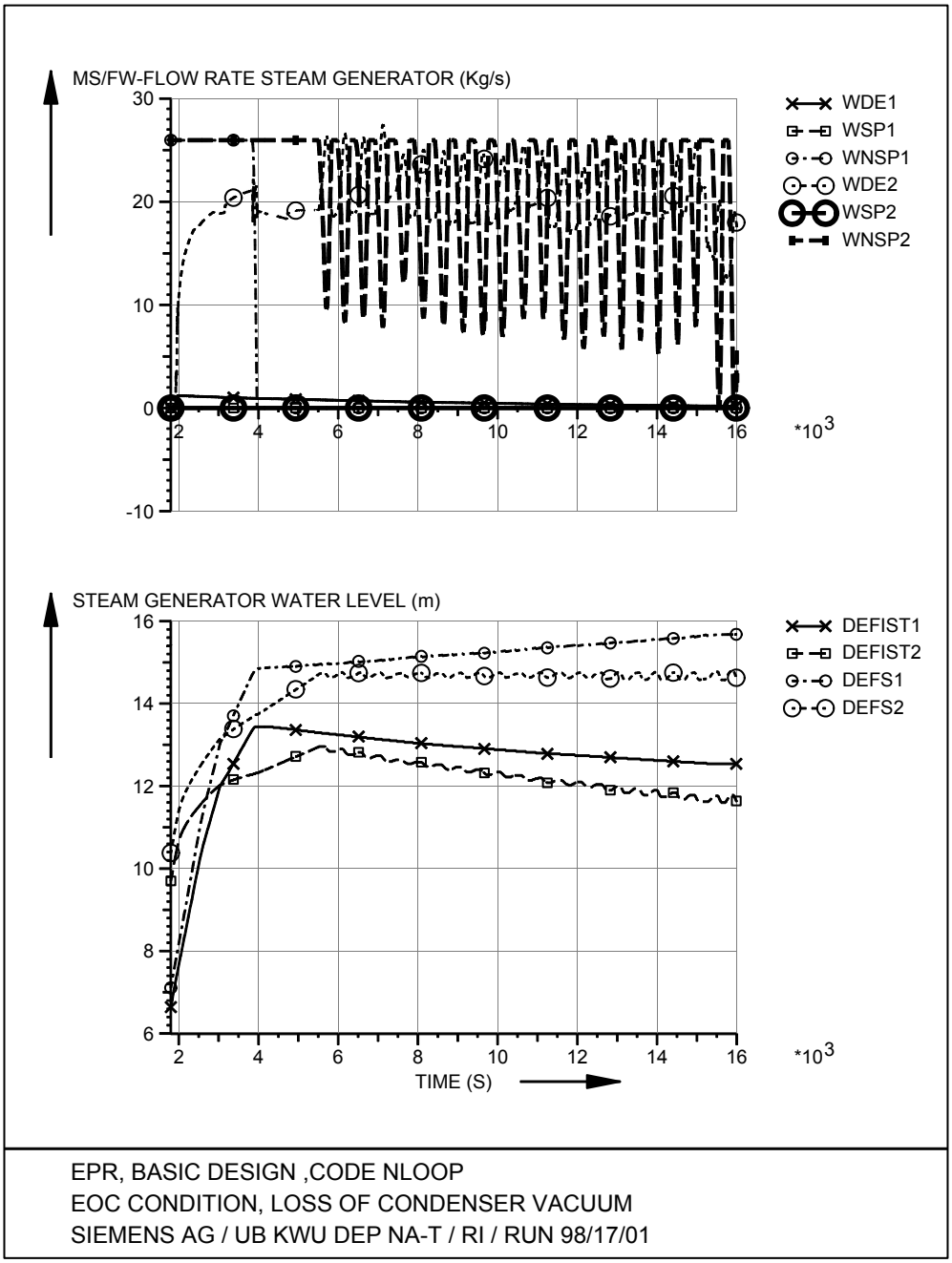
11.11.98 14:40:52 TAPE4_LAUF_102198_103421_01 Page 3



EPR, BASIC DESIGN ,CODE NLOOP
EOC CONDITION, LOSS OF CONDENSER VACUUM
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (12/16)

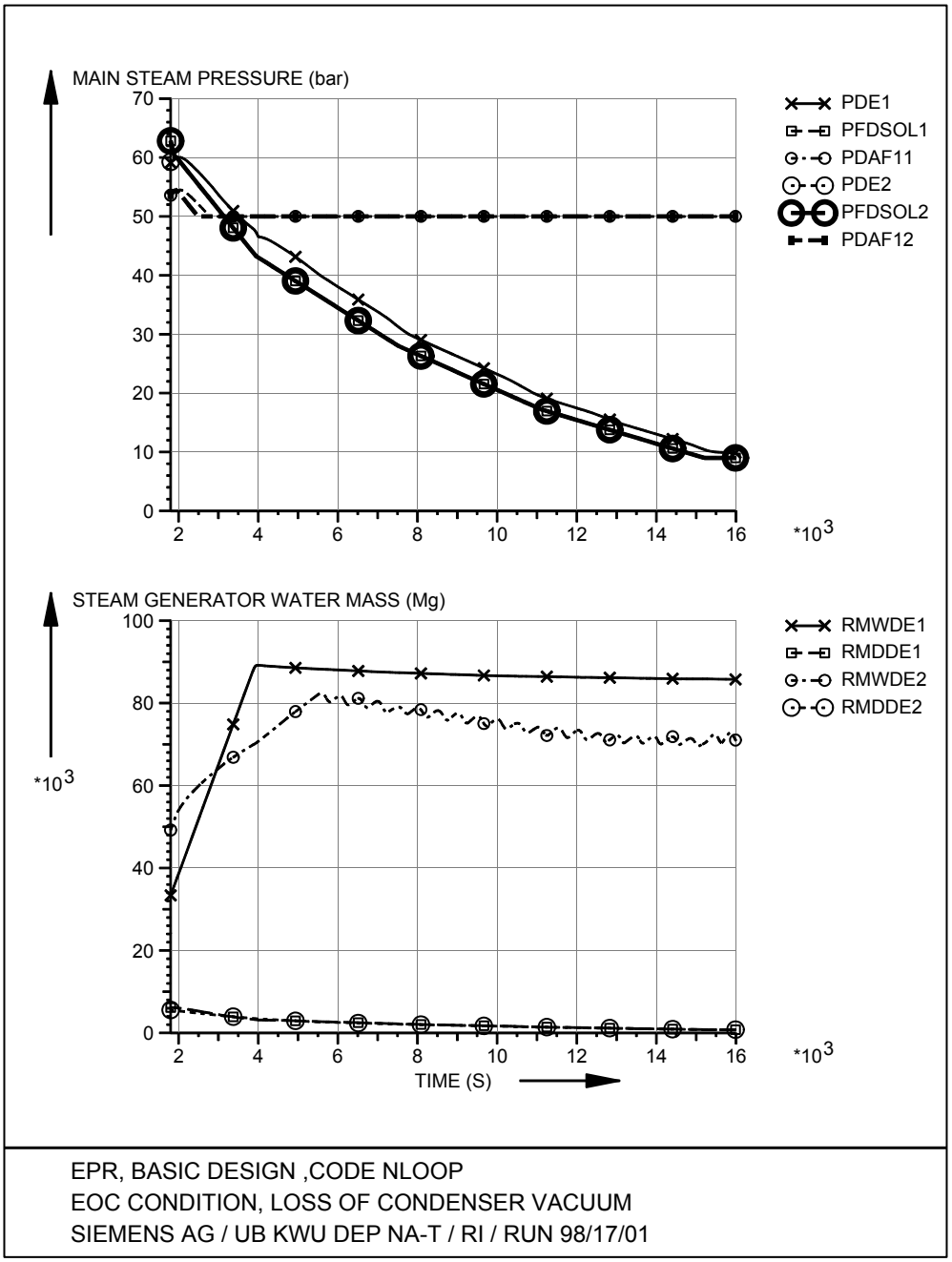
11.11.98 14:40:57 TAPE4_LAUF_102198_103421_01 Page 4



EPR, BASIC DESIGN ,CODE NLOOP
EOC CONDITION, LOSS OF CONDENSER VACUUM
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (13/16)

11.11.98 14:41: 4 TAPE4_LAUF_102198_103421_01 Page 5



EPR, BASIC DESIGN ,CODE NLOOP
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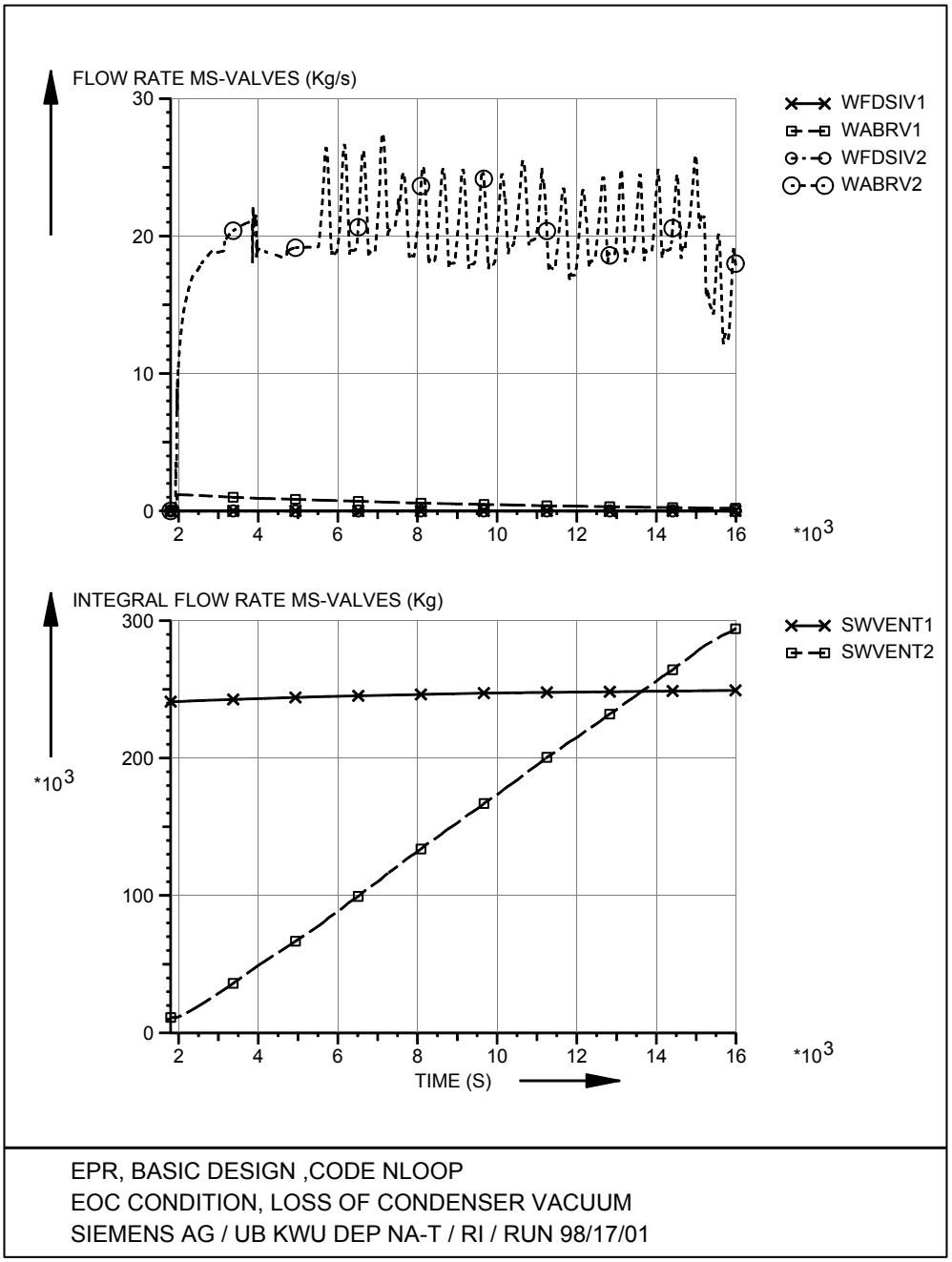
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (14/16)

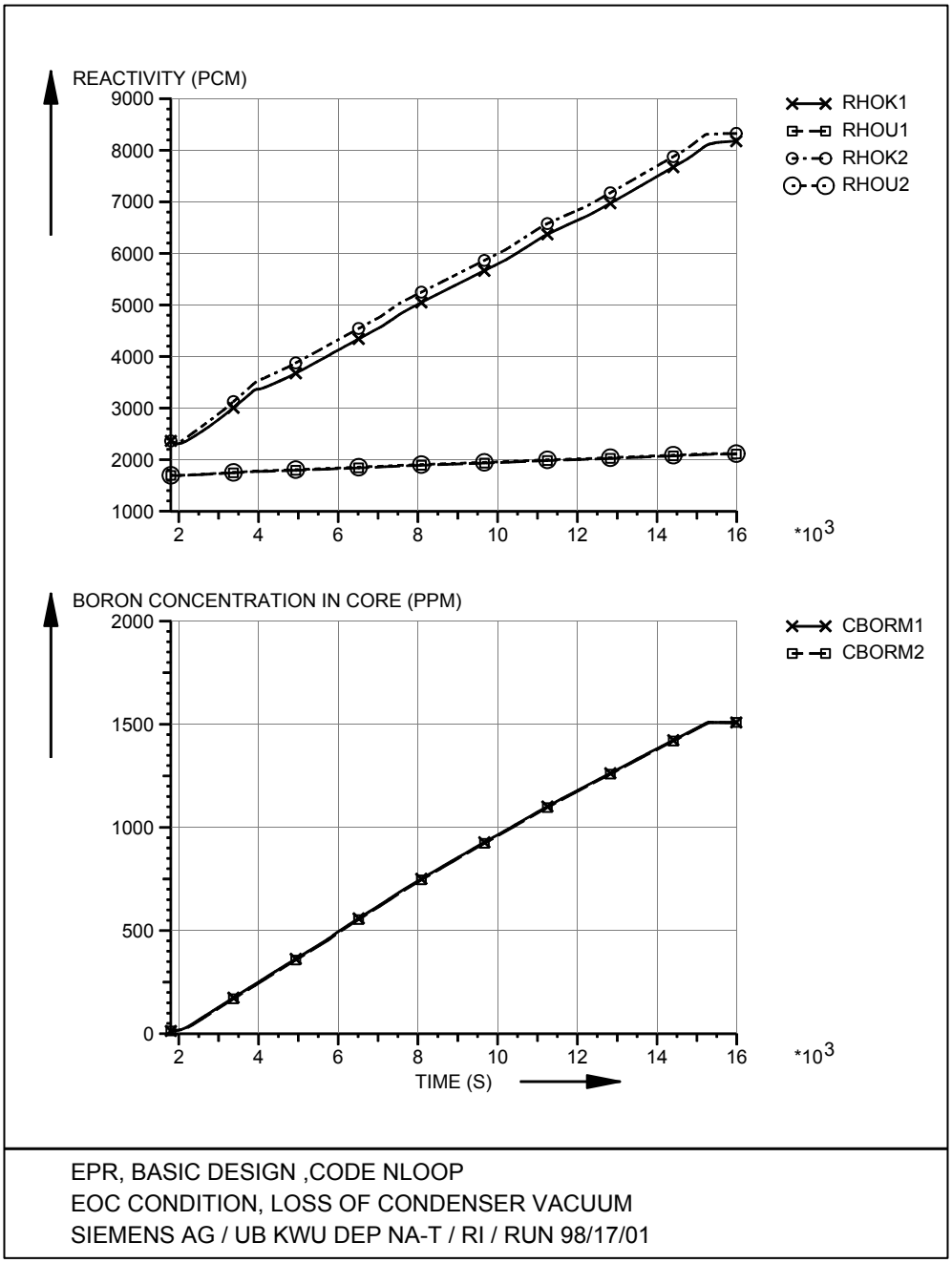
11.11.98 14:41:25 TAPE4_LAUF_102198_103421_01 Page 6



