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SUB-CHAPTER 16.1 – RISK REDUCTION ANALYSIS (RRC-A)

0. INTRODUCTION AND SAFETY OBJECTIVES

0.1. REQUIREMENTS

In the EPR defence-in-depth approach, the risk reduction category RRC-A is introduced to complement the deterministic design basis analysis (PCC categories) by considering a set of Design Extension Conditions (DECs) due to multiple failure events. The analysis of DECs is performed using both deterministic and probabilistic considerations and leads to the identification of additional safety features (or ‘RRC-A features’), which make it possible to prevent the occurrence of severe accident in these complex situations. Additional RRC-A features are classified consistent with EPR safety classification principles (see Sub-chapter 3.2).

An **RRC-A feature** is a specific system, device or function used to mitigate event sequences that are not covered by PCC analysis. Thus, an RRC-A feature is used to manage multiple-failure sequences, which could result from the loss of an F1 classified safety function (mainly due to common cause failures) or from a combination of independent events affecting the fuel in the reactor building or in the fuel building.

The RRC-A studies lead to a manageable list of **RRC-A sequences** by grouping event sequences with similar functional characteristics and for which core damage is reduced by the implementation of the same RRC-A feature.

A list of DECs was determined during the Basic Design Phase of the project, using deterministic considerations and experience feedback. This list is provided in section 0.2. Afterwards, this list has been consolidated mainly by using the GDA Step 3 Level 1 and Level 2 PSA for internal initiating events.

For the purpose of consolidating the RRC-A list with the PSA, each RRC-A sequence core damage frequency (CDF) is checked against RRC-A probabilistic criteria depending on the consequence for containment integrity:

- RRC-A sequence with potential early containment failure (C1): CDF near 10^{-8}/ry
- RRC-A sequence with potential late containment failure (EVU [CHRS] unavailable) (C2): CDF between 10^{-7}/ry and 10^{-8}/ry
- RRC-A sequence with severe accident mitigation features available (i.e. no potential containment failure) (C3): CDF near 10^{-7}/ry

These internal targets can be seen as decoupling criteria, which are defined in line with the probabilistic safety objectives of the EPR (see Chapter 15). Indeed, it is convenient to introduce a decoupling criterion, which can be used in the analysis of each RRC-A sequence. This internal target should be considered as a guideline value rather than a strict limit. If met, it offers a strong indication that the global probabilistic EPR design targets will be met at the end of the process.

The RRC-A sequences are studied in a deterministic manner through best estimate **RRC-A accident analysis,** to analyse the design of RRC-A features. In order to reduce the number of transient calculations to a manageable number, representative scenarios are chosen either by considering their contribution to PSA results for automatic RRC-A features, or on a conservative basis for manual RRC-A features (e.g. scenarios that lead to the shorter maximum acceptable delay time for operator action).

0.2. LIST OF RRC-A FEATURES AND SEQUENCES

RRC-A Feature	RRC-A Sequence
ATWS signal (automatic RBS [EBS] actuation)	ATWS through the mechanical blocking of the rods (state A)
Diverse Reactor Trip signals	ATWS due to the failure of the Protection System (RPR [PS]) (state A)
Manual switch to SBO diesel generators	Total loss of offsite power and failure of the four Emergency Diesel Generators (state A)
Manual Feed & Bleed	Total loss of the water supply to the steam generators (state A)
Automatic switchover of the cooling of the LHSI pumps 1 and 4 to their diverse cooling system	Total loss of the cooling chain and failure of the stand still seal system (DEA [SSSS]) leading to a loss of primary coolant (state A)
Diverse automatic activation of Safety injection and partial cool-down	LOCA up to 20 cm^2 and failure of the PS for the Safety Injection signal activation (state A)
Manual activation of fast secondary cool down	LOCA up to 20 cm^2 without MHSI (state A)
Cooling of the IRWST by the EVU [CHRS]	LOCA up to 20 cm^2 without LHSI (state A)
Diverse automatic activation of the RIS [SIS]	Uncontrolled level drop (ULD) and failure of the RPR [PS] for the activation of the RIS [SIS] (state Cb and D)
Manual switchover of the cooling of the LHSI pumps 1and 4 to their diverse cooling system	Total loss of the cooling chain (state D).
Manual re-supply of the ASG [EFWS] tanks	Total loss of the ultimate heat sink for 100 hours (states A to C)
Manual actuation of the LHSI for injection	LOCA without MHSI (states C and D)
Manual actuation of RIS/RRA [SIS/RHRS] train	LOCA outside containment on RIS/RRA [SIS/RHRS] train (states C and D)
Manual isolation of RIS/RRA [SIS/RHRS] train	LOCA outside containment on RIS/RRA [SIS/RHRS] train and failure of the automatic isolation signal (states C and D)
Automatic activation of the RBS [EBS]	Non-isolatable homogenous boron dilution outside the Volume Control Tank (VCT) and failure of the operator's actions (states Cb and D)
Third Spent Fuel Pool Cooling diverse train	Loss of the two main trains of the Spent Fuel Pool Cooling System (PTR [FPCS]) during core refuelling (state F)

1. USE OF PSA FOR RRC-A STUDIES

1.1. BACKGROUND

The approach presented in this section consists of checking the list of EPR RRC-A sequences (see section 0.2) by using the GDA Step 3 Level 1 and Level 2 PSA. This means that the combinations of faults and system failures or unavailabilities, which are not addressed in PCC analysis and which are identified as RRC-A sequences, are studied probabilistically to determine their related core damage frequency (and if needed their related large early release frequency).

The grouping of the fault sequences is assessed against probabilistic design targets, as defined in section 0.1.

1.2. INPUT DATA

The analysis of RRC-A sequences is carried out based on the UK EPR Level 1 (and Level 2 if needed) GDA Step 3 PSA model presented in Chapter 15 of the Step 3 UK EPR GDA PCSR.

The long-term RRC-A sequences are analysed on the basis of specific sensitivity analyses also based on the GDA Step 3 UK EPR Level 1 PSA.

1.3. METHODOLOGY AND ASSUMPTIONS

1.3.1. Characterisation of an RRC-A feature using the PSA

From a PSA point of view, the RRC-A features are subject to the following considerations:

1) An existing RRC-A feature is required to reduce the core damage frequency (CDF) from specific groups of PSA sequences. This means that the associated CDF would not meet the probabilistic objectives without the implementation of the RRC-A feature. The CDF reduction requirement is assessed according to the possible consequence of the multiple failure sequence for containment integrity.

2) The use of PSA for RRC-A analysis is limited to the core damage and to the modelling performed in the PSA. For defence-in depth, a specific RRC-A feature (the diverse third PTR [FPCS] train that is used following loss of the two main PTR [FPCS] trains to limit the risk of steam discharge in the fuel building) is added to the list to cover accident sequences in the spent fuel pool during refuelling. Another RRC-A feature (automatic boration by RBS [EBS]) is also introduced to limit the consequences of homogeneous dilution of the boron in the primary coolant, which are not yet assessed in the PSA. As a consequence, these RRC-A features are not assessed using PSA and not compared to probabilistic design criterion, as described in section 0.1.

3) The PSA may also show that systems, devices or functions introduced into the design for other reasons[1] (for example, RCV [CVCS], start-up and shutdown system, Main Steam Bypass, or turbine bypass) significantly contribute to meeting the probabilistic objectives. However, these are not treated as possible RRC-A features.

1.3.2. Description of the RRC-A methodology

The method of examining RRC-A sequences using the PSA is divided into the following stages:

Stage 1:

This stage consists of identifying the PSA system missions associated with RRC-A features - specific non-F1 systems or specific actions used in the Level 1 PSA. Examples include EVU [CHRS], SBO-diesels (Station Blackout), feed-and-bleed mode, etc.

Stage 2:

This stage consists of identifying the ‘functional sequences’ (or ‘RRC-A sequences’) associated with potential RRC-A features.

A functional sequence consists of a group of Level 1 PSA sequences with similar functional characteristics and for which the core damage frequency is reduced by the implementation of the same RRC-A feature.

At the end of this stage, each potential RRC-A feature is associated with a Level 1 PSA functional sequence.

Stage 3:

This stage consists of determining the actual RRC-A features that are required to reduce the core damage frequency.

To achieve this objective:

- firstly, the most onerous consequence for the functional sequence is determined (i.e. C1, C2, C3, see section 0.1) considering the potential challenge of the containment integrity in case of core damage;

- secondly, the functional sequence CDF with the RRC-A feature omitted (failure probability of the RRC-A feature set to unity) is compared to the internal probabilistic targets given in section 0.1 for the determined consequence.

If the internal target is met without the implementation of the RRC-A feature, the requirement for the potential RRC-A feature in the reduction of the CDF is not considered as proven. The conclusion is that the potential RRC-A feature does not belong in the list.

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[1] Such systems are introduced in the design to cope with normal transients (PCC1). They are considered in the PSA if it has been shown that they may be used in case of fault conditions.

1.3.3. Scope of analysis

This section deals with the examination of the list of RRC-A sequences of section 0.2 using the method presented in section 1.3.2. As the results of the PSA presented in Chapter 15 show that the probabilistic objectives are achieved, no other RRC-A feature need to be defined and examined here.

The support studies associated with the enhancement of RRC-A features are presented in section 3.

Examination of the RRC-A list is based on analyses using the GDA Step 3 UK EPR Level 1 PSA model for internal events. PSA sequences associated to Loss Of Off-site Power (LOOP) and Loss of Ultimate Heat Sink (LUHS) considered as external hazards are also studied because the PSA functional sequences are similar to internal event sequences.

Clearly, only the sequences analysed in the PSA model can be examined using this method. Sequences which are not analysed in the PSA cannot necessarily be considered not to belong to the RRC-A list: it can only be inferred that the probabilistic analysis itself does not provide any conclusive information about such sequences. This is the case for the sequences, ‘Non-isolatable homogenous boron dilution outside the Volume Control Tank (VCT) and failure of the operator's actions (states Cb and D)’ and 'Loss of two PTR [FPCS] trains at the time of shutdown for refuelling (state F)'.

1.4. CHARACTERISTICS OF RRC-A FEATURES

The characteristics of the RRC-A sequences, based on the probabilistic studies, allow the specification of the list of functional sequences and the associated RRC-A features.

The reactor standard states are defined in Chapter 15.

In line with the definition given in section 1.3.1, a functional sequence consists of a group of several Level 1 PSA core damage accident sequences with similar functional characteristics and for which the core damage frequency (CDF) is reduced by the implementation of the same RRC-A feature. The list of the PSA accident sequences belonging to a functional sequence is not provided in this analysis.

The CDF without (failure probability set to 1) and with (failure probability set to its nominal value) the RRC-A feature associated to its functional sequence is given in Section 16.1.1 - Table 1

1.4.1. ATWS signal (RBS [EBS] actuation) following rod failure

This RRC-A sequence is concerned with Anticipated Trip Without Scram (ATWS) due to the mechanical seizure of a number of rods in an at-power state (state A). The associated RRC-A feature is automatic boration by the RBS [EBS] (Extra Boration System) activated by the ATWS signal.

After an initiating event, which leads to activation of the Automatic Reactor Trip signal, several rods are assumed to seize. To mitigate the rod seizure, an ATWS signal is generated by the Protection System (RPR [PS]) and this leads to automatic fast boration by the RBS [EBS]. Boration ensures the reactivity of the core is controlled and that no core damage occurs due to failure of reactivity control.

In the PSA, automatic boration is required in case of mechanical blockage of control rods after the following initiating events:

- excessive increase of secondary side steam flow rate ,
- total loss of the normal feedwater system,
- loss of main electrical power grid,
- spurious actuation of pressuriser spray,
- spurious safety injection,
- turbine trip with GCT [MSB] available.

The number of control rods that need to fail to prevent reactor trip is different for different initiating events (see Sub-chapter 15.1).

The RRC-A feature is modelled in the PSA through I&C signals for ATWS actuation by the protection system and a 'boration' mission fulfilled by the two RBS [EBS] trains.

As the core damage would not lead to containment failure, the consequences associated with this functional sequence are C3.

1.4.2. Diverse RT following RPR [PS] failure

This RRC-A sequence is concerned with the failure of Automatic Reactor Trip following the failure of the protection system (RPR [PS]) in the at-power state (state A). The associated RRC-A feature is the automatic actuation of the diverse automatic reactor trip signal (PAS signal).

The failure of the automatic reactor trip following the failure of the protection signal is detected (no opening of the main trip breakers) and the diverse automatic reactor trip signal is activated to switch off the power supply to the control rod drive mechanisms (trip contactors and pilots of the RGL [CRDM]).

In the PSA, diverse reactor trip (RT) is required in the case of RPS failure after the following initiating events:

- excessive increase of secondary side steam flow rate,
- total loss of the normal feedwater system,
- loss of main electrical power grid,
- spurious actuation of pressuriser spray,
- spurious safety injection
- turbine trip with GCT [MSB] available,
- SGTR,
- primary breaks.

The RRC-A feature is modelled in the PSA through I&C signals for RT actuation by the PAS and diverse switching off of power to the RGL [CRDM].

As core damage would lead to potential early containment failure, the consequences associated with this functional sequence are C1.

1.4.3. SBO diesel generators following loss of offsite power and failure of EDGs

This RRC-A sequence is concerned with the Loss Of Offsite Power (LOOP) (due to grid failures or induced by transients) with a recovery period of 24 hours, combined with the total failure of the four Emergency Diesel Generators (EDGs), whilst at-power (state A). The RRC-A features associated with this functional sequence are the two Station Black Out (SBO) diesel generators which supply electrical power to the emergency supply system for the Emergency Feed Water System (ASG [EFWS]), trains 1 and 4 and the EVU/SRU [CHRS/UCWS]. The operator switches to the SBO diesel generators manually.

In the PSA, the SBO diesel generators are required for short-term (< 2 hours, in the case of failure of batteries) and long-term (up to 24 hours, in the case of EDG failure) LOOP events in all modes of reactor operation.

Thanks to diversification measures taken during the design and operation, in the PSA no common cause failure is assumed between the initial conditions of this RRC-A sequence and the potential failure of the SBO diesel generators (in the case of failure of batteries, the SBO diesel generators are started manually by local action).

As the core damage would lead to potential late containment failure, the consequences associated with this functional sequence are C2.

1.4.4. Manual Feed & Bleed following total loss of the water supply to the steam generators

The RRC-A sequence is concerned with the total loss of the steam generator feedwater system when the reactor is at power (state A or B).

In the PSA, this situation corresponds to accident sequences where the residual heat removal function with SG is lost. This function can be fulfilled either by the Main Feedwater System (ARE [MFWS]), the Startup and Shutdown System (AAD [SSS]) or the Emergency Feedwater System (ASG [EFWS])

The most onerous scenario of total loss of the steam generator feedwater system corresponds to a total failure of the main feedwater ARE [MFWS] in state A, including the AAD [SSS] due to a common cause failure with ARE [MFWS], and then the total failure of the emergency feedwater ASG [EFWS] due to I&C failure or internal common cause failure (CCF).

The associated RRC-A feature is the feed-and-bleed mode started by the operator. The feed-and-bleed mode ensures reactor heat removal by the opening of the dedicated pressuriser valves, RCP [RCS] injection with RIS [SIS] and IRWST cooling. The IRWST cooling can be ensured by the RIS/RRA [SIS/RHRS] or EVU [CHRS] connected to their RRI/SEC [CCWS/ESWS] heat exchangers.

As core damage would lead to potential early containment failure, the consequences associated with this functional sequence are C1.

1.4.5. Automatic switchover of the cooling of the LHSI pumps 1 and 4 to their diverse cooling system, following TLOCC (state A) and failure of the reactor coolant pump seals

The RRC-A sequence is concerned with the total loss of cooling chain RRI/SEC [CCWS/ESWS] (TLOCC) leading to a small Loss of Coolant Accident (LOCA) due to the failure of the reactor coolant pump seals (loss of seal injection and thermal barrier). The associated RRC-A feature is the automatic switchover of the cooling of legs 1 and 4 of the Low Head Safety Injection (LHSI) pumps to their diverse cooling system, DEL (Chilled Water System).

In state A, after the total loss of cooling chain, the reactor coolant pumps stop and a failure of the Standstill Seal System DEA [SSSS] would lead to a loss of primary coolant (small LOCA) through the primary pump seals.

The loss of cooling chain leads to the unavailability of the Medium Head Safety Injection (MHSI) pumps and fast secondary cooling is then necessary to depressurise the reactor coolant system (RCP [RCS]). The pressure of the RCP [RCS] is then sufficiently low to ensure an injection of water via the LHSI pumps 1 and 4, which are automatically connected to their diverse cooling channel DEL (Chilled Water System), independently of the RRI [CCWS].

In the Level 1 PSA, this scenario is bounded by a LOCA due to a small break of size lower than 20 cm^2.

As core damage would lead to potential late containment failure, the consequences associated with this functional sequence are C2.

1.4.6. Diverse automatic activation of Safety Injection and partial cool-down following LOCA and failure of the Safety Injection by the PS

This functional sequence involves the combination of a LOCA due to a small primary break at power and the failure of the automatic safety injection activated by the protection system (RPR [PS]). The associated RRC-A feature is the diverse activation of safety Injection and of partial cool-down: automatic actuation by the SAS of the safety injection and resetting of the atmospheric dump valves setpoint of four main steam relief trains, (VDA [MSRT]).

The loss of coolant leads to the depressurisation of the RCP [RCS], which initiates an automatic reactor trip and the safety injection signal by the protection system. Partial cool-down is required in case of safety injection to reach the injection pressure of the MHSI but is not activated on the signal due to the failure of instrumentation and control. The diverse signal activates safety injection and partial cool-down. The primary pressure drops allowing the MHSI pumps to inject into the RCP [RCS].

In the Level 1 PSA, a LOCA due to a small break of size less than 20 cm^2 in state A represents the most onerous scenario.

As core damage would lead to potential early containment failure, the consequences associated with this functional sequence are C1.

1.4.7. Manual activation of fast secondary cool-down following small LOCA and total failure of MHSI

This functional sequence combines a LOCA due to a small primary break at power with the unavailability of all the MHSI pumps. The associated RRC-A feature is the activation by the operator of fast secondary cool-down with all VDA [MSRT] and ASG [EFWS] trains available.

The LOCA leads to the depressurisation of the RCP [RCS], the activation of the Safety Injection signal and partial cool-down. The MHSI is unavailable or cannot inject due to the failure of the partial cooldown. Due to the RCP [RCS] pressure, the LHSI cannot inject. To avoid core uncovery, the RCP [RCS] pressure has to decrease quickly to permit accumulator and LHSI pump injection into the cold legs. RCP [RCS] depressurisation is achieved through fast secondary cool-down.

In the Level 1 PSA, LOCA due to a small break of size less than 20 cm^2 in state A represents the most onerous scenario.

As core damage would not lead to containment failure, the consequences associated with this functional sequence are C3.

1.4.8. Cooling of the IRWST by the EVU [CHRS] following small LOCA and total failure of LHSI

This functional sequence is concerned with the combination of a LOCA due to a small primary break at power with the unavailability of all the LHSI pumps. The associated RRC-A feature is the manual cooling of the IRWST (In-containment Refuelling Water Storage Tank) water by the EVU [CHRS] initiated by the operator.

The LOCA leads to the depressurisation of the reactor coolant system, which initiates the automatic reactor trip and the safety injection (SI) signal. Partial cool-down is activated by the SI signal followed by injection into the RCP [RCS] by the MHSI pumps. Operator action ensures cool-down of the RCP [RCS] to reduce the pressure and connect the LHSI in RHR mode to remove residual heat. The LHSI pumps are unavailable. Removal of residual heat and RCP [RCS] makeup are initially ensured by the steam generators and injection of water into the primary system via the MHSI but the temperature of the IRWST increases. In the long term, the MHSI pumps could fail due to the high temperature of the IRWST and uncovery of the core would be inevitable. The operator should activate the EVU [CHRS] and its dedicated cooling chain to cool the IRWST to an acceptable temperature.

In the Level 1 PSA, LOCA due to a small break of size less than 20 cm^2 in state A represents the most onerous scenario.

As core damage would lead to potential late containment failure, the consequences associated with this functional sequence are C2.

1.4.9. Diverse automatic activation of the MHSI pumps following ULD and failure of Safety Injection signal

This functional sequence combines the Uncontrolled Level Drop (ULD) during RHR operation and the failure of the automatic activation of the MHSI on Low Loop Level (LLL) by the protection system. The associated RRC-A feature is the diverse activation of the MHSI by another signal.

The ULD leads to the activation of the MHSI to ensure the makeup of the RCP [RCS] on the basis of the F1A safety injection (SI) signal LLL. If this signal fails, the Safety Automation System (SAS) ensures the automatic activation of the MHSI using diverse information (RCS Level < MIN1).

As core damage would lead to potential early containment failure, the consequences associated with this functional sequence are C1.

1.4.10. Manual switchover of the cooling for the LHSI pumps 1 and 4 to their diverse cooling system following TLOCC in state D

This functional sequence is concerned with the total loss of the cooling chain TLOCC in state D, (mainly due to the simultaneous failure of the four water-cooling trains RRI/SEC [CCWS/ESWS]). The associated RRC-A feature is the manual switchover from the two LHSI trains 1 and 4 to the diverse cooling system, DEL (Chilled Water System).

In state D, total loss of the cooling chain leads to the tripping of the LHSI pumps operating in RHR mode (on the basis of their component protection signal). As the MHSI pumps are unavailable and the primary system is open, the only way to avoid core uncovery is to use LHSI injection to compensate for loss of primary inventory through steaming.

The RRC-A sequence corresponds to the TLOCC in state D with the success of the automatic trip of the LHSI train in operation and the manual switchover of LHSI pumps 1 and 4 to their diverse cooling system DEL (Chilled Water System). This permits injection by the LHSI pumps despite the total loss of the cooling chain.

As core damage would not lead to containment failure, the consequences associated with this functional sequence are C3.

1.4.11. Re-supply of the ASG [EFWS] tanks following LUHS for 100 hours

This functional sequence combines sequences arising from the total loss of the ultimate heat sink (LUHS) in states A to C over a period of 100 hours without leakage at the primary pump seals. This leads to the depletion of the steam generator water reserves (emergency feedwater system ASG [EFWS]).

The associated RRC-A feature is the re-supply of the ASG [EFWS] tanks. The failure of the operator to implement the re-supply of the tank is not considered because of the long period available (approximately 30 hours) and the simple nature of the operation.

In the case of loss of the cooling chain in states A to C, the removal of residual heat is ensured automatically after the automatic trip of the reactor by the atmospheric dump system (VDA [MSRT]) and by the supply to the steam generators by the ASG [EFWS]. However, the inventory of the ASG [EFWS] tanks is limited and is not sufficient over an extended operating period. The re-supply of the ASG [EFWS] tanks allows residual heat removal to be achieved until the heat sink returns. The re-supply feature consists of two dedicated ASG [EFWS] pumps, backed up by emergency diesel generators and SBO diesels and taking suction into one fire protection (JAC) tank.

This specific RRC-A sequences is calculated on the basis of a 100 hour mission time in the PSA, therefore this is considered as a sensitivity study.

As core damage would not lead to containment failure, the consequences associated with this functional sequence are C3.

1.4.12. Manual activation of the LHSI for injection following ULD or LOCA with failure of MHSI in states C and D

This functional sequence combines the Uncontrolled Level Drop (ULD) or a small LOCA during RHR operation and the mechanical failure of the MHSI trains. The associated RRC-A feature is the manual activation by the operator of the LHSI.

The decrease of the RCP [RCS] level leads to the activation of the MHSI to ensure the makeup of the RCP [RCS] on the basis of F1A safety injection signals. In the event of failure of MHSI, the LHSI is manually started by the operator to provide makeup.

As core damage would lead to late containment failure, the consequences associated with this functional sequence are C2.

1.4.13. Manual actuation of RIS/RRA [SIS/RHRS] train following LOCA outside containment in states C and D

This functional sequence is related to a significant LOCA on the LHSI/RRA [RHRS] during shutdown states caused by a leakage or rupture on the injection lines outside the containment (from 20 cm² up to a double guillotine break on RHR line). The isolation of the break (trip of the RHR train in operation) and the makeup of the RCP [RCS] by the RIS [SIS] are actuated automatically or manually. The LHSI on the non-affected trains in RHR mode is manually actuated.

1.4.14. Manual isolation of RIS/RRA [SIS/RHRS] train following LOCA outside containment in states C and D and failure of the automatic isolation

This functional sequence is related to LOCA on LHSI/RRA [RHRS] during shutdown states caused by a small or significant leakage or a rupture on the injection lines outside the containment.

The leak leads to a loss of primary water inventory and to an increase in the sump level outside the containment of the affected train. The high sump level signal which actuates isolation of the affected train will be reached and leads to the closure of the motor-operated valves on the injection line. If the automatic isolation fails the water level and pressure decrease in the RCP [RCS]. In this case the isolation is performed manually.

As core damage would lead to early containment failure (containment bypass), the consequences associated with this functional sequence are C1.

SECTION 16.1.1 – TABLE 1

Core Damage Frequency of RRC-A sequences
(this analysis is carried out based on the UK EPR GDA Step 3 PSA model)

RRC-A features	RRC-A sequence core damage frequency (/ry)		Cons.
	Without RRC-A feature	With RRC-A feature	
ATWS signal (RBS [EBS] actuation)	1.9 E-6	1.2 E-8	C3
Diverse Reactor Trip	7,1 E-6	8.0 E-9	C1
Manual switch to SBO diesels	4.8 E-6	6.6 E-8	C2
Manual Feed & Bleed	1.5 E-6	3.6 E-8*	C1
Automatic switchover of the cooling for the LHSI pumps	7.3 E-7	1.8 E-10	C2
Diverse activation of SI and partial cool-down	6.6 E-7	5.8 E-8**	C1
Manual activation of fast secondary cool-down	1.1 E-6	6.4 E-8	C3
Cooling of the IRWST by the EVU [CHRS]	1.1 E-6	6.9 E-10	C2
Diverse activation of the RIS [SIS] (shutdown states)	1.0 E-5	2.5 E-8***	C1
Manual switchover of the cooling for the LHSI pumps	1.2 E-5	2.8 E-8	C3
Manual re-supply of the ASG [EFWS] tanks	4.7 E-6	1.7 E-8	C3
Manual actuation of the LHSI for injection	2.7 E-7	1.6 E-8	C2
Manual actuation of RIS/RRA [SIS/RHRS] train	1.1 E-6	7.2 E-9	C2
Manual isolation of RIS/RRA [SIS/RHRS] train	2.1 E-8	5.4 E-12	C1

* LERF contribution of the sequence: 9.2 E-9/ry
** LERF contribution of the sequence: 8.7 E-9/ry
*** LERF contribution of the sequence: 8.1 E-8/ry

2. RULES FOR RRC-A STUDIES

2.1. ASSUMPTIONS AND REQUIREMENTS FOR SAFETY ANALYSIS

2.1.1. Rules for the analysis of RRC-A sequences

2.1.1.1. RRC-A sequence Study

The RRC-A sequences cover multiple failure conditions not considered in the PCC analyses of Chapter 14, but which are taken into account in the design in order to meet the probabilistic design objectives. The RRC-A sequences result from:

- either the total loss of an F1 function following an initiating event considered in PCC analysis or
- a combination of independent events.

They are studied in a deterministic manner to justify the design.

The adequacy of the design with respect to the RRC-A sequences is illustrated through RRC-A accident analyses, i.e.:

- a thermal-hydraulic calculation of the plant transient associated with a sequence, where a code calculation is required, or
- an appropriate technical justification, where a code calculation is not considered necessary.

This demonstration aims to justify the appropriate design of the systems and functions that are necessary to achieve the RRC-A function. These systems and functions are as follows:

- the “RRC-A features”. These features are specifically included in the design to meet the objectives of the RRC-A sequence (see the definition in sub-section 2.1.1.4 and the list in section 0.2) or
- the F1 and operational features already included in the design for operational and PCC-mitigation reasons, which are required in addition to the “RRC-A feature” to achieve the RRC-A function and operating in their design range.

For the RRC-A sequences that result from the total loss of an F1 function, the RRC-A function is defined as the back-up function of the failed F1 function (for example, ultimate emergency diesel generators that backup the main diesel generators in the event of a total power loss).

For the RRC-A sequences that result from multiple failures, the RRC-A function is the specific function required to mitigate this specific event.

The RRC-A function is adequately designed when the RRC-A acceptance criteria are met (see sub-sections 2.1.1.3 and 2.1.1.8).

When a manual action to limit the consequences is required during the RRC-A sequence and when the delay time to perform this action is of importance in the probabilistic assessment of the sequence, the RRC-A analysis indicates the maximum acceptable delay time. (For example, the maximum time delay to manually actuate fast-cool down in a small break loss of coolant accident (SB-LOCA) without medium head safety injection (MHSI) is calculated). This is an example of a case where a short time delay will significantly affect the reliability of the action.

2.1.1.2. Initial conditions

The initial conditions for RRC-A accident analysis correspond to steady-state operation in the defined plant state.

The safety analysis of RRC-A events must cover the initial reactor states determined by the probabilistic studies.

Note: When the initial reactor state is not indicated, it is assumed that the event is analysed at power or at hot shutdown.

An RRC-A sequence is defined with respect to a standard state of the reactor, as described in Sub-chapter 14.0.

In the relevant standard state of the reactor, the most conservative operation mode is considered with regard to the fulfilment of the RRC-A acceptance criteria (for example, full power operation for LOCA in state A, the maximum pressure of the primary cooling system of 30 bar for LOCA in state C).

The plant parameters considered are best-estimate (see section 0.1). For example:

- at full power, the initial power is equal to 100% of the rated capacity,
- the initial thermal-hydraulic parameters of the reactor coolant system (RCP [RCS]) and the steam generators (SG) are considered at their nominal setpoint, without uncertainty.

2.1.1.3. Final state for RRC-A analysis

The RRC-A accident analysis must be performed up to a safe state called the “RRC-A final state”, and defined as follows:

- Achieving long-term core sub-criticality,
- Decay heat removal,
- Activity discharges in accordance with the acceptance criteria (see section 2.1.1.8).

2.1.1.4. Definition and classification of RRC-A features

The RRC-A feature can be briefly characterised as follows:

- without the RRC-A sequence, this feature would not be implemented in the design,
- without the RRC-A feature, the RRC-A sequence would not meet the acceptance criteria defined in sections 2.1.1.3 and 2.1.1.8

The RRC-A features are F2 safety grade and are listed in section 0.2.

Systems, devices or functions introduced in the design for other reasons, for example operational reasons or to mitigate PCCs, and which are used in the RRC-A sequence within their design range (normal or PCC operating conditions, with no need for re-design), do not need a specific classification for RRC-A events.

2.1.1.5. Failure assumptions

The definition of the RRC-A sequence identifies the failure combination to be taken into account in the RRC-A accident analysis. An additional failure to this combination has not been assumed. The specific RRC-A feature must have the degree of redundancy that is necessary to meet the probabilistic decoupling target (given by the definition of the RRC-A sequence).

The unavailability of the system due to preventive maintenance is not included in the RRC-A accident analysis.

Loss of offsite power (LOOP) is not included in the RRC-A sequence (i.e. it is not assumed if not accounted for in the failure combination that defines the RRC-A sequence).

2.1.1.6. Manual actions

Manual actions carried out from the main control room are not considered in the RRC-A accident analysis in the 30 minutes following the first significant information provided to the operator.

The manual actions carried out at local level are not taken into account in the RRC-A accident analysis in the first hour following the first significant information provided to the operator.

2.1.1.7. Boundary conditions

The RRC-A sequence study is based on a realistic approach. All the systems (safety classified and non-safety classified) are considered available (provided they do not have to operate outside their design range) except for those assumed unavailable due to the event sequence. In particular, the operational systems and the instrumentation and control functions that are not affected by the event sequence and that operate as normal are assumed available.

The boundary conditions regarding the operation of the systems are treated as follows:

- the operational parameters of systems that are not involved in the RRC-A functions are defined by realistic assumptions,

- the operational parameters of systems that perform the RRC-A function (back-up function of the failed F1 function) are defined by conservative assumptions (i.e. realistic assumptions plus uncertainties).

The boundary conditions include the system thermal-hydraulic characteristics (flow rate, etc.) and the associated instrumentation and control for automatic operation.

Particular attention is given to the uncertainties that can cause a “cliff edge” effects in the probabilistic safety analysis (PSA) results on a case-by-case basis.

2.1.1.8. Acceptance criteria

For the RRC-A sequence, it must be demonstrated that the decoupling criteria of PCC-4 accidents are met. Thus, the radiological consequences of the RRC-A sequences are implicitly limited to the radiological limits of the PCC-4 events. Therefore, a specific radiological assessment for RRC-A sequences generally need not be performed (see section 4).

2.2. PLANT CHARACTERISTICS

General plant characteristics accounted for in the RRC-A analysis presented in this sub-chapter are as given in Sub-chapter 14.1.

3. RRC-A SITUATION ANALYSIS

3.1. ATWS SIGNAL (RBS [EBS] ACTUATION) FOLLOWING ROD FAILURE

This section addresses the following PCC-2 events, assuming the mechanical failure of all control/shutdown rods on RT demand:

- excessive increase of secondary side steam flow (GCT [MSB] opening),
- loss of main feedwater flow (LOFW),
- loss of offsite power to the station auxiliaries (LOOP),
- uncontrolled boron dilution,
- uncontrolled RCCA bank withdrawal.

The above five events are representative of the most limiting RRC-A sequences of "Anticipated Transient Without Scram" (ATWS).

They are analysed with respect to their consequences on:

- the RCS integrity (overpressure protection aspect),
- the core integrity (fuel protection aspect).

The present analysis of "ATWS following rod failure" in this section, is complemented by the analysis of "ATWS by RPR [PS] failure" in section 3.2. The events considered are the same, except that the failure of the scram at the time of the F1A reactor trip signal results from the failure of the F1A signal itself (all control/shutdown rods are available, but F1A RT signal is not actuated) rather than the mechanical failure of the rods (F1A RT signal available, but all rods fail).

The new reference configuration uses SEMPELL Pressuriser Safety Valves, the old one used SEBIM Pressuriser Safety Valves. The most significant impacts of this change are a modification to the opening setpoint of the PSV and to the maximum opening stroke time. The stroke time is reduced from 1.5 seconds to 0.1 seconds which is much less onerous. Consequently, the overpressure results are impacted. The assessment of the impact is presented below, and utilises the results of previous studies.

3.1.1. Excessive increase in secondary steam (opening of the GCT [MSB])

3.1.1.1. Identification of the causes

A total failure of the automatic shutdown system of the reactor on demand from the reactor's protection system may be caused by:

- a failure of the automatic reactor shutdown F1A signals (i.e. that none of the signals sent by the RPR [PS] de-energises the rod drive coils) as discussed in section 3.2,

- a failure of the control and shutdown rods to insert into the core after the drive coils are de-energised. In this case, actuation of the rods due to control or limitation signals also fails.

This section only deals with this second cause of ATWS, due to rod failure.

The three main reasons for an excessive increase in steam flow at full power are:

- the spurious opening of one or several condenser main steam bypass valves GCT [MSB],
- the spurious opening of one or several main steam relief train (VDA [MSRT]) valves,
- the spurious opening of one main steam safety valve (MSSV).

The spurious opening of one MSSV or one VDA [MSRT] valve need not be combined with the total loss of RT due to their low frequency (the initiating event belongs to the PCC-3 category). In addition, the spurious opening of a VDA [MSRT] isolation valve, is rapidly terminated by the steam line pressure control function, that closes the corresponding control valve.

Thus the RRC-A sequence considered is the spurious opening of the GCT [MSB] valves that lead to the maximum overcooling and power increase.

3.1.1.2. Description of the accident

After the GCT [MSB] valves have been fully opened, the F1A signal "major steam generator (SG) pressure drop" partially isolates the normal feedwater by closing the high load ARE [MFWS] lines (via the RT signal) and isolates the main steam lines by closing the main steam isolation valves (VIV [MSIV]). The RT signal is also actuated, but the rods are assumed not to insert. The low load ARE [MFWS] lines remain open, and 30% of nominal feedwater flow is delivered to the SG.

As a consequence of the isolation of the main steam lines and the high load ARE [MFWS] lines, the core temperature increases and, due to the effects of moderator temperature reactivity feedback, the power decreases (after an initial increase until the isolation of the main steam lines and the high load ARE [MFWS] lines). The isolation of the main steam lines leads to a large imbalance between the RCP [RCS] power and the secondary system heat removal rate. Consequently, the RCP [RCS] temperature and pressure rapidly increase and this triggers the opening of the pressuriser safety valves (PSV).

Twenty seconds after the RT on F1 signal on a "SG pressure drop", the ATWS signal is triggered on "RT signal and rods in high position (or high flux) after an appropriate delay”. The ATWS signal (and the related actions) is an RRC-A function, especially designed for the “ATWS through rod failure” RRC-A sequences, and is safety classified F2. It causes (see PCSR Sub-chapter 14.1)**:**

- Boration with 7000 ppm enriched boron (corresponding to 11200 ppm natural boron) via the extra borating system (RBS [EBS]) pumps. The boration provided by the RCV [CVCS] injection is not taken into account (i.e. the water injected is at the initial concentration of the primary inventory)
- automatic isolation of the RCV [CVCS] tank,

- all RCP [RCS] pumps are shut down on the "SG level < MIN2" signal. This prevents a rapid increase in the temperature and pressure of the primary RCP [RCS] due to the degradation of heat transfer to the secondary (e.g. if the SG water inventory is exhausted). In parallel, the emergency feedwater system (ASG [EFWS]) is activated by the ‘SG level < MIN2” F1A signal. Due to these actions, the reactor power decreases, limiting the increase of the RCP [RCS] pressure.

In the long term, the RCP [RCS] temperature increase is limited by the capacity of the VDA [MSRT], ASG [EFWS] and low load ARE [MFWS]. After injecting borated water from the RBS [EBS] and the isolation or the reduction of the ARE [MFWS] low load and ASG [EFWS] flows, the plant is in a “final state” (i.e. a safe shutdown condition).

3.1.1.3. Safety criteria

It must be demonstrated that "the final state for the RRC-A studies" can be reached, i.e.:

- Achieving long term core sub-criticality,
- decay heat removal
- the integrity of the three barriers that ensure the containment of radioactive materials, is guaranteed by meeting the following criteria:
    - The departure from nucleate boiling ratio (DNBR) remains higher than 1.21 (Sub-chapters 4.4 and 14.1),
    - the integrity of the RCP [RCS] is not compromised (the preliminary acceptance criteria is that the peak pressure in the RCP [RCS] must not exceed 130% of the Design Pressure (DP) (i.e. 228.5 bar abs). Refer to Sub-chapter 3.4 on overpressure protection analyses.

3.1.1.4. Protection and mitigation actions

These criteria are achieved through various normal operation and safety systems:

Three trains of the pressuriser pressure relief valves (F1A) are available to limit the pressure in the primary coolant system and ensure the integrity of the RCP [RCS]. Each PSV has a capacity of 300 t/h of saturated vapour and 450 t/h of water at a pressure of 176 bar.

Normal pressuriser spray[1] is available, as long as the pressure difference between the RCP [RCS] cold leg and the top of the pressuriser is sufficient to maintain the spray efficiency.

The long term sub-criticality of the core is provided by the RBS [EBS], and potentially by the RCV [CVCS] with an automatic switch at a boron concentration of 7000 ppm.

For the secondary side heat removal, a four-train system is available that consists of the following:

- one VDA [MSRT] (F1A) per loop (the minimum capacity of the main steam bypass is 50% of nominal SG flow, at 100 bar pressure),

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[1] The normal pressuriser spray has minimal impact on the RCP [RCS] peak pressure.

- two MSSV (F1A) per loop (minimum capability is 2 x 25% of the nominal SG flow, at 100 bar pressure)

- one ASG [EFWS] train (F1A) per loop,(the design flow is 90 t/h at 97 bar SG pressure, the ASG [EFWS] tank capacity is 1680 te)

In addition to the above systems, the low load ARE [MFWS] lines remain available for heat removal during the accident.

3.1.1.5. Methods of analysis

The analysis is performed using the following internally coupled codes:

- the MANTA V3.7 code (Appendix 14A) for overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), accounting for control/F2/F1 system operations,

- the SMART V4.5 / FLICA-IIIF V3 codes (Appendix 14A) for calculating the core neutronic and thermal hydraulic behaviour.

The DNBR calculation is performed by the FLICA code (Appendix 14A).

The analysis methodology is based on the following approach:

- identification of the dominant events,

- verification of the adequacy of the codes to simulate these events.

The dominant events in this transient are:

a) turbine trip, VIV [MSIV] isolation, High Load ARE [MFWS] (ARE [MFWS]-HL) isolation, and RCP [RCS] pump trip without reactor scram,

b) reactivity changes, which influence the reactor power. Firstly, the primary temperature decreases until the response of the MS-pressure drop signal, and then the RCP [RCS] temperature increases because of the loss of secondary heat removal due to TT and VIV [MSIV]/ARE [MFWS]-HL closure. Secondly, core coolant flow decreases by Reactor Coolant Pump cut-off. Finally core boration occurs by RBS [EBS] injection,

c) rapid SG pressure increase after VIV [MSIV] isolation with response of VDA [MSRT] and MSSV,

d) rapid RCP [RCS] pressure increase due to RCP [RCS] heat-up and PZR filling and leading to the opening of one or more PSV (including fast overpressure while the pressuriser is in liquid phase),

e) large SG water inventory decrease, until the reactor power is reduced such that the SG boil-off is compensated by ASG [EFWS] and ARE [MFWS] low-load flow rate.