EPR, BASIC DESIGN ,CODE NLOOP
EOC CONDITION, LOSS OF CONDENSER VACUUM
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (15/16)

11.11.98 14:41:30 TAPE4_LAUF_102198_103421_01 Page 7



EPR, BASIC DESIGN ,CODE NLOOP
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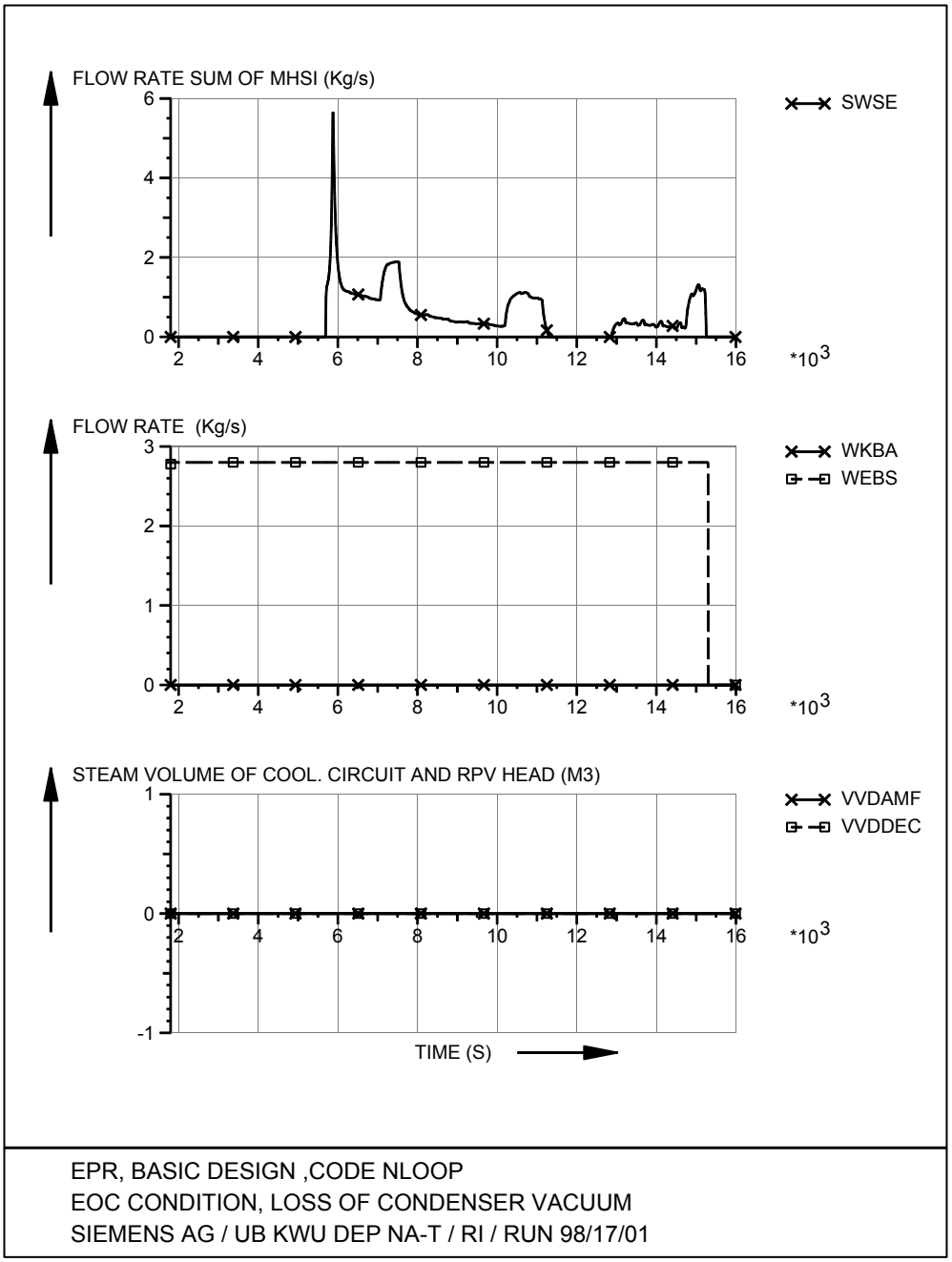
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 1 (16/16)

11.11.98 14:41:36 TAPE4_LAUF_102198_103421_01 Page 8



EPR, BASIC DESIGN ,CODE NLOOP
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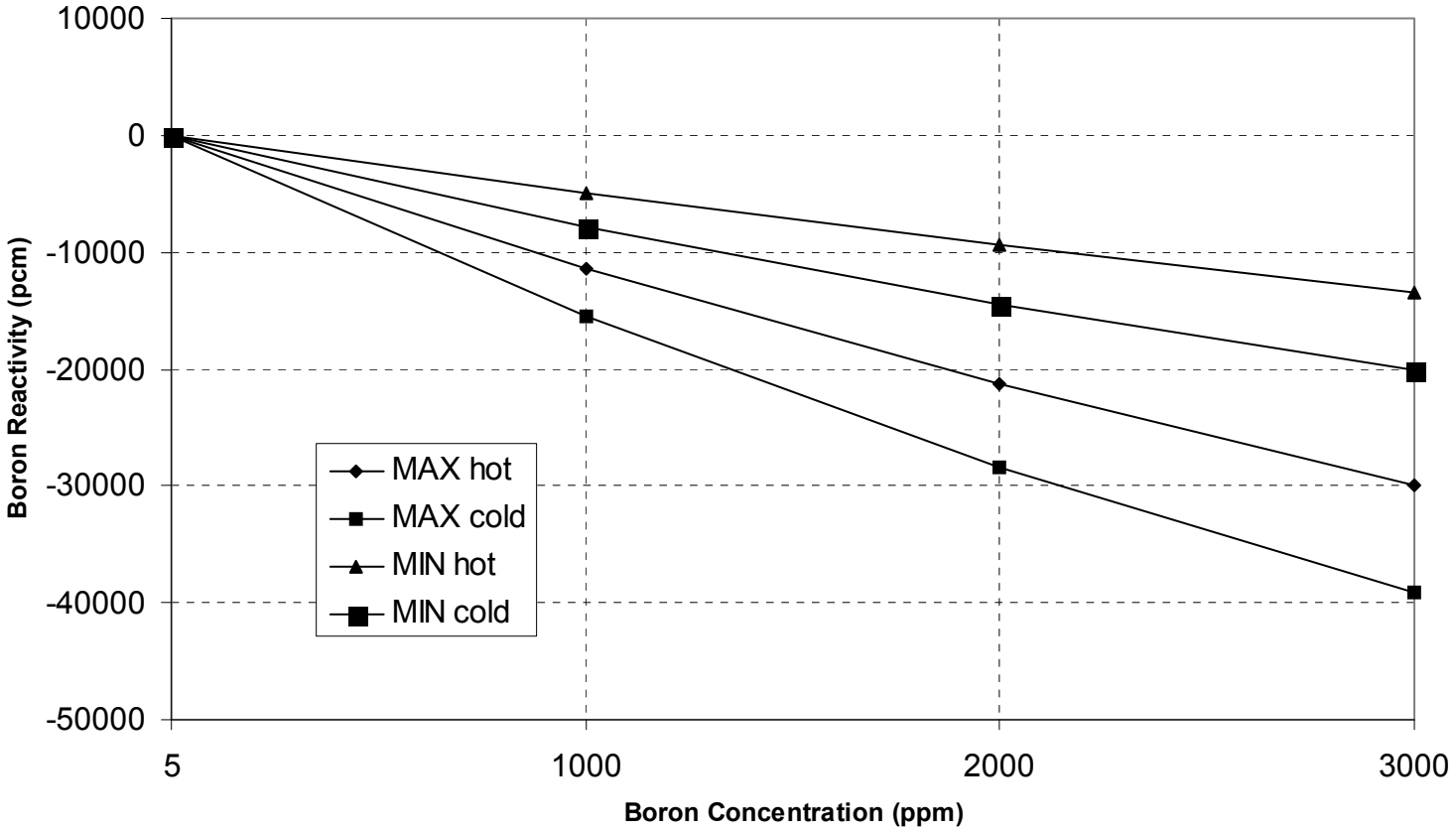
SIEMENS AG / UB KWU DEP NA-T / RI / RUN 98/17/01

APPENDIX 14B.2.5.2 - FIGURE 2



2.6. DECREASE OF FORCED REACTOR COOLANT FLOW (PCC-3)

2.6.1. Identification of causes and accident description

A - Definition, causes and description of the transient.

A complete loss of forced reactor coolant flow may result from a simultaneous failure in the electrical supplies to all reactor coolant pumps resulting for example from a fast decay of frequency in the external grid.

The enveloping case studied corresponds to an electrical supply frequency decrease of 4 Hz per second over a significant time period.

The typical sequence of events in case of a forced decrease of reactor coolant flow is as follows:

From the initiating event to attainment of the controlled state:

Frequency decay at high rates leads to a reversal of motor torque which reduces Reactor Coolant Pump speed and coolant flow more rapidly than voltage decay transients (limited by the moment of inertia).

If the reactor is at power at the time of the incident, the heat flux transferred from the fuel to the coolant is almost unchanged. The margin to DNB is reduced below its value during normal operation, resulting in a potential for DNB with consequential fuel damage if the reactor is not tripped promptly.

The reactor trip is actuated by the low Reactor Coolant Pump speed protection function which is F1A classified.

It is noted that frequency decay at a rate of 4 Hz/s leads to a complete loss of offsite electrical power supply.

The controlled state corresponds to the following conditions:

- Nuclear power = 0% full power
- Reactor Inlet temperature = 301°C
- Pressure = 155 bar
- Boron concentration of the initial power state
- Xenon level higher than or equal to the initial Xenon level
- RCCAs fully inserted
- Natural circulation

The shutdown margin ensures core subcriticality.

This state is similar to the corresponding state in LOOP (see section 2.5.1 of this appendix where more details are provided).

From the controlled state to the safe shutdown state:

The safe shutdown state corresponds to the following conditions:

- Nuclear power = 0% full power
- Hot leg temperature = 180°C
- Pressure = 30 bar
- Residual heat removed by the steam generator or the RRA [RHRS]
- Boron concentration ensuring the core subcriticality, even after the Xenon depletion
- RCCAs fully inserted
- Natural circulation

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

- RCP [RCS] cooldown and depressurisation (F1 classified)
- boration via the RBS [EBS] (F1 classified)

With regard to activity release, the decrease of forced reactor coolant flow rate event is bounded by the loss of condenser vacuum event (see section 2.5.2 of this appendix). With regard to subcriticality it is bounded by the uncontrolled boron dilution event. With regard to heat removal capability it is bounded by the feedwater line break event (section 2.16.1 of this appendix) since all four steam generators remain available in the case of a decrease of forced reactor coolant flow rate.

The subsequent analysis concentrates on the phase of the transient from the initiating event to reactor trip.

B - Safety and decoupling criteria

A complete loss of forced reactor coolant flow is a plant condition category 3 event (PCC-3).

The relevant safety criteria are the dose equivalent limits applicable in case of radioactivity release to atmosphere as discussed in section 1 of this appendix.

The relevant decoupling criteria relate to the behaviour of barriers i.e.:

- the number of fuel rods experiencing DNB must not exceed 10% of the total core,
- the average cladding hot spot temperature must not exceed 1482°C,
- the fraction of fuel melting at the hot spot must not exceed 10% of the fuel pellet cross-section.

C - Reactor protection system actions

C1 - Precautions taken to reduce probability of occurrence

The probability of occurrence of the initiating event is minimised by operating the plant in a stable grid environment and ensuring a high reliability of protection channels.

C2 - Reactor protection system actions

Following a decrease of forced reactor coolant flow rate reactor trip may be generated by the following signals:

- low Reactor Coolant Pump speed signal:

  The Reactor Coolant Pumps are fitted with speed measuring devices. A reactor trip signal is generated when the measured speed in 2 out of 4 Reactor Coolant Pumps is below a low threshold value.

- low reactor coolant flow rate (single loop) signal:

  Loss of a Reactor Coolant Pump is detectable using the corresponding loop flow rate measurement. A reactor trip signal is generated when 2 out of 4 of the loop flow rate measurements fall below a threshold value. If the nuclear power level is below a low threshold, the low reactor coolant flow rate trip channel is inhibited in order to allow the plant to be operated with 1 of the 4 Reactor Coolant Pump stopped.

When the loss of flow rate affects all four Reactor Coolant Pumps (e.g. in the case of a voltage or frequency disturbance) protection is provided by both the above signals. Reactor trip would be expected to be actuated by the low Reactor Coolant Pump speed signal due to its higher threshold and shorter response time. When the loss of flow rate is in one primary loop only, protection is provided by the low reactor coolant flow rate signal.

2.6.2. Methods and assumptions

A - Selection of single failure and maintenance state

In accordance with the general safety rules defined in section 1 of this appendix, single failure and maintenance assumptions are applied to F1 systems in the most conservative way with respect to the design criterion to be met.

Consideration of the safety systems invoked during the transient leads to the conclusion that the worst case single failure is a stuck rod of the highest worth following reactor trip.

In this transient, preventative maintenance is not relevant.

B - Method of analysis

The computer code SMART coupled with a thermal hydraulic simulation of the behaviour of the primary system are used to calculate:

- the core flow rate
- the nuclear power
- the heat flux

- the reactor coolant system pressure
- the reactor coolant system temperature

during the transient.

The use of a 3D kinetic neutronic model allows a consistent treatment of the most adverse axial power shape taking account of the opposing effects on DNBR and shutdown efficiency.

The results are used to perform an analysis of the number of rods experiencing DNB, using the thermal-hydraulic code FLICA.

C - Initial conditions (see Appendix 14B.2.6 - Table 1)

The initial operating conditions assumed are the most severe with respect to DNBR, i.e.:

- the maximum steady state power level is assumed, taking into account an uncertainty of 2% on the thermal balance, (i.e. 102% of the nominal power level),
- the maximum steady state average coolant temperature is assumed, taking into account measurement uncertainties, the average coolant temperature control deadband and the control system effectiveness, i.e. 311.25° C + 2.5° C,
- the minimum steady - state coolant pressure is assumed, taking into account measurement uncertainty and the control system effectiveness, i.e. 155 bar – 2.5 bar,
- the vessel flow rate is assumed equal to 100% of the nominal thermalhydraulic flow rate,
- for the 3D kinetic neutronic simulation, the assumed axial power distribution and the nuclear $F\Delta H$ value are such that the initial DNBR value is equal to the low DNBR limiting condition of operation (LCO) value. The initial core A0 value is assumed to be 18%.
- for the DNBR calculations performed to confirm that the design DNBR criterion is met, the axial power distribution and the nuclear $F\Delta H$ value are chosen such that the initial DNBR value is equal to the DNBR limiting value. The initial core A0 value is 18%.

D - Core related assumptions

- Moderator temperature coefficient
  - A minimum initial value for the moderator density coefficient (0.038×10^5 pcm/g cm^{-3}) is assumed, since this results in a maximum hot-spot heat flux in the initial period of the transient when the minimum DNBR is reached.
- Doppler coefficient :
  - The Doppler coefficient is set to its maximum absolute value (-3.6 pcm/°C) to reduce the negative reactivity addition following reactor trip, thereby increasing the heat flux at the time of minimum DNBR.