All these events are applicable to the MANTA, SMART and FLICA codes. The qualification of these codes is based on the following:

a) MANTA

- The use of recognised and tested correlations.
- The overall validation of the code by simulation of PWR plant transients [Ref-1]:
    - o The MANTA simulation of the loss of SG feedwater flow without reactor scram in the MB2 loop, showed a good prediction of the SG dryout time, and a good simulation of the RCP [RCS]/SG heat transfer. The SG secondary side depressurisation transient calculated by MANTA closely agreed with the measured values, thus demonstrating the applicability of the secondary side model for these conditions.
    - o The MANTA simulation of the opening of a VDA [MSRT] on the PALUEL-3, 4-loop plant, showed good agreement with measured values. The calculated loop temperatures compared well with measured values, thus validating the entire RCP [RCS] hydraulic calculation.
    - o The MANTA simulation of an overpressure on the BUGEY 4 plant showed a good prediction of the increase in the pressuriser level, and thus a good simulation of the swell of the RCP [RCS] fluid.

b) SMART and FLICA

- Qualification of the SMART code under overheating conditions relies mainly on EPICURE experiments. Some experiments have a high void fraction in the central part of the assembly. Power distributions measured in those conditions were accurately simulated by the SMART code [Ref-2].
- The critical heat flux correlation used in FLICA was developed based on experimental data. The experimental data consists mainly of results of the tests for FRAMATOME fuel assemblies performed at the University of COLUMBIA loop in US, and OMEGA loop of the CENG test facility in Grenoble. The tests cover the steam line break conditions down to a pressure of 30 bar [Ref-3].

c) Coupling between the thermal-hydraulic and the neutronic codes

The thermal-hydraulic and neutronic codes are coupled in this analysis:

- for each time step, the thermal-hydraulic conditions of the RCP [RCS] and SG are calculated by the MANTA code. The thermal-hydraulic conditions at the core inlet (temperature map, flow, pressure and boron concentration) are transferred to the FLICA code, to calculate initial core thermal-hydraulics:
    - o from the core thermal-hydraulic conditions, SMART calculates the neutronic parameters and transfers them to FLICA, on convergence,
    - o from the neutronic parameters, FLICA calculates core thermal-hydraulics, and transfers them back to SMART,
- SMART returns to MANTA the power generated in each of the 241 core assemblies, for MANTA to redistribute in the four core quadrants modelled (one corresponding to each primary loop).

Qualification of MANTA-SMART-FLICA coupling relies on the qualification of each element.

A functional validation [Ref-4] shows that the result of each code is the same when used on a stand-alone basis or in the coupled mode.

3.1.1.6. Specific assumptions

3.1.1.6.1. Cases examined

Given that the neutron flux decreases through the moderator reactivity feedback following the heat-up of the primary coolant, it is conservative to assume the initial power state at the beginning of life (BOL), where the moderator coefficient is at its minimum absolute value (0 pcm/K or negative but close to 0 pcm/K).

To cover the entire life of the plant, two transients are examined:

a) one transient with an initial power of 60% nominal power (NP) with a zero moderator temperature coefficient covering all the power levels between 60% and 0% NP,

b) a second transient with an initial power of 100% NP and a moderator temperature coefficient covering all power levels between 100% and 60% NP.

3.1.1.6.2. Single failure and maintenance

Single failure or maintenance does not have to be included in the RRC-A studies.

3.1.1.6.3. Initial conditions and characteristics of the systems

The initial conditions are the “Best Estimate” conditions at 60% and 100% NP.

A conservative thermal-hydraulic primary flow rate is assumed, even though this assumption is not strictly required by the accident analysis rules for RRC-A sequences.

All instrumentation and control systems are available, except for those lost by definition in the RRC-A sequence (for example, no rod insertions). Some systems are not considered to simplify the analysis (for example automatic boration via the RCV [CVCS] pumps).

The system characteristics are generally based on realistic assumptions.

Minimum or maximum values are conservatively taken into account for the systems that participate directly in meeting the safety or acceptance criteria, including the dedicated RRC-A function: for example, the PSV setpoints that limit RCP [RCS] overpressure, the definition of the ATWS signal and RBS [EBS] flows for the core boration.

Special assumptions (i.e. not included in the paragraphs above or in Sub-chapter 14.1) are as follows:

- no automatic re-closing of the GCT [MSB] by dedicated isolation valves,

- after reaching the MS-pressure drop signal, the ARE [MFWS] high-load lines are closed and the ARE [MFWS] low-load lines remain open with a flow capacity of 30%. The SG nominal level of 15.7 m is maintained in the long term by Low Load ARE [MFWS] (ARE [MFWS]-LL) or/and ASG [EFWS] (SG level control is not simulated in the transient calculations),

- the ATWS signal is actuated 20 seconds after reactor trip signal on "SG pressure drop > MAX1", which starts the RBS [EBS] with 7000 ppm enriched boron acid injection (corresponding to 11200 ppm natural boron) and isolates the RCV [CVCS] tank. When the SG level (wide range) is lower than MIN2, all main coolant pumps are tripped,

- automatic boration via RCV [CVCS] pumps is not credited in the analysis.

3.1.1.6.4. Assumptions related to the systems

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

The F2[2] functions assumed to operate are: RBS [EBS] boration, RCV [CVCS] tank isolation, RCP [RCS] pump.

The Non-Safety Classified (NC) systems assumed to operate are: normal pressuriser spray, pressuriser heater, RCV [CVCS].

3.1.1.6.5. Neutronic data

The neutronic data are calculated at the beginning of life (BOL). The moderator coefficient is -10 pcm/°C and 0 pcm/°C [Ref-1] at the beginning of the transient for the 100% FP case and the 60% FP case, respectively.

The Doppler temperature coefficient is maximised to minimise the core power decrease after isolation of the steam lines. The Doppler temperature coefficient is set to -3.38 pcm/°C at the beginning of the transient [Ref-1] (maximum in absolute value, covering all fuel cycle management).

Reference fuel cycle management is used to maximise the F∆H while the moderator and Doppler temperature coefficients are set at their given values. The calculation of the DNBR is thus conservative.

The initial RCP [RCS] boron concentration is 1280 ppm for the 100% FP case, and 1350 ppm for the 60% FP case (expressed as natural boron) [Ref-1].

3.1.1.6.6. DNBR calculation

DNBR calculation is performed considering the axial distribution in the hot channel and the F∆H provided by 3D core calculation (SMART, Appendix 14A) at minimum DNBR.

For RRC-A analyses, no penalty on the local thermal power is considered for DNBR calculation.

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[2] The RBS [EBS] is a F1A system started automatically on the F2 safety class ATWS signal. The RBS [EBS] automatic boration function is therefore F2 safety classified.
This also applies to the RCP [RCS] pumps trip function and the isolation of the RCV [CVCS] tank.

3.1.1.7. Results and conclusions

3.1.1.7.1. Results for EPR at 4250 MW [Ref-1]

Initial state 60% NP, moderator coefficient of 0 pcm/^{o}C at the beginning of the transient

Section 16.1.3.1.1 - Table 3 presents the sequence of events for this transient. The main parameters are presented in Section 16.1.3.1.1 - Figures 1 to 7.

The maximum pressure at the RCP [RCS] pump outlet is 182 bar (103% of the design pressure). It is reached just after the initial opening of the first PSV (less than one minute after the initiating event). The acceptance criterion of 130% of the design pressure is met with a large margin.

The second and third PSV are not activated during the transient. The first PSV opens twice following isolation of the steam and ARE [MFWS]-HL (before the complete opening of the VDA [MSRT] and the MSSV).

The MSSV and the VDA [MSRT] are activated during the transient.

The automatic reactor trip signal is triggered on “SG pressure drop >MAX1”. It causes turbine trip and isolation of the high load ARE [MFWS] lines, but not rod insertion.

The ATWS signal causes the start of the RBS [EBS] boration and the isolation of the RCV [CVCS] tank. The “very low SG level” signal is generated after approximately 5 minutes, and all the RCP [RCS] pumps are tripped.

The minimum DNBR is reached during the RCP [RCS] pump coast down phase. The value is significantly higher than the 1.21 limit and is bounded by the minimum DNBR for the transient at 100% NP (>3.19).

The decay heat and residual heat are adequately removed by the low load ARE [MFWS], the ASG [EFWS] and the VDA [MSRT].

There are no radioactive releases during the accident, because none of the barriers (fuel and RCP [RCS]) are damaged.

Initial state 100% NP moderator coefficient of -10 pcm/^{o}C at the beginning of the transient:

Section 16.1.3.1.1 - Table 4 presents the sequence of events for this transient. The main parameters are presented in Section 16.1.3.1.1 - Figures 8 to 14.

The maximum pressure downstream of the RCP [RCS] pumps is 187.2 bar (106% of the design pressure). It is reached just after the initial opening of the second and third PSV. The acceptance criterion of 130% of the design pressure is met with a large margin.

The second and third PSV are activated only once during the transient, while the first PSV is activated twice (before the complete opening of the VDA [MSRT] and the MSSV).

The MSSV and the VDA [MSRT] are all activated during the transient.

The RT signal on "major SG pressure drop" causes the turbine trip and isolation of the high load ARE [MFWS] lines, but not rod insertion.

The minimum DNBR is obtained during the initial RCP [RCS] cool down due to the spurious opening of the GCT [MSB] valves. This value is significantly higher than the 1.21 limit.

The ATWS signal causes the start of the RBS [EBS] boration and the isolation of the RCV [CVCS] tank. The “SG level < MIN2” signal is triggered at approximately 3 minutes, and all the RCP [RCS] pumps are tripped.

The decay and residual heat is removed by the low load ARE [MFWS], the ASG [EFWS] and the VDA [MSRT].

There are no radioactive releases during the accident, because none of the barriers (fuel and RCP [RCS]) are damaged.

The calculation results show that for the RRC-A sequence "ATWS through rods failure – Excessive increase in secondary steam flow (opening of the GCT [MSB])”, the criteria are met and the required final state is reached.

The analysis results show that the safety criteria (integrity of the RCP [RCS] and minimum DNBR) are met.

3.1.1.7.2. Results for EPR at 4500 MW.

The accident was not analysed at 4500 MW. The fulfilment of the acceptance criteria is deduced from the results of the analysis performed for EPR 4250 MW.

Impact of the new design of the pressuriser safety valves

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

The two PSV models have the same discharge flow rate.

The main differences between the two valve designs which can impact the results of this analysis are:

- The opening setpoint of the first SEMPELL-model valve is 1 bar higher than the first SEBIM-model valve setpoint.
- The opening setpoint of the third SEMPELL-model valve is 3 bar higher than the third SEBIM-model valve setpoint.
- The PSV opening time for the SEMPELL-model valve, 0.1 seconds, is shorter than that for the SEBIM-model valve, 1.5 seconds.

The hysteresis of the SEMPELL-model valve, 16% of the opening setpoint, 28 bar for the first PSV, is higher than that for the SEBIM-model valve, 10 bar.

Meeting the acceptance criterion with regard to the maximum RCP (RCS) pressure.

The differences between EPR 4250 MW and EPR 4500 MW which have an impact on the RCP [RCS] pressure are as follows:

- Power level increased by 6%

- Average primary temperature increased by 1°C

- The remaining neutronic and thermal-hydraulic data are identical with those for EPR 4500 MW with the exception of the initial boron concentration, which was increased to accommodate the new power level. The value was increased by approximately 240 ppm for the case at 100% NP (assumption used in section 3.1.5)

- The flow rate capacity of the pressuriser safety valves and the spray line are identical.

- One of the consequences of the spurious opening of one or several condenser main steam bypass valves GCT [MSB] is the increase of the primary pressure due to an imbalance between the primary and secondary power level. As for EPR 4250 MW, the setpoint for opening the pressuriser safety valves is reached. Since the capabilities for the discharge valves on the primary and secondary system for EPR 4500 MW are identical to the EPR 4250 MW, the increase in the primary pressure will be limited. The peak primary pressure may be slightly higher. However, taking into account the large margin calculated for EPR 4250 MW (106% of the design pressure versus an acceptance criteria of 130% design pressure) and the increase in power of only 6%, it can be concluded that the criterion of not exceeding 130% of the design pressure is met for EPR 4500 MW.

- For the case initiated at 60% NP, the first PSV discharge capacity is sufficient to depressurise the RCP [RCS]. The maximum pressure occurs 2.5 seconds after reaching the first PSV opening setpoint of 175.5 bar and is 0.6 bar higher at the pressuriser. The 2.5 second period consists of 0.5 seconds of delay time, 1.5 seconds for valve opening plus a further 0.5 seconds to reach the maximum pressure) As the opening setpoint of the first SEMPELL-model valve is 1 bar higher than that for the SEBIM-model valve, it will be reached approximately 4 seconds later. However as the opening time of the SEMPELL-model valve is nearly instantaneous, 0.5 seconds of delay time and 0.1 seconds opening time, the PSV opening will be effective about 2.6 seconds later than in the case with the SEBIM-model valves. The discharge flow is the same for the two valve models so the maximum pressure peak will also occur within the same time interval after its opening. Thus the maximum pressure peak will be less than 1 bar higher. Taking into account the margin demonstrated for the EPR_{4250} with SEBIM-model valves, 103% of the design pressure compared to the criterion of 130%, and the power increase, leading to a further pressure increase limited to 6%, it can be concluded that the 130% DP acceptance criterion will be met for EPR_{4500} with the SEMPELL-model valves installed. For the case initiated at 100% NP, the discharge capacity of the 3 PSVs is needed to depressurise the RCP [RCS]. The maximum pressure occurs just after the opening of the second and the third PSV. With SEMPELL-model valves the maximum pressure will be reached a little later and after the opening of the third PSV. It will be less than 3 bar higher than the value calculated with the SEBIM-model valves.

- Taking into account the margin demonstrated for the EPR_{4250} with SEBIM-model valves, 106% of the design pressure compared to the criterion of 130%, and the effect of the power increase, leading to a further pressure increase limited to 6%, it can be concluded that the acceptance criterion of 130% DP will be met for the EPR_{4500} with SEMPELL-model valves.

3.1.1.7.3. Meeting the acceptance criterion of minimum DNBR

The differences between EPR 4250 MW and EPR 4500 MW that have an impact on the DNB criterion are:

- Power level increased by 6%
- Average primary temperature increased by 1°C
- The remaining neutronics and thermal-hydraulic data are identical with those for EPR 4500 MW, with the exception of the initial boron concentration which was increased to accommodate the new power level. The value was increased by approximately 240 ppm for the case at 100% NP (assumption used in section 3.1.5)
- The increase in power and the increase in the average primary temperature will have a negative impact on the DNBR margin. This will lead to a decrease in the margin to the DNBR limit. However, given that the increase in power is only 6%, the value calculated for EPR 4250 MW (3.19 versus an acceptance criterion of 1.21) gives confidence that the acceptance criteria of 1.21 will be met for EPR 4500 MW.
- For case initiated at 100% NP, the minimum DNBR occurs early in the transient during the initial RCP [RCS] cooldown. At this time, the PSV are still closed. Thus, the change to the SEMPELL-model valves will have no impact on the results. Therefore it can be concluded that the criterion will be met for EPR_{4500} with SEMPELL-model valves.

3.1.1.8. Systems sizing

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

SECTION 16.1.3.1.1 - TABLE 1

EPR 4250 MW - Initial Conditions
ATWS Excessive Increase in Secondary Steam Flow
(Spurious Opening of GCT [MSB])
Initial States 60% NP and 100% NP

Parameter	Initial value
Initial state= 60%	
RCS [RCP] flow rate	Thermal-hydraulic
Power	60% FP
Pressure	155 bar (nom)
Average temperature	311.8°C (nom)
PZR level	55% (nom)
SG pressure	87.7 bar (nom)
ARE [MFWS] inlet temperature	203.5°C (nom)
SG level	49% NR (nom)
Initial state= 100%	
RCS [RCP] flow rate	Thermal-hydraulic
Power	100% FP
Pressure	155 bar (nom)
Average temperature	311.8°C (nom)
PZR level	56% (nom)
SG pressure	78 bar (nom)
ARE [MFW] inlet temperature	230°C (nom)
SG level	49% NR (nom)

SECTION 16.1.3.1.1 - TABLE 2

EPR 4250 MW - Main Assumptions
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial States 60% NP and 100% NP

Parameter	Value
Reactor trip signal on SG pressure drop > MAX1 Setpoint Delay	 - -2 bar/min - -variable limit, setpoint 7 bar below the actual pressure in steady state - -maximum value 75 bar 0.9 s
VIV [MSIV] closure on SG pressure drop > MAX1 Setpoint Delay	 as for Reactor trip signal 0.9 + 5 s = 5.9 s
ATWS signal (reactor trip signal, and high rods position or high flux), actuates RBS [EBS] boration: Delay (after RT signal) Capacity (per RSC loop) RBS [EBS] boron concentration (natural boron)	 20 s 1.4 kg/s 11200 ppm
ASG [EFWS] actuation on SG level (wide range cold side) < MIN2 Setpoint Delay Capacity	 7.85 m (40% WR) 1.5 + 15 = 16.5 s 90 te/h per SG at 97 bar
MS relief train actuation on SG pressure > MAX1 Setpoint Delay MSRT closing delay Capacity	 95.5 bar 0.9 + 1.5 + 0.5 = 2.9 s 40 s 1150 te/h under 100 bar
MS safety valves Setpoint Accumulation Capacity	 105 bar 3% 575 te/h under 100 bar, 2 per SG

SECTION 16.1.3.1.1 - TABLE 2 (CONT'D)

EPR 4250 MW - Main Assumptions
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial States 60% NP and 100% NP

Parameter	Value
PZR safety valves Setpoint dead time opening time Capacity Hysteresis	 175.5 / 179.5 / 179.5 bar 0.5 s 1.5 s steam 300 te/h, liquid 450 te/h under 176 bar 10 bar
PZR normal spray (3 stages) Setpoint opening time Capacity per stage (min) Capacity per line (min/max)	 156 / 158 / 160 bar 10 / 2 / 2 s 2 x 10 / 25 / 25 kg/s 25 / 35 kg/s
PSV [CVCS][1] Net RCP [RCS] injection flow rate PZR level < reference + dead band PZR level > reference + dead band	 + 10 kg/s 0 kg/s

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

SECTION 16.1.3.1.1 - TABLE 3

EPR 4250 MW - Sequence of Events
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 60% NP

EVENT	TIME (second)
Spurious opening of MS-bypass	0.5
PZR heating on/off	2.5 / 29.5
'MS pressure drop > MAX1" signal reached (gradient 2 bar/min and 7 bar initial margin to SG pressure) RT/TT signal VIV [MSIV] closure signal	17.5
Turbine trip	17.8
Closing of VIV [MSIV]s (5 s delay)	22.5
Maximum reactor power is reached (62.3% FP)	30.5
1st response of PZR spraying	32.5
ARE [MFWS] high-load line isolation in all SG (15 s after RT signal)	32.5
ATWS signal arises (20 s after RT signal) • RBS [EBS] injection (5.6 kg/s) • VCT isolation	37.5
MS-relief isolation valve opening	41
1st PSV opening / closing	43.5 / 50.5
Primary pressure peak: • at PZR 176.1 bar abs • at RCP [RCP] outlet 182 bar abs	44
PSV [CVCS] lines isolation (PZR level above 63%R)	47

SECTION 16.1.3.1.1 - TABLE 3 (CONT'D)

EPR 4250 MW - Sequence of Events
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 60% NP

EVENT	TIME (second)
Primary pressure peak: • at PZR 175.5 bar abs • at RCP [RCP] outlet 180.9 bar abs	84.5
1st PSV opening / closing	85 / 92
MS-safety valves opening/closing	95.5 / 309.5
RCP [RCP] trip (ATWS and SG level MIN2 signal)	291.5
ASG [EFWS] injection (SG level MIN2 signal)	306.5
Minimum DNBR (> 3.19)	323
SG minimum water level reached (28.6% WR = 5.7 m)	358
PZR filled up	373
RBS [EBS] boron arrival in the core (initial concentration + 1ppm)	375
PZR heating on	537

SECTION 16.1.3.1.1 - TABLE 4

EPR 4250 MW - Sequence of Events
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 100% NP

EVENT	TIME (second)
Spurious opening of MS-bypass	0.5
PZR heating on/off	2.5/16.5
"MS pressure drop > MAX1" signal reached (gradient 2 bar/min and 7 bar initial margin to SG pressure) • RT/TT signal • VIV [MSIV] closure signal	9.5
Turbine trip	9.8
Minimum DNBR (3.19)	13.2
Closing of VIV [MSIV]s (5 s delay)	14.5
Maximum reactor power is reached (105.1% FP)	16
1st response of PZR spraying	17.5
1st PSV opening / closing	24 / 31
ARE [MFW] high-load line isolation in all SG (15 s after RT signal)	24.5
MS-relief isolation valve opening	24.5
RCV [CVCS] lines isolation (PZR level above 63%R)	24.7
2nd and 3rd PSV opening/closing	25.7 / 29.8
Primary pressure peak: • at PZR 180.1 bar abs • at RCP [RCP] outlet 187.2 bar abs	26
PZR heating on/off	29/32

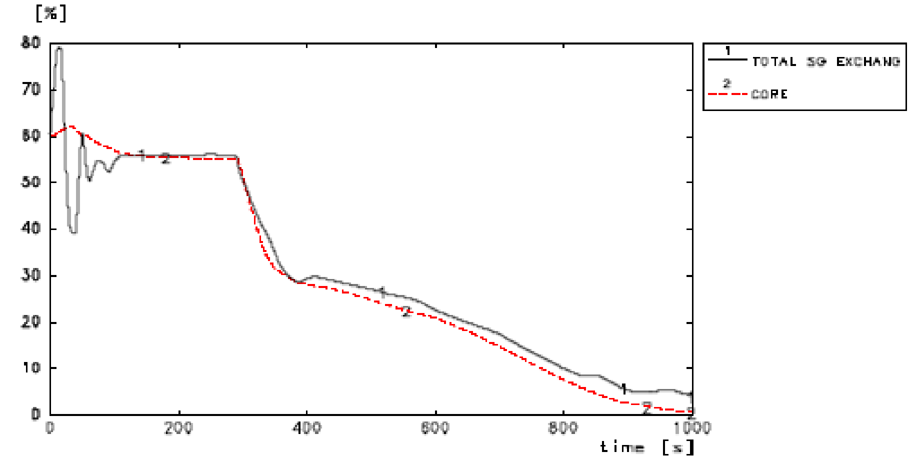
SECTION 16.1.3.1.1 - TABLE 4 (CONT'D)

EPR 4250 MW - Sequence of Events
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 100% NP

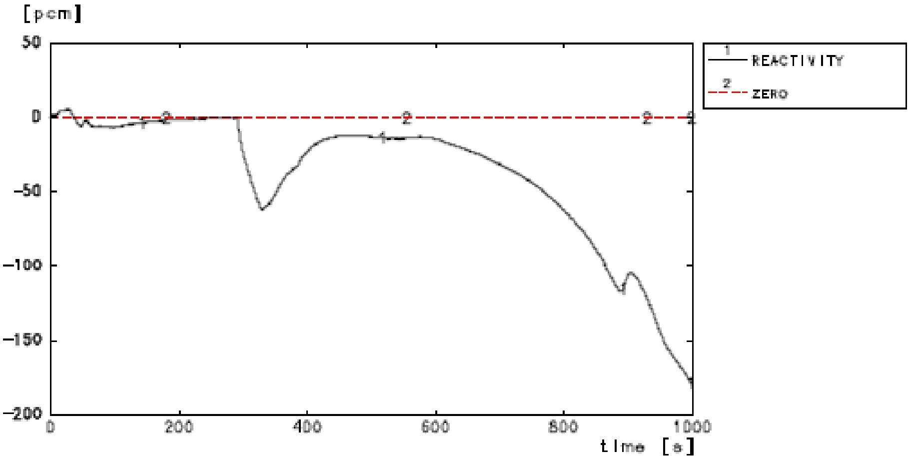
EVENT	TIME (second)
ATWS signal arises (20 s after RT signal) • RBS [EBS] injection (5.6 kg/s) • VCT isolation	29.5
1st PSV opening / closing	44 / 58
MS-safety valves opening/closing	46 / 200
PZR filled up	108
RCP [RCP] trip (ATWS and SG level MIN2 signal)	176.5
ASG [EFWS] injection (SG level MIN2 signal)	191.5
SG minimum water level reached (23.8% WR = 4.8 m)	259
RBS [EBS] boron arrival in the core (initial concentration + 1 ppm)	349
PZR heating on	582

SECTION 16.1.3.1.1 - FIGURE 1

EPR 4250 MW Power (Core and SG) – Reactivity
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 60% NP



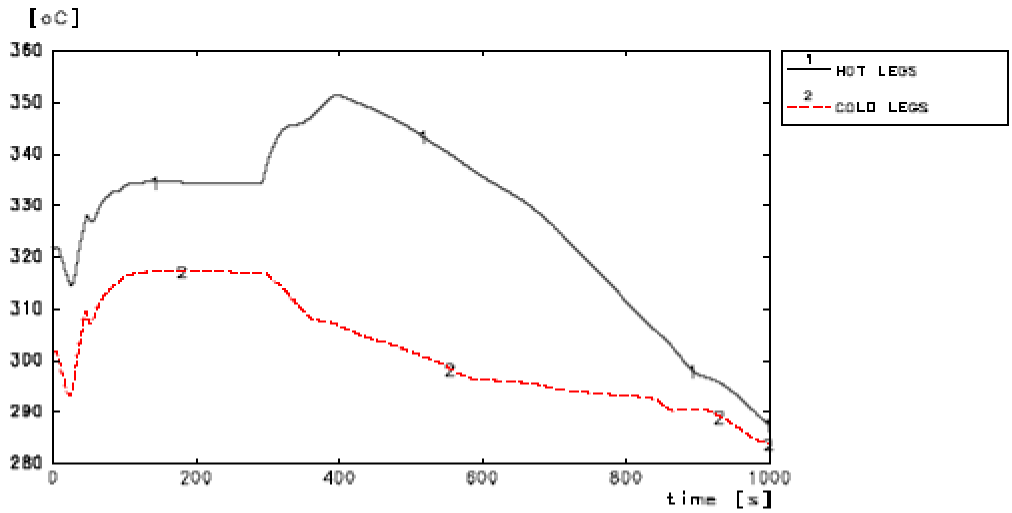
POWER (CORE AND SG)



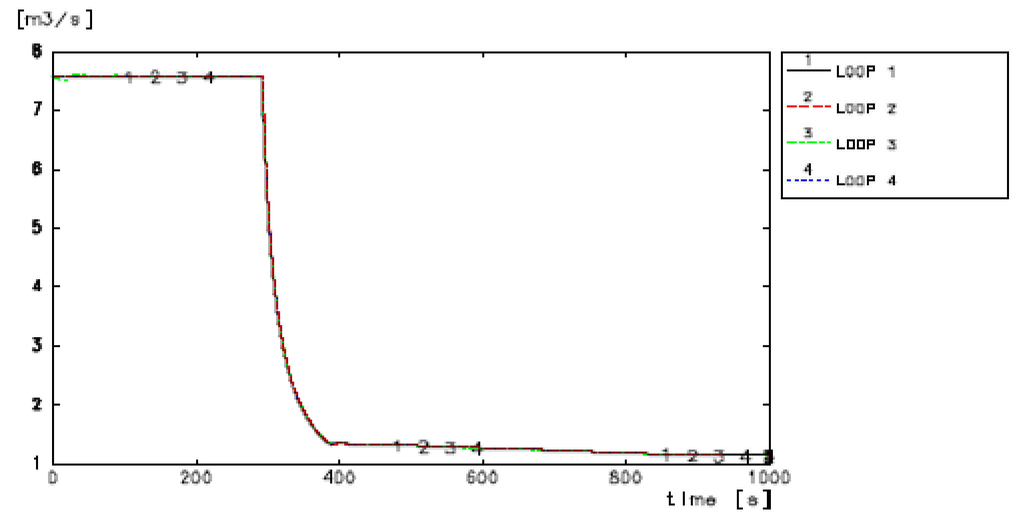
REACTIVITY

SECTION 16.1.3.1.1 - FIGURE 2

EPR 4250 MW RCP [RCS] Temperatures – RCP [RCS] Flow Rate
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 60% NP



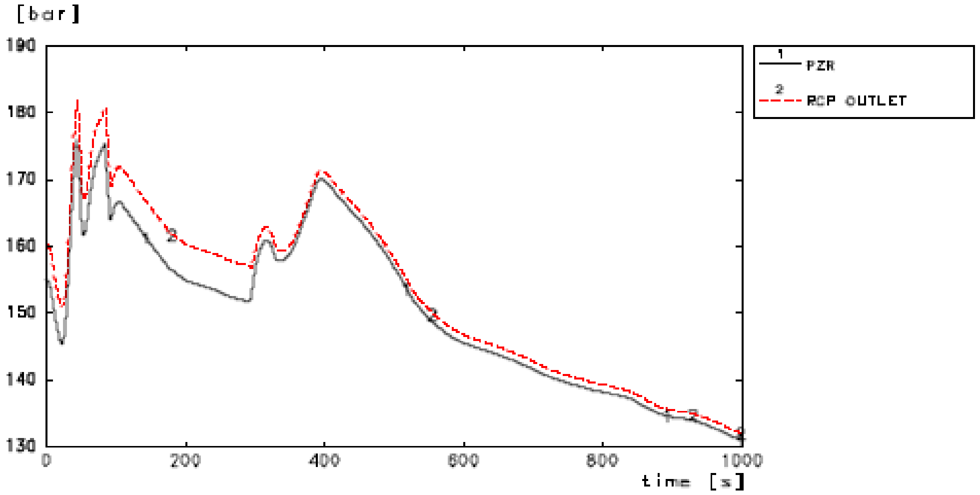
RCS TEMPERATURES



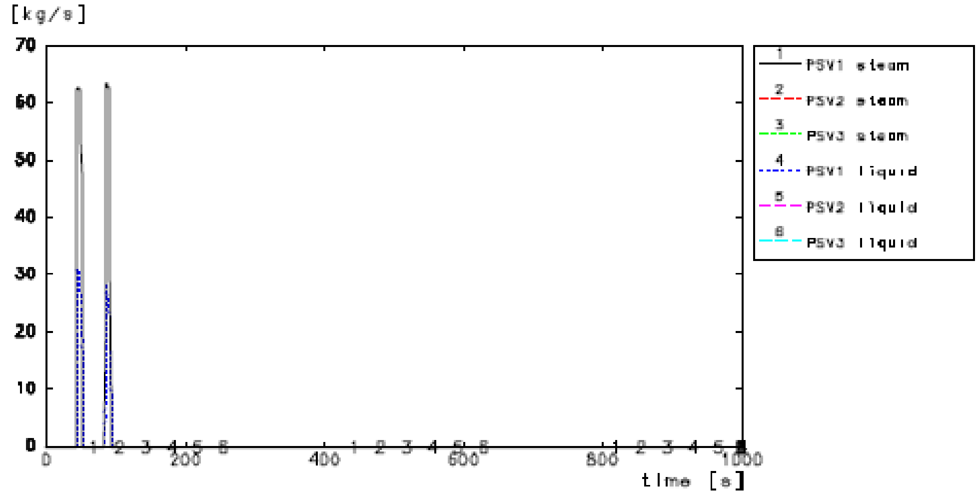
RCS FLOWRATE

SECTION 16.1.3.1.1 - FIGURE 3

EPR 4250 MW Primary Pressure – PSV Flow Rate
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 60% NP



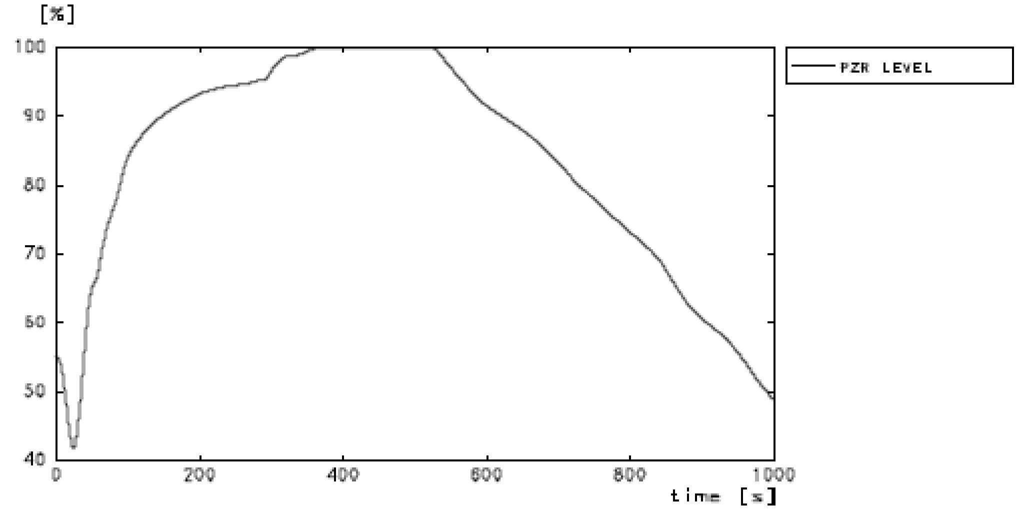
PRIMARY PRESSURE



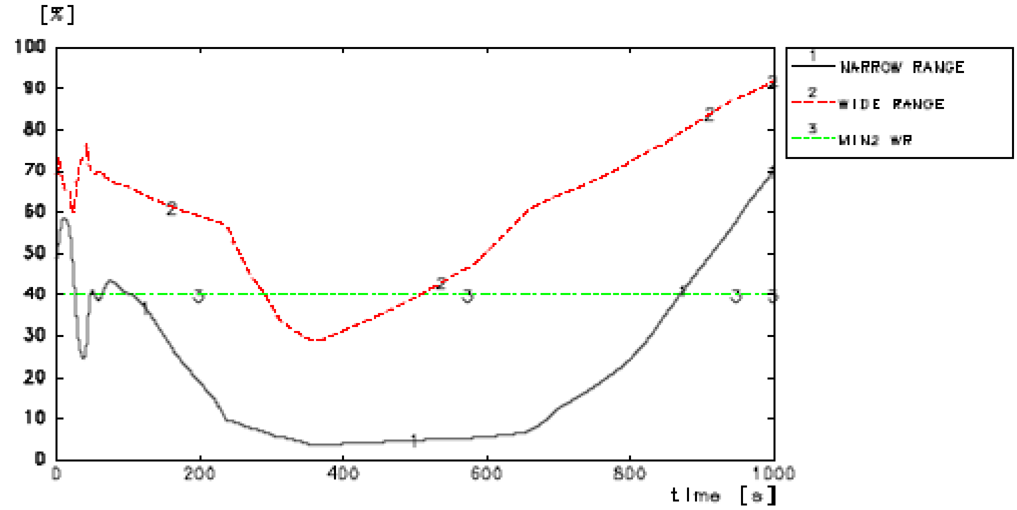
PSV FLOWRATE

SECTION 16.1.3.1.1 - FIGURE 4

EPR 4250 MW PZR Level – SG Levels
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 60% NP



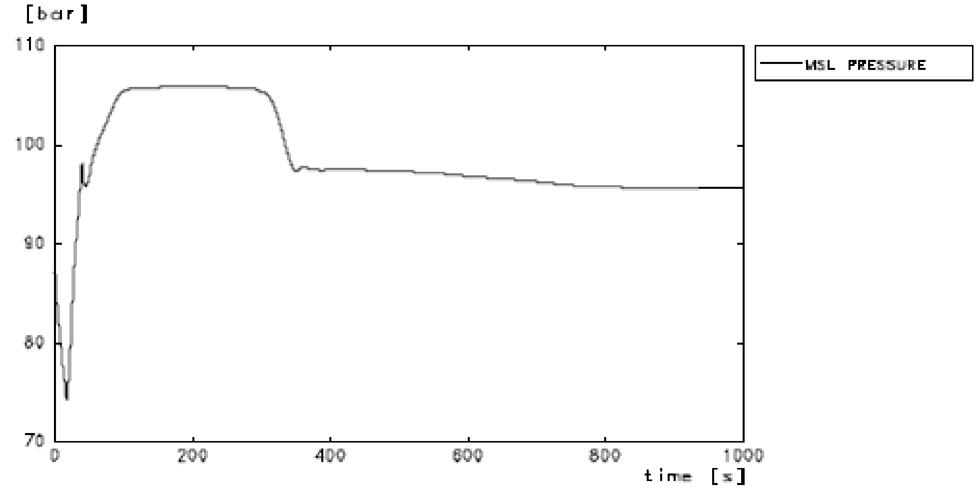
PZR LEVEL



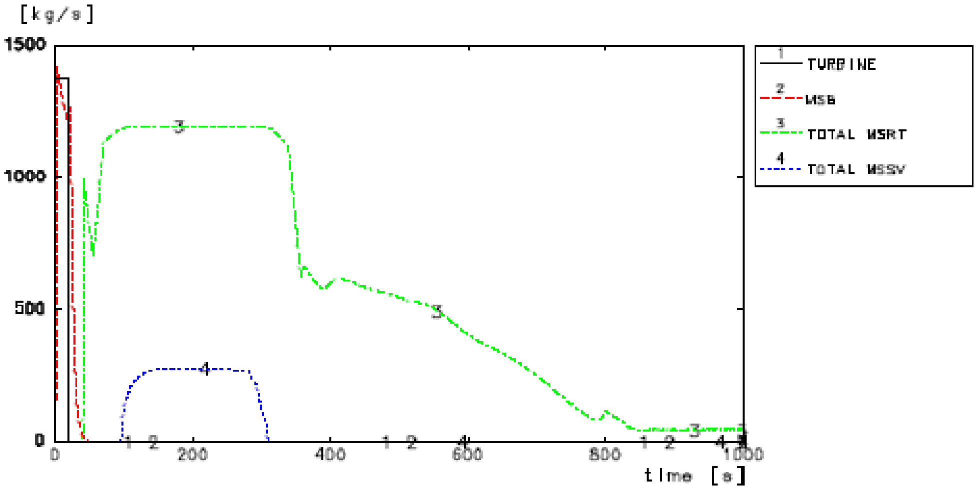
SG LEVELS

SECTION 16.1.3.1.1 - FIGURE 5

EPR 4250 MW Main Steam Pressure – Steam Flow Rate
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 60% NP



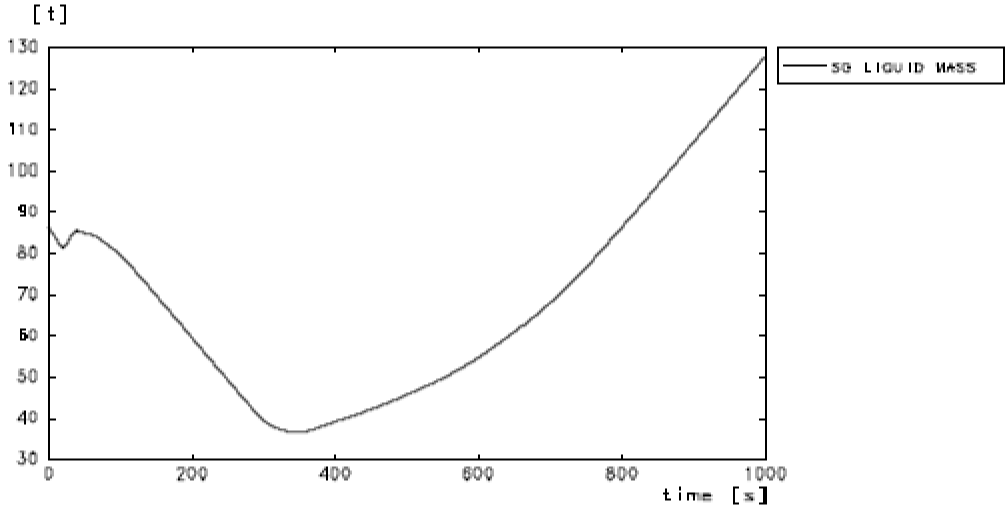
MAIN STEAM PRESSURE



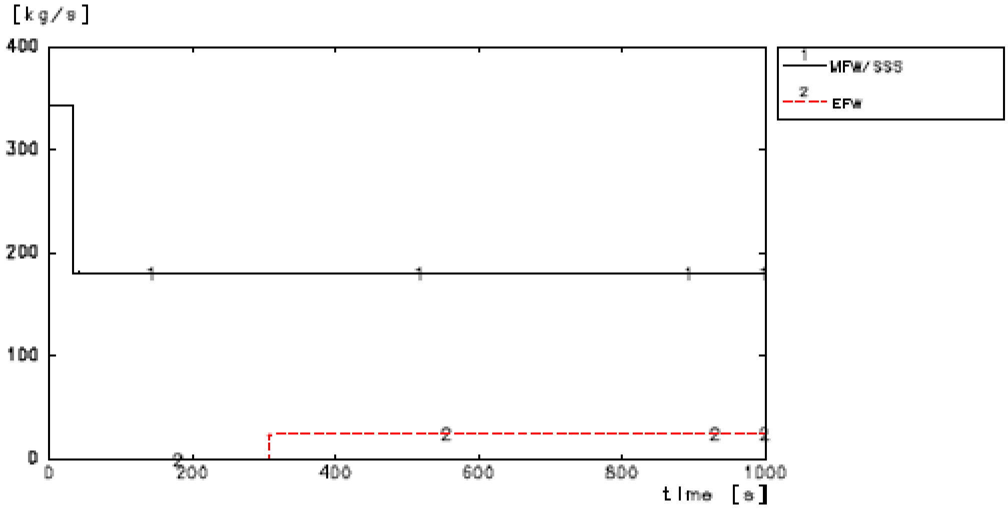
STEAM FLOWRATE

SECTION 16.1.3.1.1 - FIGURE 6

EPR 4250 MW SG Liquid Mass – Feedwater Flow Rate per SG
ATWS Excessive Increase of Secondary Side SteamFlow
(Spurious Opening of MS Bypass)
Initial State 60% NP



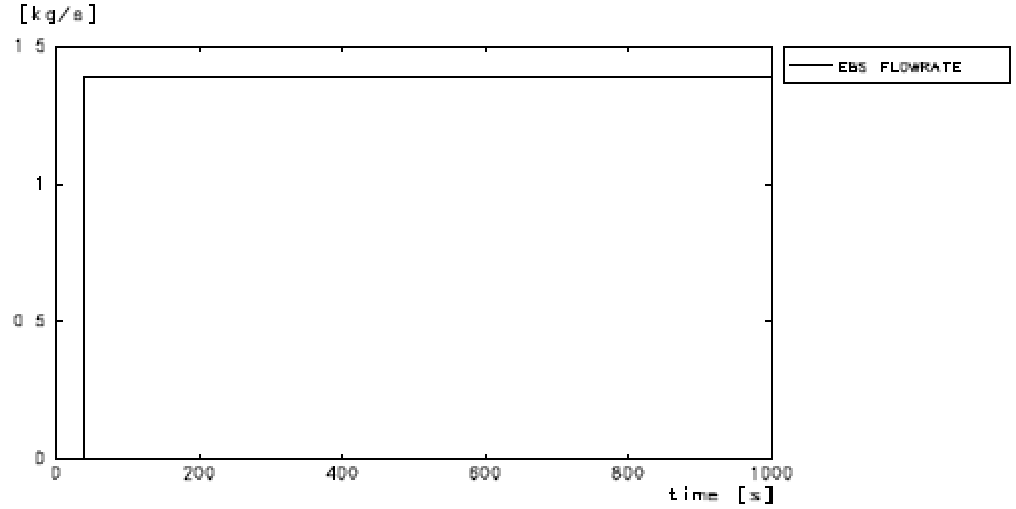
SG LIQUID MASS



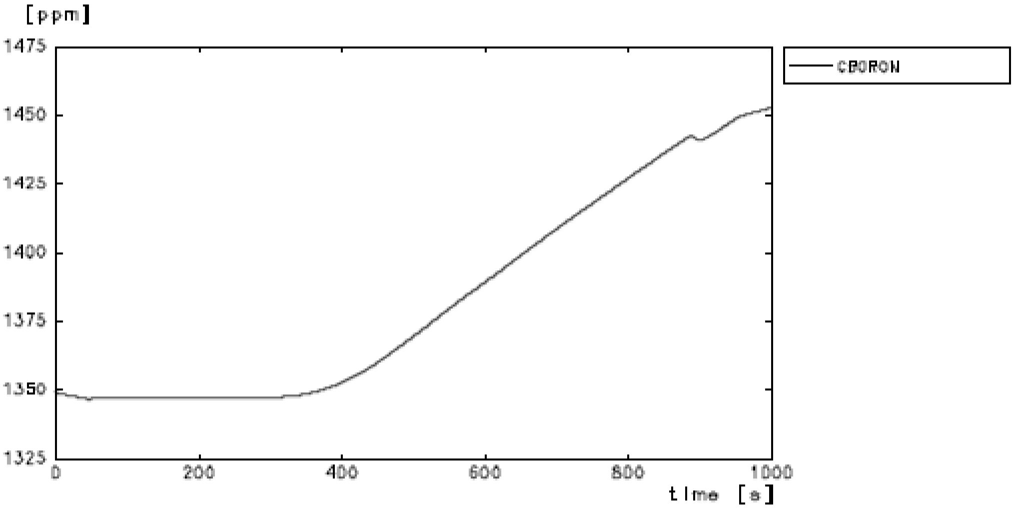
FEEDWATER FLOWRATE PER SG

SECTION 16.1.3.1.1 - FIGURE 7

EPR 4250 MW
RBS [EBS] Flow Rate per Loop – Core Boron Concentration (Natural Boron)
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 60% NP



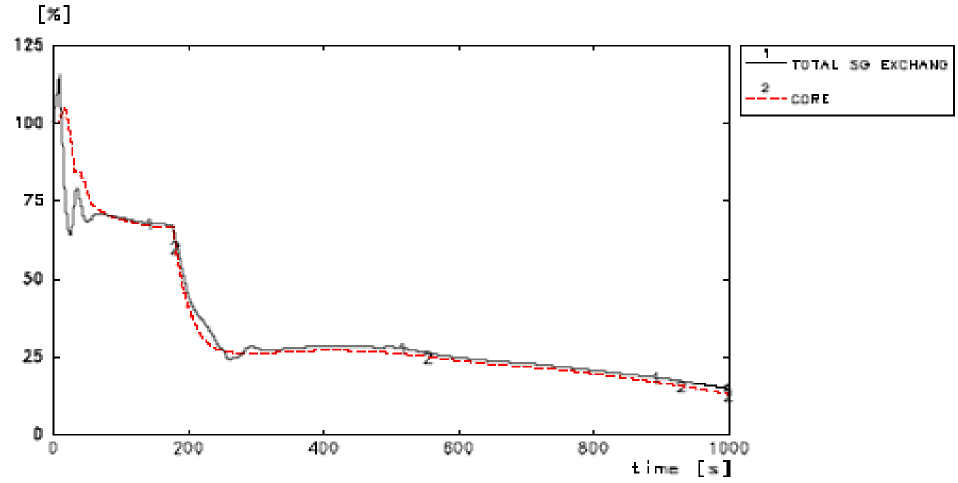
EBS FLOWRATE PER LOOP



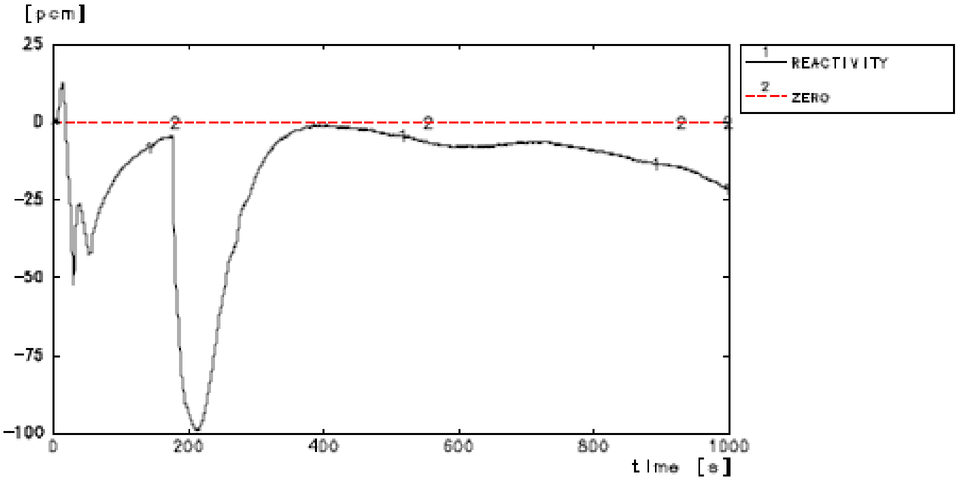
CORE BORON CONCENTRATION (NATURAL BORON)

SECTION 16.1.3.1.1 - FIGURE 8

EPR 4250 MW Power (Core and SG) – Reactivity
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 100% NP



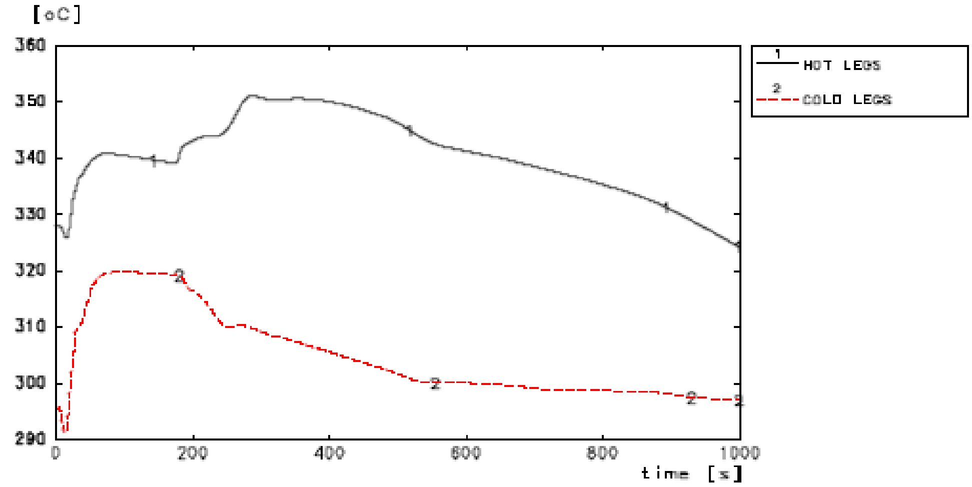
POWER (CORE AND SG)



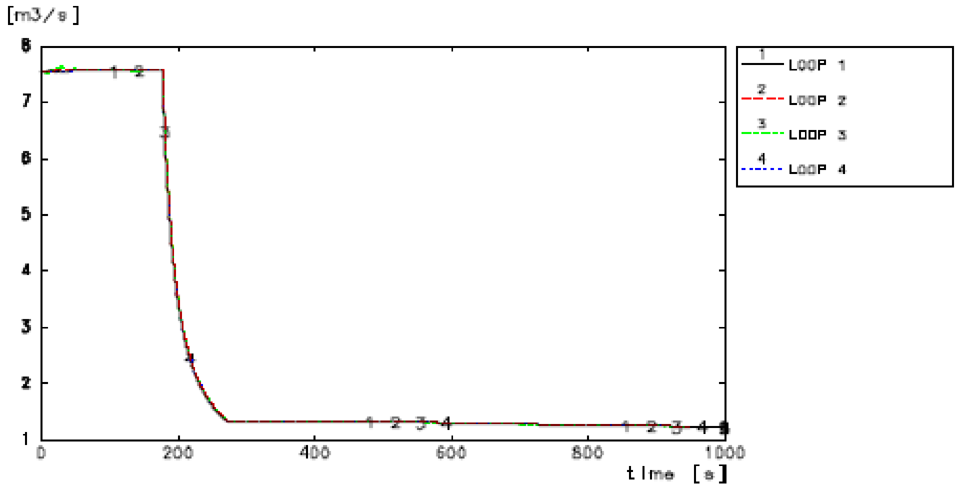
REACTIVITY

SECTION 16.1.3.1.1 - FIGURE 9

EPR 4250 MW RCP [RCS] Temperatures – RCP [RCS] Flow Rate
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 100% NP



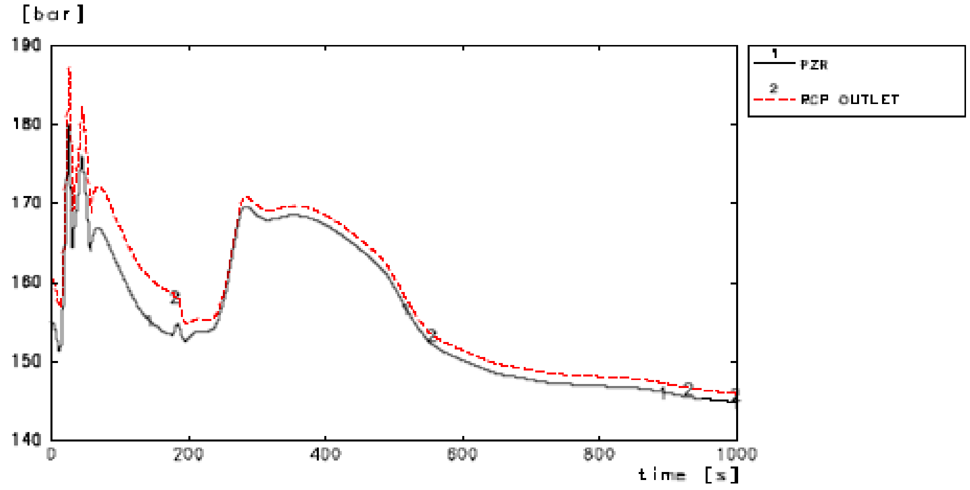
RCS TEMPERATURES



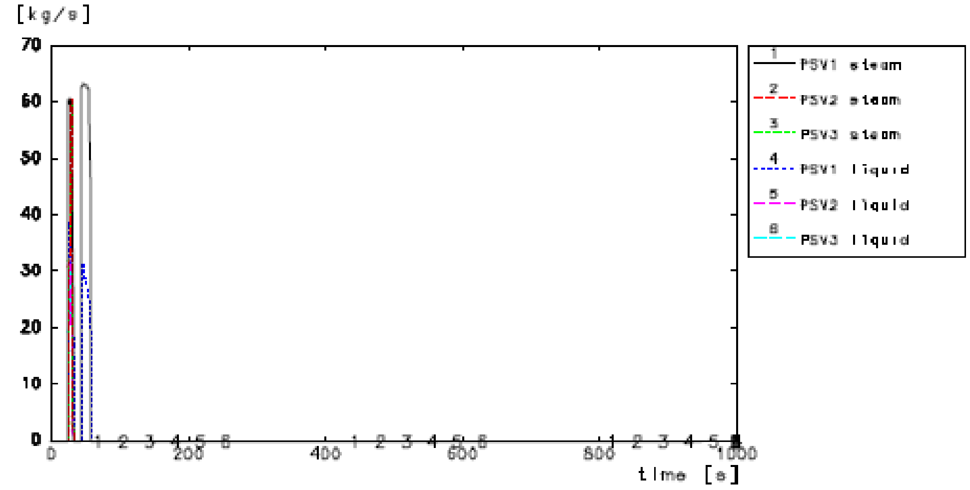
RCS FLOWRATE

SECTION 16.1.3.1.1 - FIGURE 10

EPR 4250 MW Primary Pressure – PSV Flow Rate
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 100% NP



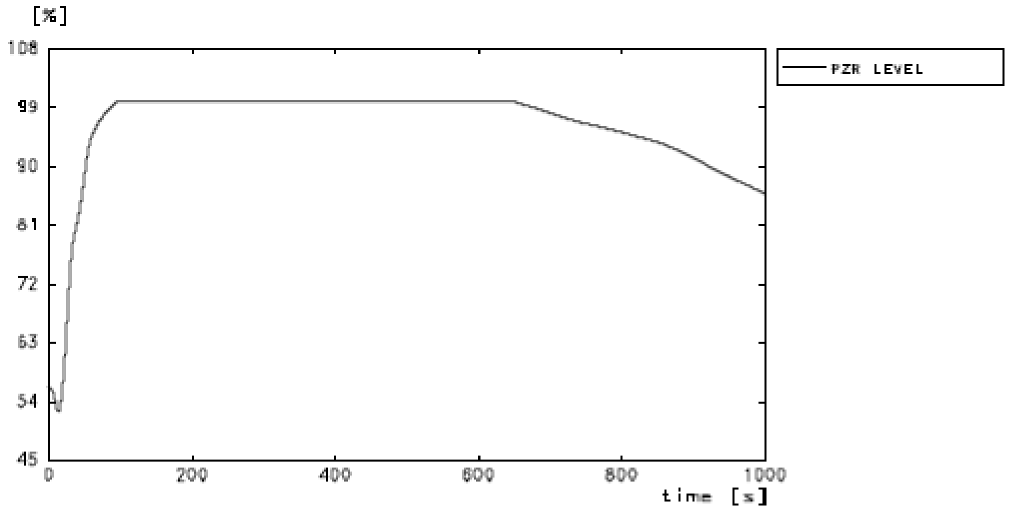
PRIMARY PRESSURE



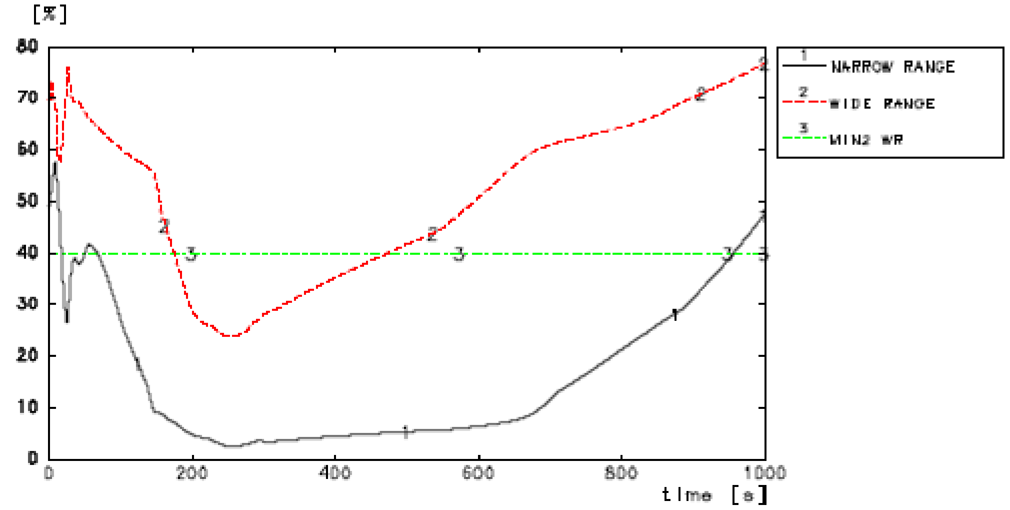
PSV FLOWRATE

SECTION 16.1.3.1.1 - FIGURE 11

EPR 4250 MW - PZR Level – SG Levels
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 100% NP



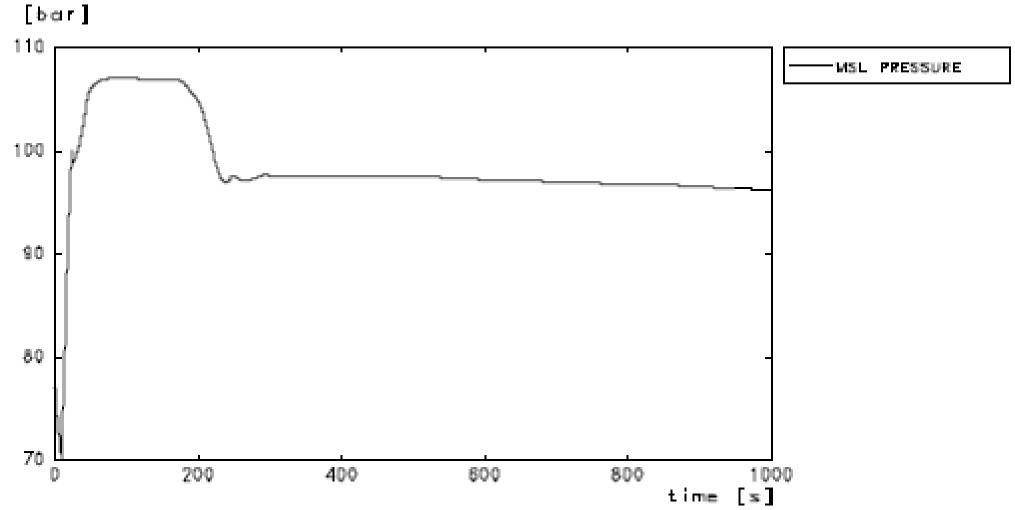
PZR LEVEL



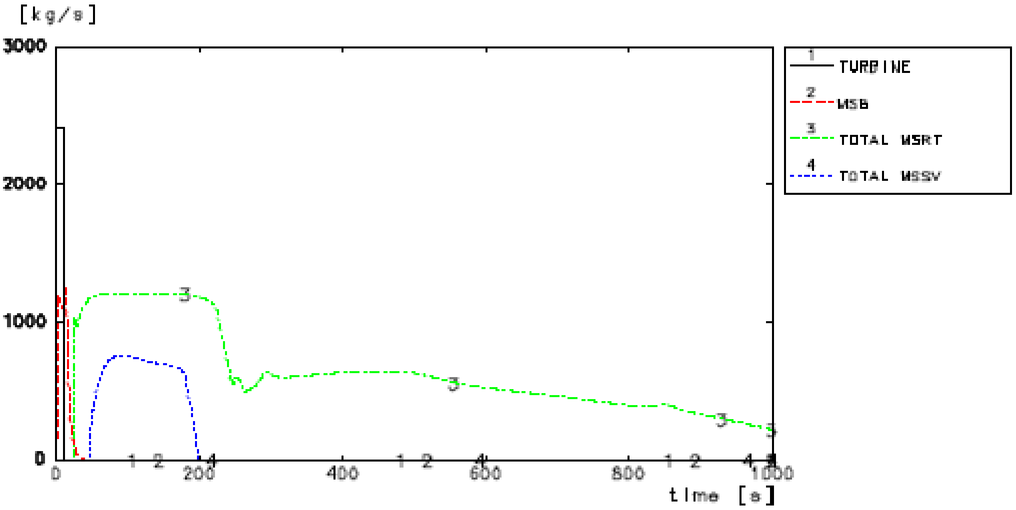
SG LEVELS

SECTION 16.1.3.1.1 - FIGURE 12

EPR 4250 MW Main - Steam Pressure – Steam Flow Rate
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 100% NP



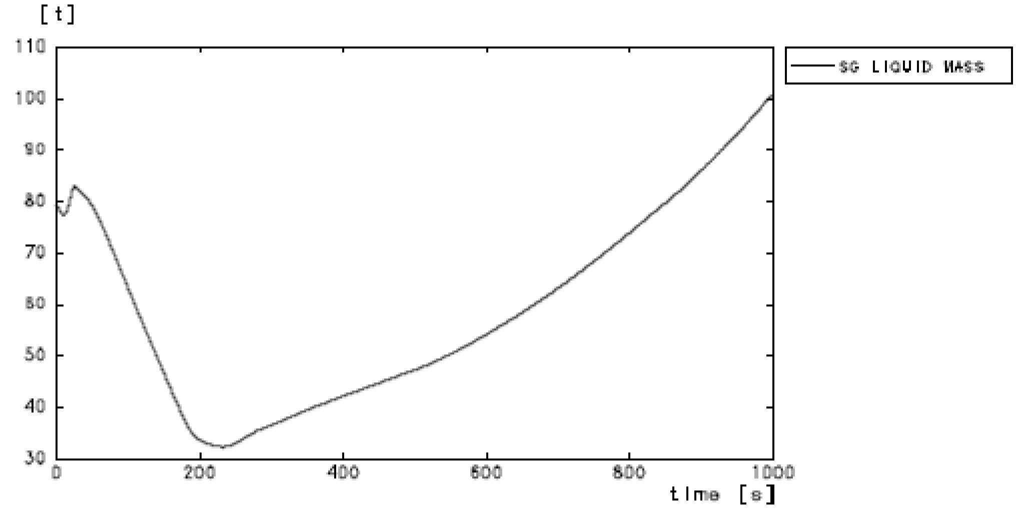
MAIN STEAM PRESSURE



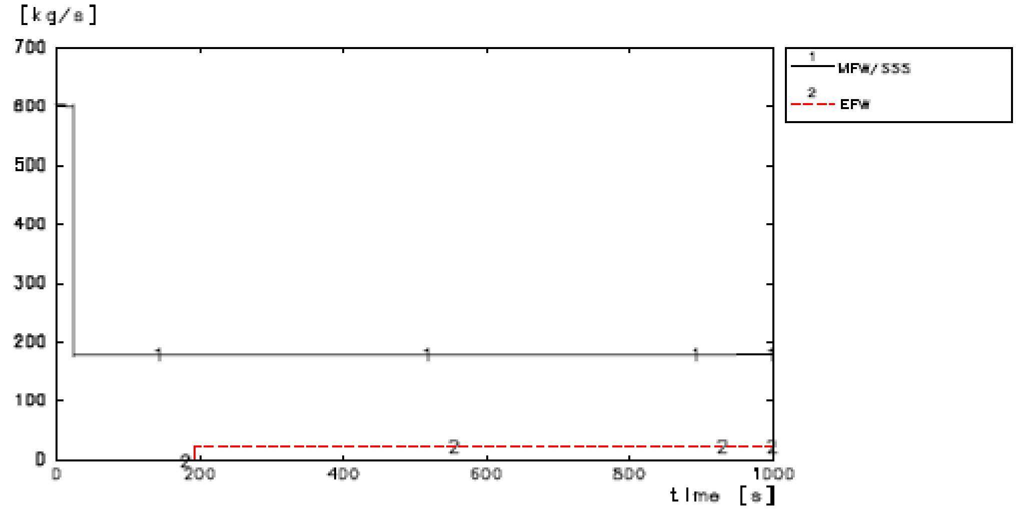
STEAM FLOWRATE

SECTION 16.1.3.1.1 - FIGURE 13

EPR 4250 MW - SG Liquid Mass – Feedwater Flow Rate per SG
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 100% NP



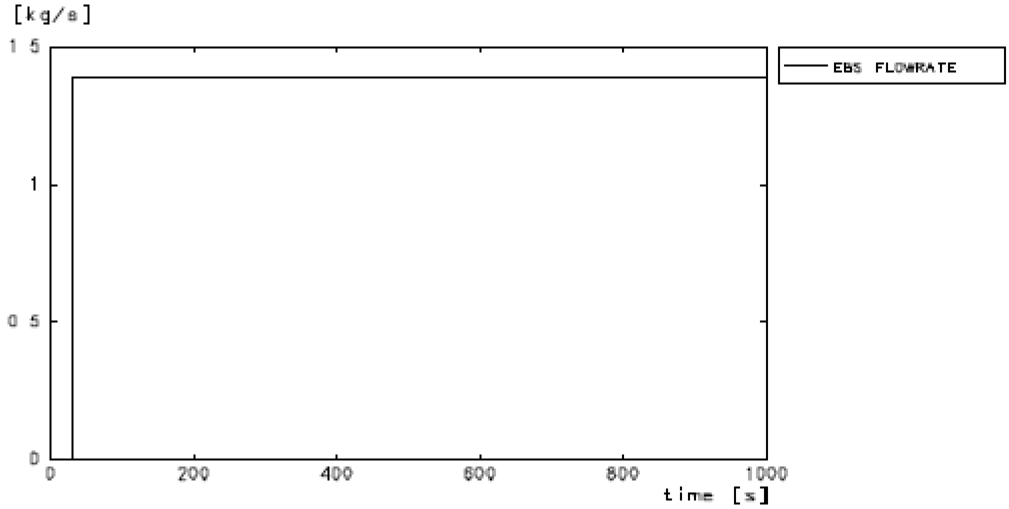
SG LIQUID MASS



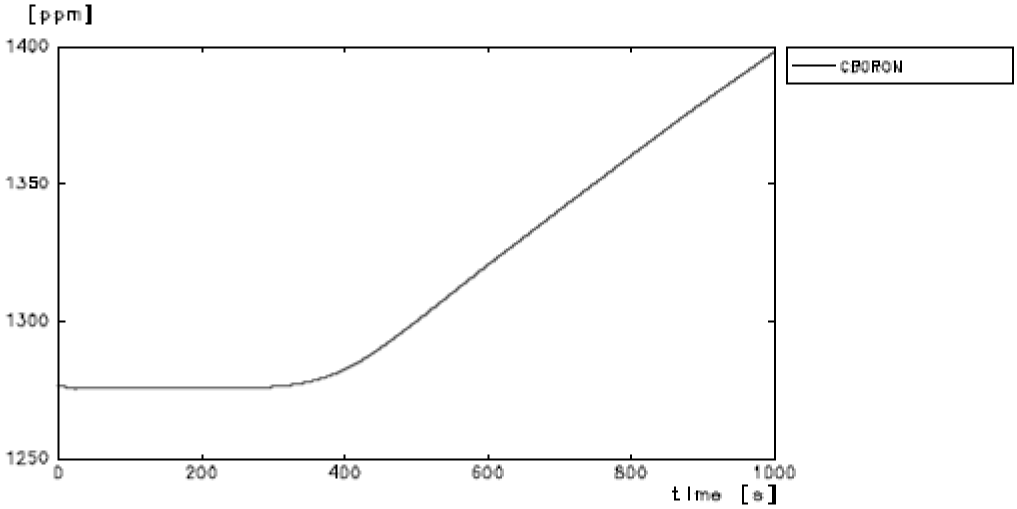
FEEDWATER FLOWRATE PER SG

SECTION 16.1.3.1.1 - FIGURE 14

EPR 4250 MW -
RBS [EBS] Flow Rate per Loop – Core Boron Concentration (Natural Boron)
ATWS Excessive Increase of Secondary Side Steam Flow
(Spurious Opening of MS Bypass)
Initial State 100% NP



EBS FLOWRATE PER LOOP



CORE BORON CONCENTRATION (NATURAL BORON)

3.1.2. ATWS “LOSS OF MAIN FEEDWATER”

3.1.2.1. Identification of Cause

A complete failure of the reactor trip system on demand from the Reactor Protection System (RPR [PS]) can result:

- either from the failure of the F1A reactor trip signals (i.e. all trip signals do not de-energise the drive coils),

- or from the mechanical failure of the control/shutdown rods after de-energizing all control/shutdown rod drive coils. In this case, the actuation of the rods due to control or limitation signals also fails.

The current sub-section addresses the second cause of ATWS related to rods failure. The first cause related to RPR [PS] failure is addressed in section 3.2.

3.1.2.2. Typical sequence of events

The RRC-A sequence considered is initiated by the total loss of all ARE [MFWS] pumps.

With the loss of main feedwater supply, reactor trip/turbine trip signals are actuated. As the control/shutdown rods have failed, the reduction of the reactor power can result from the inherent reduction of reactivity by decrease of moderator density in the short-term. The decrease of moderator density can be achieved by the following means:

- due to the secondary pressure increase (secondary side heat removal via VDA [MSRT] or MSSV), the primary temperature increases due to the thermal coupling via the SG tubes,

- the heat transfer capability of the steam generators decreases (depletion of steam generators) and the primary temperature increases as long as the reactor power is higher than the power removed by the steam generators,

- by reducing the primary coolant flow (cut-off of RCP [RCS] pumps) the temperature in the core increases.

Following the turbine trip, the secondary side heat removal is supported by the main steam bypass system and additionally via the main steam relief trains, (GCT [MSB] capacity is about 50% full power).

PSV open to limit the primary side overpressure.

Without any additional actions, this state would be stable as long as the steam generators have enough water inventories to remove the primary power (core plus RCP [RCS] pumps). Afterwards a strong reduction of heat transfer capability, down to about 10% (heat removal capability of ASG [EFWS] and AAD [SSS] systems) would occur with a sharp increase of primary temperature and pressure.

To avoid this intense pressure peak, a dedicated ATWS signal is implemented. The ATWS signal is triggered in the RPR [PS], from information of "RT signal and rods out (or flux high) after temporisation". This ATWS signal (and associated actions) is a RRC-A feature and it is specifically implemented to cope with the RRC-A sequences "ATWS by rods failure", being F2 classified. It actuates all RCP [RCS] pumps cut-off on very low SG-water level signal (SG level WR < MIN2) before the SG depletion occurs. By this action, the reactor power is reduced more smoothly with decreasing coolant flow rate, which leads to a reduced pressure increase on the primary side.

The ATWS signal also automatically initiates RBS [EBS] injection with 7000 ppm enriched boron (corresponding to 11200 ppm natural boron) thus ensuring core sub-criticality automatically in the long term. The RCV [CVCS] is also available for this boration function, but is not considered in this study. In addition, ATWS signal causes isolation of the Volume Control Tank (VCT).

3.1.2.3. Safety criteria

It must be demonstrated that the "final state for RRC-A analyses" can be reached, i.e.:

- attainment of long term core sub-criticality,
- decay heat removal ensured,
- activity release under control with the integrity of the barriers in accordance with the PCC-4 acceptance criteria.

For this demonstration, the following acceptance criteria are considered:

- the DNBR remains above 1.21 (Sub-chapters 4.4 and 14.1),
- the RCP [RCS] integrity is not impaired (as an acceptance criterion, the pressure at the most loaded point of the RCP [RCS] does not exceed the 1.3 fold design pressure, i.e. 228.5 bar abs (Sub-chapter 3.4).

3.1.2.4. Protection and mitigation actions

The following I&C functions provide protection and mitigation following loss of Main Feedwater accompanied by a reactor trip failure due to mechanical failure of the rods:

- reactor trip / turbine trip signal on "SG level (narrow range) < MIN1" (F1A),
- ATWS signal on reactor trip signal and high rods position (or high flux) after temporisation (F2),
- AAD [SSS] pumps start after ARE [MFWS] loss (NC),
- Pressuriser normal spray actuation on "Pressuriser pressure > MAX1" (NC),
- RBS [EBS] actuation on ATWS signal (F2),
- VCT isolation on ATWS signal (F2),
- RCP [RCS] pumps trip on "SG level (wide range) < MIN 2" if the ATWS signal has been obtained (F2),

- GCT [MSB] opening on "header pressure > MAX" (NC),
- MS relief train opening on "SG pressure > MAX1" (F1A),
- ASG [EFWS] actuation on "SG level (wide range) < MIN2" (F1A).

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

3.1.2.5. Methods of analysis

The analysis is carried out using the internal coupling of:

- the MANTA V3.7 code (Appendix 14A) for overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), accounting for NC/F2/F1 systems operations,
- the SMART V4.5 /FLICA-IIIF V3 codes (Appendix 14A) for neutronic and thermal-hydraulic behaviour of the core.

The DNBR calculation is performed with the FLICA code (Appendix 14A).

The analysis methodology is based on the following approach:

- identification of the dominant events,
- verification of the adequacy of the code to simulate those events.

The dominant events of this transient are:

a) trip of ARE [MFWS] pumps, turbine trip and RCP [RCS] pumps trip without reactor scram,

b) RCP [RCS] heat-up and Pressuriser filling, with consequential RCP [RCS] pressure increase (including fast overpressure while Pressuriser in liquid phase),

c) reactor power decrease due to moderator density effect, and later due to boron injection,

d) large SG water inventory decrease until sufficient power reduction for accommodation of boil-off by ASG [EFWS] and AAD [SSS] flow rate.

All these events are applicable to the MANTA, SMART and FLICA codes.

3.1.2.6. Specific assumptions

3.1.2.6.1. Transients analyses / initial power states

Considering the power reduction due to the moderator effect resulting from the RCP [RCS] heat-up, it is conservative to analyse the event at Beginning Of Life (BOL), where the absolute value of the moderator coefficient is at a minimum (0 pcm/°C, or negative but close to 0 pcm/°C).

To cover the entire life of the plant, the analysis is performed in two transients [Ref-1] or:

a) one transient with 60% NP initial state and no moderator effect, covering all power levels between 60% NP and 0% NP (for which the moderator coefficient is, at least 0 pcm/°C, in absolute value),

b) one transient with 100% NP initial state and -10 pcm/°C moderator coefficient [Ref-2], covering all power levels between 100% NP and 60% NP.

3.1.2.6.2. Single failure and maintenance

Single failure and maintenance do not have to be taken into account in RRC-A analyses.

3.1.2.6.3. Initial and boundary conditions

Typical initial conditions are at 60% NP and 100% NP (Section 16.1.3.1.2 - Table 1).

RCP [RCS] thermal hydraulic design flow rate is chosen (not set by accident analyses rules).

All systems and I&C functions are available, except those lost by definition of the RRC-A sequence (e.g. no rod drop). For simplicity, some systems are not credited.

Boundary conditions defining system efficiency are generally based on realistic characteristics.

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

3.1.2.6.4. Neutronic data

Neutronic data referring to BOL conditions.

The moderator coefficient is -10°C/pcm and 0°C/pcm at the beginning of the transient for the 100% NP case and the 60% NP case, respectively [Ref-1]. The Doppler temperature coefficient is maximised in absolute value to minimise the decrease in core power. The Doppler temperature coefficient is set to -3.38 pcm/°C at the beginning of the transient (maximum in absolute value, covering all fuel management cycles) [Ref-1].

The reference fuel management is retained to maximise the FΔH when the moderator and the Doppler temperature coefficients have been set to their target values. This is conservative for DNBR calculations.

The bounding initial RCP [RCS] boron concentration is 1280 ppm for the 100% NP case, and 1350 ppm for the 60% NP case (expressed in natural boron) [Ref-1].

3.1.2.6.5. Assumptions related to controls

SG level and RCP [RCS] temperature control are not relevant as ARE [MFWS] is unavailable.

Control rods unavailable.

Pressuriser pressure control via normal spray is not taken into account. At the end of the transient, Pressuriser pressure control via the Pressuriser heaters is accounted for to slow down the decrease of primary pressure.

SG pressure control via GCT [MSB] is taken into account.

Pressuriser level control is not modelled in the present analysis.

3.1.2.6.6. Assumptions related to systems

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

The F2 functions[3] assumed to operate are: RBS [EBS] boration, RCP [RCS] pumps cut-off.

The NC systems assumed to operate are: Pressuriser heaters, GCT [MSB], AAD [SSS].

The setpoints, delays and flow capacities are listed in Section 16.1.3.1.2 - Table 2.

3.1.2.6.7. DNBR calculation

DNBR calculation is performed by considering the axial distribution in the hot channel and the F∆H provided by 3D core calculation (SMART, Appendix 14A) at minimum DNBR. For RRC-A analyses, no penalty on the local thermal power is considered for DNBR calculation.

3.1.2.6.8. Special assumptions not included in the above

- AAD [SSS] pump starts automatically after loss of main feedwater pumps (20 seconds start-up delay) with a flow capacity of 350 m^3/h,
- the ATWS signal is actuated 20 seconds after reactor trip signal, starting the RBS [EBS] with 7000 ppm boron acid injection (corresponding to 11200 ppm natural boron). When the SG level wide range is lower than MIN2, all main coolant pumps are tripped,
- automatic boration via RCV [CVCS] pumps is not credited in the analysis.

3.1.2.7. Results and Conclusions

3.1.2.7.1. Results for EPR at 4250 MWth [Ref-1]

Initial state 60% NP:

Section 16.1.3.1.2 - Table 3 gives the typical sequence of events.

Changes in typical parameters with time are shown on Section 16.1.3.1.2 - Figures 1 to 7.

The maximum pressure peak downstream of the RCP [RCS] pumps is reached just before the initial opening of the first PSV. It does not correspond to the Pressuriser pressure peak, which is achieved at 363 seconds, when the RCP [RCS] pumps have been tripped.

The MSSV and VDA [MSRT] are not active during the transient.

---

[3] RBS [EBS] is an F1A system, automatically actuated by the F2 classified ATWS signal. The automatic RBS [EBS] boration function is then F2 classified.

Second and third PSV are activated once during the transient, at 365 seconds.

The reactor trip signal is reached 77 seconds after the LOFW occurrence.

The ATWS signal actuates the RBS [EBS] boration at 97 seconds. The "SG level WR < MIN2" signal is reached at 134 seconds and all RCP [RCS] pumps are tripped.

The ASG [EFWS] starts injecting at 148 seconds.

The Pressuriser is filled at 236 seconds.

The minimum DNBR occurs during the RCP [RCS] pumps coast down. However, since the RCP [RCS] pump trip is initiated at a time when the core power is reduced, the RRC-A sequence "ATWS – LOOP" remains the bounding case concerning DNBR. It shows that the DNBR limit of 1.21 is not reached.

Initial state 100% NP:

Section 16.1.3.1.2 - Table 4 gives the typical sequence of events.

Change in typical parameters versus time is presented on Section 16.1.3.1.2 - Figures 8 to 14.

The maximum pressure peak downstream of the RCP [RCS] pumps is reached just after the initial opening of the first PSV.

The MSSV are not activated during the transient.

The VDA [MSRT] are activated only once as a result of turbine trip.

Second and third PSV are not activated during the transient.

The reactor trip signal is reached 44 seconds after the LOFW occurrence.

The ATWS signal actuates the RBS [EBS] boration at 64 seconds. The "SG level WR < MIN2" signal is reached at 83 seconds and all RCP [RCS] pumps are tripped.

The ASG [EFWS] starts injecting at 97 seconds.

The Pressuriser is filled at 156 seconds.

The initial DNBR at the beginning of the transient is the lowest DNBR. The minimum DNBR reached during the RCP [RCS] pumps coast down is higher than the DNBR at the beginning of the transient.

The decay heat is safely removed by AAD [SSS]/ASG [EFWS] injection and GCT [MSB].

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

3.1.2.7.2. Conclusion for EPR at 4500 MWth

The accident was not analysed for 4500 MWth power. The fulfilment of the acceptance criteria is deduced from the result of the analysis from the EPR at 4250 MWth.

Impact of the pressuriser safety valves new design

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

The two PSV valves have the same discharge flow rate.

The main differences between the two valve designs that can have an impact on the results of the analysis are:

- The opening setpoint of the first SEMPELL-model valve is 1 bar higher than that for the first SEBIM-model valve.
- The opening setpoint of the third SEMPELL-model valve is 3 bar higher than that for the third SEBIM-model valve.
- The PSV opening time for the SEMPELL-model valve is shorter, 0.1 seconds, than that for the SEBIM-model valve, 1.5 seconds.

The hysteresis of the SEMPELL-model valve, 16% of the opening setpoint, 28 bar for the first PSV, is higher than that for the SEBIM-model valve, 10 bar.

Fulfilment of the acceptance criteria of the maximum RCP [RCS] pressure

The differences between EPR_{4250} and $EPR_{4500,}$ which have an impact of the primary pressure are as follows:

- the power level increased by 6%
- the average primary temperature increased by 1°C

The remaining neutronic and thermal-hydraulic inputs used in the analysis are identical to the EPR_{4250} inputs with the exception of the initial boron concentration in the core, which was adapted to the new power level. The boron concentration was increased by about 240 ppm for the case of 100% Nominal Power (assumptions that were used for the analysis are shown in section 3.1.5).

The discharge and injection flow capacities are identical.

One of the consequences of the inadvertent opening of GCT [MSB] is the increase in primary pressure due to the non-equilibrium of the power between primary and secondary. The same setpoints to open the PSV are reached for EPR_{4250}. The discharge flow capacity in the primary and secondary sides is identical to the EPR_{4250}. Hence, the increase in the primary pressure is limited. Taking into account the margin found for EPR_{4250} (106% calculated pressure for the criteria of 130%) and the increase in pressure limited to 6%, it is concluded that the criteria of not exceeding 130% DP will be met for EPR_{4500} MW.

For cases initiated at 60% NP and 100% NP, the first PSV discharge capacity is sufficient to remove a significant amount of energy and this results in a decrease in the primary pressure. The maximum pressure occurs just before the initial opening of the first PSV for the case initiated at 60% NP. Therefore the SEMPELL-model valves will have no impact on the peak pressure as the first PSV does not discharge until after the peak pressure has occurred.

For the case initiated at 100% NP the maximum pressure occurs just after the initial opening of the first set PSV. As the opening setpoint of the first set SEMPELL-model valve is 1 bar higher than that for the equivalent SEBIM-model valve, it will be reached slightly later. The discharge flow capacity is the same for the two models so the maximum pressure will also occur within the same time interval after the opening of the valve. However as the opening time of the SEMPELL-model valve is very short, 0.1 seconds, the pressure peak will be less than one bar higher.

By considering the margin obtained for the EPR_{4250} with SEBIM-model valves, a maximum of 106% DP compared to the criterion of 130% DP, and the power increase which leads to a further pressure increase limited to 6%, it can be concluded that the acceptance criterion of 130% DP will be met for EPR_{4500} with SEMPELL-model valves.

Fulfilment of the acceptance criteria of the minimum DNBR

The differences between EPR_{4250} and $EPR_{4500,}$ which have an impact on DNBR are as follows:

- the power level increased by 6%
- the average primary temperature increased by 1°C

The remaining neutronic and thermal-hydraulic data used in the analysis are identical to the EPR_{4250} inputs with the exception of the initial boron concentration in the core, which was adapted to the new power level. The boron concentration was increased by about 240 ppm for the case of 100% Nominal Power (assumptions that were used for the analysis are shown in section 3.1.5).