- Fuel-coolant heat transfer coefficient:
    - A minimum value is assumed (9700 W/m^{2}K) in order to maximise the heat flux during rod drop.
- Shutdown margin:
    - The RCCA with the highest worth is assumed stuck above the core. Negative reactivity insertion following trip is thus minimised, resulting in a minimum final shutdown margin. The assumed RCCA rod worth versus time is a code result obtained assuming an unfavourable top peaked core initial condition with 18% axial offset.

E - Protection system actions

The reactor trip is actuated on a low Reactor Coolant Pump speed signal.

The setpoint value is 91% of the nominal speed value.

The total delay between the generation of the reactor trip signal and the beginning of RCCA drop is conservatively assumed to be 0.6 seconds.

F - Control system actions

The transient generates a reactor coolant system heat up whose consequences are mitigated by the average coolant temperature and the pressure controls.

Since the effect of the average coolant temperature control is beneficial it is not taken into account in the analysis.

To minimise the pressure increase the pressuriser spray flow rate is assumed to be fixed at its maximum value.

2.6.3. Results and conclusions

The results are presented in the following figures:

Appendix 14B.2.6 - Figure 1: Core flow rate versus time
Appendix 14B.2.6 - Figure 2 : Nuclear power and heat flux versus time
Appendix 14B.2.6 - Figure 3 : Pressuriser pressure versus time

Appendix 14B.2.6 - Figure 4 : DNBR versus time:

- DNBR 2 - curve which shows the DNBR variation resulting from a 3D coupled core calculation starting from the LCO DNBR value.
- DNBR 3 – curve which shows the DNBR variation deduced from DNBR 2 by increasing $F_{\Delta H}$ in a decoupled manner to allow for uncertainties in the low DNBR surveillance channel.

Appendix 14B.2.6 - Figure 5 : Vessel average temperature versus time
Appendix 14B.2.6 - Figure 6 : Core reactivity balance versus time
Appendix 14B.2.6 - Figure 7 : Neutronic and thermal A0 versus time

The calculated sequence of events is shown in Appendix 14B.2.6 - Table 2.

Appendix 14B.2.6 – Figure 3 shows hat the design DNBR criterion is exceeded. Therefore the number of fuel rods potentially experiencing DNB is calculated, taking into account the probability distribution associated with DNBR calculation uncertainties.

From this calculation, it is deduced that less than 1% of the total number of fuel rods (actually about 0.2%) could experience DNB, which is well below the decoupling criterion of 10%.

The thermal-hydraulic conditions at the time of minimum DNBR occurrence are presented in Appendix 14B.2.6 - Table 3.

All the decoupling acceptance criteria are met.

APPENDIX 14B.2.6 – TABLE 1

Forced decrease of reactor coolant flow
Assumed input data values used in analysis

Parameter	Value
- Frequency decay rate (Hz/s)	4
- Initial conditions :	
Power (% of full power)	100 + 2 = 102
Reactor coolant average temperature (° C)	311.25 + 2.5 = 313.75
Reactor coolant pressure (bar) (for DNBR calculation)	155 - 2.5 = 152.5
Vessel flow rate (% of nominal value)	100
- Core related assumptions :	
Moderator density coefficient $\Delta k/k$ (g/cm^3)	0.038
Shutdown margin (pcm)	4000
- Protection system actuations :	
Low reactor coolant pump speed setpoint (% of nominal speed)	91
Delay between the low Reactor Coolant Pump speed protection channel actuation and the opening of the breakers (s)	0.6

APPENDIX 14B.2.6 – TABLE 2

Forced decrease of reactor coolant flow transient
Sequence of events

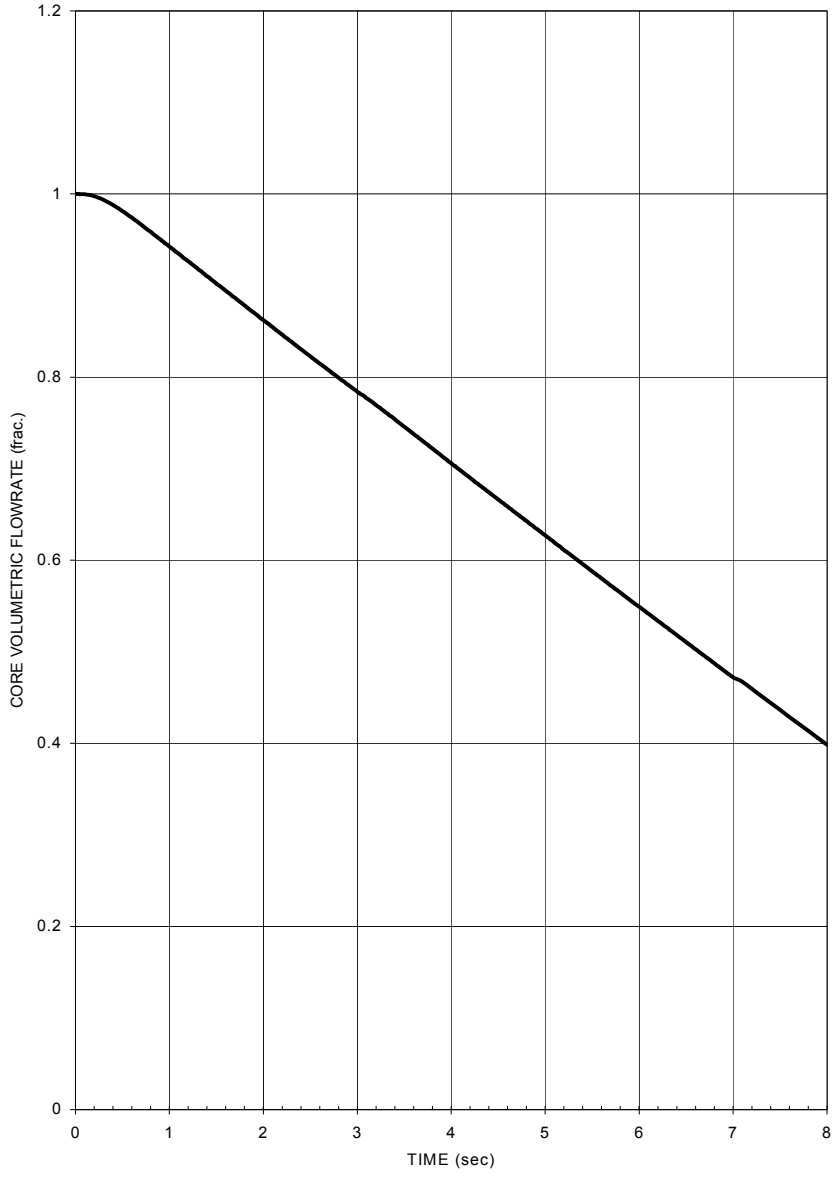
Event	Time (s)
Frequency begins to decay	0
Low reactor coolant pump speed reactor trip	1.26
Rods begin to drop	1.86
Minimum DNBR occurs	2.4

APPENDIX 14B.2.6 – TABLE 3

Thermal-hydraulic conditions at the minimum DNBR occurrence

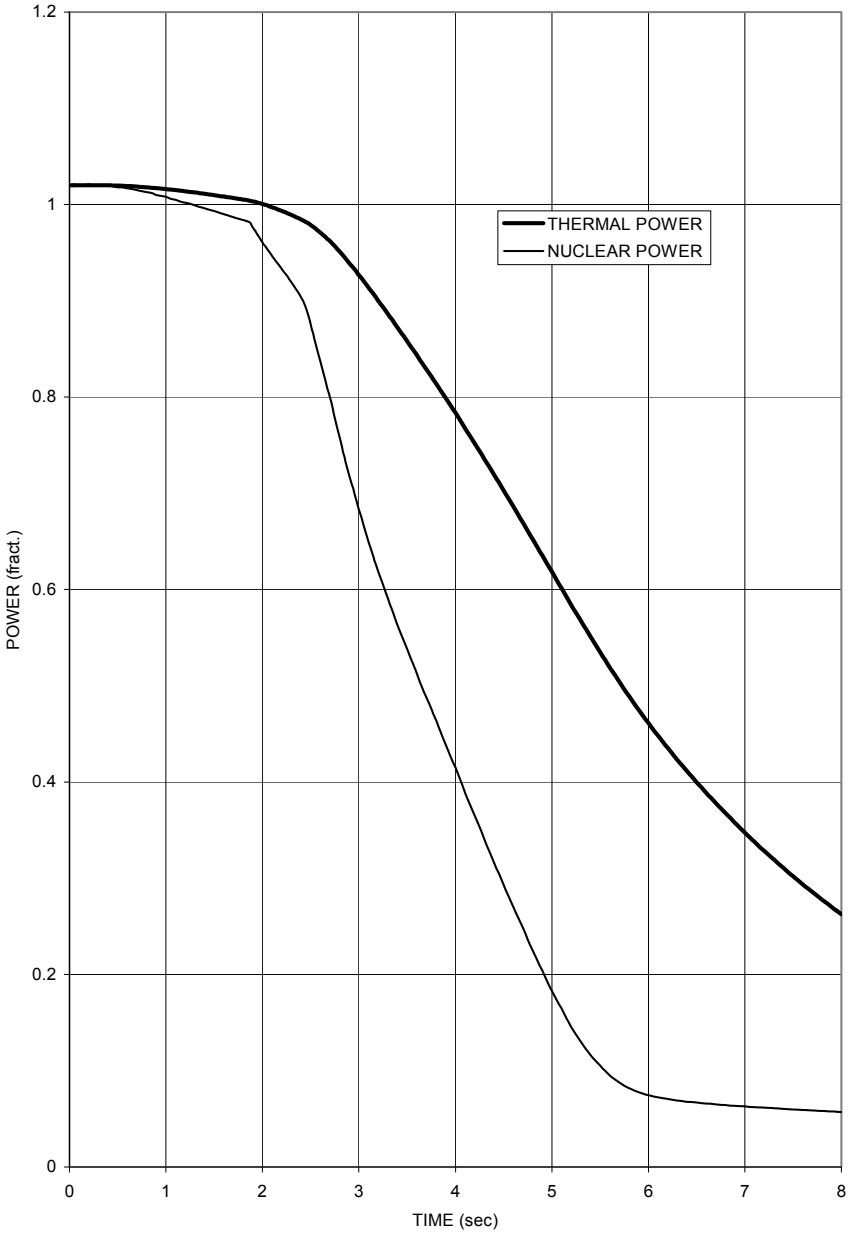
Parameter	Value
Time (seconds)	2.4
Thermal power (%)	98.5
Core inlet temperature (°C)	295.1
Pressuriser pressure (bar)	154.7
Volumetric flow rate (%)	83.1
Percentage of rods experiencing DNB	< 1%

APPENDIX 14B.2.6 – FIGURE 1



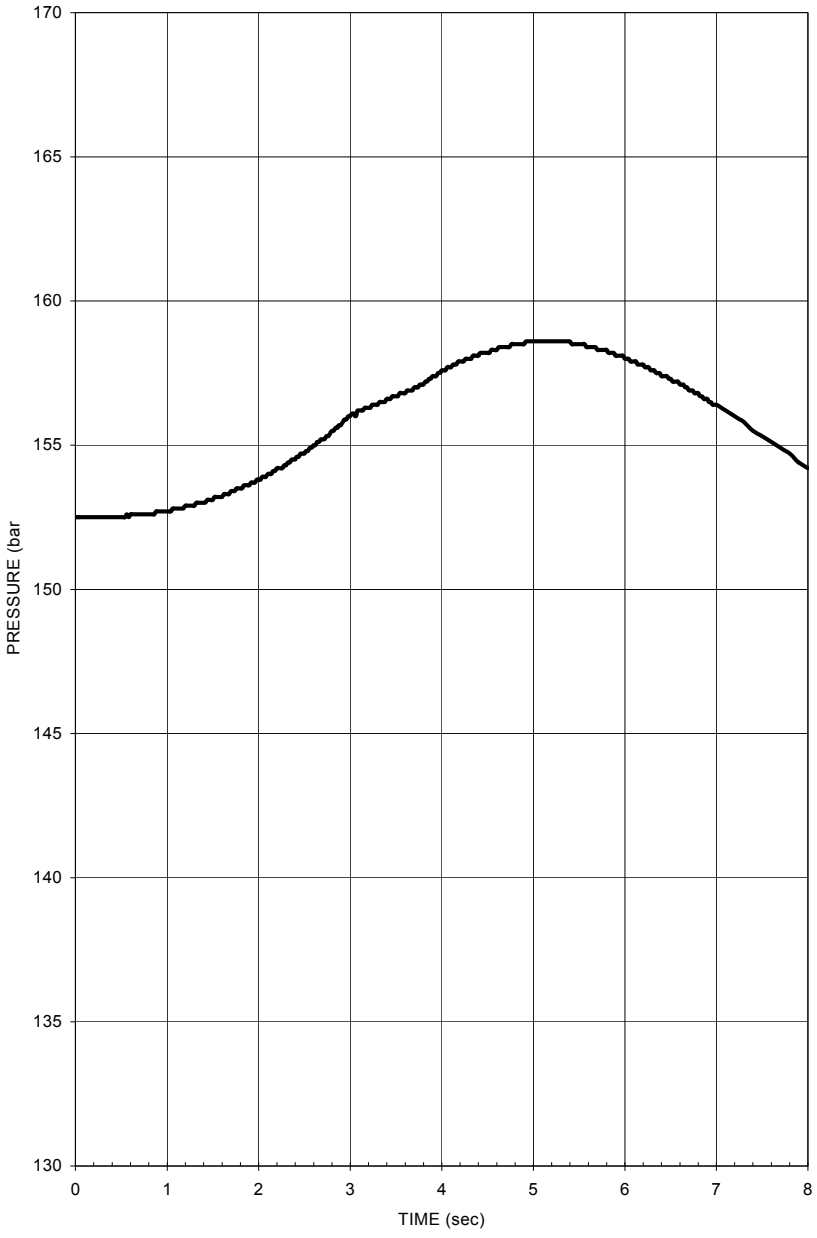
Forced decrease of reactor coolant flow transient
Core flow rate

APPENDIX 14B.2.6 – FIGURE 2



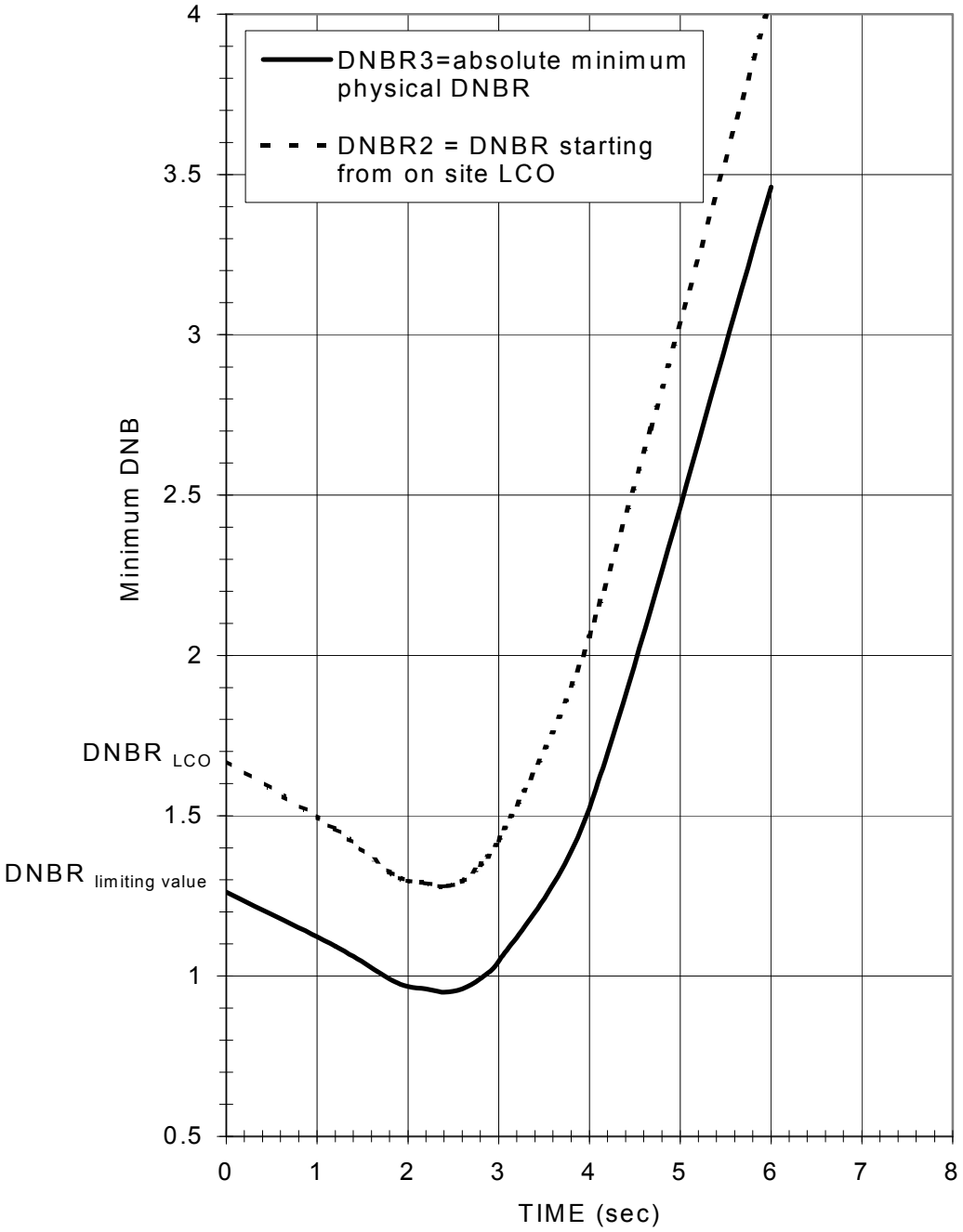
Forced decrease of reactor coolant flow transient
Nuclear power and heat flux

APPENDIX 14B.2.6 – FIGURE 3



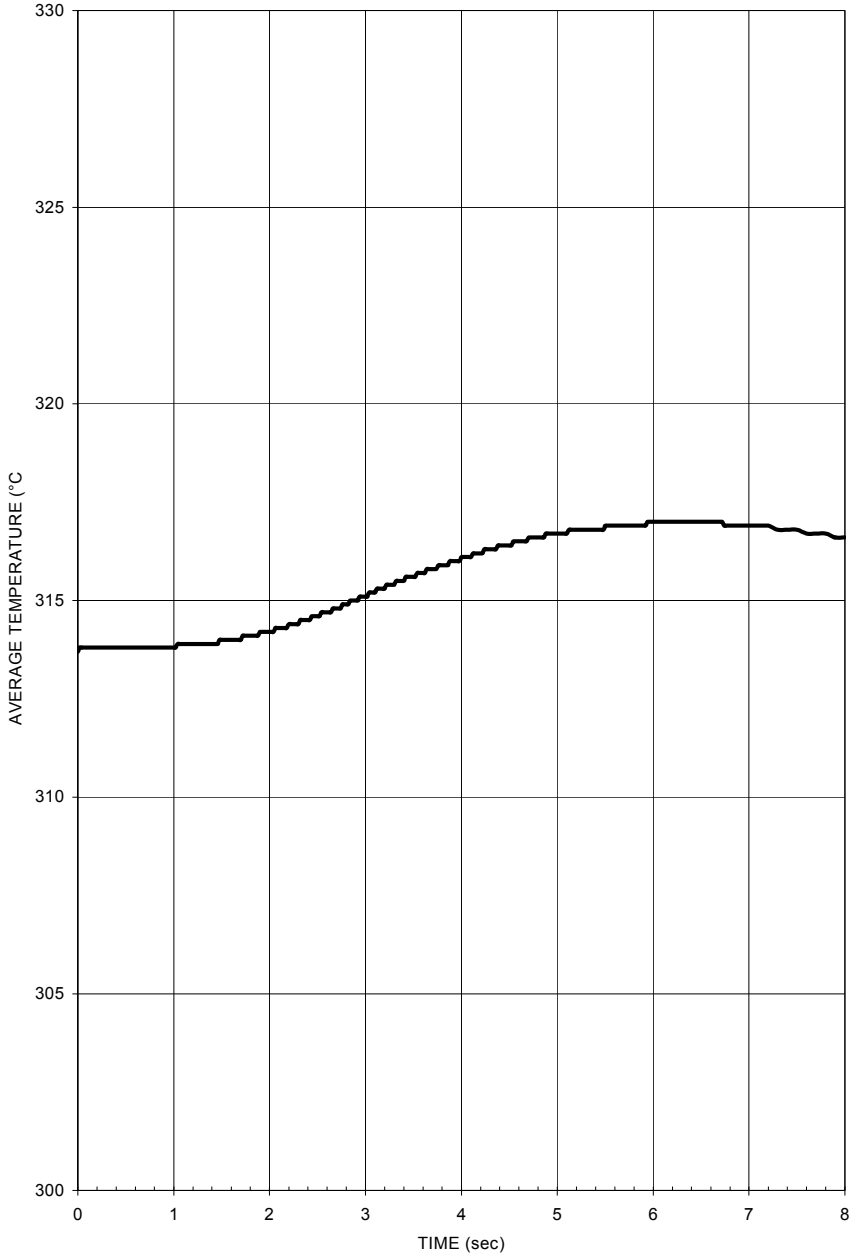
Forced decrease of reactor coolant flow transient
Pressuriser pressure

APPENDIX 14B.2.6 – FIGURE 4



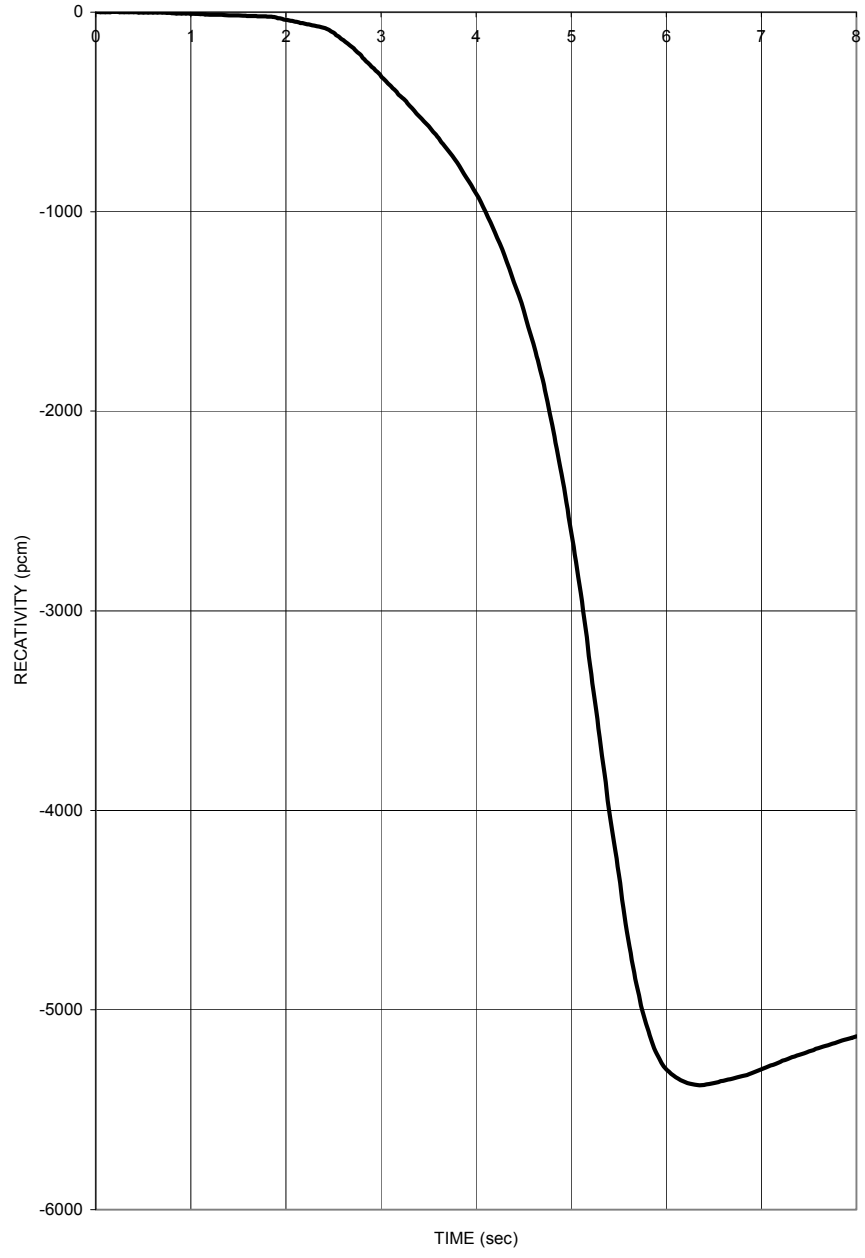
Forced decrease of reactor coolant flow transient
DNBR

APPENDIX 14B.2.6 – FIGURE 5



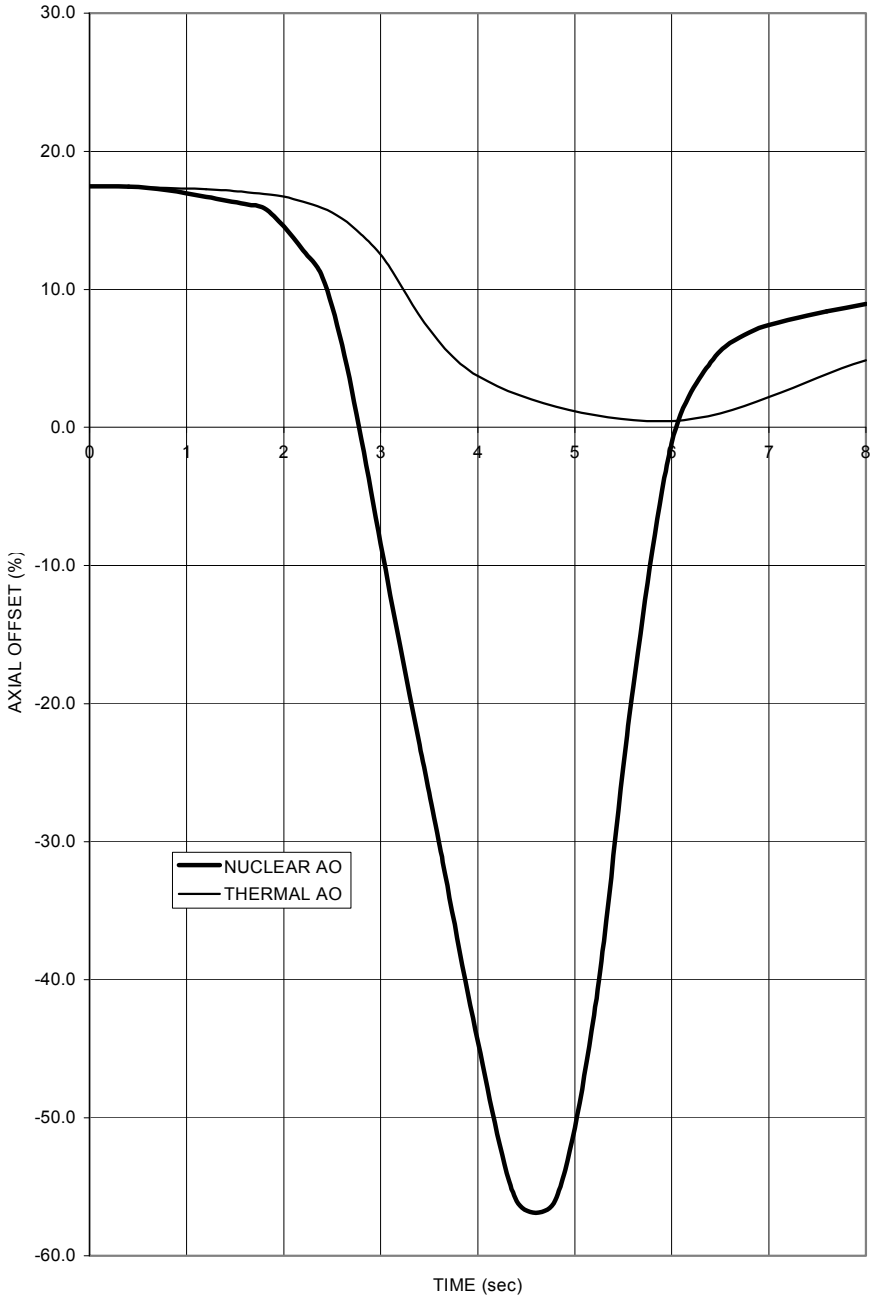
Forced decrease of reactor coolant flow transient
Vessel average temperature

APPENDIX 14B.2.6 – FIGURE 6



Forced decrease of reactor coolant flow transient
Core reactivity balance

APPENDIX 14B.2.6 – FIGURE 7



Forced decrease of reactor coolant flow transient
Axial offset

2.7. PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW (PCC-2)

This event is analysed in reactor state A only.

2.7.1. Identification of Causes and Accident Description

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

The bounding event considered for the partial loss of reactor coolant flow is the loss of one Reactor Coolant Pump which it is classified as a PCC-2 event (event of moderate frequency) as defined in section 0.1 of this appendix.

Reactor trip protection in the partial loss of coolant flow event is provided by the following signals:

- a partial trip signal at < 91% Reactor Coolant Pump speed, which reduces the reactor power to about 50%. This countermeasure is not taken into account because it is not F1A classified.
- a low reactor coolant flow signal < 25% in conjunction with reactor power > 75% (preliminary value)

Following loss of more than one Reactor Coolant Pump, reactor trip would be actuated immediately on Reactor Coolant Pump speed < 91% on 2 out of 4 Reactor Coolant Pumps.

Controlled and Safe Shutdown State

It must be demonstrated that the controlled state can be attained with F1A automatic functions only and the safe shutdown state can be attained using only F1B and F1A functions without violating relevant safety/decoupling criteria (see below):

- Safety Criteria :
  - Radiological limits for normal operation.
- Decoupling Criteria:
  - Avoidance of DNB (DNBR limit =1.0)

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

- Reactor Trip:
  - The reactor trip is initiated from the following signal: Loop coolant flow < 25%

- Systems:
  - 4 MS relief trains
  - 4 ASG [EFWS] trains
- Signals:
  - Actuation of MS relief trains at SG pressure of 93 bar
  - Actuation of ASG [EFWS] on “SG water level < 8 m”.

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

- 4 MS relief trains for cooldown to RHR connection conditions (manual action).
- 2 RBS [EBS] trains for boration
- 4 ASG [EFWS] trains for feedwater supply to the SGs (automatic or manual action)

2.7.2. Methods and Assumptions

The NLOOP code is used to calculate the main plant parameters and the PANBOX/COBRA-3-CP code is used to calculate core behaviour and the DNBR. The results are obtained by an iterative calculation using NLOOP and PANBOX/COBRA-3-CP.