The increase in initial power and the primary temperature will have a negative impact on the DNBR value. It is predicted that there is a decrease in the margin compared to the criteria. However, taking into the account the margin found for EPR_{4250} (3.9 for criteria of 1.21) and increase in the power by 6%, it can be concluded that criteria of 1.21 will be met for EPR_{4500}.

The hysteresis of the SEMPELL-model valves is higher than that of the SEBIM-model valves. Therefore the initial opening of PSV would remove more energy and the primary pressure would decrease to about 147 bar, compared to about 164 bar with the SEBIM-model valve, before increasing again once the PSV has closed. For the two cases, the RCP [RCS] pressure at the time of pump coast down will be lower which will have a adverse impact on the DNBR margin. However, taking into account the significant margin to the minimum DNBR calculated for EPR_{4250}, it can be concluded that the acceptance criterion of 1.21 will be met for the EPR_{4500} with SEMPELL-model valve.

3.1.2.7.3. Criteria fulfilment

The maximum pressure peak downstream of the RCP [RCS] pumps is obtained for the 100% NP initial state LOFW case. The value of the peak is smaller than the acceptance criterion of 130% DP, by a high margin.

The initial DNBR at the beginning of the transient for the 100% NP case is the lowest DNBR during the entire transient. This is explained by the increase of the primary pressure and the power reduction that occur over the temperature increase. As a result, the DNBR remains higher than the acceptance criterion of 1.21.

Due to higher initial margin, there is no risk of DNBR during the 60% NP case transient.

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

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

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

3.1.2.8. System Sizing

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

SECTION 16.1.3.1.2 - TABLE 1

EPR 4250 MW - Initial State
ATWS LOFW
Initial State 60% NP and 100% NP

Parameter	Initial value
Initial state= 60%	
RCP [RCS] flow rate	T/H design flow rate
Power	60% NP (nom)
Pressure	155 bar (nom)
Average temperature	311.8°C (nom)
Pressuriser level	55% (nom)
SG pressure	87.8 bar (nom)
ARE [MFWS] inlet temperature	203.5°C (nom)
SG level	49% NR (nom)
Initial state= 100%	
RCP [RCS] flow rate	T/H design flow rate
Power	100% NP
Pressure	155 bar (nom)
Average temperature	311.8°C (nom)
Pressuriser level	56% (nom)
SG pressure	78 bar (nom)
ARE [MFWS] inlet temperature	230°C (nom)
SG level	49% NR (nom)

SECTION 16.1.3.1.2 - TABLE 2

EPR 4250 MW - Main Assumptions
ATWS LOFW
Initial State 60% NP and 100% NP

PARAMETER	VALUE
Reactor trip / turbine trip signals on SG level < MIN1 (NR)	
Setpoint	13.8 m (20%)
Delay	1.8 s
ATWS signal (reactor trip signal and high rod position (or high flux) after delay, actuates RBS [EBS] boration:	
Delay (after RT signal)	20 s
Capacity (per RCP [RCS] loop)	1.4 kg/s
RBS [EBS] boron concentration (natural boron)	11200 ppm
AAD [SSS] actuation on loss of main feedwater pumps	
Delay	20 s
Capacity	350 te/h
ASG [EFWS] actuation on SG level (wide range cold side) < MIN2	
Setpoint	7.85 m (40%)
Delay	1.5 + 15 = 16.5 s
Capacity	90 te/h per SG at 97 bar
4 ASG [EFWS] tanks water content	1680 te
MS bypass actuation on SG pressure > MAX	
Setpoint (with uncertainty)	91.5 bar
Delay	0.9 + 1.5 = 2.4 s
Capacity	4600 te/h saturated steam under 75 bar (50% of full load flow rate)
MS relief train actuation on SG pressure > MAX1	
Setpoint	95.5 bar
Delay	0.9 + 1.5 + 0.5 = 2.9 s
VDA [MSRT] closing delay	40 s
Capacity	1150 te/h under 100 bar (50% of full load flow rate/SG)
MS safety valves	
Setpoint	105 bar
Accumulation	3%
Capacity	575 te/h under 100 bar, 2 per SG (2 x 25% of full load flow rate/SG)
Pressuriser safety valves	
Setpoints (with uncertainty)	175.5 / 179.5 / 179.5 bar
dead time	0.5 s
opening time	1.5 s
Capacity	steam 300 te/h, liquid 450 te/h under 176 bar
Hysteresis (preliminary value)	10 bar

SECTION 16.1.3.1.2 - TABLE 3

EPR 4250 MW - Typical Sequence of Events
ATWS LOFW
Initial State 60% NP

EVENT	TIME (second)
LOFW:	
ARE [MFWS] cut off	0
AAD [SSS] actuation	20
1st PSV opening/closing	45 / 52
Primary pressure peak	45
GCT [MSB] opening	50.5
Reactor trip signal (SG level < MIN1)	
Turbine trip	77
GCT [MSB] full opening	78
ATWS signal / RBS [EBS] actuation	97
1st PSV opening/closing	112 / 119
RCP [RCS] pumps trip (ATWS and SG level < MIN2 signal)	134
ASG [EFWS] injection (SG level < MIN2 signal)	148
1st PSV opening/closing	156 / 164
1st PSV opening/closing	182 / 192
1st PSV opening/closing	202 / 211
1st PSV opening/closing	228 / 243
Pressuriser filled	236
1st PSV opening/closing	273 / 491
2nd and 3rd PSV opening/closing	365 / 413
RBS [EBS] boron arrival in the core	388
SG minimum water mass	472

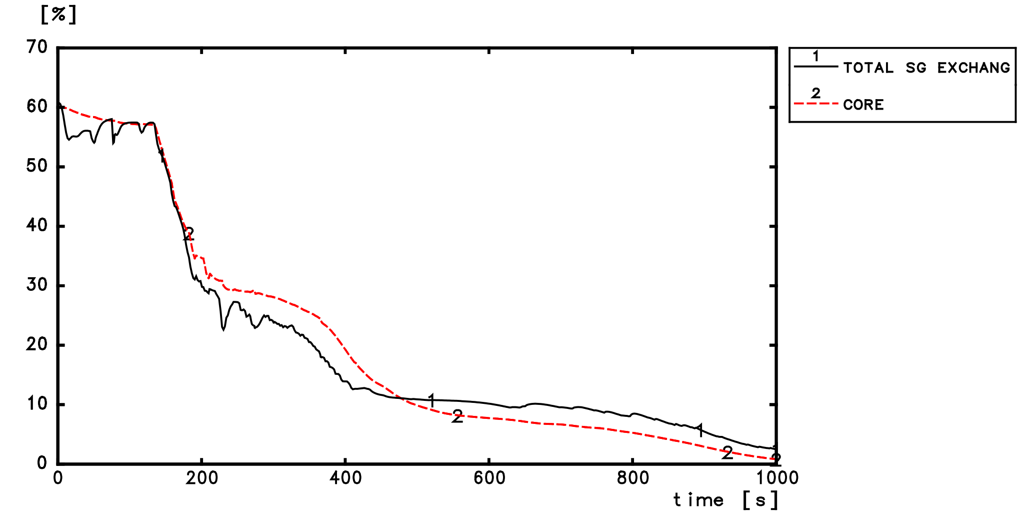
SECTION 16.1.3.1.2 - TABLE 4

EPR 4250 MW - Typical Sequence of Events ATWS LOFW
Initial State 100% NP

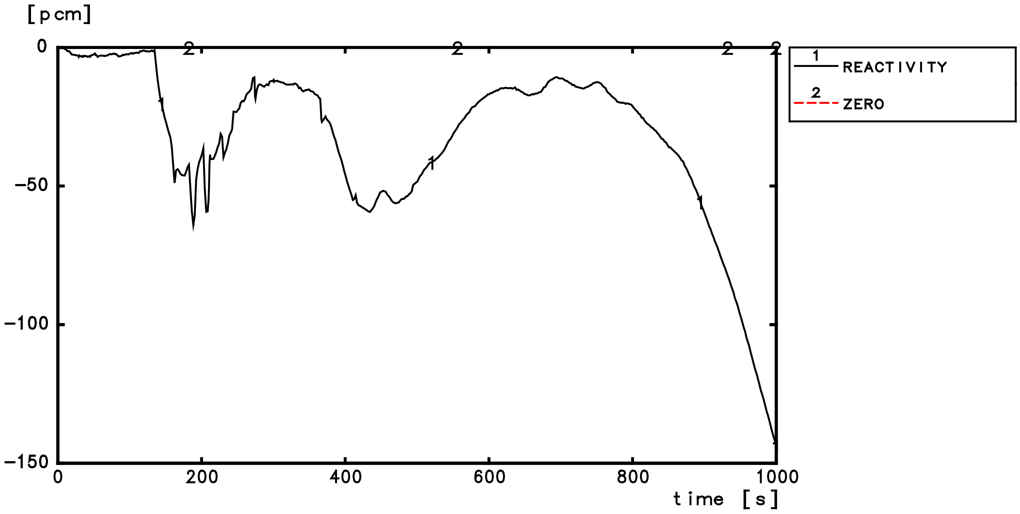
EVENT	TIME (second)
LOFW:	
ARE [MFWS] cut off	0
AAD [SSS] actuation	20
Reactor trip signal (SG level < MIN1)	
Turbine trip	44
GCT [MSB] opening	46
VDA [MSRT] opening / closing	62 / 105
1st PSV opening/closing	46 / 67
Primary pressure peak	53
ATWS signal / RBS [EBS] actuation	64
RCP [RCS] pumps trip (ATWS and SG level < MIN2 signal)	83
1st PSV opening/closing	93 / 99
ASG [EFWS] injection (SG level < MIN2)	97
1st PSV opening/closing	133 / 142
1st PSV opening/closing	153 / 161
Pressuriser filled	156
1st PSV opening/closing	171 / 391
RBS [EBS] boron arrival in the core	352
SG minimum water mass	392

SECTION 16.1.3.1.2 - FIGURE 1

EPR 4250 MW - Typical Power (Core and SG) – Reactivity
ATWS LOFW
Initial State 60% NP



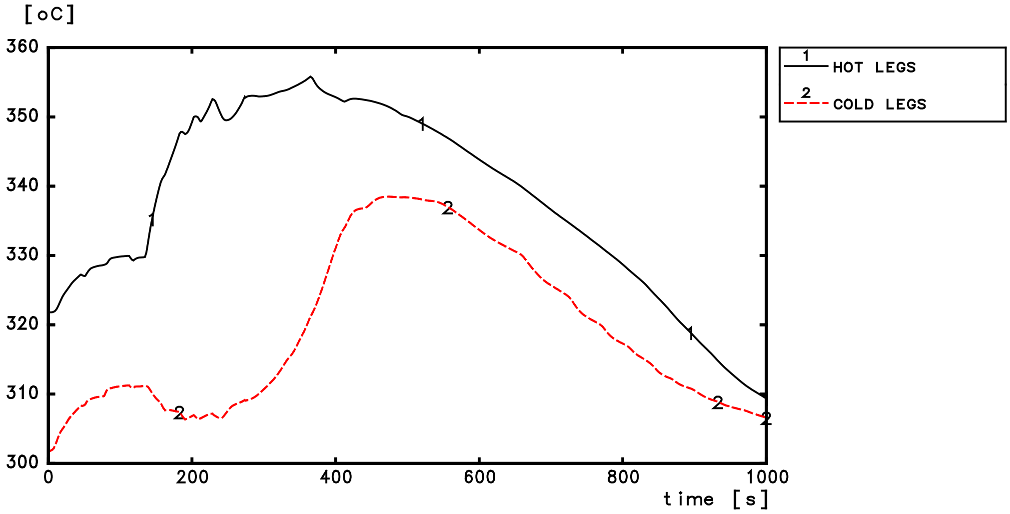
POWER (CORE AND SG)



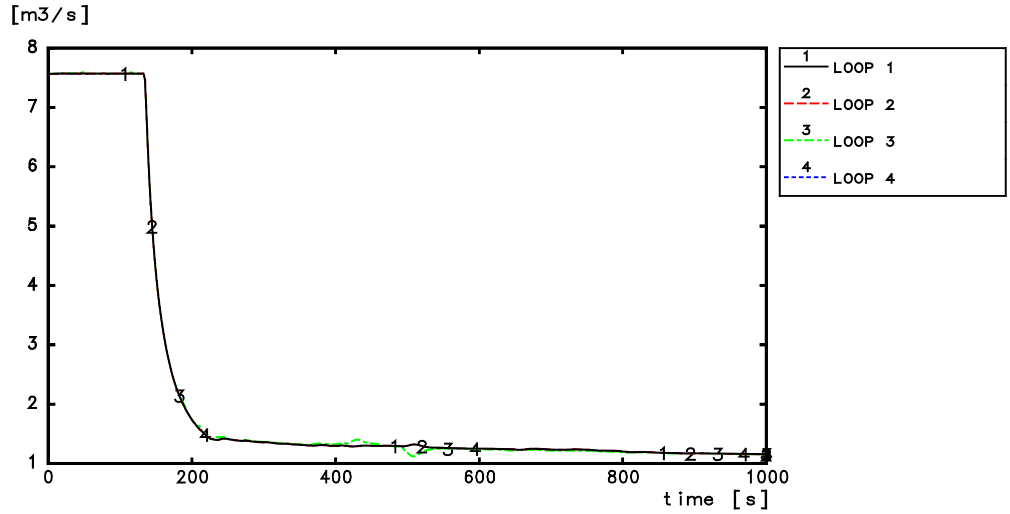
REACTIVITY

SECTION 16.1.3.1.2 - FIGURE 2

EPR 4250 MW - Typical RCP [RCS] Temperatures – RCP [RCS] Flow Rate
ATWS LOFW
Initial State 60% NP



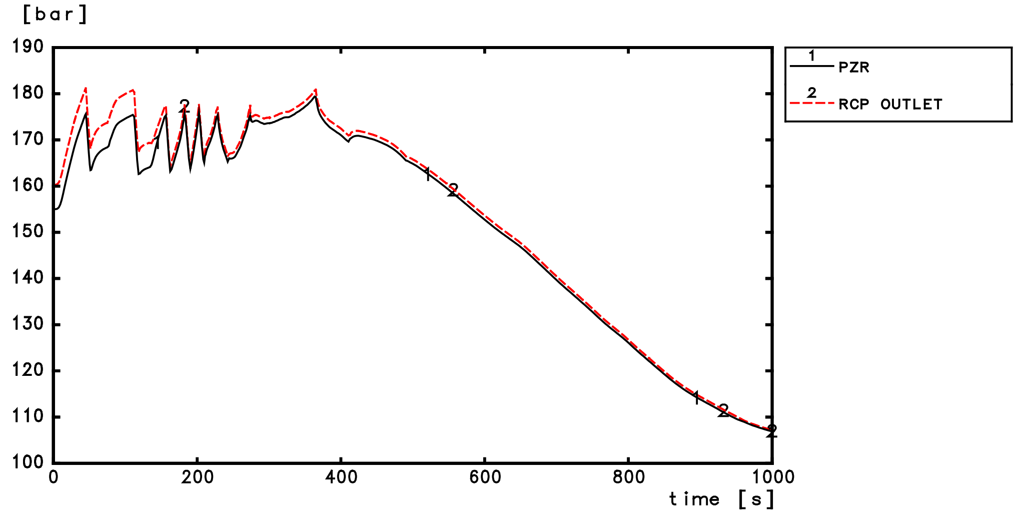
RCS TEMPERATURES



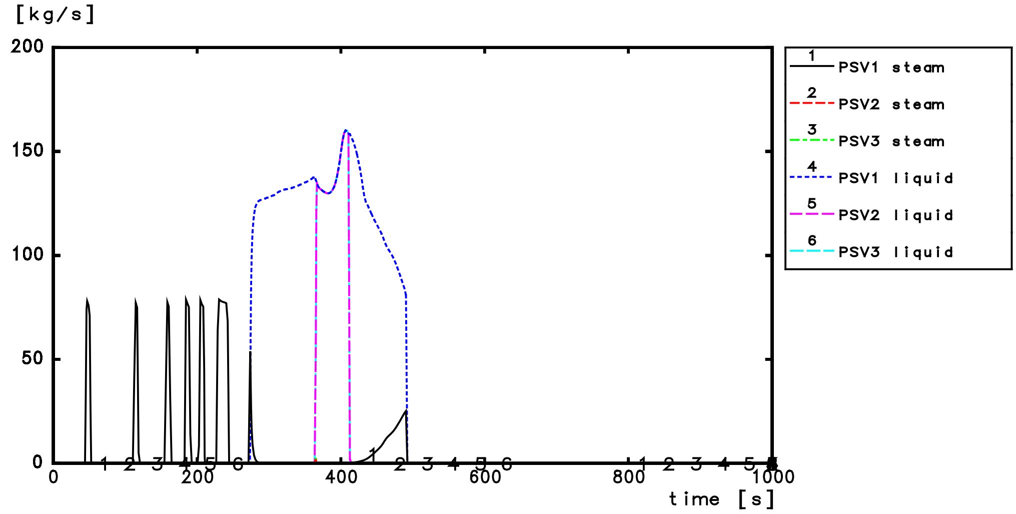
RCS FLOWRATE

SECTION 16.1.3.1.2 - FIGURE 3

EPR 4250 MW - Typical Primary Pressure – PSV Flow Rate
ATWS LOFW
Initial State 60% NP



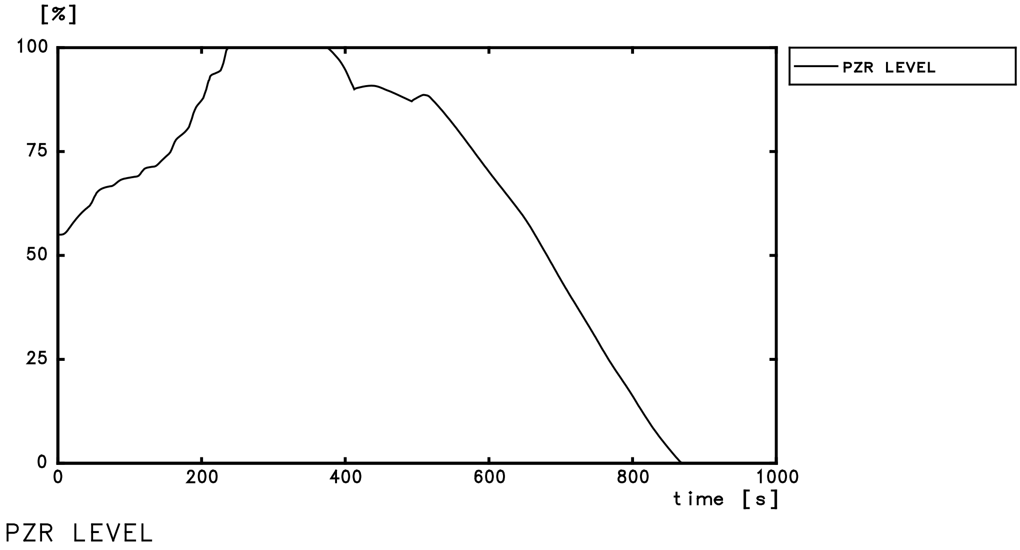
PRIMARY PRESSURE

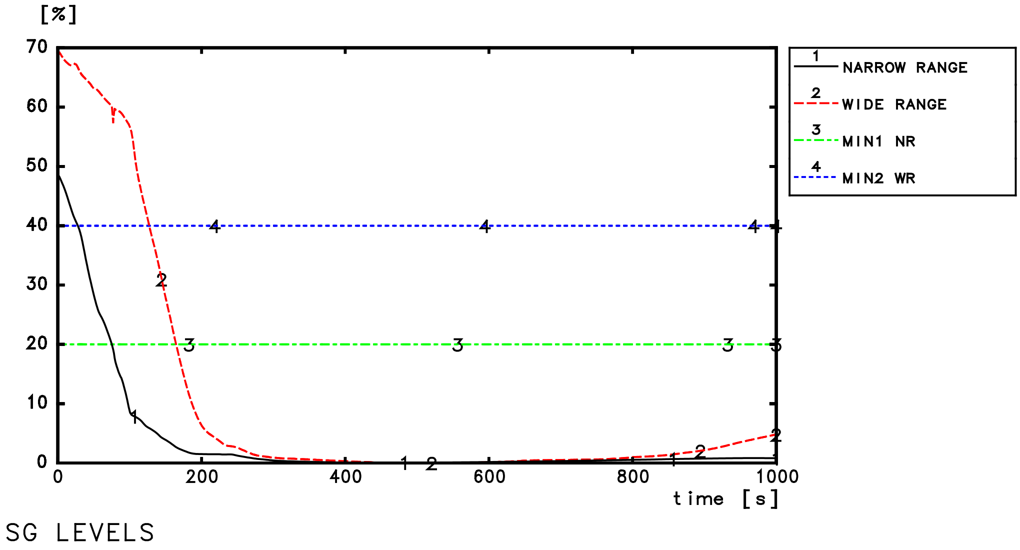


PSV FLOWRATE

SECTION 16.1.3.1.2 - FIGURE 4

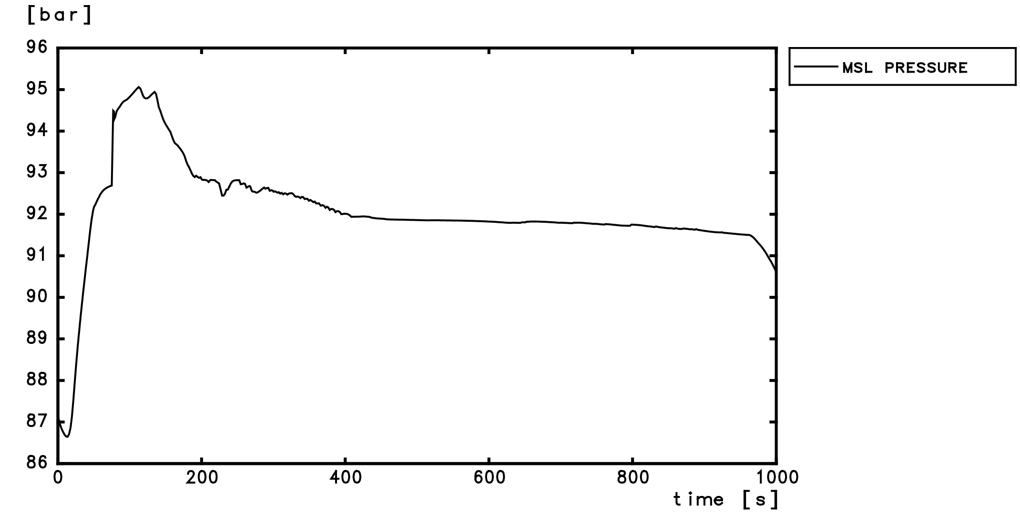
EPR 4250 MW - Typical Pressuriser Level – SG Levels
ATWS LOFW
Initial State 60% NP



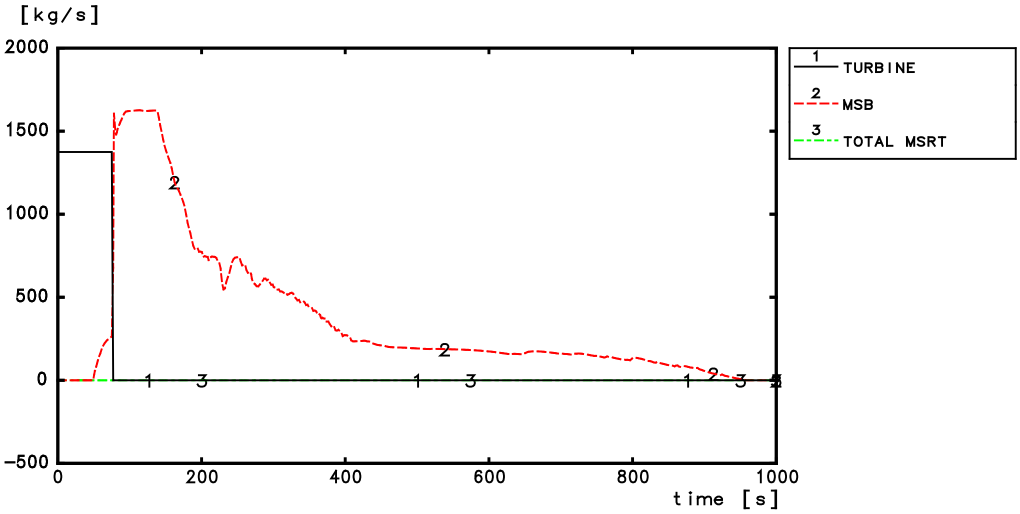


SECTION 16.1.3.1.2 - FIGURE 5

EPR 4250 MW - Typical Main Steam Pressure – Steam Flow Rate
ATWS LOFW
Initial State 60% NP



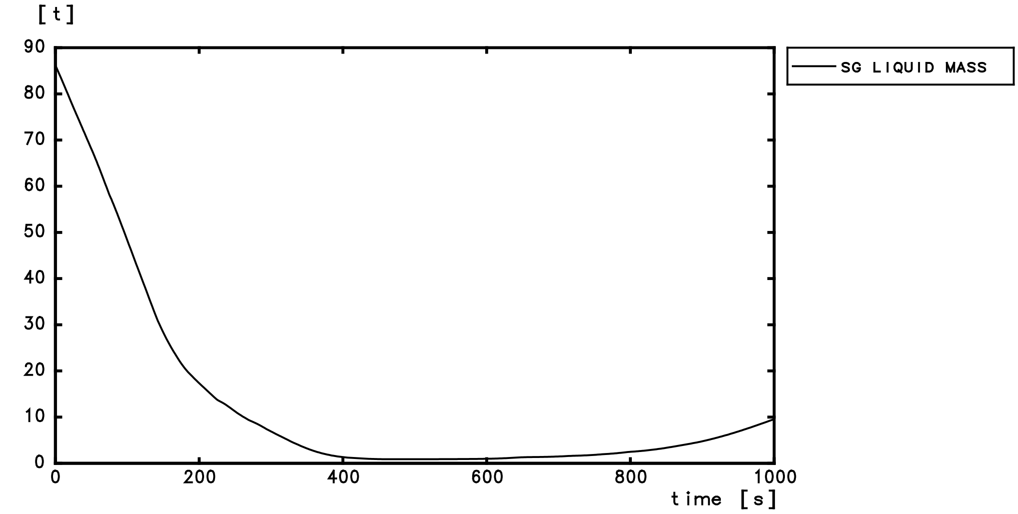
MAIN STEAM PRESSURE



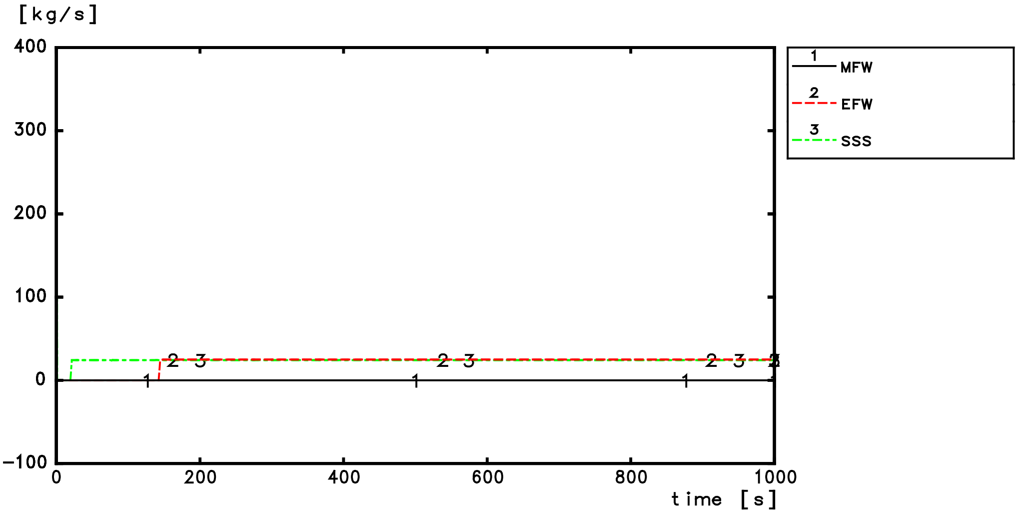
STEAM FLOWRATE

SECTION 16.1.3.1.2 - FIGURE 6

Typical SG Liquid Mass – Feedwater Flow Rate per SG
ATWS LOFW
Initial State 60% NP



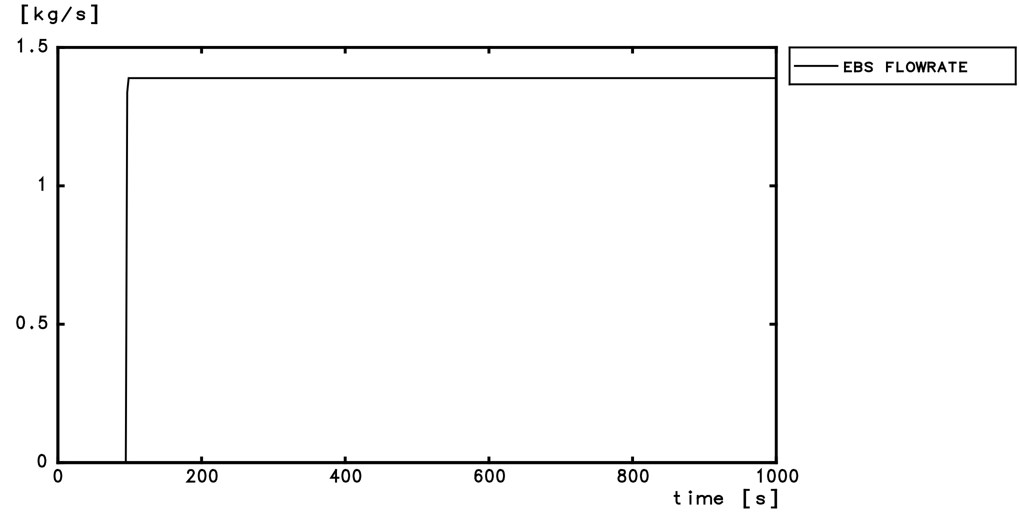
SG LIQUID MASS



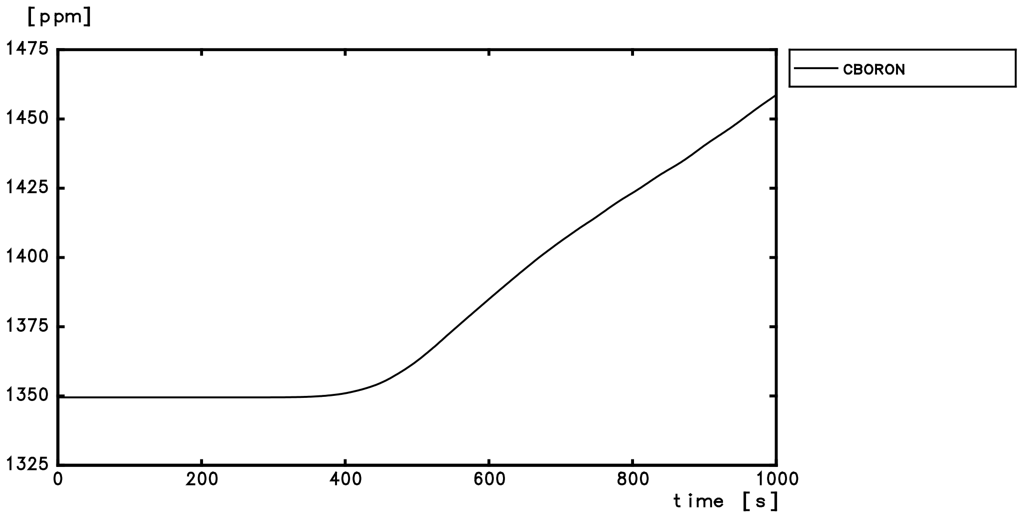
FEEDWATER FLOWRATE PER SG

SECTION 16.1.3.1.2 - FIGURE 7

EPR 4250 MW -
Typical RBS [EBS] Flow Rate per Loop – Core Boron Concentration (Natural Boron)
ATWS LOFW
Initial State 60% NP



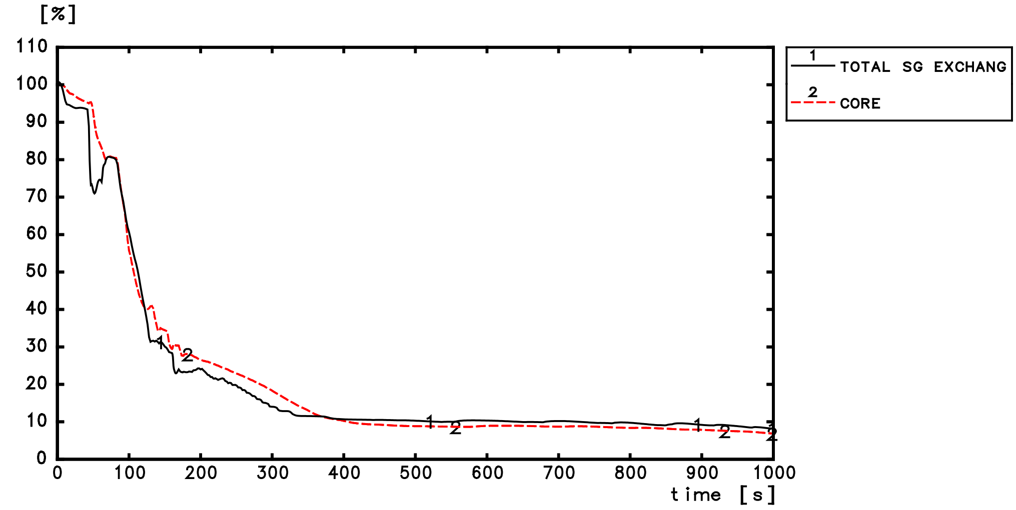
EBS FLOWRATE PER LOOP



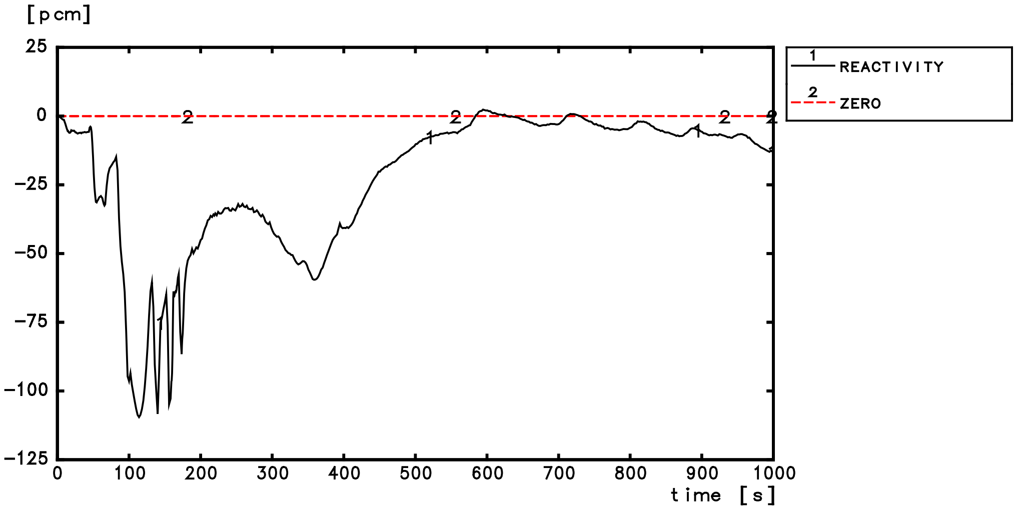
CORE BORON CONCENTRATION (NATURAL BORON)

SECTION 16.1.3.1.2 - FIGURE 8

EPR 4250 MW - Typical Power (Core and SG) – Reactivity
ATWS LOFW
Initial State 100% NP



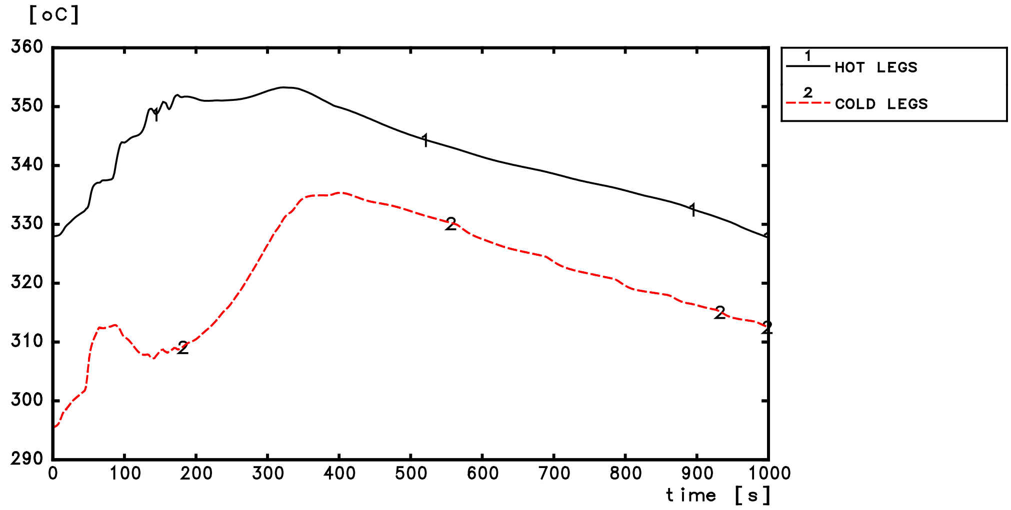
POWER (CORE AND SG)



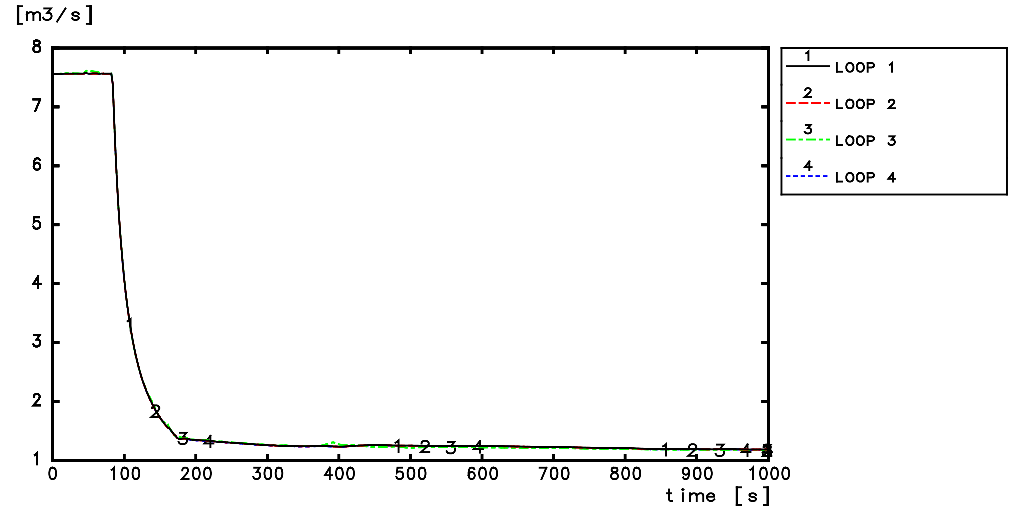
REACTIVITY

SECTION 16.1.3.1.2 - FIGURE 9

EPR 4250 MW - Typical RCP [RCS] Temperatures – RCP [RCS] Flow Rate
ATWS LOFW
Initial State 100% NP



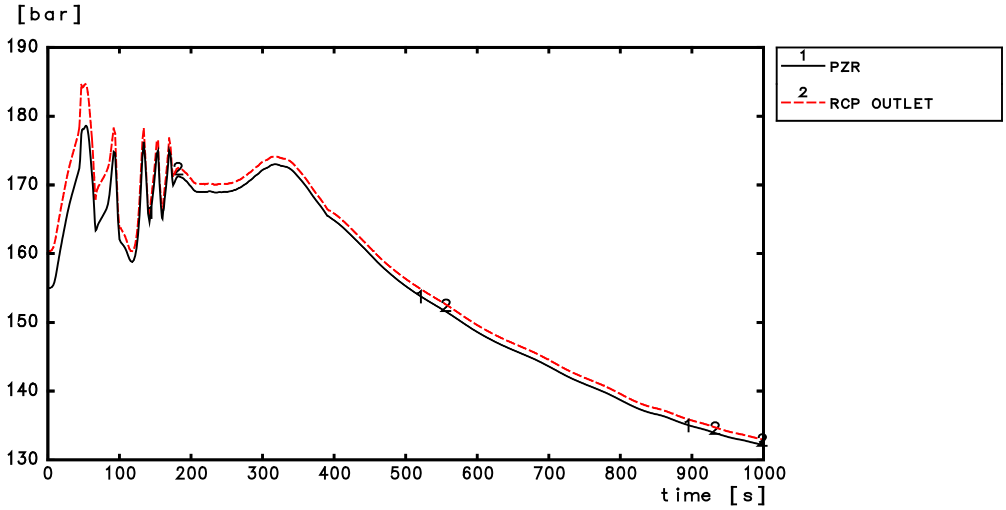
RCS TEMPERATURES



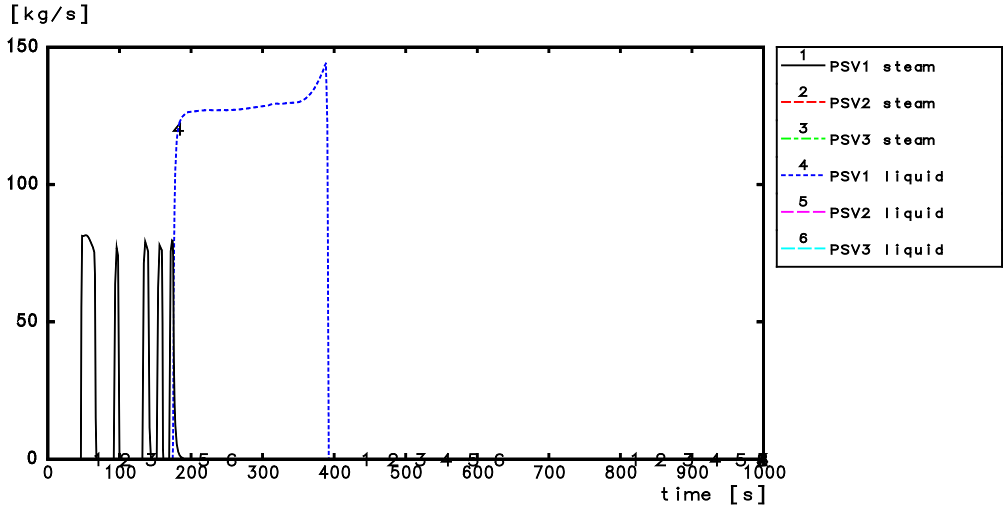
RCS FLOWRATE

SECTION 16.1.3.1.2 - FIGURE 10

EPR 4250 MW - Typical Primary Pressure – PSV Flow Rate
ATWS LOFW
Initial State 100% NP