Important phenomena and qualification of the models used in NLOOP and PANBOX/COBRA

The event belongs to the family of transients involving total or partial loss of primary forced reactor coolant flow.

Phenomena

The phenomena to be addressed here are basically the same as considered in section 2.5 of this appendix. Therefore the scope of code qualification mentioned therein also applies in the partial loss of forced flow case.

The capability of NLOOP to model phenomena specific to the partial loss of forced flow case, in particular backflow in the affected loop, was confirmed by applying the code to model a corresponding event in the KKG/BAG NPP.

2.7.2.1. Initial and Boundary Conditions

The key initial conditions are the reactor power, core average temperature, and primary pressure, with allowance for control system deadbands, measurement uncertainties and typical maximum control system deviations. The coolant flow assumed corresponds to the thermal design flow.

The initial conditions assumed are the most conservative operating conditions consistent with the minimum DNBR value allowable during steady state operation. The initial conditions are identical to those assumed in the LOOP event described in section 2.5.1 of this appendix.

The major initial and boundary conditions are listed in Appendix 14B.2.7 - Table 1.

Reactor Trip

A low flow trip setpoint of 25% loop flow rate (including measurement uncertainties) is assumed. A conservative delay is assumed between RT setpoint actuation and the beginning of rod drop.

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

- the RCCA drop characteristic, i.e. RCCA position = f (time), as described in Appendix 14B.0.2 – Table 7
- the RCCA worth as a function of RCCA position calculated by PANBOX on the basis of the actual 3D power distribution

Reactivity Feedback

a) Moderator Temperature Coefficient
A minimum initial value for the moderator temperature coefficient is assumed, since these results in minimum density feedback and hence maximum hot-spot heat flux in the initial part of the transient when the DNBR is a minimum.

b) Doppler Coefficient
A maximum absolute value of the Doppler Coefficient is assumed in order to maximise the positive reactivity addition when the power is reduced due to density feedback.

c) Density Feedback
The density reactivity feedback is a code result obtained for a core initial condition with 18% axial offset, minimum moderator temperature feedback, and maximum absolute value for the Doppler coefficient.

Flow Coast down and Cladding-Coolant Heat transfer

The core flow coast down curve is based on a conservative value for the Reactor Coolant Pump inertia (-15%) (see Appendix 14B.2.7 - Figure 1).

α-gap is assumed to be 9700 W/m^2/K

Choice of Single Failure and Preventative Maintenance

Single Failure:

- No additional single failure is assumed because an additional single failure has no impact.

Preventative Maintenance

- For the analysis of the transient up to the controlled state, preventative maintenance is ignored since it has no impact.

  For the analysis of the transient from controlled to safe shutdown state, the preventative maintenance assumptions are as for the corresponding reference cases (see sub-section 2.7.3.2 of this appendix).

2.7.3. Results and Conclusions

2.7.3.1. Initiating Event to Controlled State

Key parameters are plotted in Appendix 14B.2.7 - Figures 1 to 7. A description of plotted variables is given in Appendix 14B.2.7 - Table 3. The sequence of events is listed in Appendix 14B.2.7 - Table 2.

The flow setpoint for reactor trip is reached at 19.5 seconds. The minimum DNBR in the hot channel is 1.1 (DNBR3 uncoupled) and 1.4 (DNBR2, coupled starting from LCO). These values are above the DNBR acceptance criterion of 1.0.

Since DNB does not occur, the ability of the reactor coolant to remove heat from the fuel rods is not impaired.

The RCP [RCS] pressure increases only by a few bars and hence there is no challenge to the pressuriser safety valves. The heat removal from the secondary side is ensured by the VDA [MSRTs] and ASG [EFWS] only (no demand on the MSSVs).

Note that the total heat to be removed is less than that in the case of reactor trip with all Reactor Coolant Pumps running (see section 2.5.2 of this appendix); reactor trip occurs at about 20 seconds.

These results demonstrate that the controlled state, i.e.

- core subcritical (reactivity < 0)
- core power removed by VDA [MSRTs] and ASG [EFWS]

is achieved without violating the decoupling criteria. The integrity of barriers is not impaired.

2.7.3.2. Controlled State to Safe Shutdown State

This transition is not analysed explicitly as it is covered by analyses of other events (reference cases). The table below identifies the reference cases demonstrating safe shutdown is achieved with compliance with the three criteria: subcriticality, decay heat removal via the RHR and activity release/ barrier integrity within PCC limits:

Criteria	Reference case	Remark/Reason
Subcriticality	Uncontrolled boron dilution	Reference case more severe as one stuck rod assumed.
Max. Activity release	Loss of condenser vacuum	Reference case more severe as one SG is completely emptied.
Heat removal	Feedwater system line break	Reference case more severe as only one train is available for cooldown.

APPENDIX 14B.2.7 - TABLE 1 (1/3)

Event	Single Reactor Coolant Pump Failure
Event Category (PCC/RRC):	PCC-2
Family of Events:	Primary coolant flow reduction, decrease in primary heat removal
Safety Criteria:	Radiological limits for normal operation bounded by Loss of Condenser Vacuum case.
Decoupling Criteria:	No DNB (DNBR limit :1.0)
Purpose of Analysis	Calculation of minimum DNBR and overall plant behaviour
Safe shutdown conditions:	
controlled state	Hot shutdown with heat removal via MS relief valves
safe shutdown:	Connection to RHR system (case bounded by loss of condenser vacuum)

APPENDIX 14B.2.7 - TABLE 1 (2/3)

Initial Conditions	Best estimate	Conservative to DNBR	Overall plant behaviour	Remarks
Reactor/turbine power	100/100%	102/code res.	102/code res.	
Thermal reactor power	4900 MW	4998MW	4998 MW	
Reactor cooling pumps power		30.59 MW	30.59 MW	
Thermal steam generator power (per SG)		1256.9 MW	1256.9 MW	
Initial insertion of control rods		ARO	ARO	
• Void fraction at hot channel		0.188		At DNBR LCO = 1.66
Outlet		0.303		At DNBR = 1.26
Initial Axial Offset		18%		Core avg.
		15%		Hot channel, DNBR = 1.26
F∆H at DNBR LCO		1.745	not relevant	
F∆H at DNBR = 1.26		1.919		
Initial local power density		≤506 W/cm	not relevant	hot channel, uncoupled
Reactor cooling flow (total)	22241kg/s	code result	code result	Thermal design
Total core bypass flow	3.4%	5.5%	5.5%	hot RPV head bypass assumed in analysis: Bypass = 7%
Coolant mixing for temp. distribution at core inlet		36.7	not relevant	
Average reactor coolant temperature	311.25°C	311.25(+2.5) °C	311.25(+2.5) °C	For DNBR calc. core inlet temp. is 295 °C
Pressuriser pressure	155 bar	155(-2.5)bar	155(+2.5)bar	
Pressuriser level	6.97m	6.97 (-0.55) m	6.97(+0.55) m	
Feedwater/Main steam flow/SG	694.1 kg/s	708 kg/s	708 kg/s	
Feedwater temperature	230 °C	231°C	231 °C	
Steam generator water level	16.2 m	16.2 m	16.2 m	uncertainty not relevant
Steam generator water mass	87.6 Mg	code result	code result	(RCP [RCS] temp. and flow)
Pressure in steam generator	74.6 bar	code result	code result	(RCP [RCS] temp. and flow)

APPENDIX 14B.2.7 - TABLE 1 (3/3)

Boundary Conditions (kinetics and reactivity)	Best estimate	Conservative to DNBR	Overall plant behaviour	Remark
Burn-up state BOC/EOC		BOC	BOC	
Bank reactivity				
• Partial trip				not considered
• Reactor trip		10600 pcm at HSB	4900 pcm	rod dropping characteristic, see section 0.2.8 of this appendix. rod worth acc. PANBOX calc.
Control rod dropping time	2.5 s	3.5 s	3.5 s	
Moderator R-feed back		-9.6 pcm/°C [1)]	0.0	- 9.6 pcm/°C corresponds to 0.038 * 10^3 pcm/gcm^{-3}
Void reactivity		code result		
Fuel R-feed back		- 3.6.pcm/°C [1)]	-4.1 pcm/°C	
Boron R-coefficient/efficiency		not relevant	not relevant	
Boron concentration		not relevant	not relevant	
Decay power	ORIGEN/S	ORIGEN/S*fact.	ORIGEN/S*fact	fact. is time depending i.e. between 1.1 and 1.2
Delayed neutrons ($\Sigma\beta$)		730.5 pcm	700 pcm	Has no influence on results

[1)] at time 0s

APPENDIX 14B.2.7 - TABLE 1 (3/3)

Boundary Conditions (failure assumptions)	Best Estimate	Conservative with respect to DNBR	Overall plant behaviour	Remark
Emergency power mode		no	no	
Single failure F1A or F1B		no	no	no impact on event
Maintenance		not relevant	no	no impact on event
Consequential failure		no	no	
Considered control and limitation systems (event specific):				
• Partial trip		no	no	
• RCP [RCS] temperature control		no	no	
• RCP [RCS] pressure control				
• Pressuriser normal spray		yes	no	
• Pressuriser heater		no	yes	
• Pressuriser level control (RCV [CVCS])		no	no	
• Auxiliary spray via RCV [CVCS]		no	no	
• Turbine control		no	no	no change of state till turbine trip
• Steam dump control		no		
• MS relief control		yes	yes	
• SG level control (ARE [MFWS] and AAD [SSS])		yes	yes	no change of state till turbine trip
• Start-up/Shutdown system		no	no	

APPENDIX 14B.2.7 - TABLE 2

Sequence of Events: Single Reactor Coolant Pump Failure

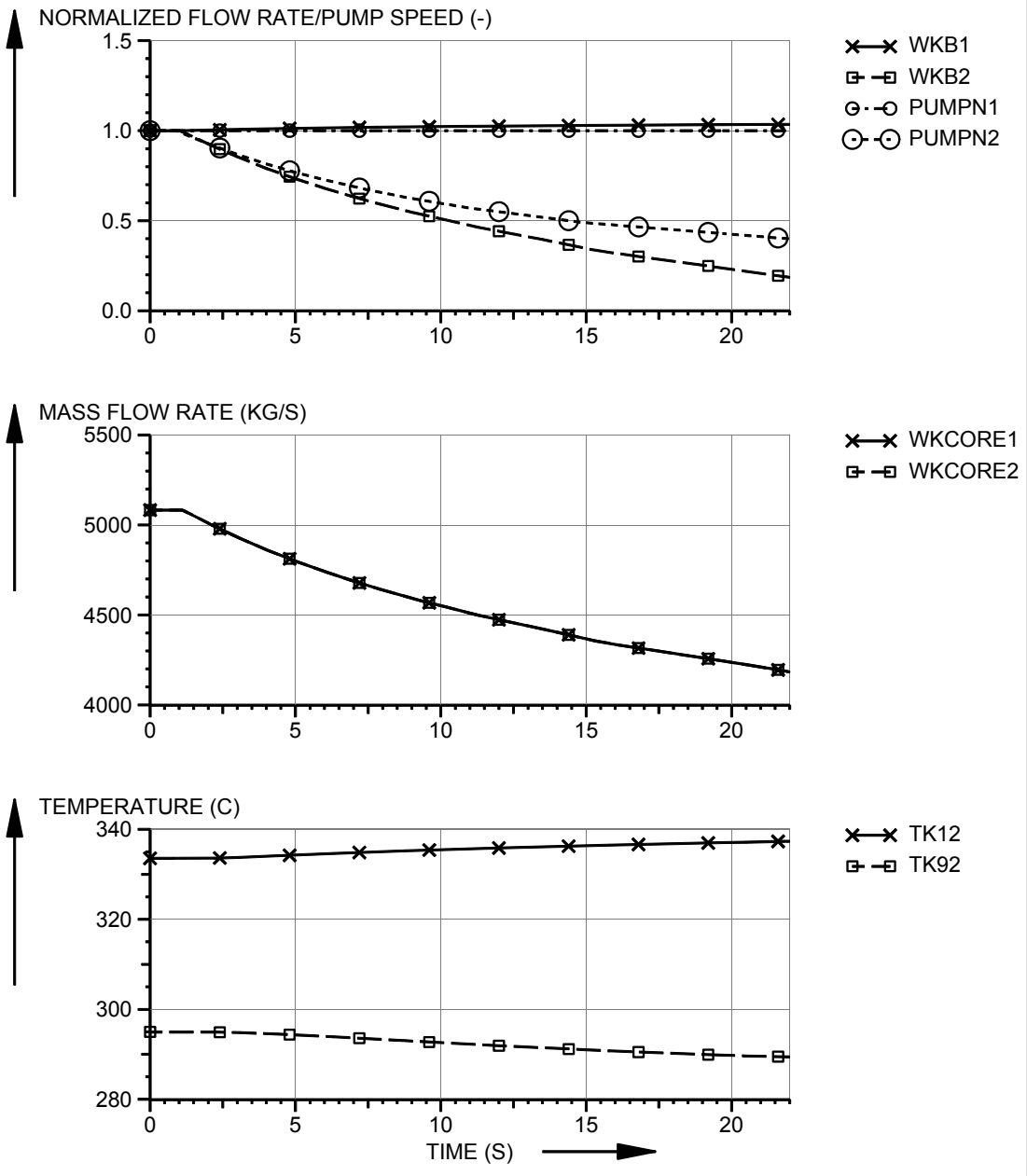
Event		**Time (s)**
Failure of one Reactor Coolant Pump		1
Increase of MFW injection to SG1 in affected loop due to level decrease (normal control function i.e. no change of state)		3
Loop flow < 25%, reactor trip		19.5
Minimum DNBR	= 1.03	
Core flow:	= 16880 (kg/s)	
Primary pressure hot leg	= 156.5 bar	
Reactor power	= 97.3 (%)	
		22
End of calculation		

APPENDIX 14B.2.7 - TABLE 3

Plot	Unit	Variable		Description of Variable
Page 1	(-)	WKB(i)	=	Normalised RC flow rate loop(i)
	(-)	PUMPN(i)	=	Normalised Reactor Coolant Pump speed loop(i)
	(kg/s)	WKCORE(i)	=	RC core flow rate loop(i)
	(°C)	TK1(i)	=	Coolant temperature RPV outlet loop(i)
	(°C)	TK9(i)	=	Coolant temperature SG outlet loop(i)
Page 2	(-)	PGBIST	=	Normalised generator power
	(-)	PBEZ	=	Normalised power (fission + decay + Reactor Coolant Pump power)
	(-)	SQKCB	=	Normalised heat transfer cladding coolant
	(-)	SQKDEB	=	Normalised total heat transfer in steam generators
Page 3	(bar)	PK	=	RC pressure RPV outlet (hot leg)
	(bar)	PDH	=	Pressuriser pressure
	(bar)	PKRCP(i)	=	RC pressure downstream of Reactor Coolant Pump (cold leg) loop(i)
	(m)	DHFIST	=	Real pressuriser water level
	(m)	DHFMESS	=	Measured pressuriser water level
	(m)	DHFSOL	=	Pressuriser water level setpoint
Page 4	(kg/s)	WDE(i)	=	Steam flow rate at outlet of SG(i)
	(kg/s)	WSP(i)	=	Main feedwater flow rate of SG(i) (kg/s)
		WNSP(i)	=	ASG [EFWS] flow rate of SG(i)
	(m)	DEFIST(i)	=	Real steam generator water level of SG(i)
	(m)	DEFS(i)	=	small range measured SG water level of SG(i)
Page 5	(bar)	PDE(i)	=	Steam pressure at top of SG(i)
	(bar)	PFDSOL(i)	=	Setpoint of MS relief valves of SG(i)
	(bar)	PDAF1(i)	=	Setpoint of “MS pressure drop rate > max“ of SG(i)
	(kg)	RMWDE(i)	=	Liquid water and steam mass in SG(i)
Page 6	(-)	DURCHSATZ	=	Normalised net core mass flow rate
	(-)	P_BOC	=	Normalised fission and decay power
Page 7	(-)	OFF_BOC	=	Axial offset core
	(-)	DNBR_BOC2	=	DNBR hot channel, FΔH increased, coupled PANBOX-COBRA calculation
	(-)	DNBR_BOC3	=	DNBR hot channel, as DNBR 2 + uncoupled power increase in COBRA calculation for covering uncertainties