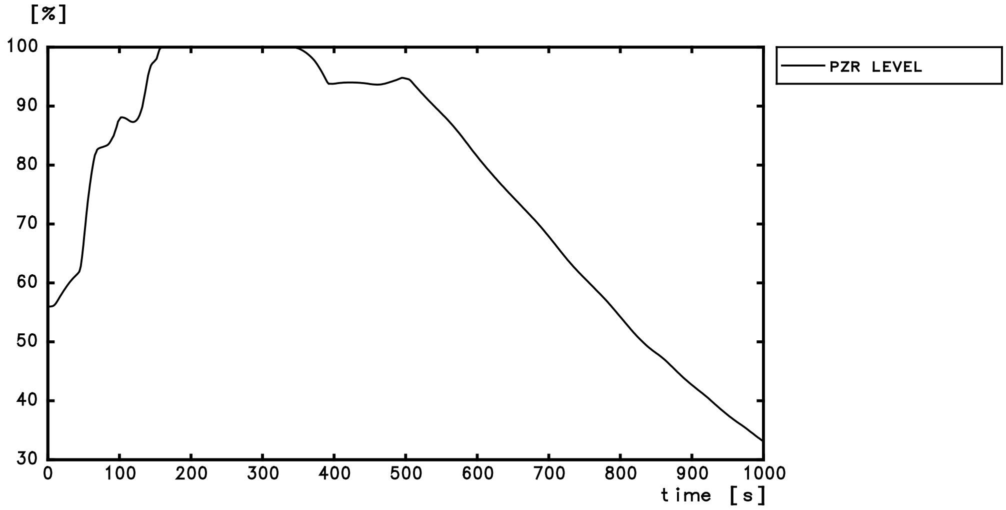
PRIMARY PRESSURE



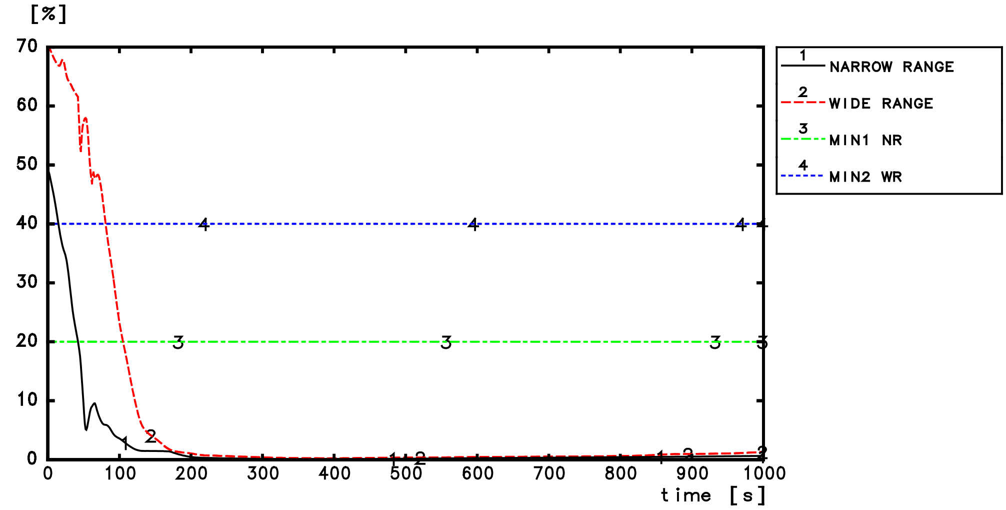
PSV FLOWRATE

SECTION 16.1.3.1.2 - FIGURE 11

EPR 4250 MW - Typical Pressuriser Level – SG Levels
ATWS LOFW
Initial State 100% NP



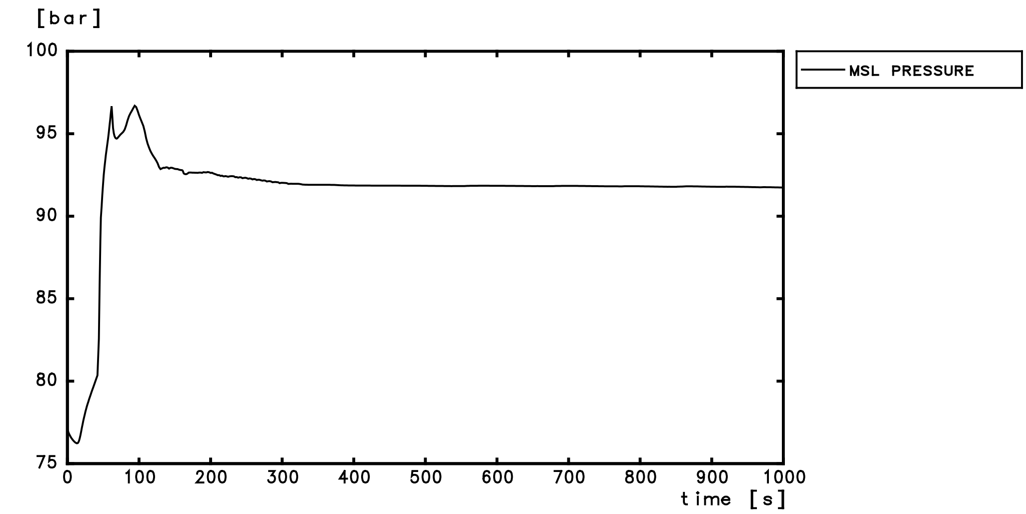
PZR LEVEL



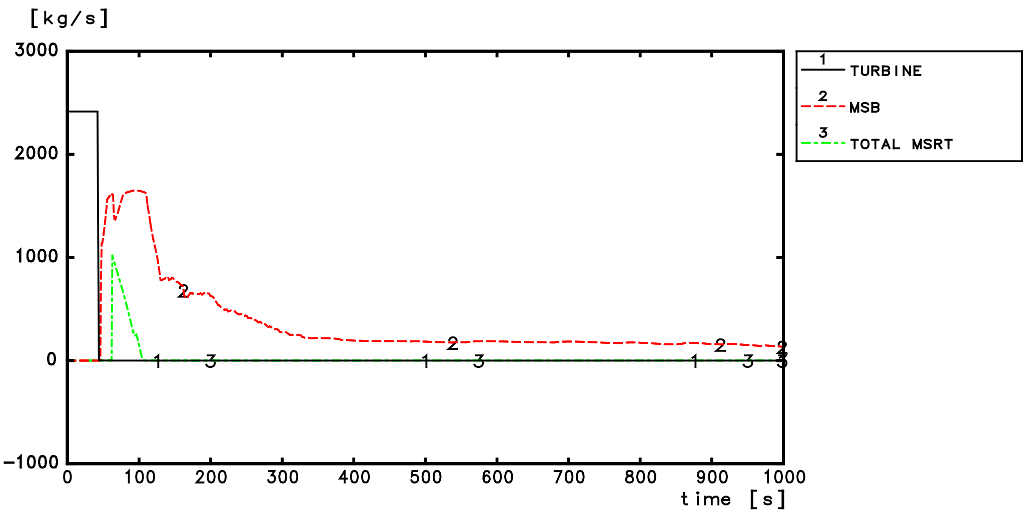
SG LEVELS

SECTION 16.1.3.1.2 - FIGURE 12

EPR 4250 MW - Typical Main Steam Pressure – Steam Flow Rate
ATWS LOFW
Initial State 100% NP



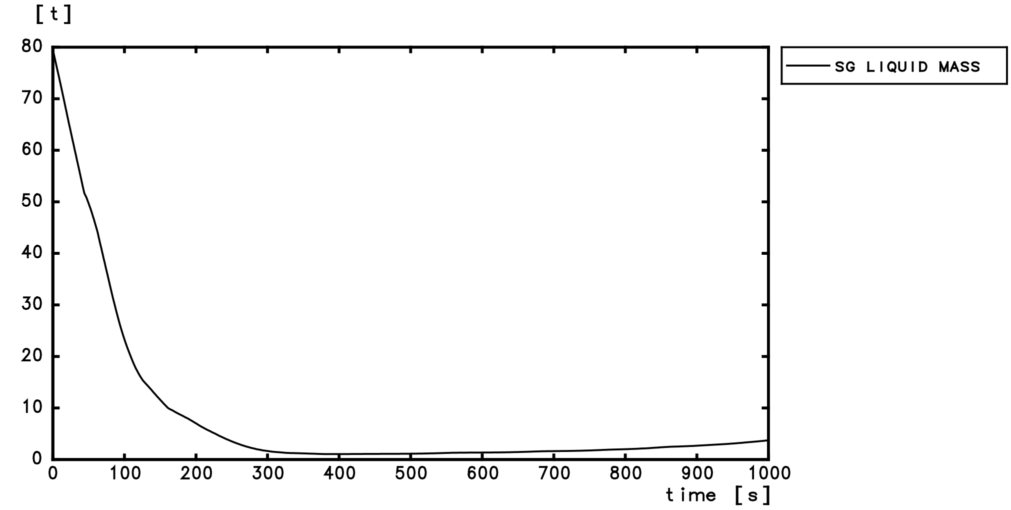
MAIN STEAM PRESSURE



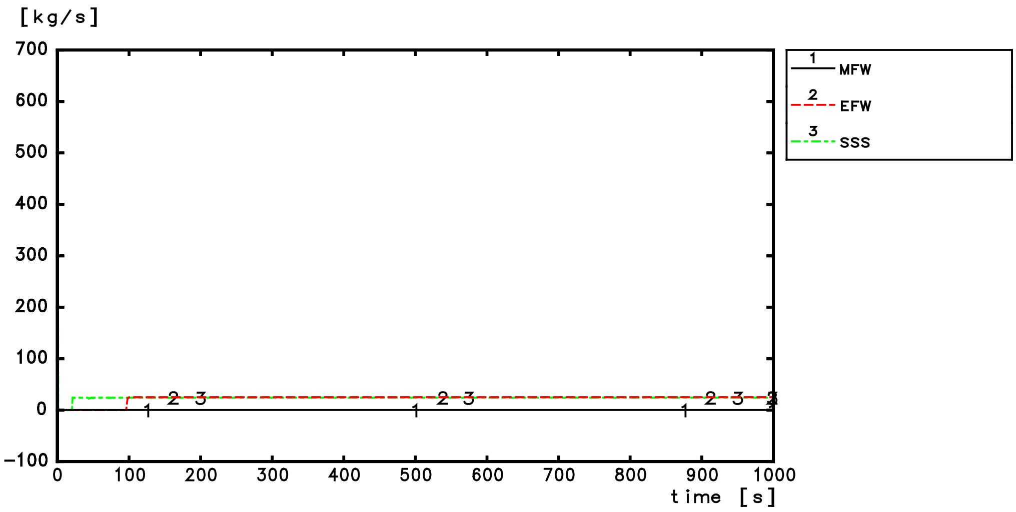
STEAM FLOWRATE

SECTION 16.1.3.1.2 - FIGURE 13

EPR 4250 MW - Typical SG Liquid Mass – Feedwater Flow Rate per SG
ATWS LOFW
Initial State 100% NP



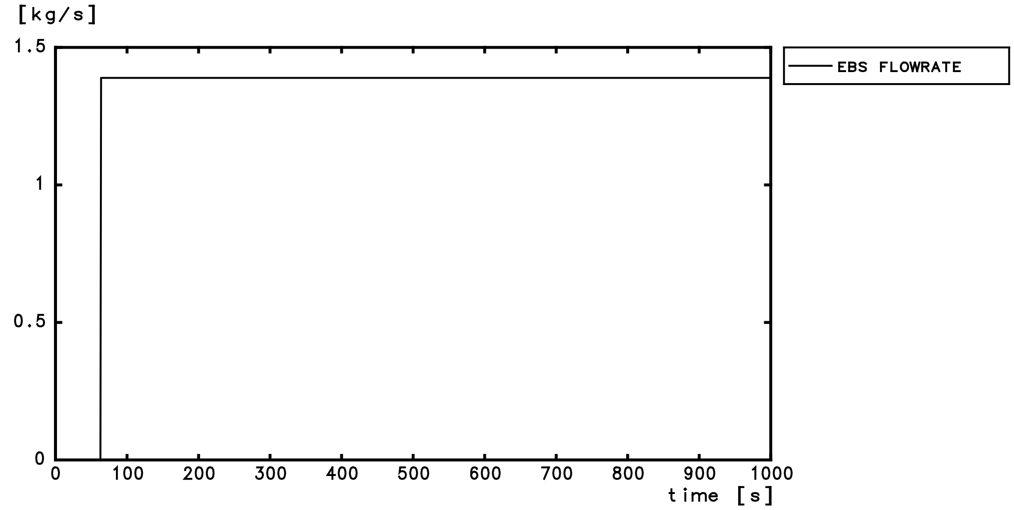
SG LIQUID MASS



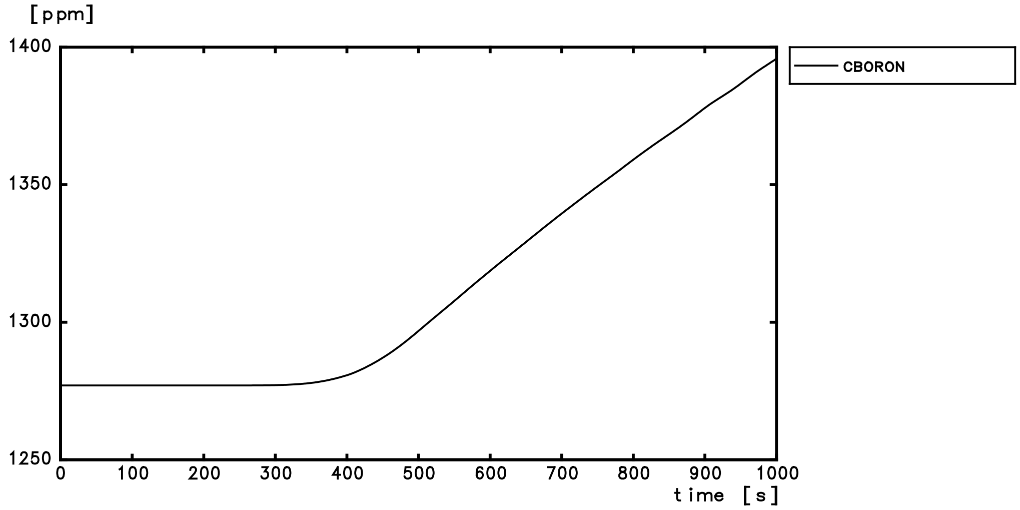
FEEDWATER FLOWRATE PER SG

SECTION 16.1.3.1.2 - FIGURE 14

EPR 4250 MW - Typical RBS [EBS] Flow Rate per Loop – Core Boron Concentration (Natural Boron)
ATWS LOFW
Initial State 100% NP



EBS FLOWRATE PER LOOP



CORE BORON CONCENTRATION (NATURAL BORON)

3.1.3. ATWS - Loss of Offsite Power (LOOP)

3.1.3.1. Identification of causes

A complete failure of the reactor trip system on demand from the Reactor Protection System (RPR [PS]) can result from:

- either the failure of the F1A reactor trip signals (i.e. all trip signals do not de-energise the drive coils),
- or the mechanical failure of the control/shutdown rods after de-energizing of all control/shutdown rod drive coils. In this case, the actuation of the rods due to control or limitation signals also fails.

The current section 3.1.3 addresses the second cause of ATWS, namely mechanical failure of the control/shutdown rods. The first cause, related to RPR [PS] failure is addressed in section 3.2.3.

3.1.3.2. Typical sequence of events

The loss of offsite power induces a turbine trip, and switches off all RCP [RCS] pumps and ARE [MFWS] pumps. In addition, the AAD [SSS] pump cannot be operated.

The ARE [MFWS] supply cut off leads to a decrease in secondary side heat removal and the primary flow coast-down further reduces the capacity of the primary coolant to remove heat from the core.

Consequently, primary and secondary pressures and temperatures increase.

Reactor scram, or any other power reduction by rod insertion, does not occur on reactor trip signal because of mechanical failure of the rods. The heating of the core causes reactivity decrease via the moderator temperature feedback effect.

The secondary side heat removal is ensured by:

- the GCT [MSB], until GCT [MSB] valves close on high condenser pressure (≈ 10 seconds after the LOOP occurrence)
- subsequently one VDA [MSRT] and two SG safety valves (MSSV) per SG, fed only by ASG [EFWS].

The PSV open to limit the primary side overpressure.

Twenty seconds after the F1A RT signal, the ATWS signal is triggered by the reactor protection system, from "RT signal and high rod position (or high flux) after temporisation". This ATWS signal (and associated actions) is an RRC-A feature, specifically implemented to cope with the RRC-A sequences "ATWS by rods failure", being F2 classified. It actuates the boration with 7000 ppm enriched boron (corresponding to 11200 ppm natural boron) via the RBS [EBS] pumps. It also causes the automatic isolation of the RCV [CVCS] tank.

The boration by the RCV [CVCS] pumps is not considered (i.e. the injected water has the initial primary boron concentration of BOL).

3.1.3.3. Safety criteria

It must be demonstrated that the "final state for RRC-A analyses" can be reached by assuring:

- long term core sub-criticality,
- decay heat removal,
- activity release control by maintaining the integrity of the barriers in accordance with the PCC-4 acceptance criteria.

For this demonstration, the following decoupling criteria are considered:

- the DNBR remains above the design limit 1.21 (see Sub-chapters 4.4 and 14.1),
- the primary system integrity is not impaired. The preliminary decoupling criterion is that the maximum pressure cannot exceed 130% of the calculated pressure (i.e. 228.5 bar abs). (Refer to Sub-chapter 3.4.)

3.1.3.4. Protection and mitigation actions

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

- reactor trip signal on "RCP [RCS] pump speed < MIN1" (F1A),
- reactor trip signal on "SG pressure > MAX1" (F1A),
- reactor trip signal on "SG level (narrow range) < MIN1" (F1A),
- reactor trip signal on "PZR pressure > MAX2" (F1A),
- ATWS signal on "reactor trip signal, and high rods position (or high flux) after temporisation" (F2),
- RBS [EBS] actuation on ATWS signal (F2),
- RCV [CVCS] isolation on ATWS signal (F2),
- Pressuriser normal spray actuation on "PZR pressure > MAX1" (NC),
- GCT [MSB] opening on "header pressure > MAX" (NC),
- VDA [MSRT] opening on "SG pressure > MAX1" (F1A),
- ASG [EFWS] actuation on "SG level (wide range) < MIN2" (F1A).

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

3.1.3.5. Methods of analysis

The analysis is carried out using internal coupling of the following computer codes:

- the MANTA V3.7 code (see Appendix 14A) for the overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), accounting for NC/F2/F1 systems operations,
- the SMART V4.5 / FLICA-IIIF V3 codes (see Appendix 14A) for the neutronic and thermal-hydraulic behaviour of the core.

The DNBR calculation is performed with the FLICA code (appendix 14A).

The method of analysis is based on the following approach:

- identification of the dominant events,
- verification of the adequacy of the code to simulate those events.

The dominant events of this transient are:

a) turbine trip, trip of RCP [RCS] pumps and ARE [MFWS] pumps without reactor scram,

b) RCP [RCS] heat-up and pressuriser filling with consequential RCP [RCS] pressure increase (including fast overpressure when the pressuriser is in liquid phase),

c) reactor power decrease due to moderator density effect, and later due to boron injection,

d) large SG water inventory decrease until sufficient power reduction allows for the ASG [EFWS] flow to replace the boil off.

All these events are applicable to the MANTA, SMART and FLICA codes.

3.1.3.6. Specific assumptions

3.1.3.6.1. Transients analyses / initial power states

Considering the core power reduction due to the moderator effect resulting from the RCP [RCS] heat-up, it is conservative to analyse the event at Beginning Of Life (BOL), when the moderator coefficient is minimised (0 pcm/°C, or negative but close to 0 pcm/°C).

To cover the entire life of the plant, two transients are analysed:

- one transient with 60% NP initial state and no moderator effect, covering all power levels between 60% NP and 0% NP (for which the moderator coefficient is, at least 0 pcm/°C in absolute value),
- one transient with 100% NP initial state and -10 pcm/°C moderator coefficient, covering all power levels between 100% NP and 60% NP (for which the moderator coefficient is at least 10 pcm/°C in absolute value).

3.1.3.6.2. Single failure and maintenance

Single failure and maintenance do not have to be taken into account in RRC-A analyses.

3.1.3.6.3. Initial and boundary conditions

The initial conditions are those typical at 60% NP and 100% NP (Section 16.1.3.1.3 - Table 1).

RCP [RCS] thermal hydraulic design flow rate is assumed (not set by accident analyses rules).

All systems and I&C functions are available, except those lost by definition of the RRC-A sequence (e.g. no rods drop). For simplicity some systems are not considered.

Boundary conditions defining system efficiency are generally based on realistic characteristics.

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

3.1.3.6.4. Neutronic data

Neutronic data are calculated at the beginning of life (BOL).

The moderator coefficient is -10°C/pcm and 0°C/pcm at the beginning of the transient for the 100% NP and the 60% NP cases, respectively [Ref-1]. The Doppler temperature coefficient is maximised to make the rate of core power decrease conservative. Doppler temperature coefficient is set to -3.38 pcm/°C at the beginning of the transient (maximum in absolute value, covering all fuel management cycles) [Ref-1].

The reference fuel management is retained to maximise the FΔH when the moderator and the Doppler temperature coefficients have been set to their target values. The DNBR calculation is thus conservative.

The bounding initial RCP [RCS] boron concentration is 1280 ppm for the 100% NP case, and 1350 ppm for the 60% NP case (expressed as natural boron) [Ref-1].

3.1.3.6.5. Assumptions related to controls

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

Pressuriser pressure control via normal spray is not accounted for. At the end of the transient, pressuriser pressure control via the pressuriser emergency heaters is taken into account to slow down the decrease of primary pressure.

SG pressure control via GCT [MSB] is not accounted for because of the early GCT [MSB] valves closure on high condenser pressure (≈10 seconds after LOOP occurrence).

Pressuriser level control is not modelled in the present analysis.

3.1.3.6.6. Assumptions related to systems

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

The F2 functions[4] used are: RBS [EBS] boration, isolation of the RCV [CVCS], and RCP [RCS] pumps trip.

The setpoints, delays and flow capacities are shown in Section 16.1.3.1.3 - Table 2.

3.1.3.6.7. DNBR calculation

The DNBR calculation takes into account the axial distribution in the hot channel and the F∆H provided by 3D core calculation (SMART, Appendix 14A) at the time of minimum DNBR. For RRC-A analyses, no penalty on the local thermal power is considered for DNBR calculation.

3.1.3.7. Results and conclusion

3.1.3.7.1. Results for EPR 4250 MWth [Ref-1]

Initial state 60% NP, moderator coefficient 0 pcm/°C at beginning of transient:

Section 16.1.3.1.3 - Table 3 shows the typical sequence of events and Section 16.1.3.1.3 - Figure 1 shows the typical advancement of the main parameters.

The maximum pressure peak downstream of the RCP [RCS] pumps is 180.4 bar and it is reached just before the initial opening of the first PSV at 13.4 seconds.

Second and third PSV are not activated during the transient.

The SG safety valves (MSSV) are not activated during the transient.

The RT signal on "RCP [RCS] pump speed < MIN1" is initiated 2.5 seconds after LOOP occurrence.

The pressuriser is filled at 95 seconds (PZR level = 100%).

The ATWS signal actuates the RBS [EBS] boration at 53 seconds.

The "SG level WR < MIN2 " signal is reached at 190 seconds and the ASG [EFWS] starts injecting at 240 seconds.

Initial state 100% NP, moderator coefficient -10 pcm/°C at beginning of transient:

Section 16.1.3.1.3 - Table 4 gives the typical sequence of events and Section 16.1.3.1.3 - Figure 2 shows the typical advancement of the main parameters.

The maximum pressure peak downstream of the RCP [RCS] pumps is 184.7 bar. It is reached just after the initial opening of the second and third PSV at 10.5 seconds.

---

[4] RBS [EBS] is an F1A system, automatically actuated by the F2 classified ATWS signal. The automatic RBS [EBS] boration function is then F2 classified.

Second and third PSV are activated initially at 10 seconds (steam release), then twice at 378 seconds and 460 seconds (water release).

The SG safety valves (MSSV) safety valves are not activated during the transient.

The RT signal on "RCP [RCS] pump speed < MIN1" is initiated at 2.5 seconds after LOOP occurrence.

The pressuriser is filled at 94 seconds (PZR level = 100%).

The ATWS signal actuates the RBS [EBS] boration at 53 seconds.

The "SG level WR < MIN2" signal is reached at 150 seconds and the ASG [EFWS] starts injecting at 200 seconds.

Criteria fulfilment

The maximum pressure at the RCP [RCS] pumps outlet occurs in the LOOP case for an initial power of 100% NP. It reaches a value of 184.7 bar (105% of the design pressure) 10.5 seconds after the occurrence of the LOOP. The decoupling criterion of 130% of the design pressure is therefore met with a large margin.

The minimum DNBR is reached 14.7 seconds after the LOOP for the 100% NP case and is equal to 2.36, significantly above the 1.21 criterion. The minimum DNBR for the transient from 60% NP will be higher than the minimum DNBR for the transient from 100% NP due to the higher initial margin.

The heat in the SG is safely removed via the GCT [MSRT] and the ASG [EFWS].

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

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

3.1.3.7.2. Results for EPR 4500 MWth

The transient is not analysed for the EPR 4500 MWth. The decoupling criteria are inferred from the analysis results for the EPR 4250 MWth case.

Impact of the new design of the primary safety valves

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

The two PSV models have the same discharge flow rate.

The main differences between the two valve models that can have an impact on the results of the analysis are:

- The opening setpoint of the first SEMPELL-model valve is 1 bar higher than that for the first SEBIM-model valve.
- The opening setpoint of the third SEMPELL-model valve is 3 bar higher than that for the third SEBIM-model valve.

- The PSV opening time for the SEMPELL-model valve is shorter, 0.1 seconds, than that for the SEBIM-model valve, 1.5 seconds.

The hysteresis of the SEMPELL-model valve, 16% of the opening setpoint, 28 bar for the first PSV, is higher than that for the SEBIM-model valve, 10 bar.

RCP [RCS] maximum pressure criteria fulfilment

The observed differences between the EPR 4250 MWth and the EPR 4500 MWth having an impact on the primary pressure are as follows:

- Power increase level 6%

- Primary average temperature increase 1°C

Other neutronic and thermodynamic data are identical with those of the EPR 4250 MWth, except for the initial boron concentration in the core, which is adjusted to the new power level. This value is increased to approximately 240 ppm for the 100% NP (assumption used in section 3.1.5).

The flow capacities of the discharge devices (pressuriser safety valves) are identical.

One of the consequences of LOOP is the increase in primary pressure caused by the reduced ability to remove heat. In the same manner as the EPR 4250 MWth case, the pressuriser safety valves setpoint is reached. Since the discharge flow capacities for the primary and secondary are identical to the EPR 4250 MWth, the peak primary pressure will be limited. A higher peak pressure could be expected, however, given the pressure margin for the EPR 4250 MWth (105% calculated pressure for a criteria of 130%) and the restriction in power level increase to 6%, it can be concluded that the criterion of 130% DP for the EPR 4500 MWth will be met.

For the case initiated at 60% NP, the discharge capacity of the first PSV is sufficient to release the excess energy in the RCP [RCS] and to cause it to depressurise. The maximum pressure occurs just before the initial opening of the first PSV.

Therefore the SEMPELL-model valves will have no impact on the peak pressure as the first PSV does not discharge at the time of the peak pressure.

Considering the results of the SEBIM-model valve assessment and the power increase leading to a further pressure increase limited to 6%, it can be concluded that the acceptance criterion of 130% DP will be met for the EPR_{4500} with the SEMPELL-model valves.

For the case initiated at 100% NP, the full discharge capacity of the 3 PSVs is needed to depressurise the RCP [RCS]. The maximum pressure occurs just after the opening of the second and the third PSV. With the SEMPELL-model valves the maximum pressure will be reached slightly later, after the opening of the third PSV, and will be less than 3 bar higher.

Based on the margin demonstrated for the EPR_{4250} with SEBIM-model valves, 106% DP compared to the criterion of 130% DP, and the power increase which results in a further pressure increase limited to 6%, it can be concluded that the acceptance criterion of 130% DP will be met for the EPR_{4500} with SEMPELL-model valves.

Minimum DNBR criteria fulfilment

The observed differences between the EPR 4250 MWth and the EPR 4500 MWth cases affecting the DNBR are as follows:

- Power increase level 6%
- Primary average temperature increase 1°C

Other neutronic and thermodynamic data are identical with those of the EPR 4250 MWth case, except for the initial boron concentration in the core, which is adjusted to the new power level. This value is increased to approximately 240 ppm for the 100% NP (assumption used in Sub-section 3.1.5).

It is noticed that the initial increase in the power level and in the primary temperature will have a negative impact on the DNBR value. A reduced margin to the DNBR criteria could be expected, however given the DNBR margin for EPR 4250 MWth (2.36 versus a criterion of 1.21) and the restriction in power increase to 6%, it is concluded that the 1.21 DNBR criterion for EPR 4500 MWth is met.

For the case initiated at 100% NP, the minimum DNBR occurs at the beginning of the transient during the RCP [RCS] pump coast down while the 3 PSVs are discharging. With the SEMPELL-model valves, the RCP [RCS] pressure at the time of minimum DNBR will be higher than the pressure calculated with the SEBIM-model valves. This increased pressure will have a slightly beneficial effect on the DNBR margin as the valves open later. Thus, it can be concluded that the DNBR acceptance criterion of 1.21 will be met for the EPR_{4500} with SEMPELL-model valves.

3.1.3.8. System sizing

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

SECTION 16.1.3.1.3 - TABLE 1

EPR 4250 MW - Initial State
ATWS LOOP
Initial States 60% NP and 100% NP

Parameter	**Initial value**
Initial State 60%	
RCP [RCS] flow rate	Thermal-hydraulic
Power	60% FP
Primary Pressure	155 bar
RCP [RCS] Average temperature	311.8°C
PZR level	55%
SG pressure	87.7 bar
ARE [MFWS] inlet temperature	203.5°C
SG level	49% NR
Initial State 100%	
RCP [RCS] flow rate	Thermal-hydraulic
Power	100% FP
Primary Pressure	155 bar
RCP [RCS] Average temperature	311.8°C
PZR level	56%
SG pressure	78 bar
ARE [MFWS] inlet temperature	230°C
SG level	49% NR

SECTION 16.1.3.1.3 - TABLE 2

EPR 4250 MW - Main Assumptions
ATWS LOOP
Initial States 60% NP and 100% NP

Parameter	Value
Reactor trip signal on RCP [RCS] pump speed < MIN1	
Setpoint	91% Nominal value
Delay	0.6 s
ATWS signal (reactor trip signal and high rods position (or high flux) after temporisation), actuates RBS [EBS] boration:	
Delay (after RT signal)	50 s
Capacity (per RCP [RCS] loop)	1.4 kg/s
RBS [EBS] boron concentration (natural boron)	11200 ppm
ASG [EFWS] actuation on SG level (wide range cold side) < MIN2	
Setpoint	7.85 m
Delay	1.5 + 50 = 51.5 s
Capacity	90 te/h per SG at 97 bar
MS relief train actuation on SG pressure > MAX1	
Setpoint	95.5 bar
Delay	0.9 + 1.5 + 0.5 = 2.9 s
VDA [MSRT] closing delay	40 s
Capacity	1150 te/h under 100 bar
MS safety valves	
Setpoint	105 bar
Accumulation	3%
Capacity	575 te/h under 100 bar, 2 per SG
PZR safety valves	
Setpoint (with uncertainty)	175.5 / 179.5 / 179.5 bar
dead time	0.5 s
opening time	1.5 s
Capacity	steam 300 te/h, liquid 450 te/h under 176 bar
Hysteresis (preliminary value)	10 bar

SECTION 16.1.3.1.3 - TABLE 3

Typical Sequence of Events
ATWS LOOP
Initial State 60% NP

EVENT	TIME (second)
LOOP:	
Turbine trip APA [MFWPS] cut off RCP [RCS] pumps trip	0
Reactor trip signal (RCP [RCS] pump speed < MIN1)	2.5
VDA [MSRT] opening	7
1st PSV opening/closing	12.5/28
Primary pressure peak	13.4
1st PSV opening / closing	35 / 44
ATWS signal / RBS [EBS] actuation	53
1st PSV opening / closing	57 / 65
1st PSV opening / closing	85 / 102
PZR filled	95
ASG [EFWS] injection	240
1st PSV opening / closing	272 / 300
RBS [EBS] boron arrival in the core	347
1st PSV opening / closing	376 / 617
SG minimum water mass	645

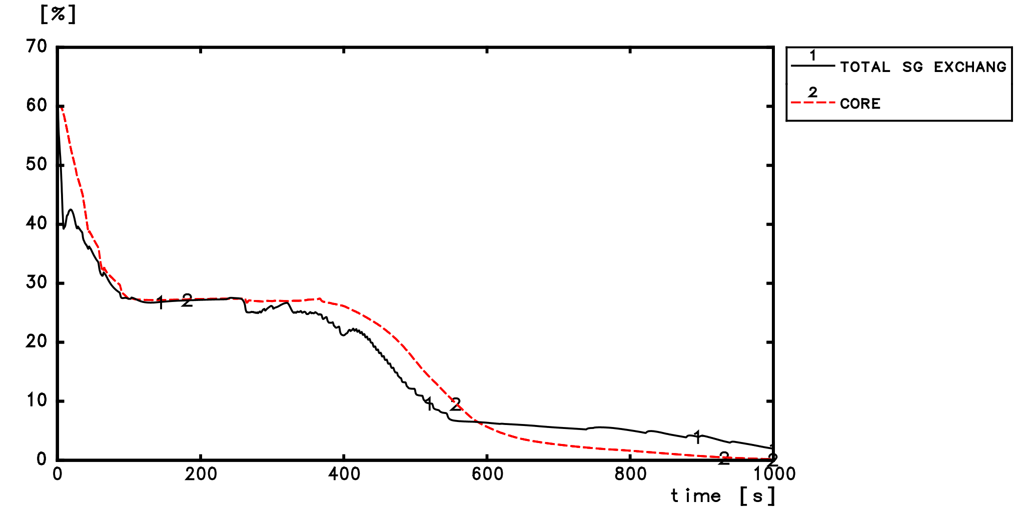
SECTION 16.1.3.1.3 - TABLE 4

Typical Sequence of Events
ATWS LOOP
Initial State 100% NP

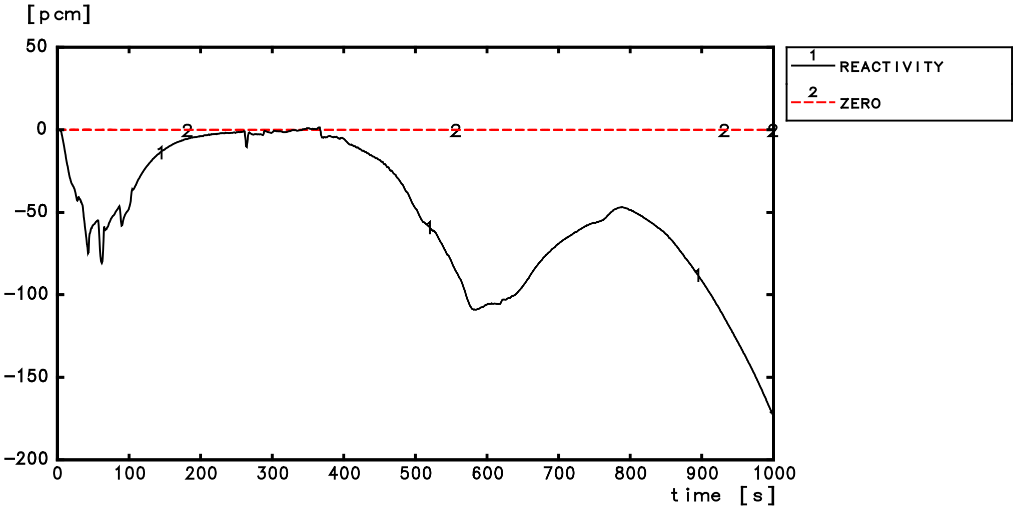
EVENT	TIME (second)
LOOP:	
Turbine trip ASG [MFWS] cut off RCP [RCS] pumps trip	0
Reactor trip signal (RCP [RCS] pump speed < MIN1)	2.5
1st PSV opening / closing	9 / 16.5
VDA [MSRT] opening	9.5
2nd and 3rd PSV opening/closing	10 / 15.3
Primary pressure peak	10.5
1st PSV opening / closing	45 / 52
ATWS signal / RBS [EBS] actuation	53
1st PSV opening / closing	86 / 99
PZR filled	94
ASG [EFWS] injection	200
1st PSV opening / closing	239 / 257
1st PSV opening / closing	313 / 339
RBS [EBS] boron arrival in the core	341
1st PSV opening / closing	368 / 411
2nd and 3rd PSV opening/closing	378 / 386
1st PSV opening / closing	436 / 525
2nd and 3rd PSV opening/closing	460 / 470
SG minimum water mass	546

SECTION 16.1.3.1.3 - FIGURE 1 (1/7)

Typical Power (Core and SG) – Reactivity
ATWS LOOP
Initial State 60% NP



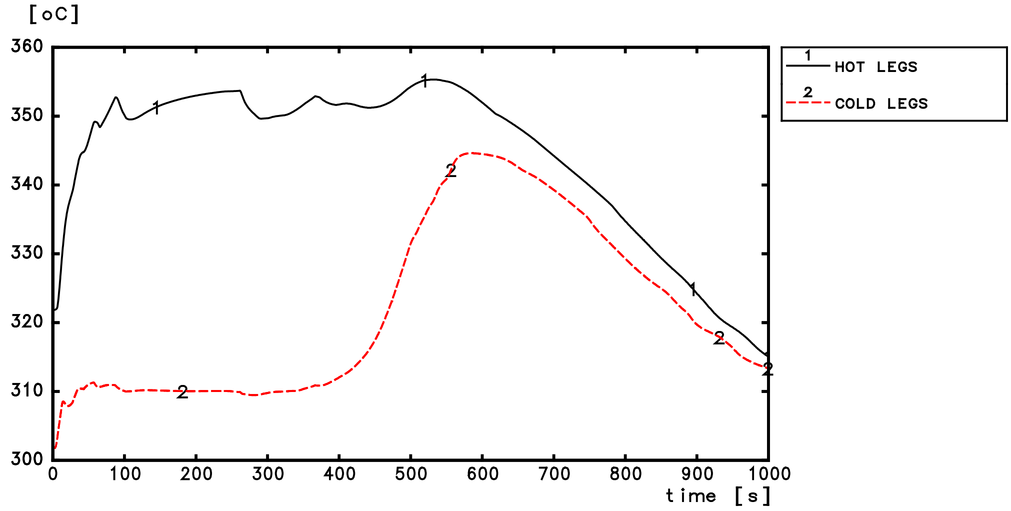
POWER (CORE AND SG)



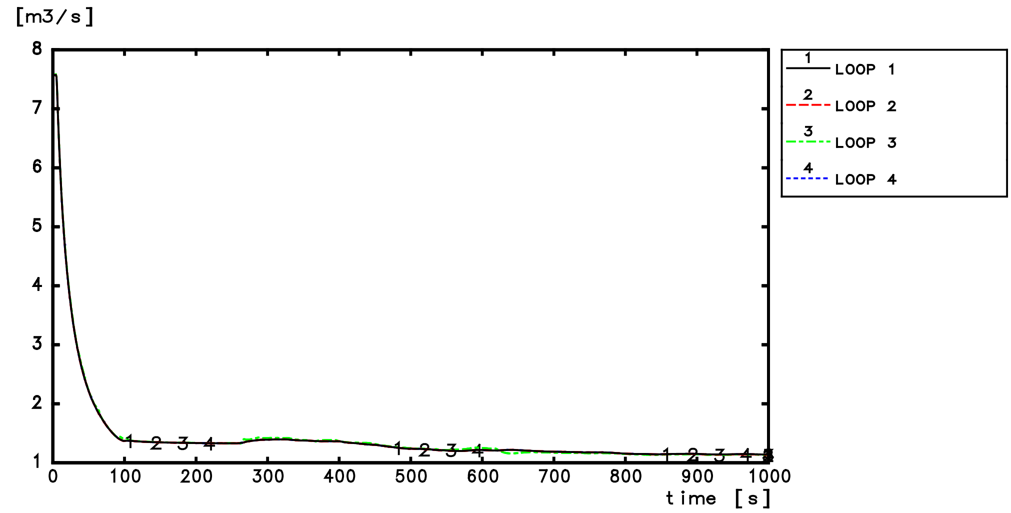
REACTIVITY

SECTION 16.1.3.1.3 - FIGURE 1 (2/7)

Typical RCP [RCS] Temperatures – RCP [RCS] Flow Rate
ATWS LOOP
Initial State 60% NP



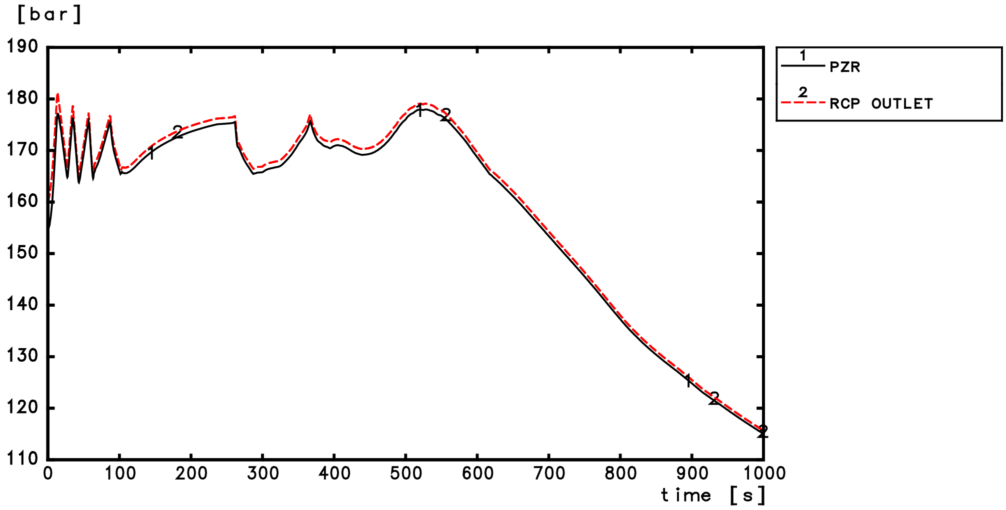
RCS TEMPERATURES



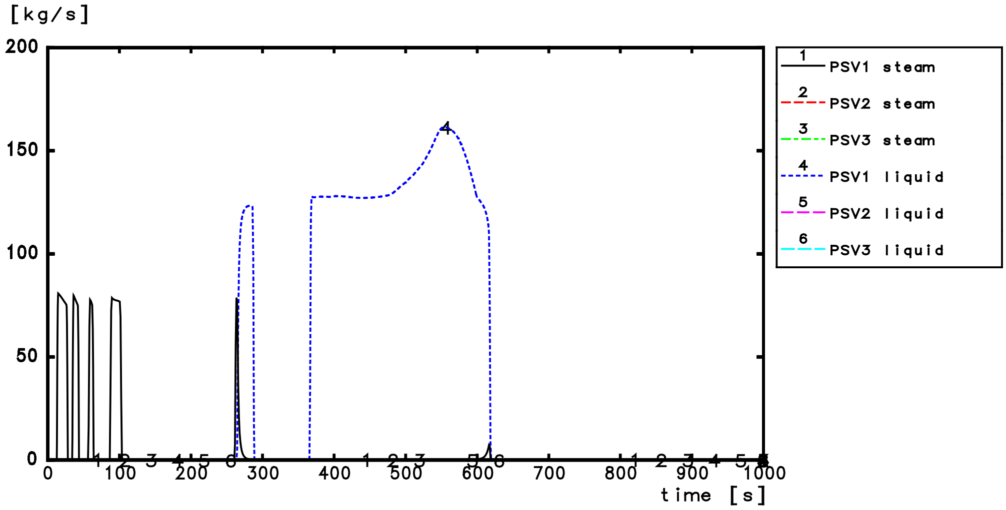
RCS FLOWRATE

SECTION 16.1.3.1.3 - FIGURE 1 (3/7)

Typical Primary Pressure – PSV Flow Rate
ATWS LOOP
Initial State 60% NP



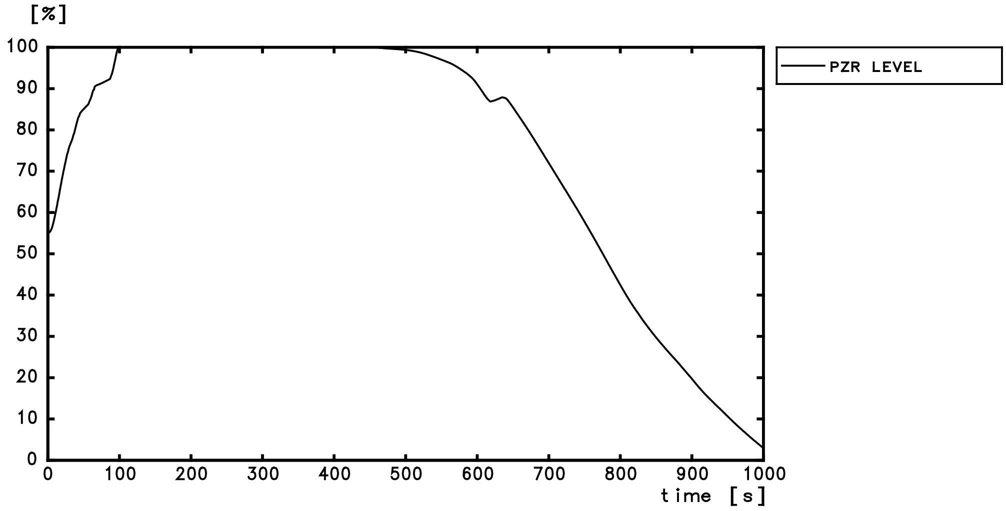
PRIMARY PRESSURE



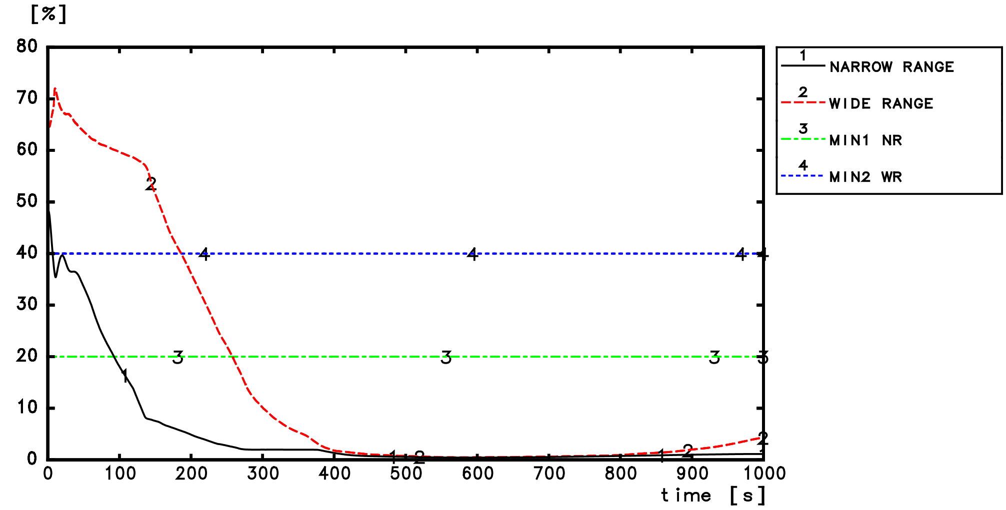
PSV FLOWRATE

SECTION 16.1.3.1.3 - FIGURE 1 (4/7)

Typical PZR Level – SG Levels
ATWS LOOP
Initial State 60% NP



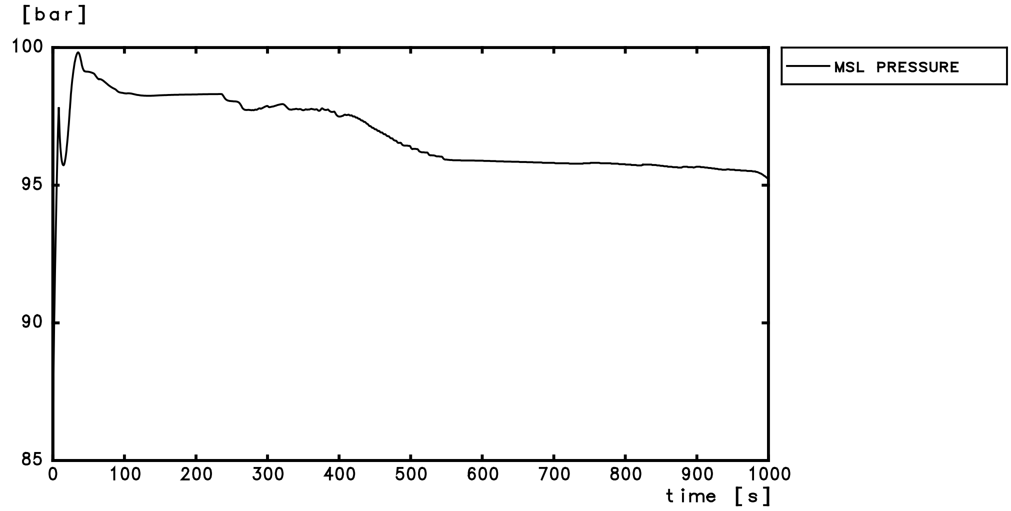
PZR LEVEL



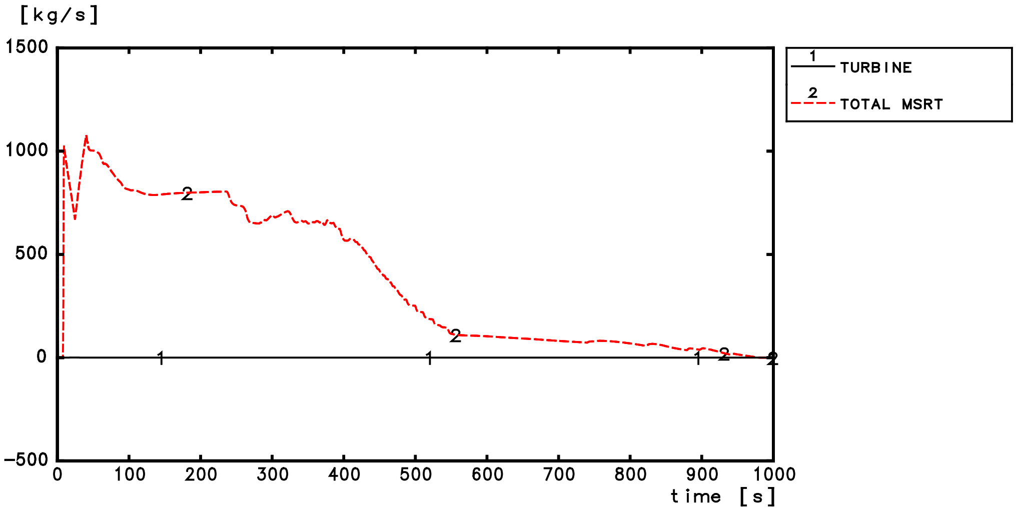
SG LEVELS

SECTION 16.1.3.1.3 - FIGURE 1 (5/7)

Typical Main Steam Pressure – Steam Flow Rate
ATWS LOOP
Initial State 60% NP



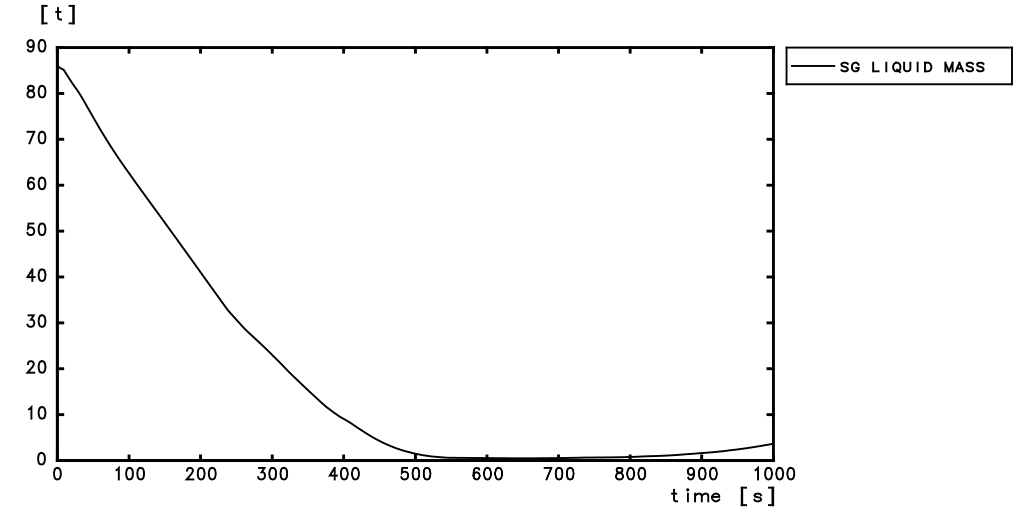
MAIN STEAM PRESSURE



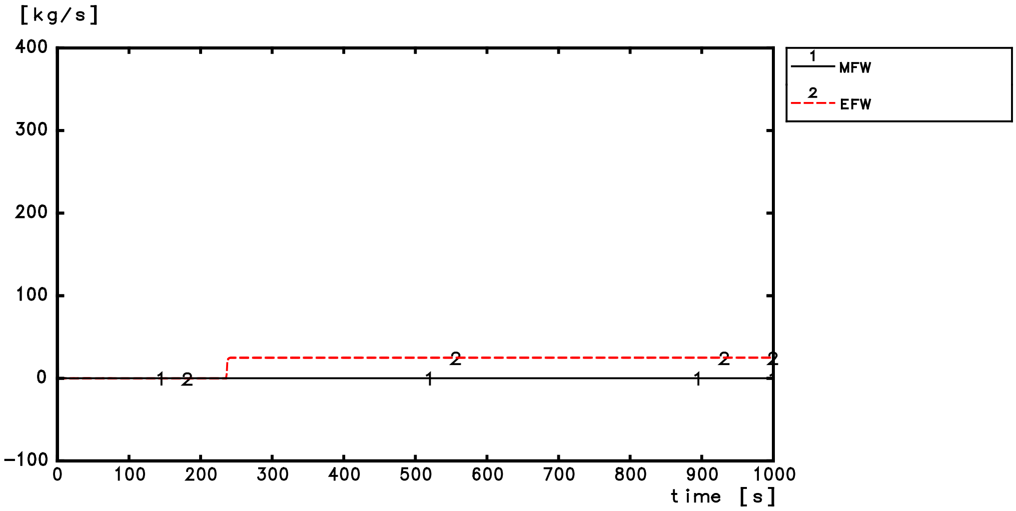
STEAM FLOWRATE

SECTION 16.1.3.1.3 - FIGURE 1 (6/7)

Typical SG Liquid Mass – Feedwater Flow Rate per SG
ATWS LOOP
Initial State 60% NP



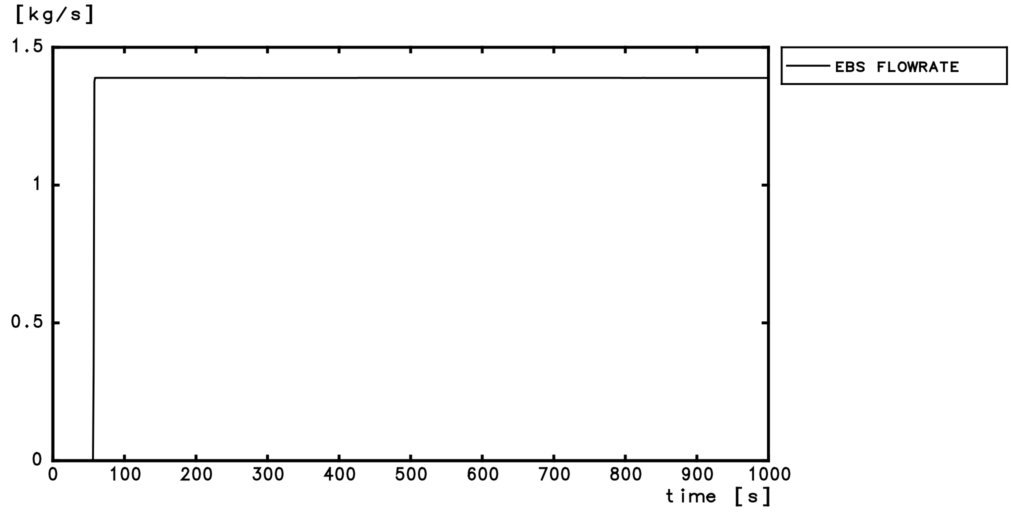
SG LIQUID MASS



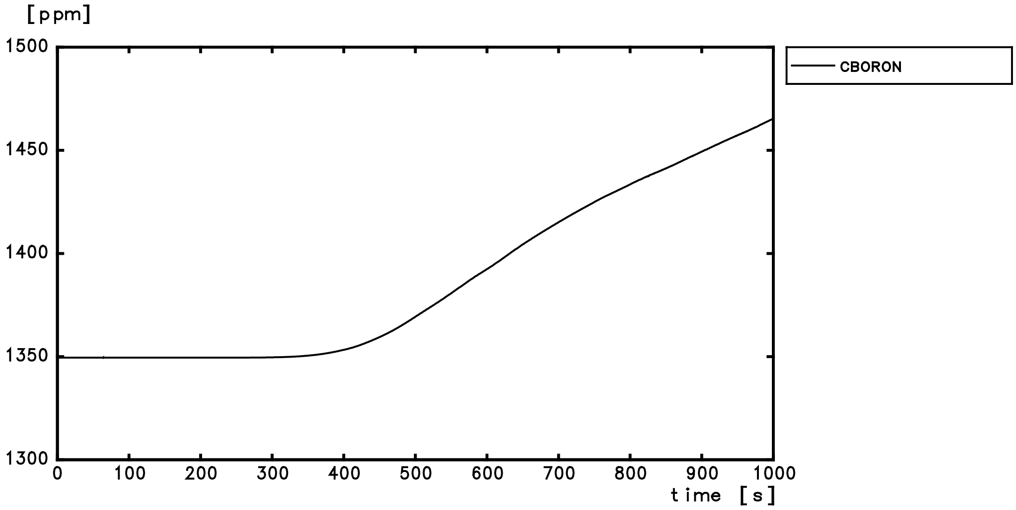
FEEDWATER FLOWRATE PER SG

SECTION 16.1.3.1.3 - FIGURE 1 (7/7)

Typical RBS [EBS] Flow Rate per Loop
Core Boron Concentration (Natural Boron)
ATWS LOOP
Initial State 60% NP



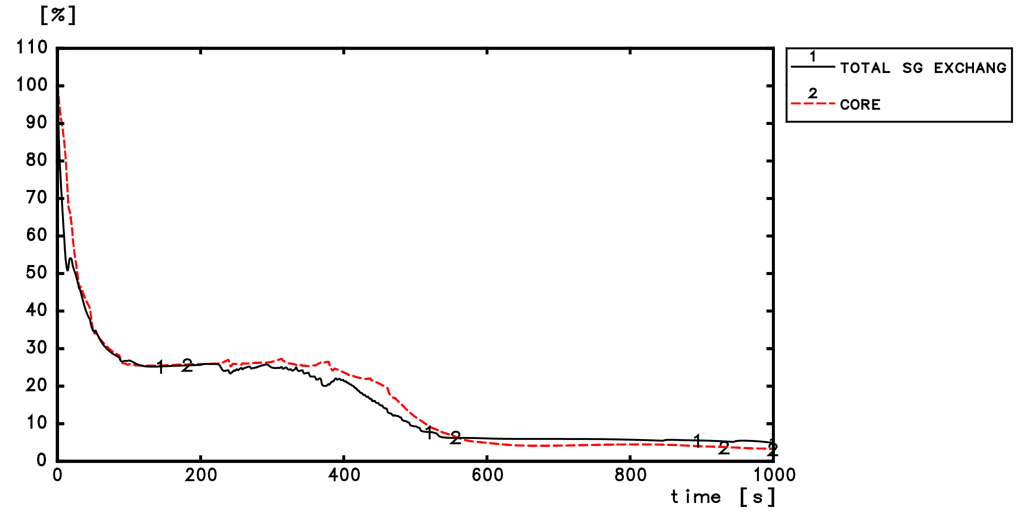
EBS FLOWRATE PER LOOP



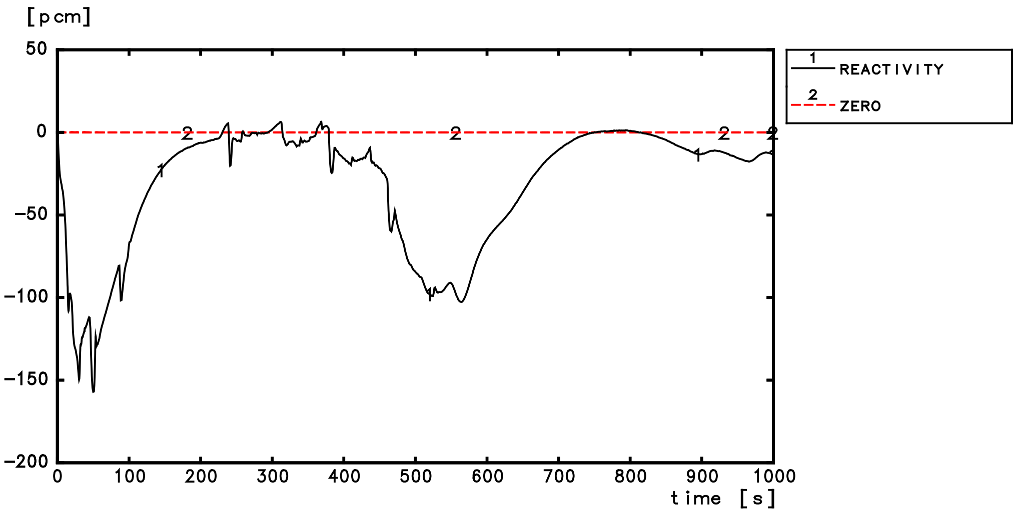
CORE BORON CONCENTRATION (NATURAL BORON)

SECTION 16.1.3.1.3 - FIGURE 2 (1/7)

Typical Power (Core and SG) – Reactivity
ATWS LOOP
Initial State 100% NP



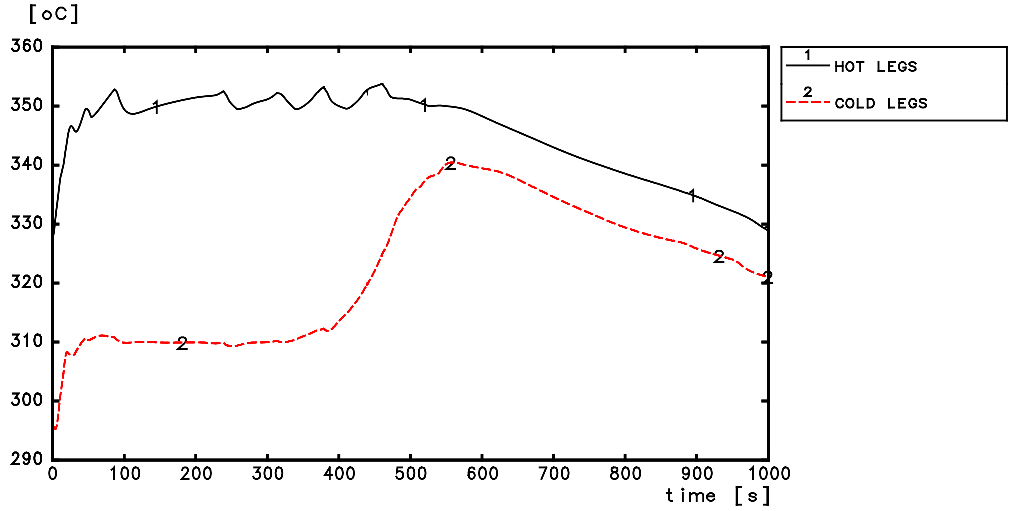
POWER (CORE AND SG)



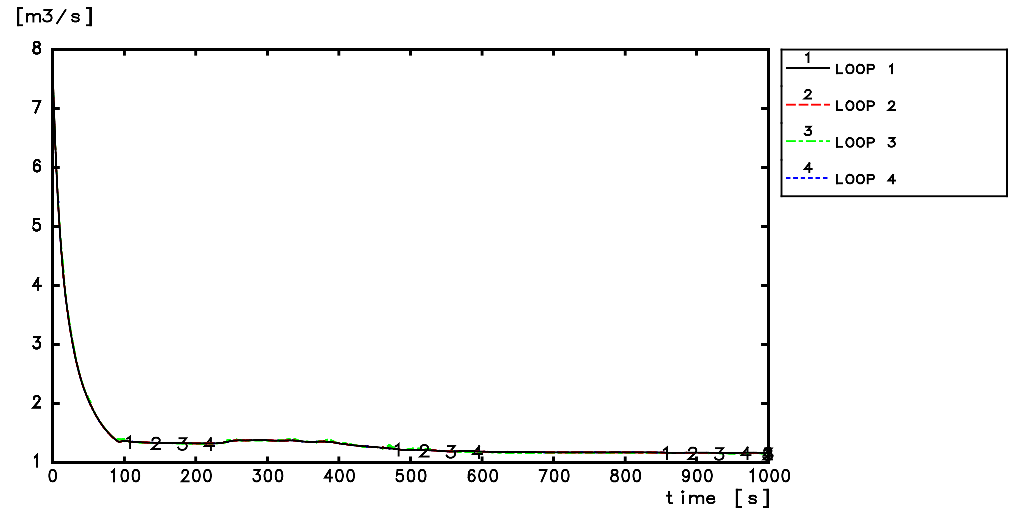
REACTIVITY

SECTION 16.1.3.1.3 - FIGURE 2 (2/7)

Typical RCP [RCS] Temperatures – RCP [RCS] Flow Rate
ATWS LOOP
Initial State 100% NP



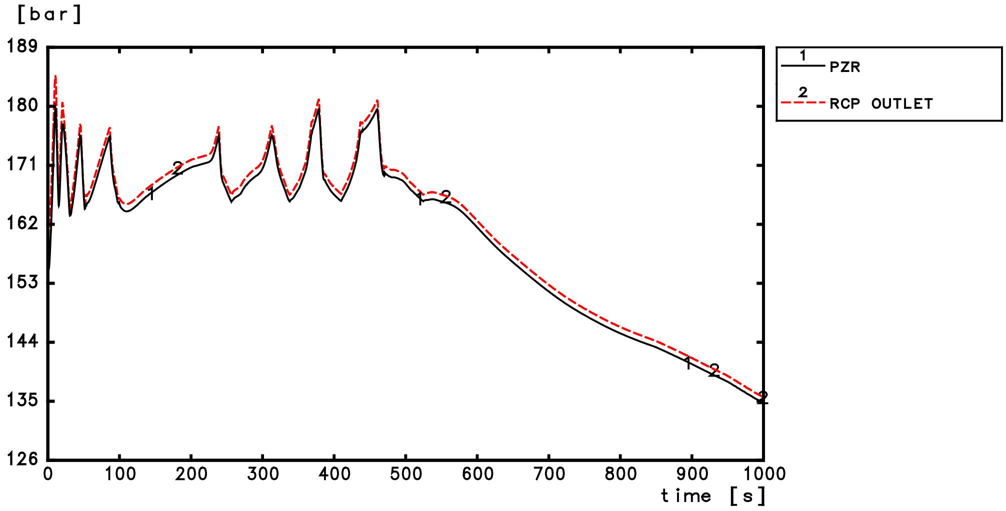
RCS TEMPERATURES



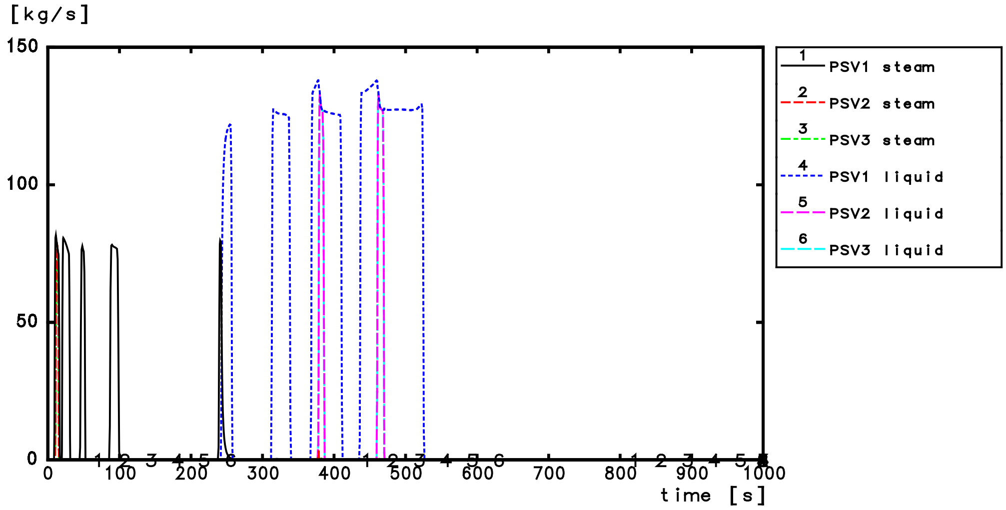
RCS FLOWRATE

SECTION 16.1.3.1.3 - FIGURE 2 (3/7)

Typical Primary Pressure – PSV Flow Rate
ATWS LOOP
Initial State 100% NP



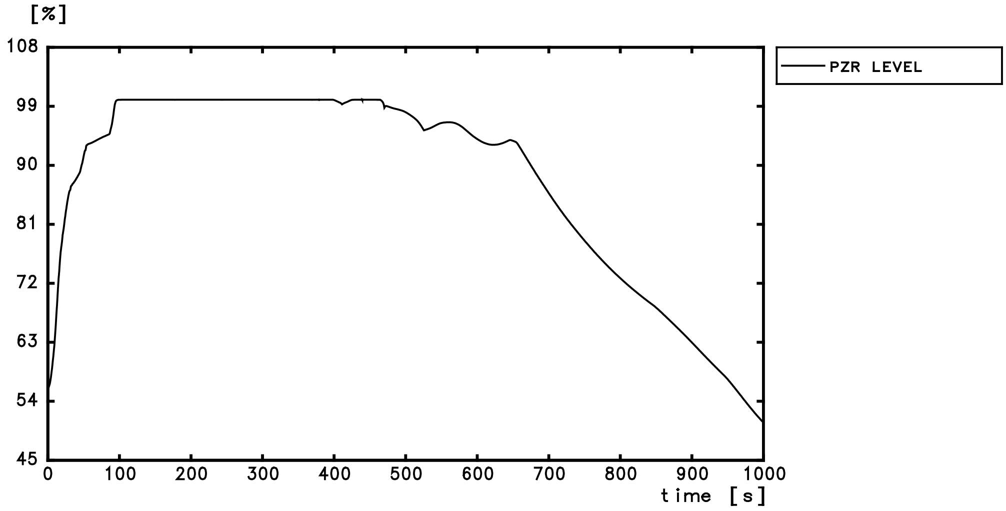
PRIMARY PRESSURE



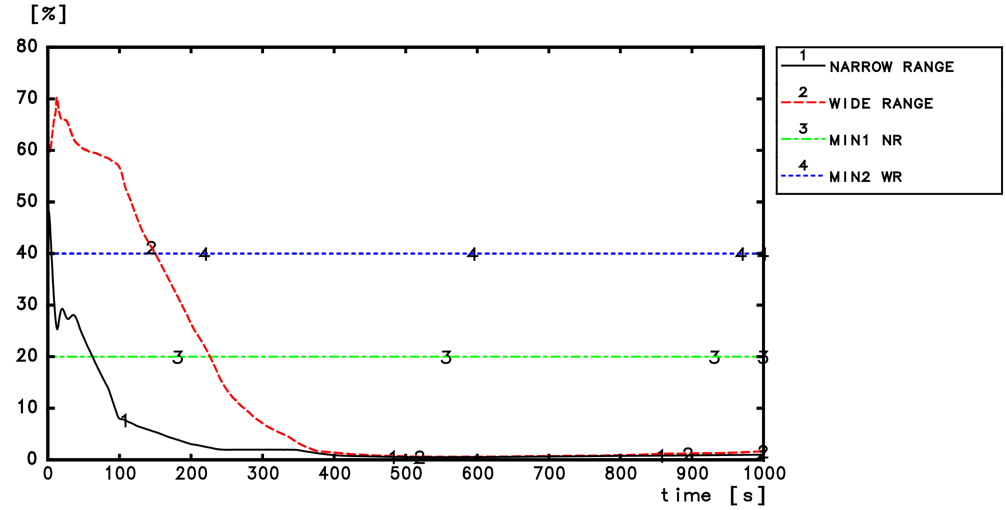
PSV FLOWRATE

SECTION 16.1.3.1.3 - FIGURE 2 (4/7)

Typical PZR Level – SG Levels PSV
ATWS LOOP
Initial State 100% NP



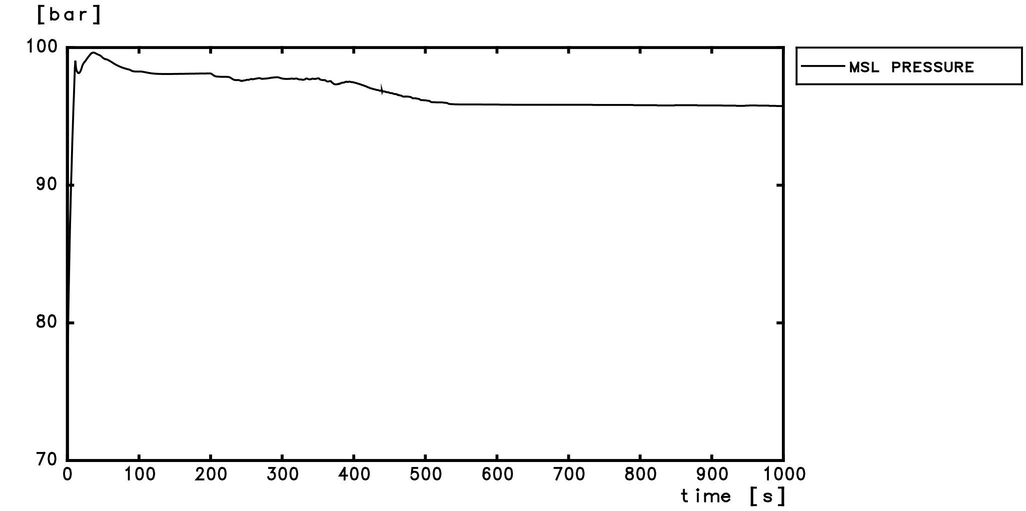
PZR LEVEL



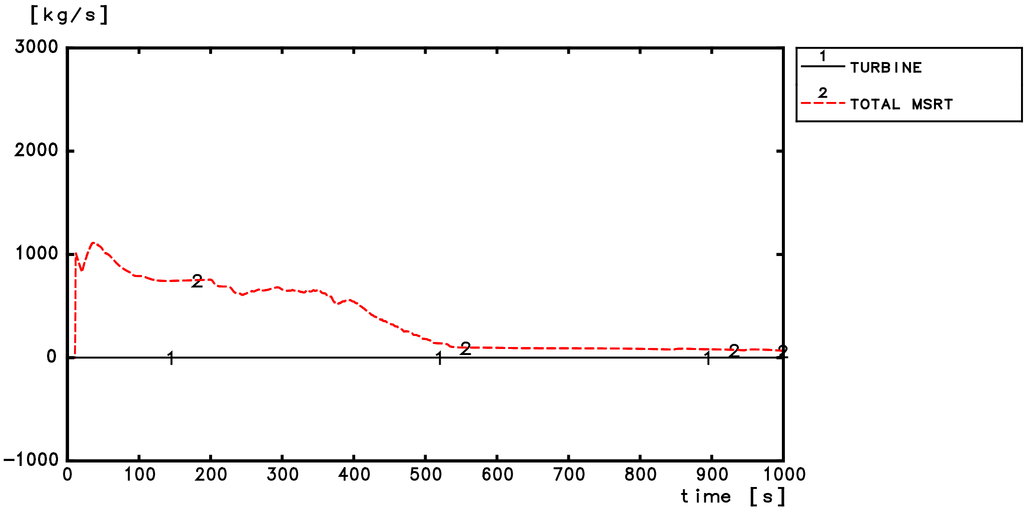
SG LEVELS

SECTION 16.1.3.1.3 - FIGURE 2 (5/7)

Typical Main Steam Pressure – Steam Flow Rate
ATWS LOOP
Initial State 100% NP



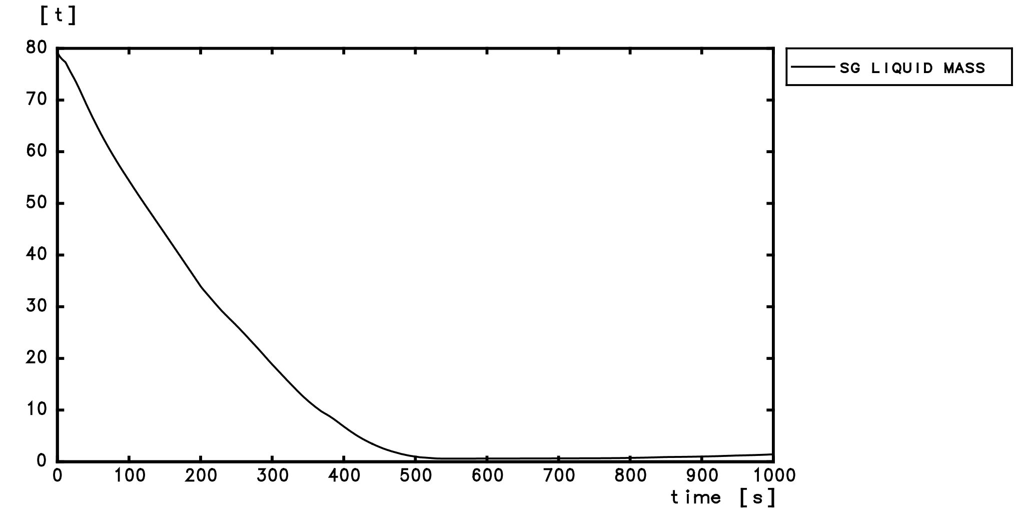
MAIN STEAM PRESSURE



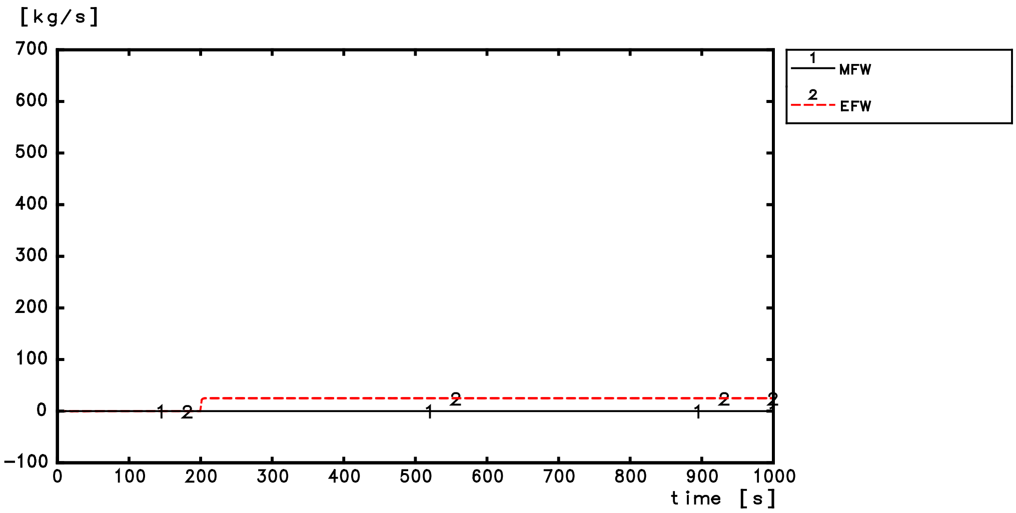
STEAM FLOWRATE

SECTION 16.1.3.1.3 - FIGURE 2 (6/7)

Typical SG Liquid Mass – Feedwater Flow Rate per SG
ATWS LOOP
Initial State 100% NP



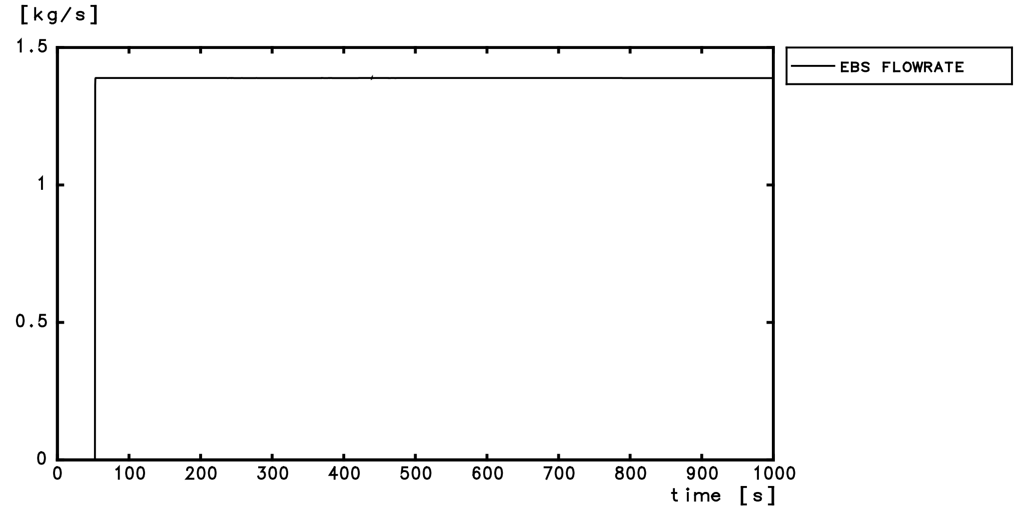
SG LIQUID MASS



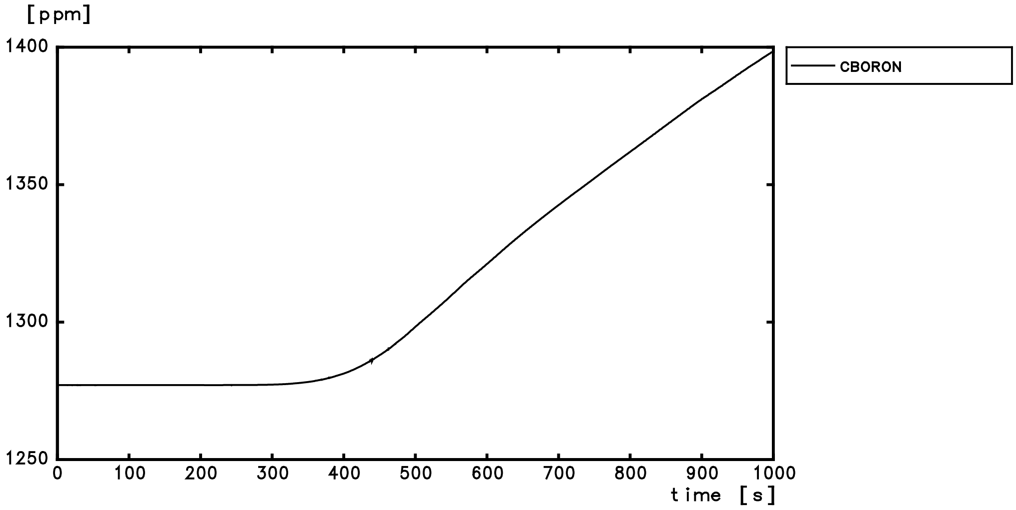
FEEDWATER FLOWRATE PER SG

SECTION 16.1.3.1.3 - FIGURE 2 (7/7)

Typical RBS [EBS] Flow Rate per Loop
Core Boron Concentration (Natural Boron)
ATWS LOOP
Initial State 100% NP



EBS FLOWRATE PER LOOP



CORE BORON CONCENTRATION (NATURAL BORON)

3.1.4. ATWS – RCV [CVCS] malfunction that leads to a decrease in the boron concentration of the primary coolant

3.1.4.1. Identification of the causes

A complete failure of the reactor trip system on demand from the Reactor Protection System (RPR [PS]) can result from:

- either a failure of the automatic reactor shutdown F1A signals, i.e. that none of the signals sent by the RPR [PS] de-energises the rod drive coils,
- or failure of the control and shutdown rods to insert into the core after de-energising of their drive coils. In this case, actuation of the rods due to control or limitation signals also fails.