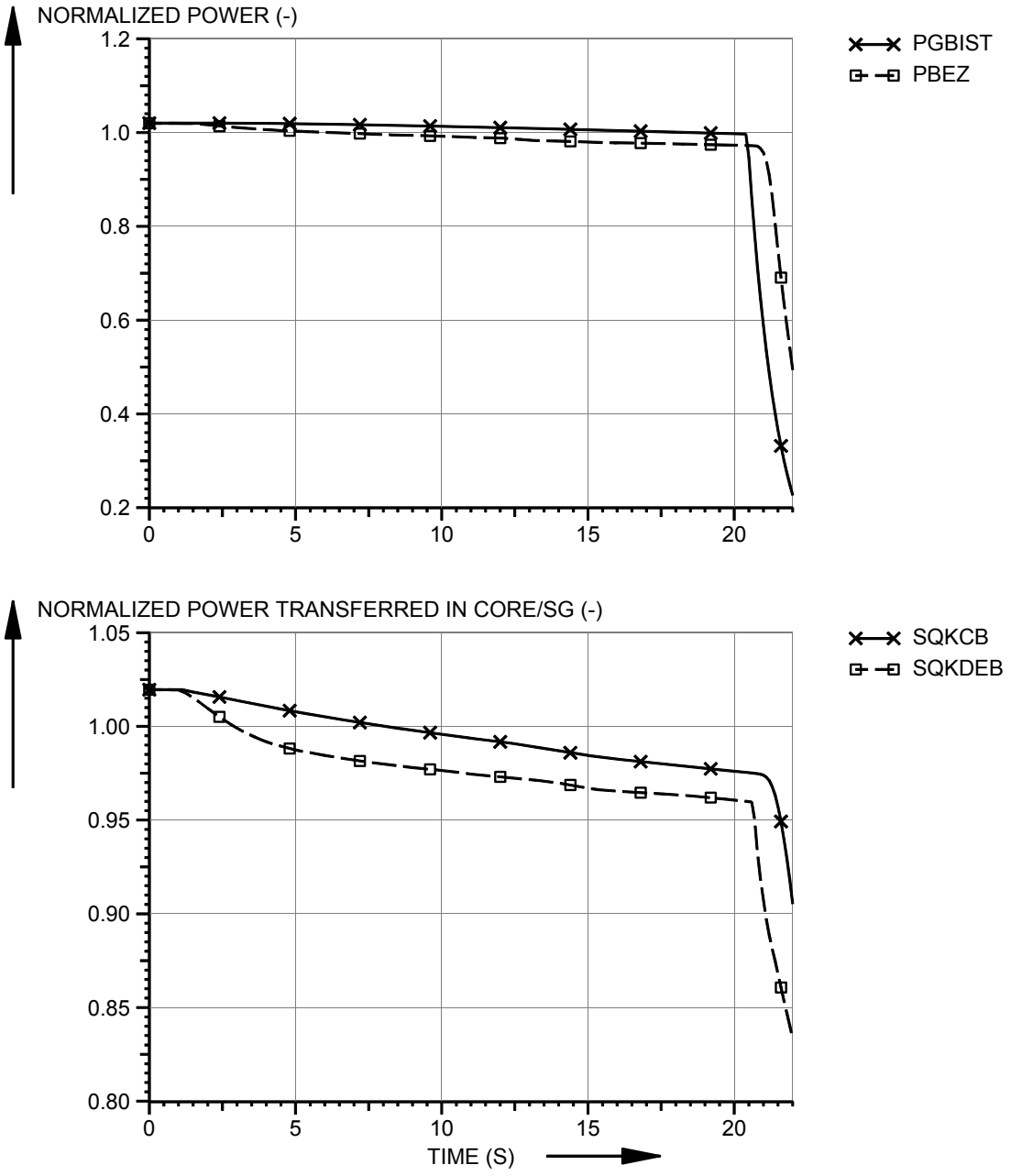
The variables with index ‘’2’’ refer to the affected loop and index ‘’1’’ to the other loops

APPENDIX 14B.2.7 - FIGURE 1



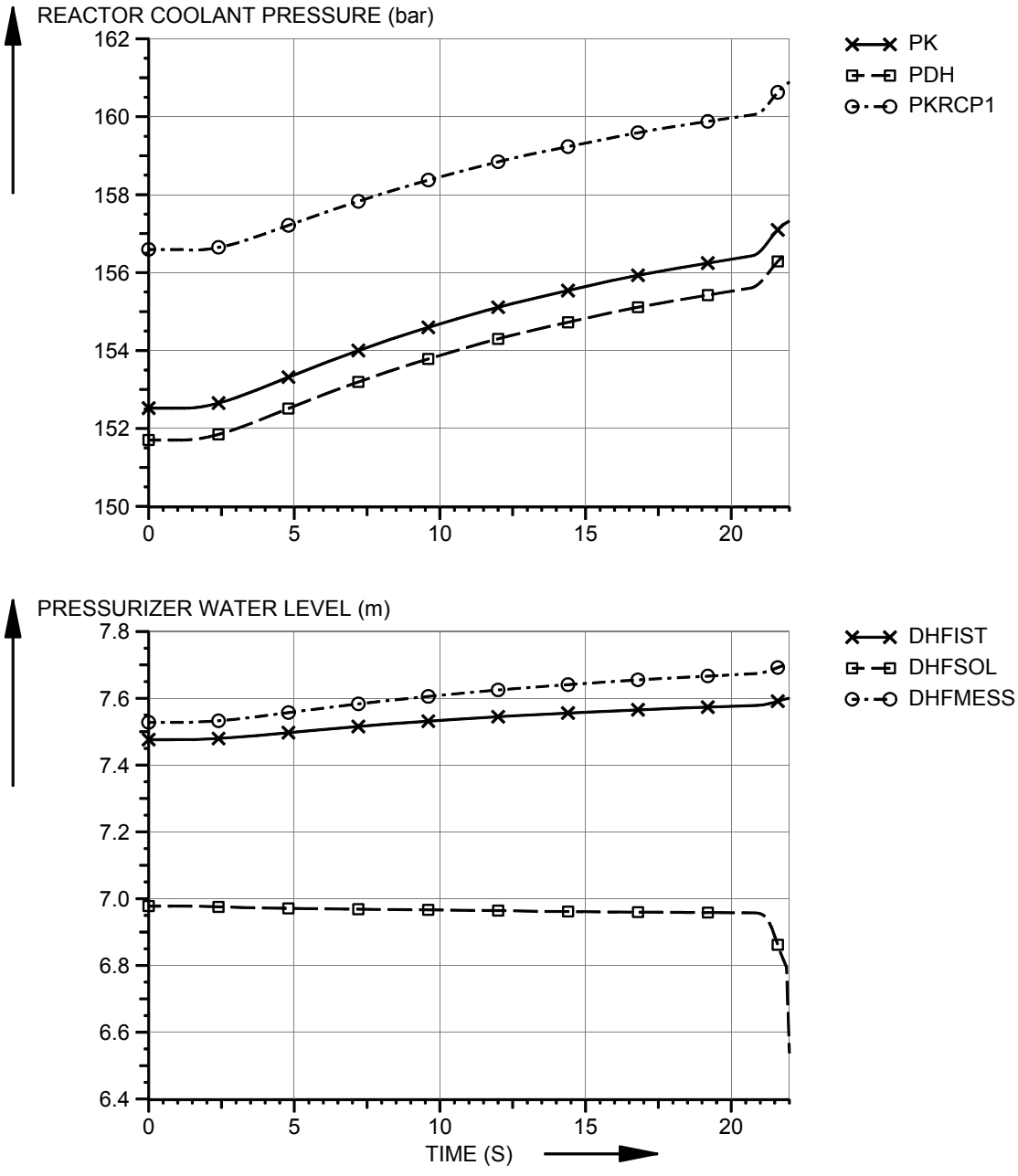
EPR, BASIC DESIGN
BOC CONDITION, ONE REACTOR COOLANT PUMP FAILURE (15.2.7)
CODE PANBOX/COBRA/NLOOP SIEMENS AG / UB KWU DEP NBTT/NA-T

APPENDIX 14B.2.7 - FIGURE 2



EPR, BASIC DESIGN
BOC CONDITION, ONE REACTOR COOLANT PUMP FAILURE (15.2.7)
CODE PANBOX/COBRA/NLOOP SIEMENS AG / UB KWU DEP NBTT/NA-T

APPENDIX 14B.2.7 - FIGURE 3



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BOC CONDITION, ONE REACTOR COOLANT PUMP FAILURE (15.2.7)
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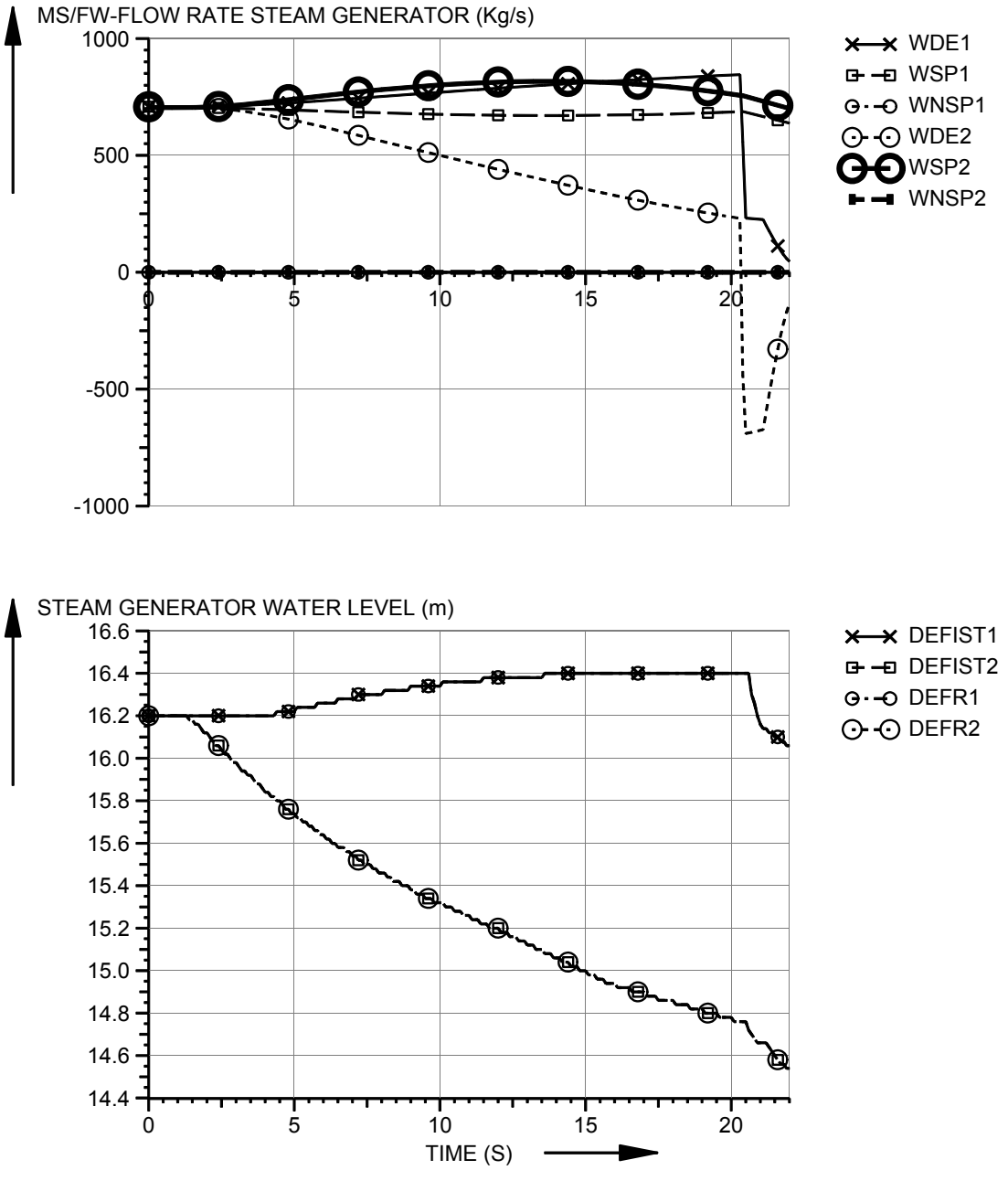
APPENDIX 14B.2.7 - FIGURE 4