This section, 3.1.4, deals with the second case, due to rods failure. The first case, due to a failure of the RPR [PS], is discussed in section 3.2.4.

3.1.4.2. Sequence of events

The decrease of the boron concentration causes an increase of core power, and consequently a primary circuit temperature and pressure increase. On the secondary side, the secondary pressure increase is potentially limited by the GCT [MSB] opening.

A reactor trip signal occurs on a high pressuriser pressure measurement. Reactor trip occurs, but control rod insertion does not occur, due to the assumed mechanical failure of the rods.

An ATWS signal is generated on a combination of a ‘reactor trip’ signal and a ‘high rod position (or high flux) after an appropriate delay’ signal. Following the ATWS signal, the RCV [CVCS] is isolated (downstream of the RCV [CVCS] volume control tank). This stops the dilution, and the RBS [EBS] starts automatically to provide boron injection. In addition, the ATWS signal causes the primary coolant pumps to trip when the low SG level setpoint is reached.

Following the reactor trip signal, the turbine trips and the ARE [MFWS] is switched from high to low flow.

The reduction in ARE [MFWS] supply leads to a decrease in secondary side heat removal. This, combined with the primary flow rate reduction due to the primary pump coast-down, causes an increase in the rate of increase in primary and secondary pressure and temperature.

The pressure control is provided by the pressuriser safety valves on the primary side and by the GCT [MSB], the VDA [MSRT] and two MSSV valves per SG on the secondary side.

The increase in temperature causes a reduction in reactivity and hence a power decrease via the moderator temperature feedback effect.

At the end of the transient, the dilution has been halted, reactivity is controlled by the RBS [EBS] and decay heat is removed by ASG [EFWS] injection. This corresponds to the "final state for RRC-A analyses" as defined in section 3.1.4.3.

3.1.4.3. Safety criteria

It must be demonstrated that the 'final state for RRC-A analyses' can be reached by assuring:

- long term core sub-criticality,
- decay heat removal,
- activity release control by maintaining the integrity of the barriers in accordance with the PCC-4 acceptance criteria.

For this demonstration, the following decoupling criteria are considered:

- the DNBR remains above the design limit 1.21 (see Sub-chapters 4.4 and 14.1),
- the primary system integrity is not impaired (the preliminary decoupling criterion is that the maximum pressure cannot exceed 130% of the calculated pressure (i.e. 228.5 bar abs). (Refer to Chapter 3.)

3.1.4.4. Protection and mitigation actions

The following I&C functions provide protection and mitigation in the case of an uncontrolled boron dilution accompanied by a reactor trip failure due to a mechanical failure of the rods:

- Reactor trip signal on "SG level (narrow range) <MIN1" (F1A),
- Reactor trip signal on "pressuriser pressure > MAX2" (F1A),
- ATWS signal on reactor trip signal and high rods position (or high flux) after an appropriate delay (F2),
- pressuriser spray on "pressuriser pressure > MAX1" (NC),
- opening of the GCT [MSB] on "header steam pressure > MAX1" (NC),
- opening of the VDA [MSRT] on "SG pressure > MAX1" (F1A),
- start-up of the ASG [EFWS] on "SG level (wide range) < MIN2" (F1A).

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

3.1.4.5. Methods of analysis

For the DNBR, the analysis is performed by comparison with the transient analysis of the uncontrolled RCCA bank withdrawal (see section 3.1.5).

The core sub-criticality at the final state is calculated with the SMART code (see Appendix 14A).

The change in boron concentration during the boration by the RBS [EBS] is calculated with the following formula:

$$BC\left(t \succ t_{EBS\ start}\right) = \left(BC\left(t_{EBS\ start}\right) - BC_{EBS}\right)\exp\left(-\frac{G}{M}\left(t - t_{EBS\ start}\right)\right) + BC_{EBS}$$

Where M is the RCP [RCS] water mass, G is the RBS [EBS] flow rate, BC_{EBS} is the RBS [EBS] boron concentration and $T_{EBS\ start}$ is the RBS [EBS] start time.

3.1.4.6. Specific assumptions

The change in boron concentration (BC) during the boration by the RBS [EBS] is calculated from the following values [Ref-1] [Ref-2]:

- The initial BC is the minimum value over the UO_2 fuel management cycles, 2239 ppm.
- The best estimate RCP [RCS] water mass is taken into account, 312 t.
- The RBS [EBS] flow rate is 2.8 kg/s. The RBS [EBS] BC is 11200 ppm. In normal conditions the RBS [EBS] tank volume is 27 m^3 and its mass is 27 te (see Sub-chapter 14.1).

3.1.4.7. Results and conclusion

3.1.4.7.1. Results for EPR 4250 MWth [Ref-1]

The dilution is equivalent to an uncontrolled slow withdrawal of RCCA banks at power, (reactivity insertion rate of less than 2 pcm/second).

The consequences of the transient are therefore bounded by those calculated in the study of the uncontrolled RCCA bank withdrawal combined with ATWS by mechanical rod failure (refer to section 3.1.5).

The moderator temperature feedback effect increases throughout the transient due to the decrease of the boron concentration. This is not the case for uncontrolled RCCA bank withdrawal. In particular when the turbine and the primary pumps are tripped, the higher moderator temperature coefficient causes a faster power decrease for the dilution, and hence a lower pressure peak and a lower temperature increase than in the case of an RCCA bank withdrawal.

The boron concentration at the RBS [EBS] initiation, 20s after the ATWS signal, is conservatively calculated at 2200 ppm. After the boration by the RBS [EBS] the final boron concentration is 2946 ppm, leading to core sub-criticality in the hot shutdown conditions of:

- 3356 pcm with xenon equilibrium
- 550 pcm without xenon (X=0)

3.1.4.7.2. Results for EPR 4500 MWth

This transient is not analysed for EPR 4500 MWth. The decoupling criteria are inferred from the results for EPR 4250 MWth.

As for EPR 4250 MW the dilution is equivalent to an uncontrolled slow withdrawal of RCCA banks at power (reactivity insertion rate of less than 2 pcm/second).

The consequences of the transient are covered by the study of the uncontrolled RCCA bank withdrawal at power combined with ATWS by mechanical rod failure. This transient is performed for EPR 4500 MW (see section 3.1.5).

3.1.4.8. System sizing

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

3.1.5. ATWS – Uncontrolled RCCA bank withdrawal

The uncontrolled rod withdrawal, a PCC-2 accident, when combined with the complete failure of the Automatic Reactor Shutdown system, is an RRC-A sequence.

3.1.5.1. Identification of the causes

A complete failure of the reactor trip system on demand from the Reactor Protection System (RPR [PS]) can result from:

- either a failure of the automatic reactor shutdown F1A signals (i.e. that none of the signals sent by the RPR [PS] de-energises the rod drive coils),
- or failure of the control and shutdown rods to insert into the core after de-energising of their drive coils. In this case, actuation of the rods due to control or limitation signals also fails.

This section 3.1.5 deals with the second case, due to rods failure. The first case, due to a failure of the RPR [PS], is discussed in section 3.2.5.

3.1.5.2. Sequence of events

The uncontrolled RCCA bank withdrawal causes a rapid increase in core power, and a primary circuit temperature and pressure increase. On the secondary side, the secondary pressure increase is potentially limited by the GCT [MSB] opening.

A reactor trip signal is generated on a high pressuriser pressure measurement. Reactor shutdown does not occur, nor does any other power reduction because of mechanical failure of the rods, thus further RCCA withdrawal is stopped.

An ATWS signal is generated on an ‘RT signal’ combined with a ‘high rod position (or high flux) after an appropriate delay’ signal. The ATWS signal automatically initiates RBS [EBS] injection with 7000 ppm enriched boron (corresponding to 11200 ppm natural boron) [Ref-1], thus automatically providing core sub-criticality in the long term. The RCV [CVCS] is also available for this boration function but is not claimed in the study.

Following the reactor trip signal, the turbine trip is successful and the ARE [MFWS] is switched from high to low flow.

The ARE [MFWS] supply reduction leads to a decrease in secondary side heat removal, and further reduces the capacity of the primary coolant to remove heat from the core.

Consequently, primary and secondary pressures and temperatures increase more rapidly. However, core power is reduced due to the moderator temperature feedback effect that decreases the overall reactivity.

The primary side pressure is controlled by:

- the pressuriser normal spray,
- three PSVs.

The secondary side pressure is controlled by:

- the GCT [MSB],
- the VDA [MSRT] and two MSSV valves per SG

When the wide range SG level setpoint MIN2 is reached, the ASG [EFWS] is started, the primary coolant pumps are tripped (by the ATWS signal) and the subsequent core heating causes a reactivity decrease from the moderator feedback effect.

The heat is removed through the ASG [EFWS] and the VDA [MSRT]. The final state, as defined in sub-section 3.1.1.3, is reached.

3.1.5.3. Safety criteria

It must be demonstrated that the "final state for RRC-A analyses" can be reached by assuring:

- long term core sub-criticality,
- decay heat removal,
- activity release control by maintaining the integrity of the barriers in accordance with the PCC-4 acceptance criteria.

For this demonstration, the following decoupling criteria are considered:

- the DNBR remains above the design limit 1.21 (see Sub-chapters 4.4 and 14.1),
- the primary system integrity is not impaired (the preliminary decoupling criterion is that the maximum pressure cannot exceed 130% of the calculated pressure (i.e. 228.5 bar abs). (Refer to Sub-chapter 3.4)

3.1.5.4. Protection and mitigation actions

In the event of an uncontrolled RCCA bank withdrawal coupled with a failure of the Automatic reactor trip system due to rods failure, the following functions will mitigate the accident consequences:

- reactor trip signal on “SG pressure >MAX1” (F1A),
- reactor trip signal on “SG level (narrow range) < MIN1” (F1A),
- reactor trip signal on “pressuriser pressure > MAX2” (F1A),
- ATWS signal on reactor trip signal and high rod position (or high flux) after an appropriate delay (F2),
- starting of pressuriser normal spray on "pressuriser pressure > MAX1” (NC),
- opening of the GCT [MSB] on “header steam pressure > MAX1” (NC),
- opening of the VDA [MSRT] on “SG pressure > MAX1” (F1A),

- start-up of the ASG [EFWS] on “SG level (wide range) < MIN2” (F1A).
- tripping the RCS [RCP] pumps on ATWS signal and "SG level (wide range) < MIN2" (F2).

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

3.1.5.5. Method of Analysis

The analysis is carried out using the following:

- MANTA computer code (see Appendix 14A) for the analyses of the thermal-hydraulic behaviour of the primary and secondary systems (RCP [RCS] and SG), accounting for F1 systems operations.
- SMART/ FLICA computer code (see Appendix 14A) for the neutronic and thermal-hydraulic behaviour of the core.

The DNBR calculation is performed with the FLICA code (see Appendix 14A).

The analysis is performed starting from an initial state of 100% nominal power with a moderator coefficient of -10 pcm/°C. This assumption covers more than 95% of plant life.

3.1.5.6. Specific assumptions

3.1.5.6.1. Single failure and maintenance

Single failure or maintenance do not have to be accounted for in the RRC-A analyses.

3.1.5.6.2. Initial conditions

The initial conditions are the ”Best Estimate” conditions at 100% NP (see Section 16.1.3.1.5 – Table 1).

The thermal-hydraulic primary flow rate is assumed, being conservative even though it is not strictly required by the accident analysis rules for RRC-A sequences.

3.1.5.6.3. Assumptions related to the rods

The initial positions assumed for the RCCA bank are conservative but realistic (see Section 16.1.3.1.5 - Figure 1, definition of control rods):

- P1 is inserted 150 steps
- The other control rods (P2 to P5) are inserted 50 steps.

A conservatively high withdrawal speed for all RCCAs is assumed for this assessment (75 cm/min).

3.1.5.6.4. Neutronic data

The neutronic data are calculated at BOL.

The moderator temperature coefficient is -10 pcm/°C [Ref-1] at the beginning of the transient. The Doppler temperature coefficient is at its minimum absolute value to make the rate of core power increase conservative. This coefficient is fixed at a value of -2.5 pcm/°C [Ref-1] at the beginning of the transient (minimum absolute value without uncertainties, and covers all fuel management schemes).

The reference fuel management is used to maximise the $F\Delta H$, when the moderator and Doppler temperature coefficients are set at their minimum values, giving a conservative DNBR calculation.

3.1.5.6.5. Assumptions related to controls

The SG level control and the primary temperature control cannot be actuated.

Pressuriser pressure control by the pressuriser spray is credited to delay the automatic reactor trip signal due to high pressuriser pressure.

Pressuriser level control is not modelled in the present analysis.

3.1.5.6.6. Assumptions related to the systems

The implemented F1A systems are: ASG [EFWS], PSV, VDA [MSRT], MSSV valves.

The implemented F2[5] function is the boration through the RBS [EBS].

Input, limits and delays are summarised in Section 16.1.3.1.5 - Table 2.

3.1.5.6.7. DNBR calculation

The DNBR calculation takes into account the axial distribution in the hot channel and the ΔH provided by the 3D core calculation (SMART, Appendix 14A) at the time of minimum DNBR. No penalty on the local thermal power is applied for analyses within the RRC-A domain.

3.1.5.7. Results and conclusion [Ref-1]

Impact of the new design of pressuriser safety valves

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

The two PSV models have the same discharge flow rate capacity.

The main differences between the two valve models that can have an impact on the results of the analysis are:

- The opening setpoint of the first SEMPELL-model valve is 1 bar higher than the first SEBIM-model valve.

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[5] The RBS [EBS] is a F1A safety class system, automatically activated during the ATWS via an F2 safety class signal. The RBS [EBS] automatic boration function is therefore F2 safety class.

- The opening setpoint of the third SEMPELL-model valve is 3 bar higher than the third SEBIM-model valve.

- The PSV opening time for the SEMPELL-model valve is shorter, 0.1 seconds, than that for the SEBIM-model valve, 1.5 seconds.

The hysteresis of the SEMPELL-model valve, 16% of the opening setpoint, is higher than that for the SEBIM-model valve, 10 bar.

The sequence of events is shown in Section 16.1.3.1.5 - Table 3 and Section 16.1.3.1.5 – Figure 2 shows the behaviour of the main parameters.

The core power increases because of rod withdrawal. The minimum DNBR occurs when the core thermal power reaches its maximum (44 seconds). When the reactor trip signal on high pressuriser pressure is actuated, withdrawal of the rods ceases, but control rod insertion does not occur due to the assumed rod mechanical failure.

Simultaneously, the turbine is tripped which leads to an increase in secondary, and consequently primary temperature and pressure. As a result, the core power reduces due to the moderator temperature feedback that decreases reactivity.

The first PSV is demanded, and the peak primary pressure is reached (184.7 bar at 51 seconds at the RCP [RCS] pumps outlet). The GCT [MSB] and the VDA [MSRT] open on the secondary side.

The first PSV discharge capacity is sufficient to remove the excess energy stored in the RCP [RCS] and to cause it to depressurise. The maximum pressure occurs 6.5 seconds after reaching the first PSV opening threshold of 175.5 bar and is 3.4 bar higher at the pressuriser. The time delay consists of 0.5 seconds of delay time and 1.5 seconds for opening plus 4.5 seconds to reach the maximum pressure.

As the opening setpoint of the first SEMPELL-model valve is 1 bar higher than that for the SEBIM-model valve, it will be reached about 2 seconds later. However as the opening time of the SEMPELL-model valve is nearly instantaneous, 0.5 seconds of delay time and 0.1 seconds for valve opening, the PSV opening will be effective about 0.6 seconds later than for the SEBIM model valve case. The discharge flow capacity is the same for the two valve models and thus the maximum pressure will also occur within the same time interval after its opening. Consequently the maximum pressure will be less than 1 bar higher.

By comparing the margin calculated with the SEBIM-model valves, 105% of the design pressure compared to a criterion of 130% DP, it can be concluded that the acceptance criterion of 130% DP will be met with the SEMPELL-model valves installed.

As the feedwater flow has been switched to a low level, SG levels decrease and, when the wide range level MIN2 is reached, the ASG [EFWS] is started and the reactor coolant pumps are tripped. The resulting core heating causes a reactivity decrease due to the moderator feedback and the core power level rapidly decreases.

However, since the level in the SG drops rapidly, the SG heat transfer decrease is temporarily faster than core power decrease.

The corresponding primary circuit temperature increase associated with the core flow reduction results in a temporary decrease in the DNBR margin (however this margin remains higher than the minimum value reached early in the transient). During the same phase of the transient, the pressuriser fills with water.

When the core power is low enough, the main feedwater low flow rate, and the ASG [EFWS] flow are sufficient for core power removal. The plant then behaves in a similar manner to the case of “ATWS – Loss of feed water” (see section 3.1.2).

The minimum DNBR occurs at the beginning of the transient while the PSV are closed.

Therefore, the SEMPELL-model valves will have no impact on the minimum DNBR calculation. Thus it can be concluded that the DNBR acceptance criterion will be met with the SEMPELL-model valves installed.

3.1.5.7.1. Criteria fulfilment

The maximum pressure downstream of the RCP [RCS] pumps occurs early in the transient, following the turbine trip. The peak value is 184.7 bar. The decoupling criteria of 130% of the design pressure (228.5 bar abs) is therefore met with a large margin.

The heat is removed via the VDA [MSRT], via the main feedwater low flow rate and ASG [EFWS] in the SG, and via the PSV in the RCP [RCS]. For core heat removal, the consequences are covered by those of the ATWS – Loss of feedwater (see section 3.1.2) for which normal feedwater is not available.

The minimum DNBR is reached in 22 seconds, and it is equal to 2.44, a value higher than the acceptance criterion of 1.21.

3.1.5.8. System sizing

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

SECTION 16.1.3.1.5 - TABLE 1

ATWS - Uncontrolled RCCA Bank Withdrawal – Initial State

Parameter	Initial value
SG heat transfer area	Nominal
RCP [RCS] flow rate	Thermal hydraulic (27180 m^3/h)
Power	100% FP
Pressure	155 bar
Average temperature	312.8°C
PZR level	56%
SG pressure	78 bar
ARE [MFWS] inlet temperature	230°C
SG level	49%

SECTION 16.1.3.1.5 - TABLE 2

Main Assumptions
ATWS - Uncontrolled RCCA Bank Withdrawal

Parameter	Value
Reactor trip signal on PZR pressure > MAX1	
Setpoint	166.5 bar Nominal value
Delay	1.2 s
ATWS signal (on RT signal and high rods position (or high flux) after temporisation) actuates RBS [EBS] boration	
Delay	20 s
Capacity (per RCP [RCS] loop)	1.4 kg/s
RBS [EBS] boron concentration (natural boron)	11200 ppm
ASG [EFWS] actuation on SG level (wide range cold side) < MIN2	
Setpoint	7.85 m
Delay	1.5 + 15 = 16.5 s
Capacity	90 te/h per SG at 97 bar
MS relief train actuation on SG pressure > MAX1	
Setpoint	95.5 bar
Delay	0.9 + 1.5 + 0.5 = 2.9 s
VDA [MSRT] closing delay	40 s
Capacity	1150 te/h under 100 bar
MS safety valves	
Setpoint	105 bar
Accumulation	3%
Capacity	575 te/h under 100 bar, 2 per SG
PZR safety valves	
Setpoint	175.5 / 179.5 / 179.5 bar
dead time	0.5 s
opening time	1.5 s
Capacity	steam 300 te/h, liquid 450 te/h under 176 bar
Hysteresis	10 bar
Reactor Coolant Pump trip on ATWS signal + SG level (wide range cold side) < MIN2	
Setpoint	7.85 m (40%)
Delay	1.5 s

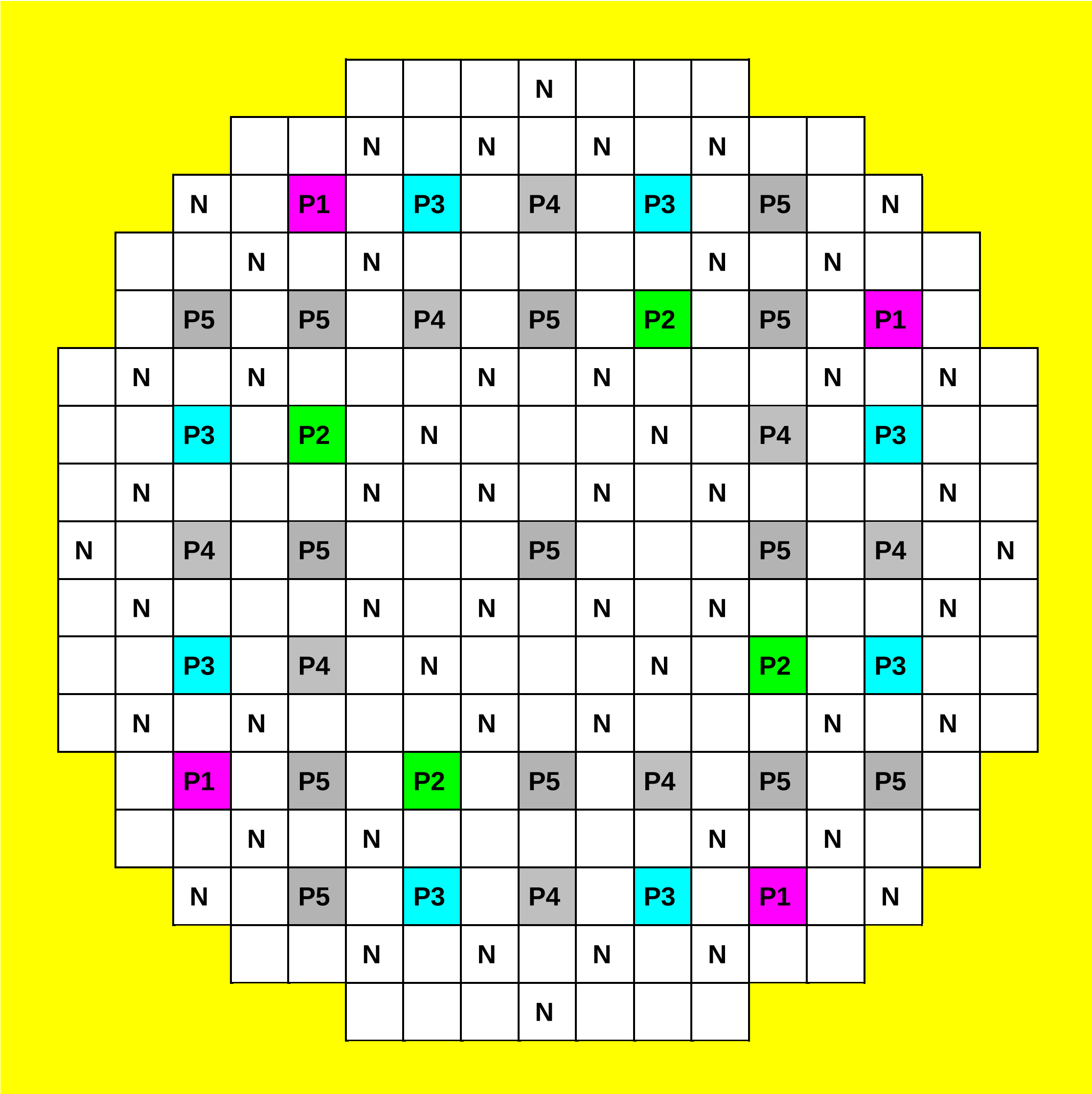
SECTION 16.1.3.1.5 - TABLE 3

Sequence of Events
ATWS - Uncontrolled RCCA Bank Withdrawal

EVENT	TIME (s)
Start of uncontrolled withdrawal of control bank at 75 step/min	0
Minimum DNBR (2.44)	22
Reactor trip signal on PZR pressure > MAX2	38.5
RCCA bank stop from withdrawing (mechanical blockage after reactor trip)	39.7
Turbine trip	
Maximum core thermal power level (113.9%)	44
1st PSV opening / closing (only first PZR valve)	45.5/64
Primary peak pressure (178.9 bar, pressuriser)	51
VDA [MSRT] opening	52.2
Feedwater flow rate switches to low level	55
RBS [EBS] start	60.3
2nd PSV opening / closing (only first PZR valve)	74.5 / 87
Trip of reactor coolant pumps	137
ASG [EFWS] actuation	150.5
VDA [MSRT] closing	182
3rd opening of first PZR valve	193.5
PZR filled	200
RBS [EBS] boron arrival in the core (initial concentration + 1 pcm)	350

SECTION 16.1.3.1.5 - FIGURE 1

ATWS - Uncontrolled RCCA Bank Withdrawal

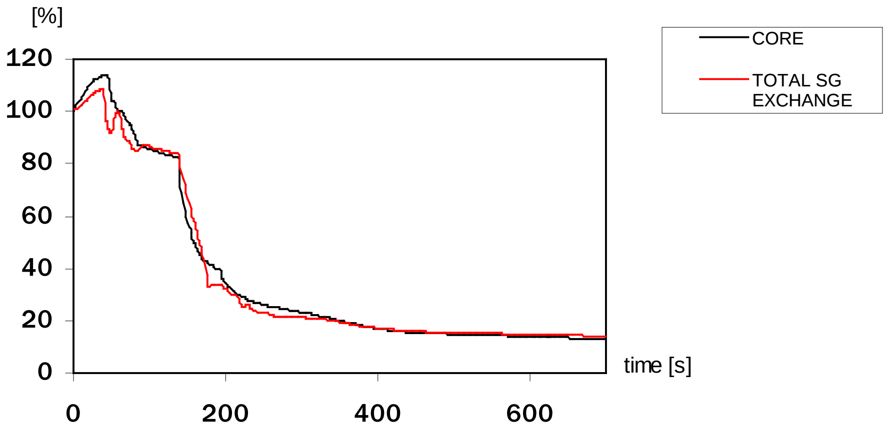


P1, P2, P3, P4, P5: Control rods

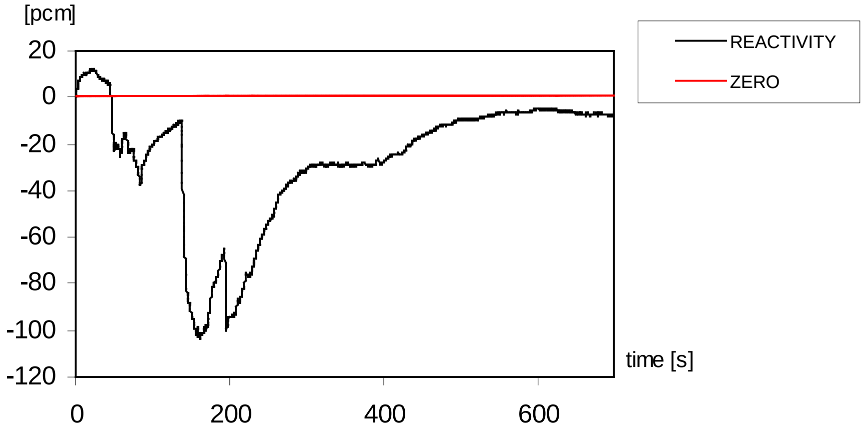
N: Shutdown rods

SECTION 16.1.3.1.5 - FIGURE 2 (1/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
Power and Reactivity



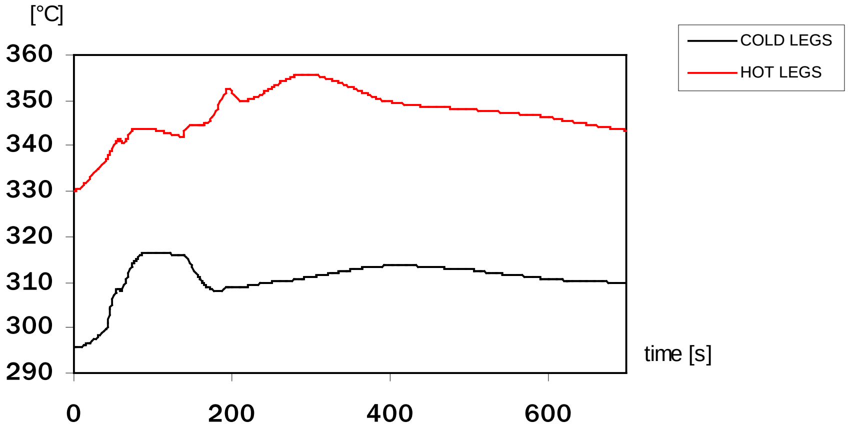
POWER (CORE AND SG)



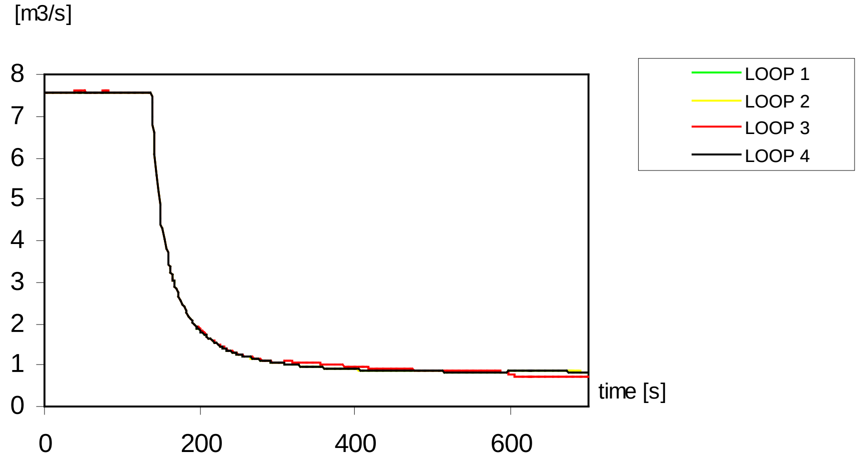
REACTIVITY

SECTION 16.1.3.1.5 – FIGURE 2 (2/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
RCP [RCS] Temperatures and Flow Rate



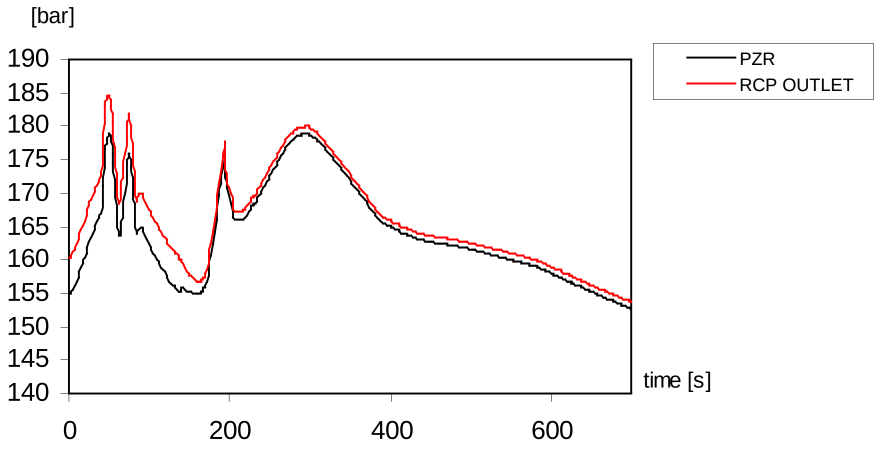
RCS TEMPERATURES



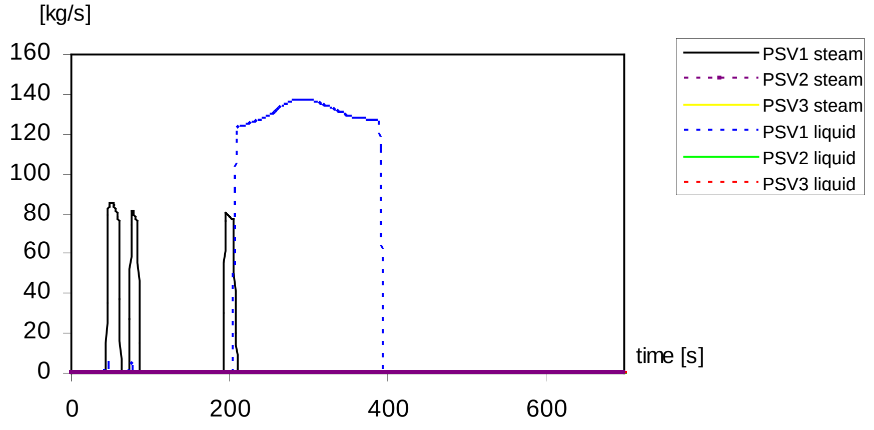
RCS FLOWRATE

SECTION 16.1.3.1.5 - FIGURE 2 (3/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
Primary Pressure and PSV Flow Rate



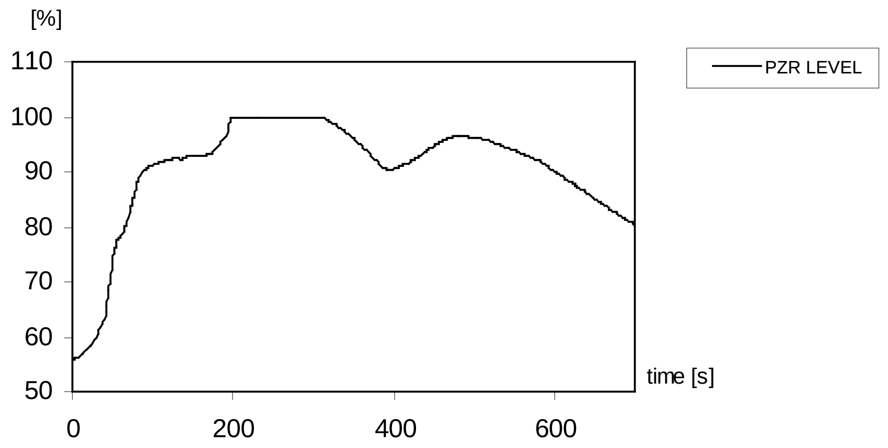
PRIMARY PRESSURE



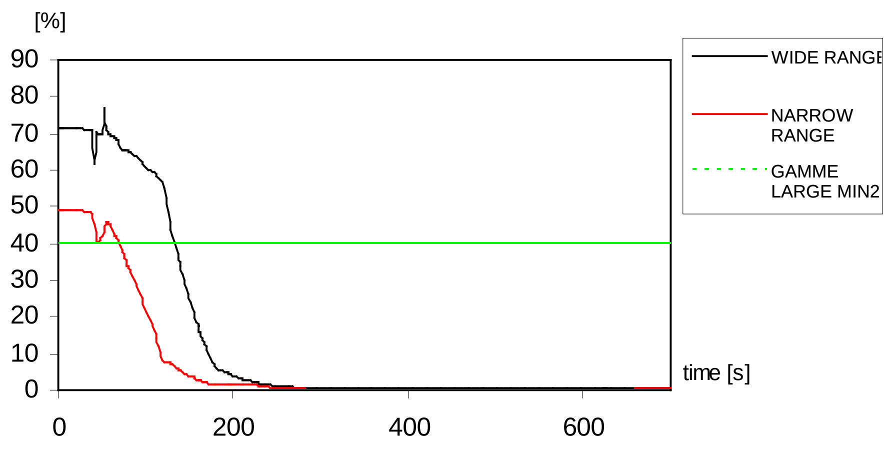
PSV FLOWRATE

SECTION 16.1.3.1.5 - FIGURE 2 (4/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
PZR Level and SG Level



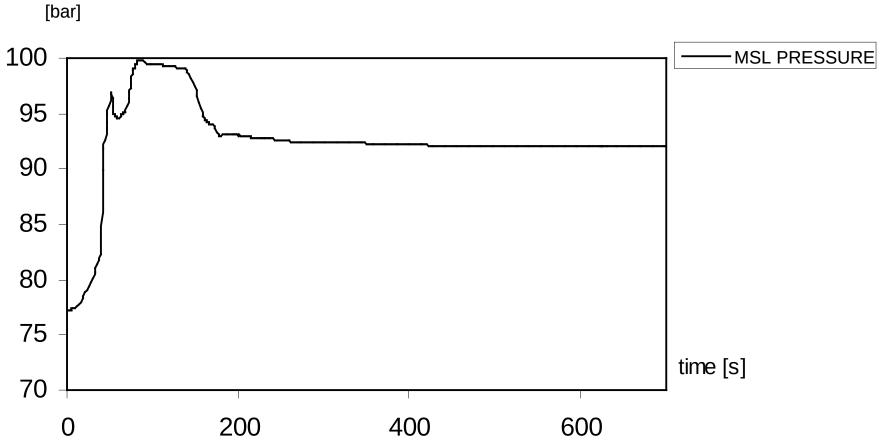
PZR LEVEL



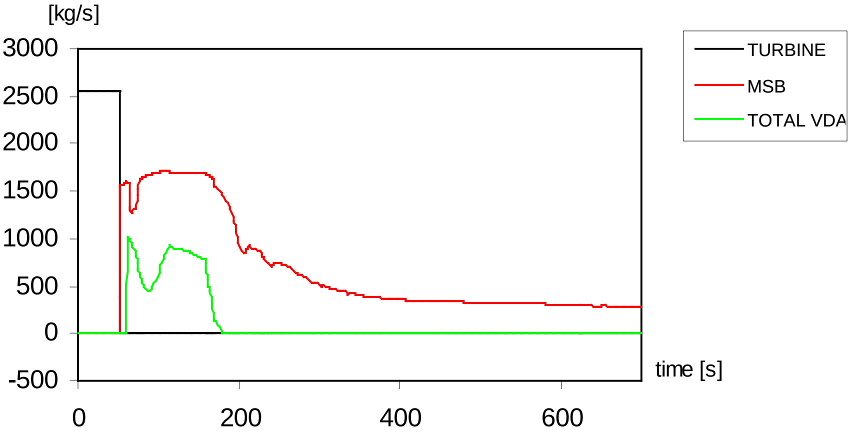
SG LEVELS

SECTION 16.1.3.1.5 - FIGURE 2 (5/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
Main Steam Pressure and Steam Flow Rate



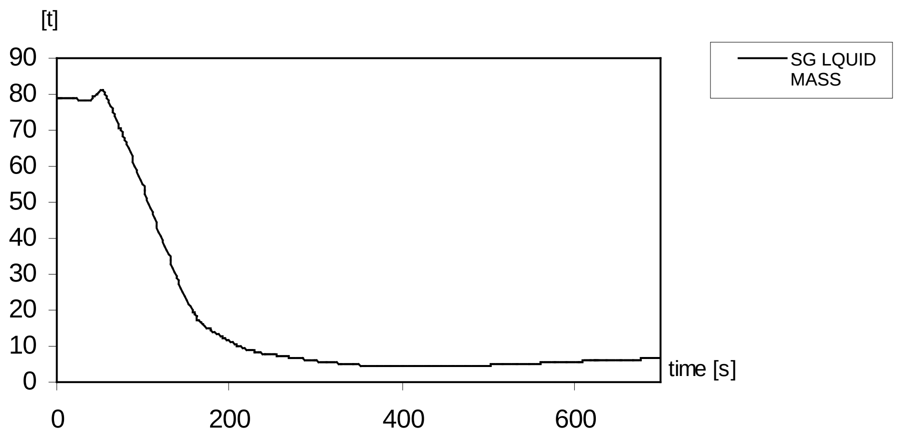
MAIN STEAM PRESSURE



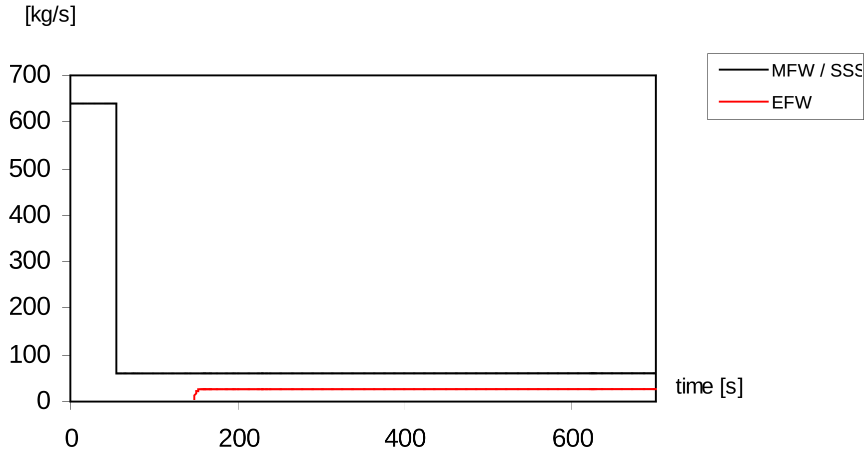
STEAM FLOWRATE

SECTION 16.1.3.1.5 - FIGURE 2 (6/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
SG Liquid Mass and Feedwater Flow Rate per SG



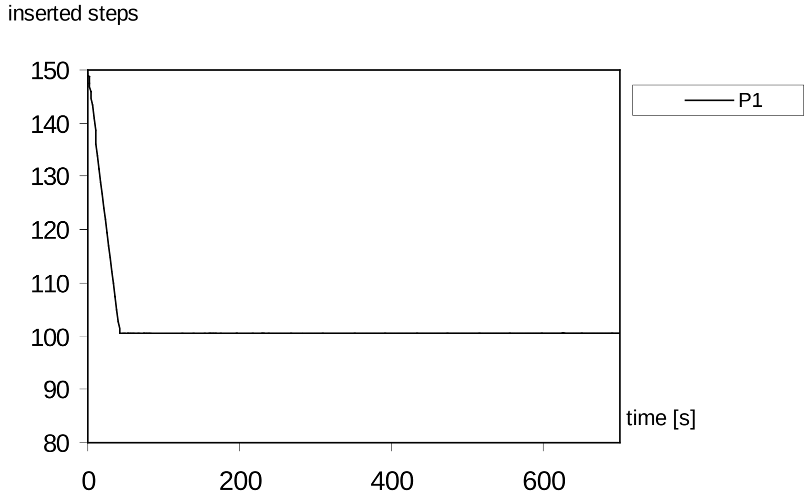
SG LIQUID MASS



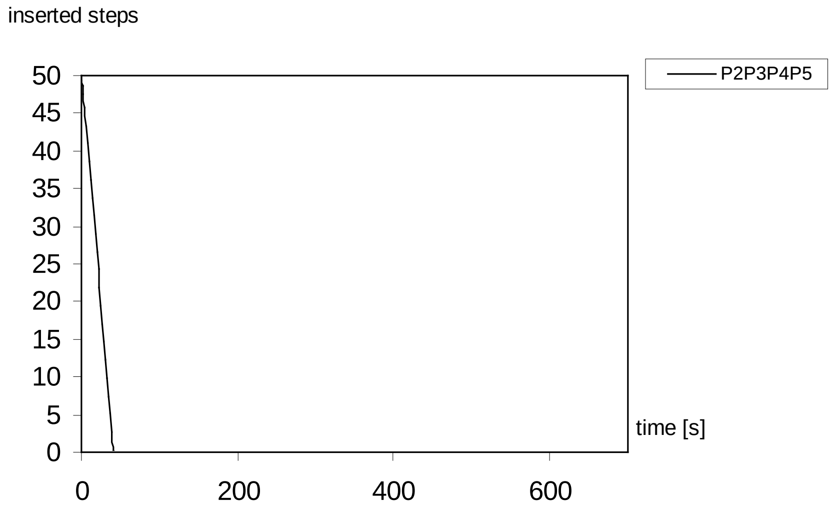
FEEDWATER FLOWRATE PER SG

SECTION 16.1.3.1.5 - FIGURE 2 (7/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
P1 P2 P3 P4 P5 Bank Insertion



P1 BANK INSERTION



P2 P3 P4 P5 BANKS INSERTION

3.2. ATWS BY RPR [PS] FAILURE [STATE A]

This section addresses the following PCC-2 events, assuming the failure of Reactor Trip signal from the Reactor Protection System:

- Excessive increase of secondary side steam flow (GCT [MSB] opening),
- Loss of main feedwater flow (LOFW),
- Loss of offsite power to the station auxiliaries (LOOP),
- Uncontrolled boron dilution,
- Uncontrolled RCCA bank withdrawal.