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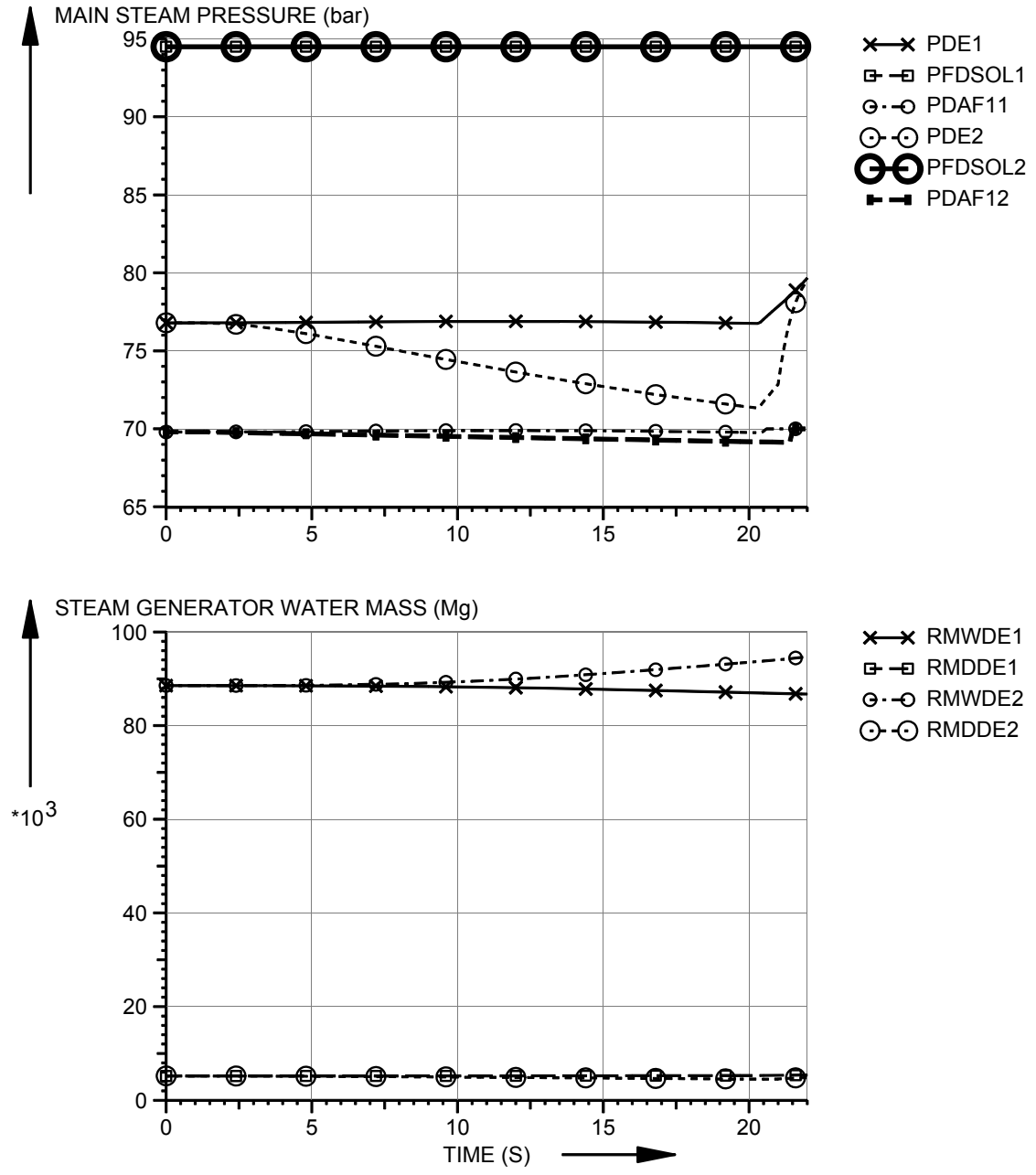
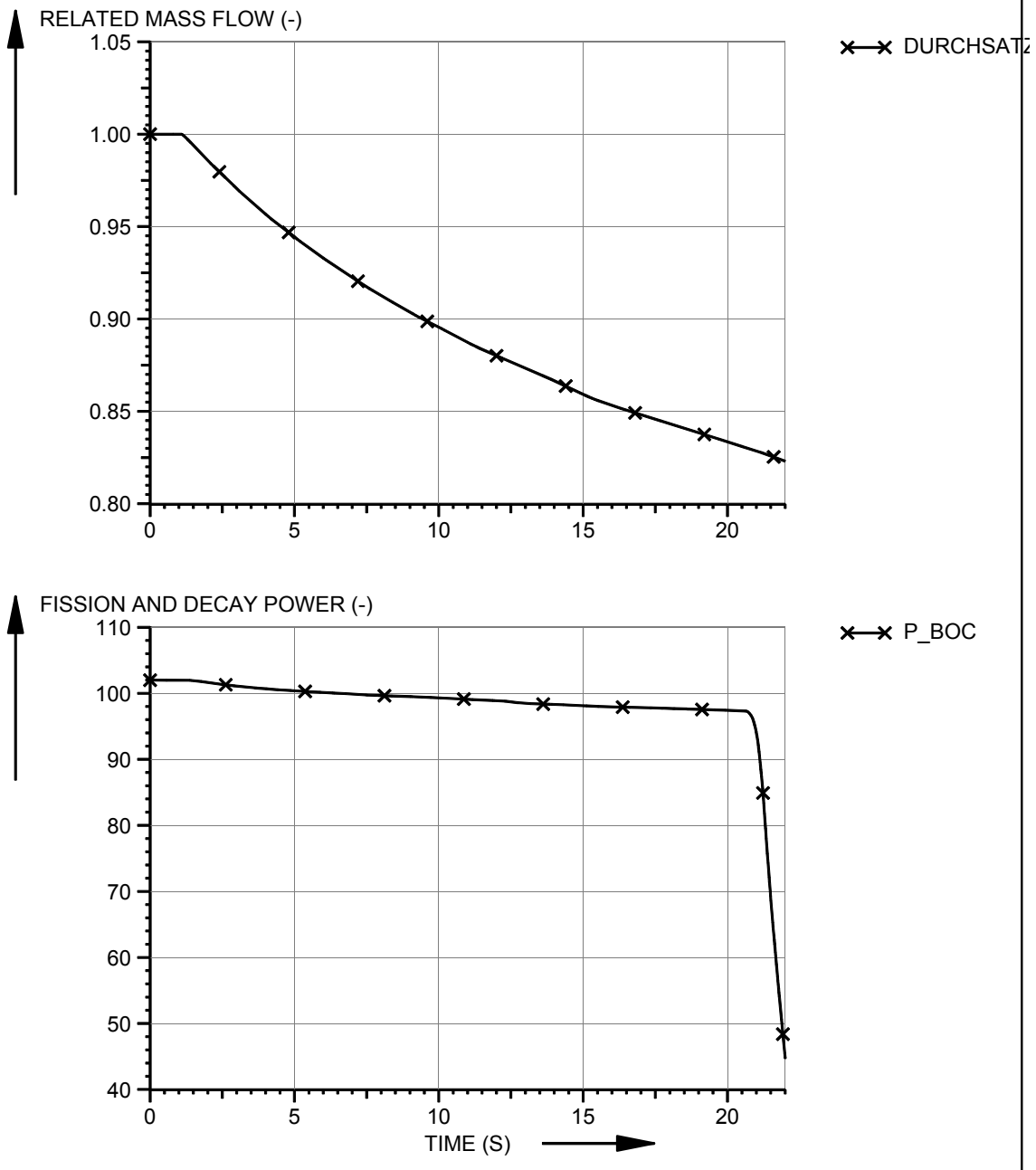
BOC CONDITION, ONE REACTOR COOLANT PUMP FAILURE (15.2.7)
CODE PANBOX/COBRA/NLOOP SIEMENS AG / UB KWU DEP NBTT/NA-T

APPENDIX 14B.2.7 - FIGURE 5



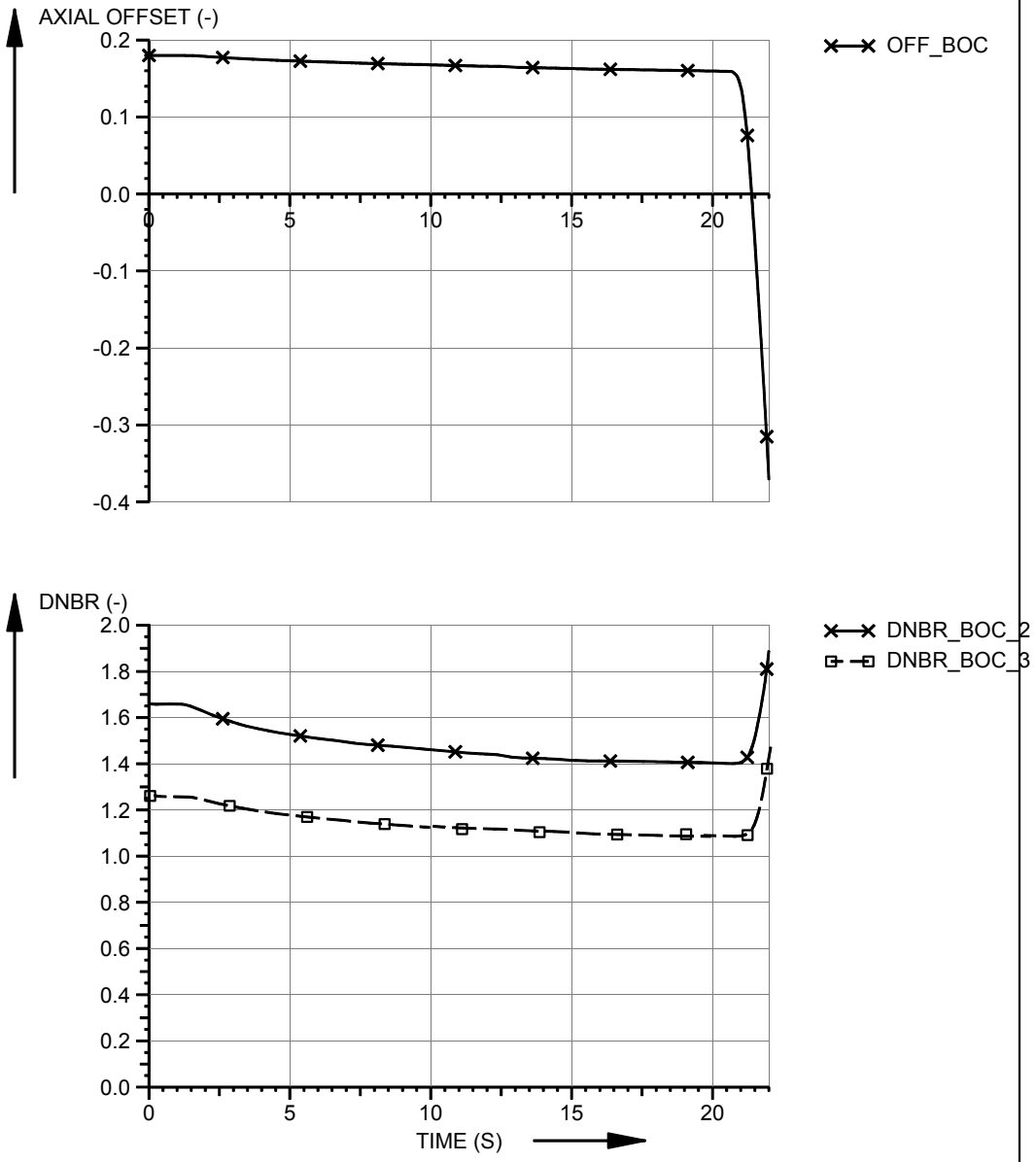
EPR, BASIC DESIGN
BOC CONDITION, ONE REACTOR COOLANT PUMP FAILURE (15.2.7)
CODE PANBOX/COBRA/NLOOP SIEMENS AG / UB KWU DEP NBTT/NA-T

APPENDIX 14B.2.7 - FIGURE 6



EPR, BASIC DESIGN
BOC CONDITION, ONE REACTOR COOLANT PUMP FAILURE (15.2.7)
CODE PANBOX/COBRA/NLOOP SIEMENS AG / UB KWU DEP NBTT/NA-T

APPENDIX 14B.2.7 - FIGURE 7



EPR, BASIC DESIGN
BOC CONDITION, ONE REACTOR COOLANT PUMP FAILURE (15.2.7)
CODE PANBOX/COBRA/NLOOP SIEMENS AG / UB KWU DEP NBTT/NA-T

2.8. REACTOR COOLANT PUMP SHAFT BREAK OR LOCKED ROTOR (PCC-4)

This event is analysed in reactor state A only.

2.8.1. Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure or shaft break of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is assumed to be rapidly reduced, i.e. reduced to zero within 1 to 4 seconds, leading to initiation of reactor trip on a low flow signal. Experience for shaft breaks shows that the flow reduction to zero in the affected loop occurs in 2 to 3 seconds. In case of a locked rotor partial trip would be actuated on low pump speed but this is ignored: thus reactor trip is assumed to occur at the same time for pump shaft break and locked rotor.

Following initiation of reactor trip, heat stored in the fuel rods continues to be transferred at a reduced rate if there is density feedback or at a constant rate if density feedback is zero, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, firstly because the reduced flow results in a decreased tube side heat transfer and secondly because the temperature of the reactor coolant in the tubes reduces while that on the shell side increases (turbine steam flow is reduced to zero on plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressuriser and a pressure increase throughout the reactor coolant system. The insurge into the pressuriser compresses the steam volume, actuating the automatic spray system and possibly opening the pressuriser safety valves which are designed for the relevant conditions.

Note that in the calculation of the maximum primary pressure, the beneficial effect of the spray system is not considered.

Controlled and Safe Shutdown State

It must be demonstrated that the controlled state can be reached using automatic F1A functions only, and that the safe shutdown state can be reached using F1A and F1B functions only, without exceeding safety and decoupling criteria as follows:

- Safety Criteria:
  - Radiological limits for PCC-3/PCC-4 events
- Decoupling Criteria:
  - The peak cladding temperature must remain below 1482°C
  - Not more than 10% of fuel rods should enter DNB.

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

- Reactor Trip:
  - The reactor trip is initiated from the following signal: “Coolant Loop flow < 25%”

- Systems:
  - 4 MS relief trains
  - 4 ASG [EFWS] trains
- Signals:
  - Actuation of MS relief trains at SG pressure of 93 bar
  - Actuation of ASG [EFWS] on “SG water level < 8 m”.

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

- 4 MS relief trains for cooldown to RHR connection conditions (manual action).
- 4 ASG [EFWS] trains for feedwater supply to the SGs (automatic or manual action)
- 2 RBS [EBS] trains for boration

2.8.2. Methods and Assumptions

2.8.2.1. Methods

The calculation of minimum DNBR and the fraction of fuel rods in DNB is performed using the PANBOX/COBRA-3CP code.

The aim of the analysis is to determine the number of rods entering in DNB starting from DNBR at the LCO condition, DNBR = 1.66 (1.26 including uncertainties).

The way the LCO DNBR value is obtained is described in section 2.5.1 of this appendix. The method of calculation of the DNBR transient (DNBR2 and DNBR3) is identical to that described in section 2.5.1 of this appendix.

The number of fuel rods which may experience DNB is calculated taking into account the probability distribution associated with DNBR calculation uncertainties.

Important phenomena and qualification of the models used in PANBOX/COBRA

The fault transient belongs to the family of transients comprising complete or partial loss of forced primary reactor coolant flow.

Phenomena

The phenomena to be addressed are basically the same as those identified in section 2.5.1 and 2.7 of this appendix. Therefore the arguments regarding code qualification presented there are also applicable to the current transient.

The ability to model phenomena specific to the present case, particularly backflow in the affected loop was verified by analysis of a corresponding event in the KKG/BAG NPP.

Based on the validation experience a dedicated analysis of the overall plant behaviour is not considered to be needed.

2.8.2.2. Initial and Boundary Conditions

The main initial and boundary conditions such as reactor power, core inlet temperature, primary pressure etc allow for dead-bands, total uncertainties and maximum typical control system deviations. The assumed coolant flow corresponds to the thermal design flow.

The initial conditions are selected to be the most conservative conditions with respect to the initial DNBR during steady state operation. Based on the results of the bounding design study for DNBR LCO i.e. 'Loss of non-emergency AC power to the plant auxiliaries' (see section 2.5.1 of this appendix) the initial DNBR for the 3D kinetics simulation is taken as 1.66, corresponding to a DNBR LCO value of 1.26 (including uncertainties).

The main initial and boundary conditions are listed in Appendix 14B.2.8 - Table 1.

The analysis assumes a pump coast down time of 1 second.

Reactor Trip

A low loop coolant flow trip set-point of 25% (including measurement uncertainties) is assumed. A conservative delay between set-point actuation and the beginning of rod drop is assumed.

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

- The RCCA drop characteristic, i.e. RCCA position = t (time), as described in Appendix 14B.0.2 – Table 7
- RCCA worth as a function of RCCA position calculated by PANBOX on the basis of the actual 3D power distribution

Moderator Temperature/Density Coefficient

Based on the fuel management schemes envisaged, the lowest absolute value of the moderator temperature coefficient is assumed, since this results in a low density feedback and hence a maximum hot-spot heat flux in the initial part of the transient when the lowest DNBR is reached.

The density reactivity feedback is a code result obtained assuming an unfavourable top peaked core initial power profile with an axial offset of 18% and an enveloping Doppler coefficient.

Doppler Coefficient

This coefficient is assumed to have its maximum absolute value including uncertainties, to minimise negative reactivity feedback

Evaluation of the Hot Spot Temperature (Transient)

The zirconium-water reaction is not considered relevant because the clad temperature remains below the permissible limit. The pellet clad gap transfer coefficient considered in the COBRA code is adjusted so as to obtain an average value for α-gap of 9700 W/m^2*K.

Choice of Single Failure and Preventative Maintenance

- Single failure

  No single failure is postulated since a single failure would have no influence (e.g. a stuck rod would be irrelevant since RT becomes effective only after the occurrence of minimum DNBR.)

- Preventative maintenance

  For the analysis up to the controlled state preventative maintenance is ignored as it has no impact on the minimum DNBR.

For the analysis from controlled to safe shutdown state, assumptions for the reference cases (see Section 2.8.3.2 of this appendix) are applicable.

2.8.3. Results and Conclusions

2.8.3.1. Initiating Event to Controlled State

Key parameters are plotted in Appendix 14B.2.8 - Figure 1/1.

The plotted parameters are described in Appendix 14B.2.8 - Table 2.

The minimum DNBR is reached before the rods begin to drop after generation of the RT signal. Appendix 14B.2.8 Figure 1/1 shows that the design DNBR criterion is exceeded (DNBR3 < 1.0): consequently the number of fuel rods which may enter DNB must be calculated.

The number of fuel rods entering into DNB is calculated as 0.6% which is less than the safety limit of 10%.

The plant parameters relevant to DNBR such as pressure and temperature remain practically unchanged until the occurrence of the minimum DNBR and therefore are maintained constant at their initial values in the calculation.

The maximum cladding temperature is 835°C which is well below the limit value.

The controlled state, i.e.

- core subcritical (reactivity < 0)
- core power removed by VDA [MSRT]s and ASG [EFWS]

is achieved without violating the decoupling criteria.

The core is clearly subcritical after RT, and heat removal requirements are identical to those after any RT.

2.8.3.2. From the Controlled State to the Safe Shutdown State

This phase of the transient is not analysed explicitly since it is covered by analyses of other events (reference cases). The table below defines the reference cases demonstrating safe shutdown with compliance with the three criteria subcriticality, decay heat removal by

RRA/RRI/SEC [RHRS/CCWS/ESWS] or LHSI/RRI/SEC [LHSI/CCWS/ESWS] and activity release/ barrier integrity within PCC limits:

Criteria	Reference case	Remark/Reason
Subcriticality	Uncontrolled boron dilution	Rod worth is sufficient considering stuck rod
Maximum Activity release	Section 2.3.1 of this appendix LBLOCA	For the LOCA case activity release 10% core damage is assumed.
Heat removal	Feedwater system pipe failure	In the reference case only one train is available for cooldown.

APPENDIX 14B.2.8 - TABLE 1 (1/4)

Definition of Conditions and Assumptions for Safety Analyses
(Reactor Coolant Pump Locked Rotor or Shaft Break)

Event	Reactor Coolant Pump Locked Rotor / Reactor Coolant Pump Shaft break
Event Category (PCC/RRC):	PCC-4
Family of Events:	Primary coolant flow reduction, decrease in primary heat removal
Safety Criteria:	Radiological limits for PCC-3/PCC-4 covered by LBLOCA
Decoupling Criteria:	Not more than 10% of rods in DNB, PCT < 1482°C
Purpose of Analysis	Analyses to determine min DNBR
Safe shutdown conditions: controlled state safe shutdown:	 Hot shutdown with removal via MS relief valves Connection to LHSI/RHR system

APPENDIX 14B.2.8 - TABLE 1 (2/4)

Definition of Conditions and Assumptions for Safety Analyses
(Reactor Coolant Pump Locked Rotor and Reactor Coolant Pump Shaft Break

Initial Conditions	Best estimate	Conservative value for DNBR	Remark
Reactor/turbine power	100/100%	102/code res.	
Thermal reactor power	4900 MW	4998MW	
Reactor cooling pumps power		30.59 MW	
Thermal steam generator power (per SG)		1256.9 MW	
Initial insertion of control rods		ARO	
□ Void fraction at hot channel		0.188	At DNBR LCO = 1.66
Outlet		0.303	At DNBR = 1.26
□ Initial Axial Offset		18%	Core avg.
		15%	Hot channel, DNBR = 1.26
F∆H at DNBR LCO		1.745	
F∆H at DNBR = 1.26		1.919	
Initial local power density		≤506 W/cm	hot channel, uncoupled
Reactor cooling flow (total)	22241kg/s	code result	Thermal design
Total core bypass flow	3.4%	5.5%	hot RPV head bypass assumed in analysis: Bypass = 7%
Core/RPV bypass	5%	7%	
Coolant mixing for core inlet temp. distribution		not relevant	
Average reactor coolant temperature	311.25°C	311.25 (+2.5)°C	for DNBR calc. core inlet temp. is 295 °C
Pressuriser pressure	155 bar	155 (-2.5)bar	
Pressuriser level	6.97m	not relevant	
Feedwater/Main steam flow/SG	694. kg/s	not relevant	
Feedwater temperature	230 °C	not relevant	
Steam generator water level	16.25 m	not relevant	
Steam generator water mass	87.6 Mg	not relevant	
Pressure in steam generator	74.6 bar	not relevant	

)

APPENDIX 14B.2.8 - TABLE 1 (3/4)

Definition of Conditions and Assumptions for Safety Analyses
(Reactor Coolant Pump Locked Rotor and Reactor Coolant Pump Shaft Break)

Boundary Conditions (kinetics and reactivity)	Best estimate	Conservative with respect to DNBR		Remark
Burn-up state BOC/EOC		BOC		
Bank reactivity Initial bank reactivity acc. to rod pos. Partial trip Reactor trip		 not relevant not relevant 10600 pcm at HSB		 rod insertion versus time see Appendix 14B.0.2 – Table 7, rod worth acc. PANBOX result
Control rod dropping time	2.5 s	3.5 s		
Moderator R-feed back		-9.6 pcm/°C		at time 0s corresponding to 0.038 10^3 pcm/gcm^3
Void reactivity		code result		
Fuel R-feed back		- 3.6 pcm/°C		at time 0s
Boron R-coefficient		not relevant		
Boron concentration		not relevant		
Decay power	ORIGEN/S	ORIGEN/S*fact.		Fact. is time depending i.e. between 1.1 and 1.2
Delayed neutrons ($\Sigma\beta$)		730.5 pcm		

APPENDIX 14B.2.8 - TABLE 1 (4/4)

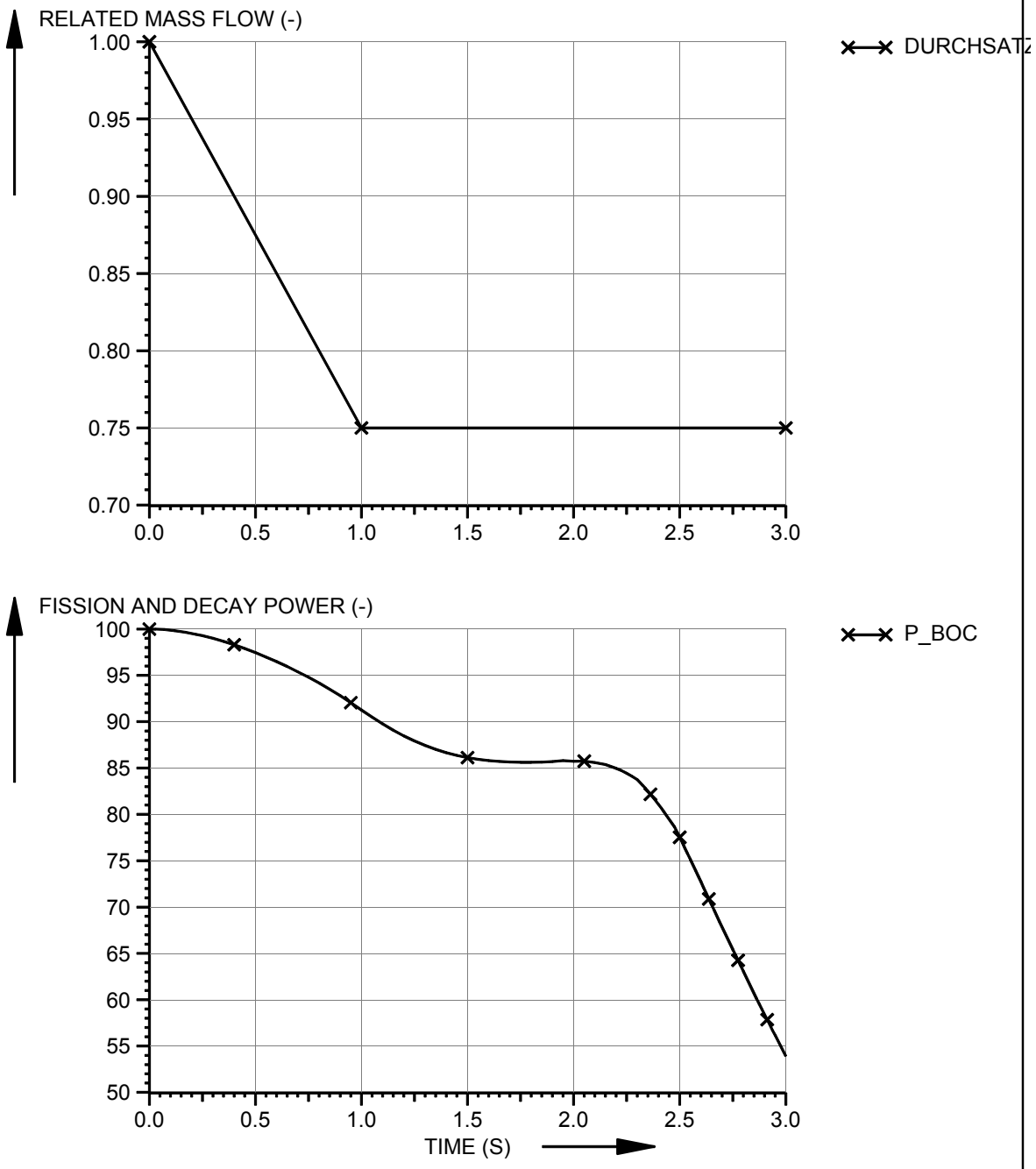
Definition of Conditions and Assumptions for Safety Analyses
(Reactor Coolant Pump Locked Rotor and Reactor Coolant Pump Shaft Break)

Boundary Conditions (failure assumptions)	Best Estimate	Conservative with respect to DNBR		Remarks
Emergency power mode		no		
Single failure F1A or F1B		no		no impact on event
Maintenance		not relevant		
Consequential failure		no		
Consideration of control and limitation systems (event specific):				
• Partial trip		not relevant		
• RCP [RCS] temperature control		not relevant		
• RCP [RCS] pressure control		not relevant		
• PZR normal spray		not relevant		
• PZR heater		not relevant		
• Pressuriser level control (RCV [CVCS])		not relevant		
• Auxiliary spray via RCV [CVCS]		not relevant		
• Turbine control		not relevant		no change of state till turbine trip
• Steam dump control		not relevant		
• MS relief control		not relevant		
• SG level control (ARE [MFWS] and AAD [SSS])		no		no change of state till turbine trip
• Startup/shutdown system		no		

APPENDIX 14B.2.8 - TABLE 2

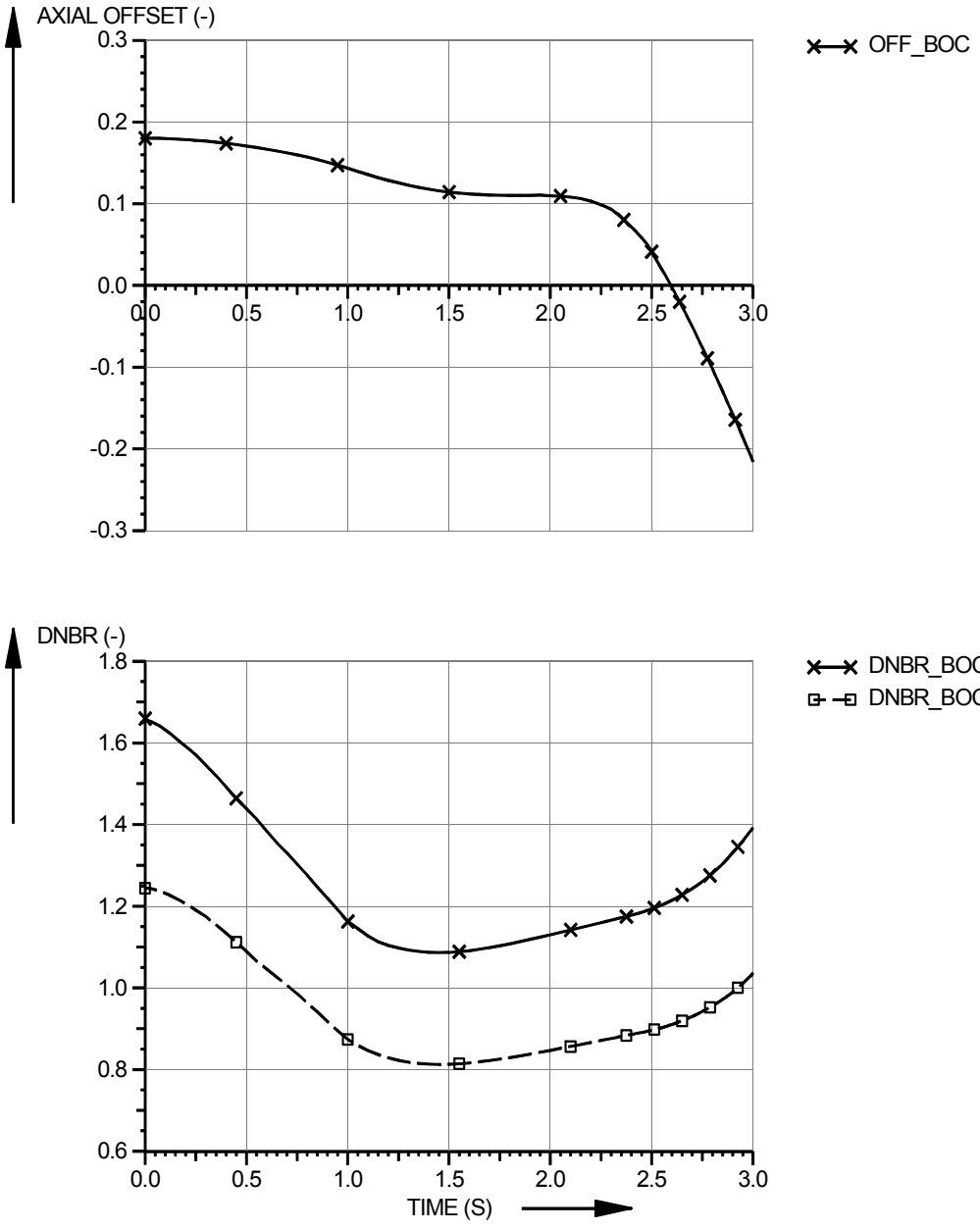
Plot	Unit	Variable	Description of Variable
Page 1	(-)	DURCHSATZ	= Normalised net core mass flow rate
	(-)	P_BOC	= Normalised fission and decay power
Page 2	(-)	OFF_BOC	= Axial offset
	(-)	DNBR_BOC2	= DNBR hot channel, FΔH increased, coupled
	(-)	DNBR_BOC3	= DNBR hot channel, as DNBR2 + uncoupled uncertainties

APPENDIX 14B.2.8 - FIGURE 1



EPR, BASIC DESIGN
BOC CONDITION, EMERGENCY POWER MODE (SHORT TERM REG. MIN. DNBR)
CODE PANBOX/COBRA SIEMENS AG / UB KWU DEP NBTT

APPENDIX 14B.2.8 - FIGURE 2



EPR, BASIC DESIGN
BOC CONDITION, ONE REACTOR COOLANT PUMP SHAFT BREAK
CODE PANBOX/COBRA SIEMENS AG / UB KWU DEP NBTT