The above five events are representative of the most limiting ATWS cases.

They are analysed with respect to their consequences on:

- The RCP [RCS] integrity (overpressure protection aspect),
- The core integrity (fuel protection aspect).

The events considered are the same as in section 3.1 except that the failure to scram at occurrence of the F1A reactor trip signal results from the failure of the F1A signal itself (all control/shutdown rods available, but F1A RT signal not actuated). In section 3.1 the failure to scram is caused by the mechanical failure of the rods (F1A RT signal available, but all rods fail).

Calculations are provided only when it is insufficient to conclude that the safety criteria are met. When a calculation is not performed, the evaluation uses the calculation performed in section 3.1 above as a basis for transient illustration.

3.2.1. ATWS "Excessive increase in secondary steam flow"

The following evaluation uses for illustration the analysis performed in section 3.1.1, related to ATWS from mechanical rod failure:

- Prior to occurrence of the F1A RT signal, the transient is identical to the one presented in section 3.1.1,
- After the occurrence of the F1A RT signal, the transient differs from the one presented in section 3.1.1 as follows:
  - the absence of "ATWS signal" and associated actions (RBS [EBS] immediate actuation, all RCP [RCS] pumps cut-off on "SG-level MIN2 Wide Range") since the F1A RT signal has not been triggered,
  - the absence of actions associated with the failed F1A RT signal, i.e. TT, ARE [MFWS] high-load lines isolation, and VIV [MSIV] isolation, and
  - if necessary to meet the safety criteria, the actuation of the dedicated RRC-A feature introduced to cope with the loss of F1A RT signal.

With respect to Overpressure Protection, a generic approach is implemented, which covers any ATWS sequence: the situation to be avoided is the emptying of SG with the core remaining at high power level. This situation is avoided by the automatic actuation of a diverse RT/TT signal on "SG-level MIN3 Wide Range" implemented out of the Reactor Protection System (RPR [PS]), e.g. in the Process Automation System (PAS). This diverse RT signal is a RRC-A feature, F2 classified.

Following "excessive increase of secondary side steam flow" by GCT [MSB] opening, the SG water content is not depleted and there is no risk of RCP [RCS] overpressure. The diverse F2 RT/TT is not actuated, and not needed.

Following "excessive increase of secondary side steam flow" by VDA [MSRT] opening (or any secondary valve leading to reduction of secondary side water inventory), the secondary side water content decreases continuously, which results in a SG water level depletion. The diverse F2 RT/TT is actuated, and protects RCP [RCS] against excessive overpressure.

With respect to Fuel Protection, the "reactor power limitation" function limits the power excursion resulting from the uncontrolled cooldown, and stabilises the core power below approximately 105% NP).This limitation first blocks control rod withdrawal, then decreases turbine power by reducing turbine inlet valve opening, and if necessary inserts control rods. During the transient, the DNBR decrease remains within the initial DNBR margin, and there is no risk of DNB. The fuel integrity is not impaired.

Following "excessive increase of secondary side steam flow" by GCT [MSB] opening, which bounds any uncontrolled cooldown in ATWS, the core power is finally removed by approximately half via the GCT [MSB] (capacity ≈ 50% nominal steam flow) and half by the turbine.

3.2.1.1. System Sizing

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

3.2.2. ATWS “Loss of main feedwater”

The following discussions uses for illustration the calculations performed in section 3.1.2 related to ATWS from mechanical rod failure:

- Prior to occurrence of the F1A RT signal, the transient is identical to section 3.1.2.
- Following the occurrence of the F1A RT signal, the transient differs from section 3.1.2 by:
    - the absence of "ATWS signal" and its associated actions such as RBS [EBS] immediate actuation and all RCP [RCS] pumps cut-off on "SG-level MIN2 Wide Range" since the F1A RT signal has not been triggered,
    - the absence of actions associated with the failed F1A RT signal (i.e. turbine trip) – the absence of ARE [MFWS] high-load lines isolation has no effect since the ARE [MFWS] is lost),
    - the actuation of the dedicated RRC-A feature introduced to cope with the loss of F1A RT signal, if necessary to meet the safety criteria..

With respect to Overpressure Protection, the situation to be avoided is the SG emptying with the core remaining at a high power level. This situation is avoided by the automatic actuation of a diverse RT/turbine trip signal on "SG-level MIN3 Wide Range" implemented by the Reactor Protection System (RPR [PS]) in the Process Automation System (PAS). This diverse RT signal is an F2 classified RRC-A feature.

With respect to Fuel Protection, compared to the "rods failure" case analysed in section 3.1, the RCP [RCS] pumps are not tripped on "SG-level MIN2" (signal not triggered since there is no ATWS signal). Consequently, there is no DNBR margin reduction. In parallel with the SG-level decrease, the RCP [RCS] average temperature control initiates control rods insertion to keep RCP [RCS] temperature at its setpoint value (RCP [RCS] / SG heat transfer efficiency decreases with SG water level). The core power decreases, which increases the DNBR margin. On "SG-level MIN3", the diverse F2 RT / turbine trip signal implemented by the RPR [PS] is actuated and scram occurs. There is no longer a risk of fuel damage. During the transient, the DNBR margin remains above the initial DNBR margin, and there is no risk of DNB. The fuel integrity is not impaired.

3.2.2.1. System Sizing

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

3.2.3. ATWS - Loss of offsite power (LOOP)

The following argument uses for illustration the calculations performed in section 3.1.3, related to ATWS from mechanical rod failure:

- Prior to occurrence of the F1A RT signal, the transient is identical to section 3.1.3.
- Subsequent to the occurrence of the F1A RT signal, the transient differs from section 3.1.3 by:
    - the absence of "ATWS signal" and associated action (RBS [EBS] actuation) since the F1A RT signal has not been triggered (note: absence of RCP [RCS] pumps cut-off on "SG-level MIN2 Wide Range" has no effect, because all RCP [RCS] pumps are lost at beginning of transient),
    - the absence of actions associated with the failed F1A RT signal, which has no effect since turbine trip and loss of ARE [MFWS] occur at beginning of transient, and
    - the actuation of the dedicated RRC-A feature introduced to cope with the loss of F1A RT signal, if necessary to meet the safety criteria.

With respect to Overpressure Protection, the situation to be avoided is the emptying of SG with the core remaining at high power level. This situation is avoided by the automatic actuation of a diverse RT/turbine trip signal on "SG-level MIN3 Wide Range" implemented out of the Reactor Protection System (RPR [PS]), e.g. in the Process Automation System (PAS). This diverse RT signal is an F2 classified RRC-A feature.

With respect to Fuel Protection, the most troublesome event is the decrease in reactor coolant flow (caused by the initiating event) that results in DNBR margin reduction. The "rod failure" case analysed in section 3.1.3 shows that the minimum DNBR occurs at the early stage of the transient, before any mitigation action has occurred. The transients "rod failure" and "PS failure" cases are similar. On "SG level MIN3" the scram is actuated in the "PS failure" case by the diverse F2 RT/turbine trip signal implemented out of the RPR [PS], which is beneficial compared to the "rod failure" case. After scram, there is no longer risk of fuel damage. During the transient, the minimum DNBR value is that in the "rod failure" case, which is higher than the DNBR limit. There is no risk of DNB, and the fuel integrity is not impaired.

3.2.3.1. System Sizing

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

3.2.4. ATWS – RCV [CVCS] malfunction leading to a decrease in the boron concentration of the primary coolant

3.2.4.1. Identification of the causes

A total failure of the automatic shutdown system of the reactor on demand from the reactor protection system can be caused by:

- either the failure of the automatic reactor shutdown F1A signals (i.e. none of the signals sent by the Reactor Protection System de-energise the rod drive coils),
- or failure of the control and shutdown rods after de-energisation of their drive coils. In this case, the actuation of the rods due to control or limitation signals also fails.

Section 3.2.4 deals with the first case due to protection system failure. The second cause, linked to the locking of the rods, is dealt with in section 3.1.4.

3.2.4.2. Sequence of events

When the reactor is at power, the decrease in boron concentration causes an increase in the primary temperature, counteracted by the insertion of the control rods.

When the P1 bank (which is inserted first) reaches its insertion limit, the reactor core surveillance system generates an alarm. When the alarm occurs, the control procedures require the operator to check the boron meter: the dilution is identified and the operator is assumed to terminate it before 30 minutes.

Meanwhile, the control rods continue to be inserted thus maintaining the primary temperature and pressure constant. Thus, the transient amounts to a slow insertion of the RCCA banks at nominal conditions.

3.2.4.3. Safety criteria

It must be demonstrated that "the final state for the RRC-A studies" can be reached, i.e.:

- achieving long term core sub-criticality,
- ensuring that the residual heat is removed,
- integrity of the three barriers that prevent release of radioactive materials, in accordance with the following criteria:
  - o The DNBR remains higher than 1.21 (see Sub-chapters 4.4 and 14.1),
  - o The integrity of the RCP [RCS] is not impaired (the preliminary decoupling criterion is that the peak pressure of the RCP [RCS] must not exceed 130% of the design pressure (i.e. 228.5 bar abs). (Refer to Chapter 3.)

3.2.4.4. Method of analysis

The analysis is carried out using the computer code SMART/ FLICA (see Appendix 14A) for the analysis of core neutronics and thermal-hydraulics for simulating the core behaviour.

The change in boron concentration during the boration and dilution phases is calculated with the following formulas:

$BC(t \prec t_{EBS\ start}) = BC(t=0)\exp\left(-\frac{D}{M}t\right)$ in the dilution phase,

$BC(t \succ t_{EBS\ start}) = \left(BC(t_{EBS\ start}) - BC_{EBS}\right)\exp\left(-\frac{G}{M}\left(t - t_{EBS\ start}\right)\right) + BC_{EBS}$ during the boration by the RBS [EBS].

Where M is the RCP [RCS] water mass, D the dilution flow rate, G is the RBS [EBS] flow rate, BC_{EBS} is the RBS [EBS] boron concentration and $T_{EBS\ start}$ is the RBS [EBS] start time.

The DNBR calculation uses the FLICA code (see Appendix 14A).

3.2.4.5. Specific assumptions

- The core is initially in 100% NP nominal conditions. The control rods are at their recommended insertion.

3.2.4.5.1. Neutronic Data

- The neutronic data are calculated at BOL for the equilibrium cycle of the UO_2, INOUT, 12 months fuel management, which presents the highest FΔH.
- The boron concentration is the maximum BOL value of all the UO_2 fuel management cycles to minimise the dilution (2239 ppm) [Ref-1].
- The Doppler coefficient is the minimum value of all the UO_2 fuel management cycles (-2.5 pcm/ºC [Ref-1]) to maximise the FΔH and to make the DNBR conservative.
- The moderator coefficient is the BE value of the current BU and cycle. This assumption leads to a more onerous axial power distribution with regard to the DNBR calculation.

3.2.4.5.2. Assumptions concerning the dilution and boration calculations [Ref-1] [Ref-2]

- The water volume in the RCP [RCS] (except the PZR, the surge line and the upper head plenum) taken into account for the dilution calculation is 336 m^3, corresponding to a mass of 235 te at nominal power conditions. It is a minimum value to maximise the dilution. The use of the best estimated primary water mass (312 te) to calculate the boration by the RBS [EBS] would be non-conservative.
- The dilution flow rate assumed is 36 te/h.
    - o The flow rate of the RBS [EBS] is 2.8 kg/s, the boron concentration is 11200 ppm, the tank volume is 27 m^3, which corresponds to 27 te in normal conditions (see Sub-chapter 14.1).

3.2.4.5.3. DNBR calculation

The DNBR calculation takes into account the axial power distribution in the hot channel and the F∆H provided by the 3D core calculation (see SMART code, Appendix 14A).

3.2.4.6. Results and conclusion

3.2.4.6.1. Results for EPR 4250 MW [Ref-1]

The continued dilution for 30 minutes after reaching the P1 bank insertion limit causes the full insertion of the P1 and P2 rod banks and the partial insertion at the top of the core of the P3 rod bank; at this time the boron concentration is 2055 ppm. If the operator actuates the RBS [EBS] to reach the final state, the boron concentration reaches 2813 ppm which leads to a sub-criticality of 3588 pcm.

The minimum DNBR occurs when the P1 and P2 rod banks are fully inserted. The insertion of the P3 rod bank, even if it causes an increase in the enthalpy rise factor, skews the axial power distribution to the bottom of the core which increases the DNBR.

The minimum DNBR is significantly higher than the 1.21 criterion (2.03).

As the temperature is kept constant by the control rods insertion, there is no increase in the primary pressure.

3.2.4.6.2. Results for EPR 4500 MW

The main differences between EPR 45000 MW and EPR 4250 MW are:

- Higher initial power (6%),
- Higher RCP [RCS] average temperature (+1ºC),
- Lower initial boron concentration
- Same bounding dilution flow rate,
- Same RBS [EBS] boration capability.

While the higher power and temperature are penalising with regard to the DNBR, the dilution has a lower impact due to the lower initial boron concentration. Consequently, for EPR 4500 MW the DNBR margin will be reduced but the DNBR will remain above the criterion.

3.2.4.7. System sizing

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

3.2.5. ATWS – Uncontrolled RCCA bank withdrawal

3.2.5.1. Identification of the causes

The PCC-2 event, uncontrolled RCCA bank withdrawal, combined with the complete failure of reactor trip system is classified as RRC-A event.

A total failure of the reactor trip on demand from the reactor's protection system can be caused by:

- either the failure of the automatic reactor shutdown F1A signals (i.e. none of the signals sent by the protection system de-energises the rod drive coils),
- or failure of the control and shutdown rods after de-energizing of their drive coils. In this case, actuation of the rods due to control or limitation signals also fails.

Section 3.2.5 deals with the first case, due to protection system failure. The second case, linked to rod failure, is dealt with in section 3.1.5.

3.2.5.2. Sequence of events

The uncontrolled withdrawal of the control rods causes a rapid increase in core power and an increase in the primary circuit temperature and pressure. On the secondary side, the pressure increase is limited by the opening of the GCT [MSB].

As no automatic shutdown occurs, nor any other power reduction, due to the protection system failure, the withdrawal of the rods is not stopped until they are fully withdrawn. Reactivity insertion and the resulting power increase can be higher than the values obtained in the case of rod failure (see section 3.1.5). However, reactivity feedback due to moderator and Doppler temperature effects limit the power excursion.

When the low SG level wide range, MINin 3, is reached, a diverse automatic shutdown signal is generated which ends the transient by the insertion of both the control and shutdown rods.

3.2.5.3. Safety criteria

It must be demonstrated that "the final state for the RRC-A studies" can be reached, i.e.:

- achieving long term core sub-criticality,
- ensuring that the residual heat is removed,
- integrity of the three barriers that prevent release of radioactive materials, in accordance with the following criteria:
  - o The DNBR remains higher than 1.21 (see Sub-chapters 4.4 and 14.1),
  - o The integrity of the RCP [RCS] is not impaired (the preliminary decoupling criterion is that the peak pressure of the RCP [RCS] must not exceed 130% of the design pressure (i.e. 228.5 bar abs, see Chapter 3).

3.2.5.4. Protection and mitigation actions

The following functions mitigate the accident consequences in the event of an uncontrolled rod withdrawal coupled with a failure of the protection system.

- reactor trip and turbine trip on “SG level, wide rage < MINin 3” (F2)

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

3.2.5.5. Method of analysis

The analysis is carried out using the following computer codes:

- MANTA (see Appendix 14A) for the analyses of the thermal-hydraulic behaviour of the primary and secondary systems (RCP [RCS] and SG), accounting for F1 systems operations

- SMART/FLICA (see Appendix 14A) for the neutronic and thermal-hydraulic behaviour of the core.

The DNBR calculation is performed with the FLICA code (see Appendix 14A).

The analysis starts from an initial state of 100% nominal power with a moderator coefficient of -10 pcm/°C. This assumption covers more than 95% of plant life.

3.2.5.6. Specific assumptions

3.2.5.6.1. Single failure and maintenance

Single failure and maintenance do not have to be taken into account in the RRC-A analyses.

3.2.5.6.2. Initial conditions

The initial conditions are ”Best Estimate” conditions at 100% NP (see Section 16.1.3.2.5 – Table 1).

The thermal-hydraulic primary flow rate is assumed, being conservative even though it is not strictly required by the accident analysis rules for RRC-A sequences.

3.2.5.6.3. Assumptions related to the rods

The initial positions assumed for the RCCAs are conservative but realistic as in section 3.1.5 (see Section 16.1.3.1.5 - Figure 1, definition of control rods):

- P1 is inserted 150 steps

- The other control rods (P2 and P5) are inserted 50 steps.

A conservatively high withdrawal speed for all RCCAs is assumed for this assessment (75 cm/min).

3.2.5.6.4. Neutronic data

The neutronic data are calculated at BOL.

The moderator temperature coefficient is -10 pcm/°C [Ref-1] at the beginning of the transient. The Doppler temperature coefficient is at its minimum absolute value to make the rate of core power increase conservative. This coefficient is fixed at a value of -2.5 pcm/°C [Ref-1] at the beginning of the transient (minimum in absolute value without uncertainties, it covers all fuel management schemes).

The reference fuel management is used to maximise the F(H, when the moderator and Doppler temperature coefficients are set at their minimum values, which makes the DNBR calculation conservative.

3.2.5.6.5. Assumptions related to controls

The SG level control and the primary temperature control cannot be actuated.

Pressuriser pressure control by the pressuriser spray is claimed to delay the automatic reactor trip signal due to high pressuriser pressure.

Pressuriser level control is not modelled in the present analysis.

3.2.5.6.6. Assumptions related to the systems

The claimed F1A systems are as follows: ASG [EFWS], pressuriser safety valves, VDA [MSRT], MSSV valves.

Input, limits and delays are summarised in Section 16.1.3.2.5 - Table 2.

3.2.5.6.7. DNBR calculation

The DNBR calculation takes into account the axial distribution in the hot channel and the ΔH provided by the 3D core calculation (SMART, Appendix 14A) at the time of minimum DNBR. No penalty on the local thermal power is applied for analyses within the RRC-A domain.

3.2.5.7. Results and conclusion [Ref-1]

The sequence of events is shown in Section 16.1.3.2.5 - Table 3. Section 16.1.3.2.5 - Figure 2 shows the behaviour of the main parameters.

The core power first increases with the rod withdrawal, the minimum DNBR occurs after 3 minutes, long after the core thermal power reaches its maximum (at 39.5 seconds).

Since no reactor trip is activated, the turbine remains available, the steam generator pressure increases up to the GCT [MSB] opening pressure, then stabilises and finally decreases slowly.

The first pressuriser safety valve is demanded, the peak primary pressure is 181.1 bar at 69.2 seconds, at the RCP [RCS] pumps outlet. The GCT [MSB] opens on the secondary side.

After the withdrawal of the P1 rod bank and given that the SG can remove the core heat, the primary parameters, and in particular the core power, stabilise at an equilibrium level. The DNBR also reaches a stable value (at approximately 200 seconds).

Given that the SG level control is not taken into account in this analysis, the SG level decreases slowly. On low SG level (MINin 3), an automatic reactor trip is initiated by the diverse protection, which stops the transient. When the SG level is controlled automatically, no automatic reactor trip occurs; however, stable conditions are reached.

3.2.5.7.1. Criteria fulfilment

With regards to the maximum pressure in the primary circuit, the results of the transient are covered by those presented in section 3.1.5 (rods failure). The decoupling criterion of 130% of the design pressure is met with a large margin.

There is no particular problem regarding the decay heat removal.

The minimum DNBR, reached at about 200 seconds, is much higher than the 1.21 criterion (2.25).

3.2.5.8. System sizing

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

SECTION 16.1.3.2.5 - TABLE 1

ATWS - Uncontrolled RCCA Banks Withdrawal – Initial State

Parameter	Initial value
SG heat transfer area	Nominal
RCP [RCS] flow rate	Thermal hydraulic (27180 m3/h)
Power	100% FP
Pressure	155 bar
Average temperature	312.8°C
PZR level	56%
SG pressure	78 bar
ARE [MFWS] inlet temperature	230°C
SG level	49% GE

SECTION 16.1.3.2.5 - TABLE 2

Main Assumptions
ATWS - Uncontrolled RCCA Bank Withdrawal

Parameter	Value
Reactor trip signal on SG level < MINin 3	
Setpoint	(1)later
Delay	(1)later
VDA [MSRT] actuation on SG pressure > MAXax 1	
Setpoint	95.5 bar
Delay	0.9 + 1.5 + 0.5 = 2.9 s
VDA [MSRT] closing delay	40 s
Capacity	1150 te/h under 100 bar
MSSV valves SG level (wide range cold side) < MIN2	
Setpoint	105 bar
Accumulation	3%
Capacity	575 te/h under 100 bar, 2 per SG
PZR safety valves	
Setpoint	175.5 / 179.5 / 179.5 bar
dead time	0.5 s
opening time	1.5 s
Capacity	steam 300 te/h, liquid 450 te/h under 176 bar
Hysteresis	10 bar

Note (1): to be defined at the detailed design stage

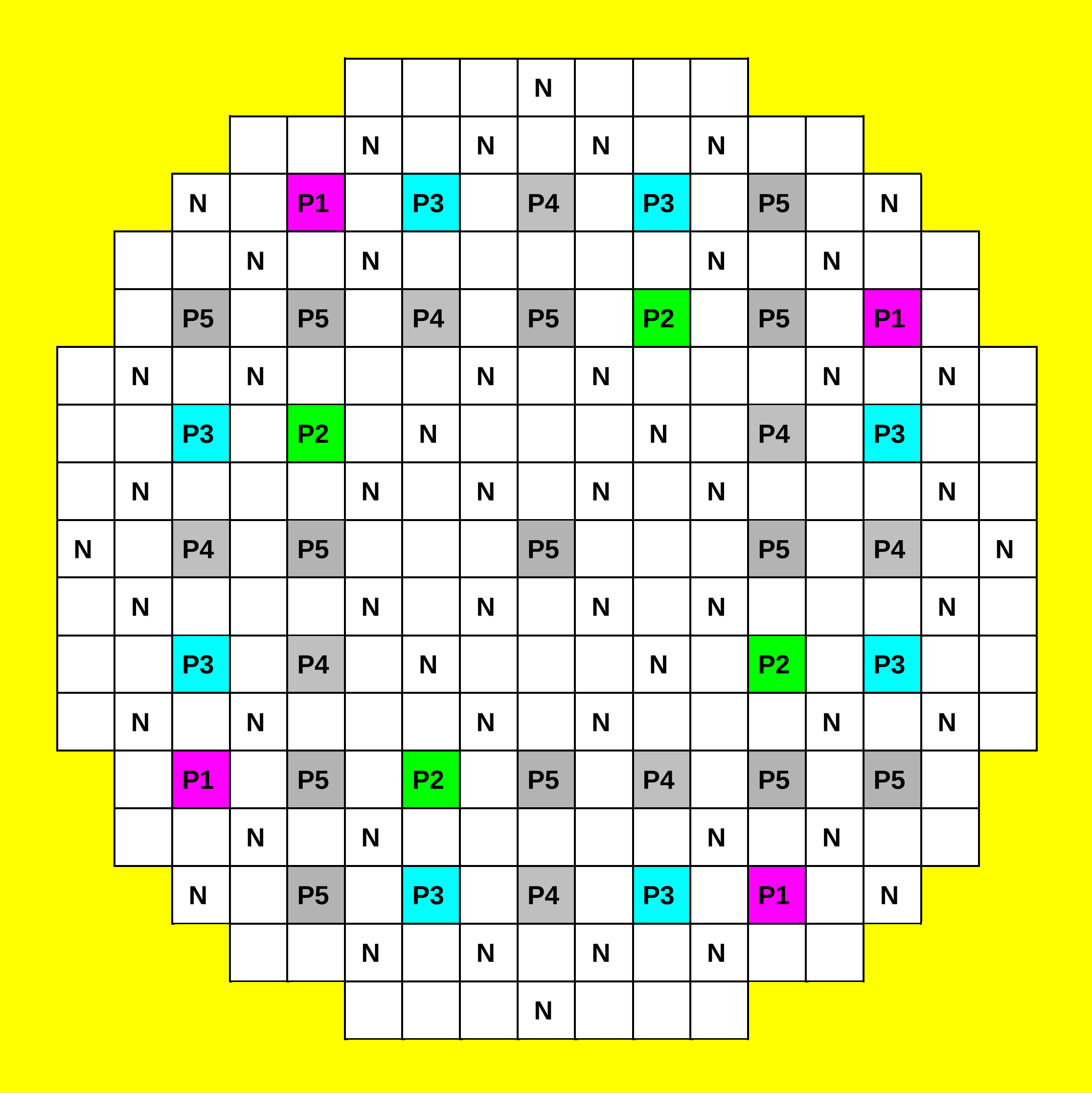
SECTION 16.1.3.2.5 - TABLE 3

Sequence of Events
ATWS - Uncontrolled RCCA Bank Withdrawal

EVENT	TIME (second)
Start of uncontrolled withdrawal of control banks at 75 step/min	0.
Maximum core thermal power level (113.6%)	39.5
1st PSV opening / closing (only first PZR valve)	69/79.5
Stable state reached	≈300s
Minimum DNBR (2.25)	

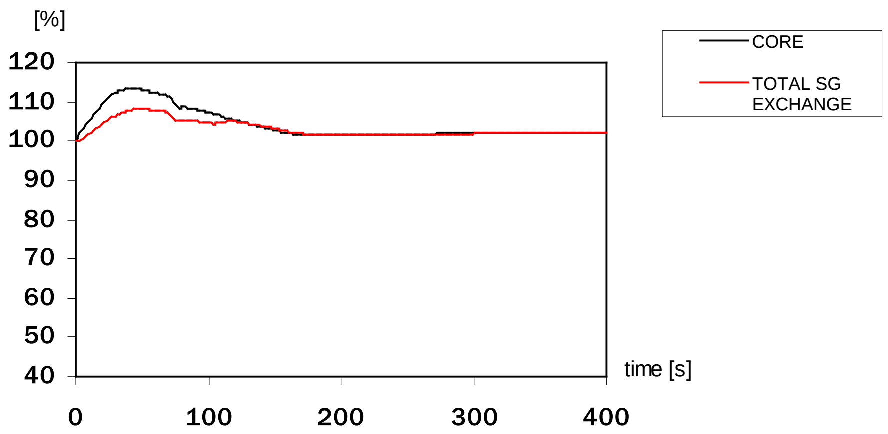
SECTION 16.1.3.2.5 - FIGURE 1

ATWS - Uncontrolled RCCA Bank Withdrawal

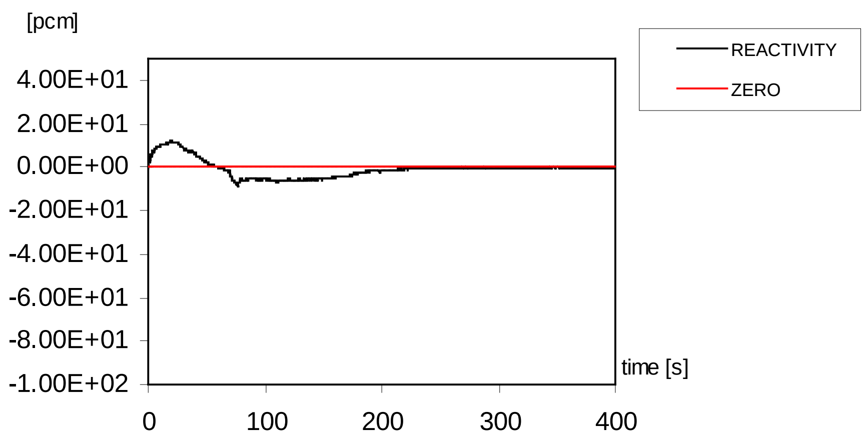


SECTION 16.1.3.2.5 - FIGURE 2 (1/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
Power and Reactivity



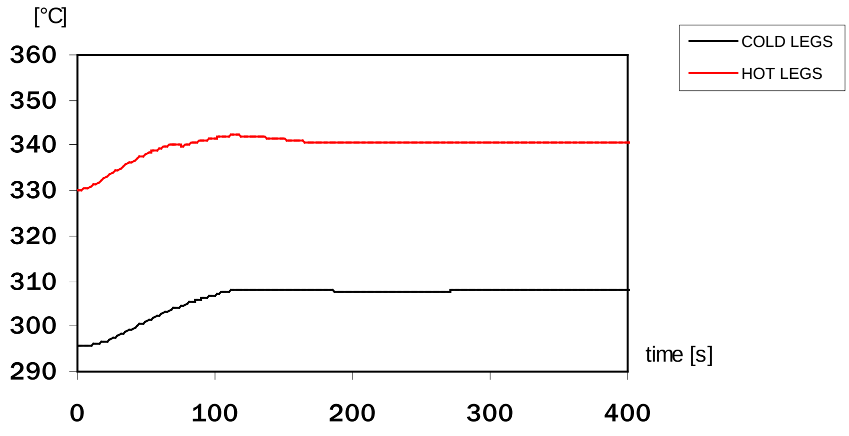
POWER (CORE AND SG)



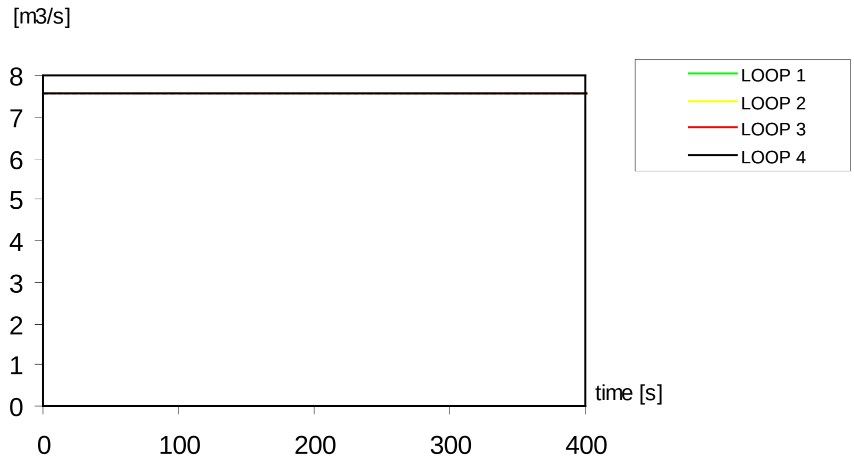
REACTIVITY

SECTION 16.1.3.2.5 - FIGURE 2 (2/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
RCP [RCS] Temperatures and Flow Rate



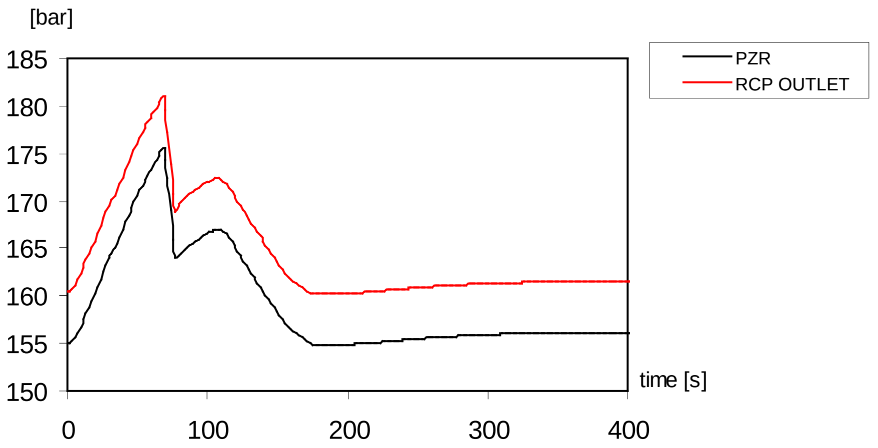
RCS TEMPERATURES



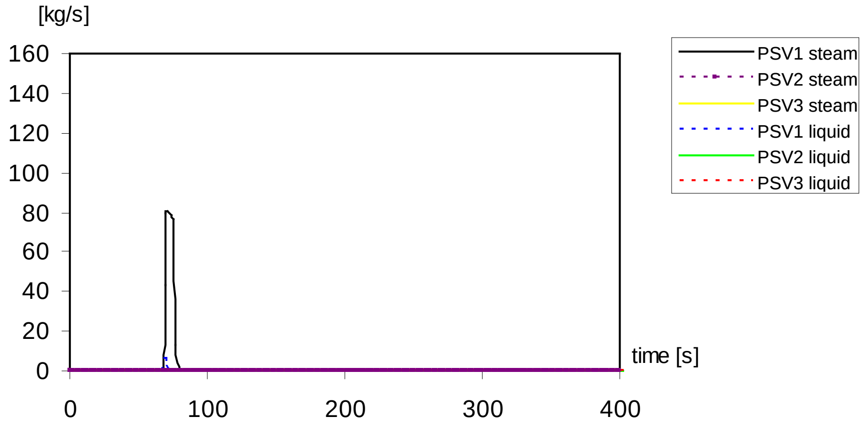
RCS FLOWRATE

SECTION 16.1.3.2.5 - FIGURE 2 (3/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
Primary Pressure and PSV Flow Rate



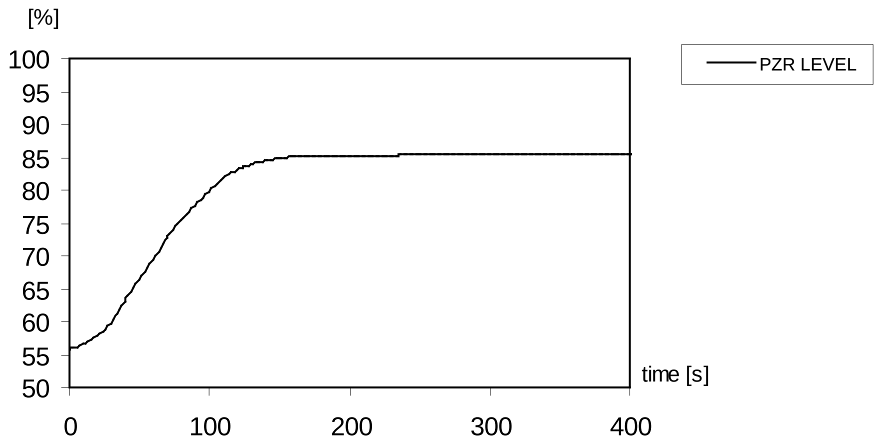
PRIMARY PRESSURE



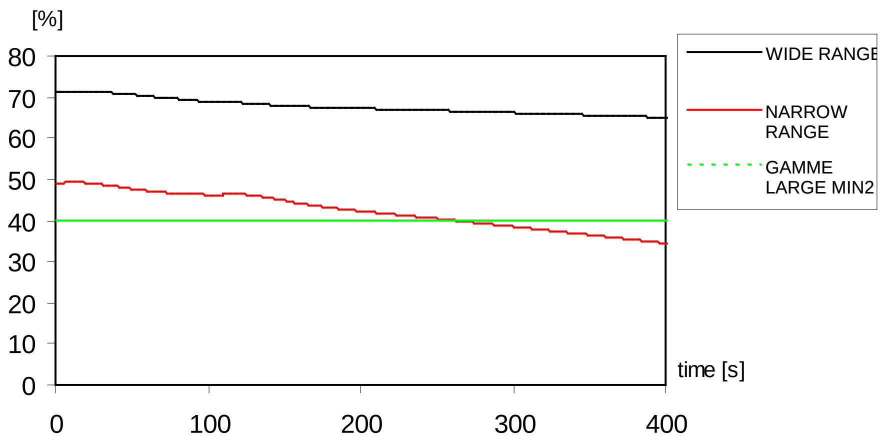
PSV FLOWRATE

SECTION 16.1.3.2.5 - FIGURE 2 (4/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
PZR Level and SG Level



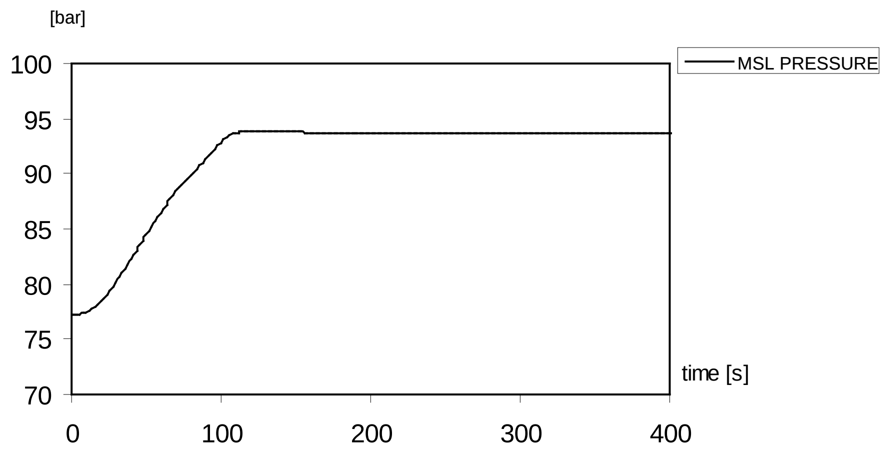
PZR LEVEL



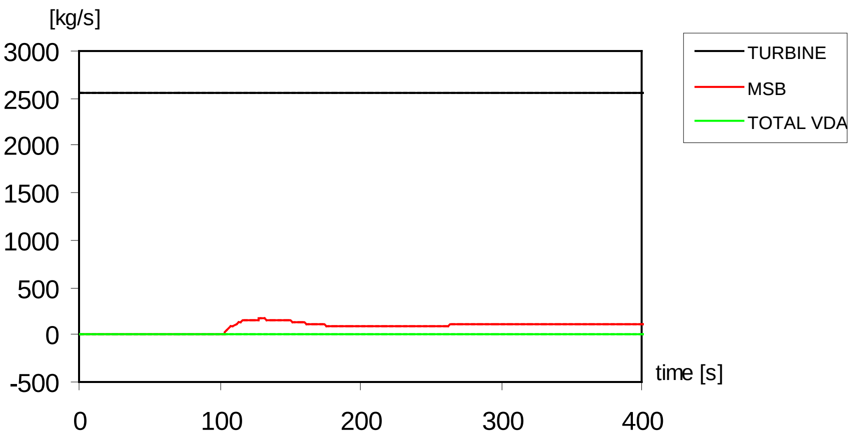
SG LEVELS

SECTION 16.1.3.2.5 - FIGURE 2 (5/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
Main Steam Pressure and Steam Flow Rate



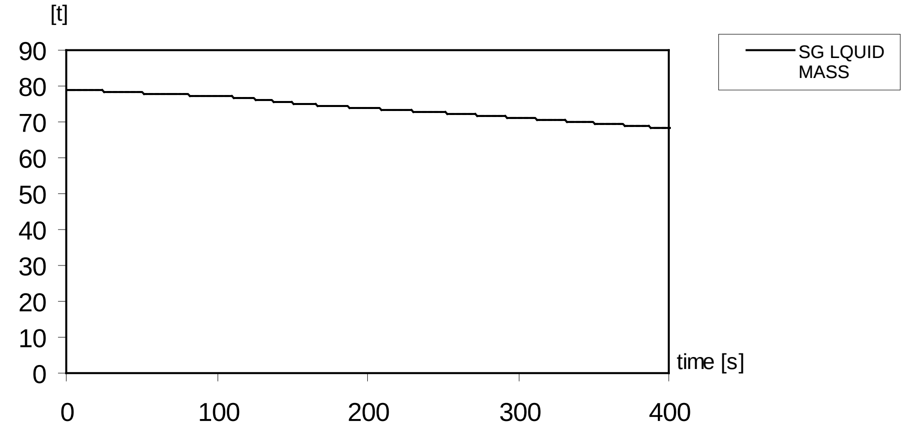
MAIN STEAM PRESSURE



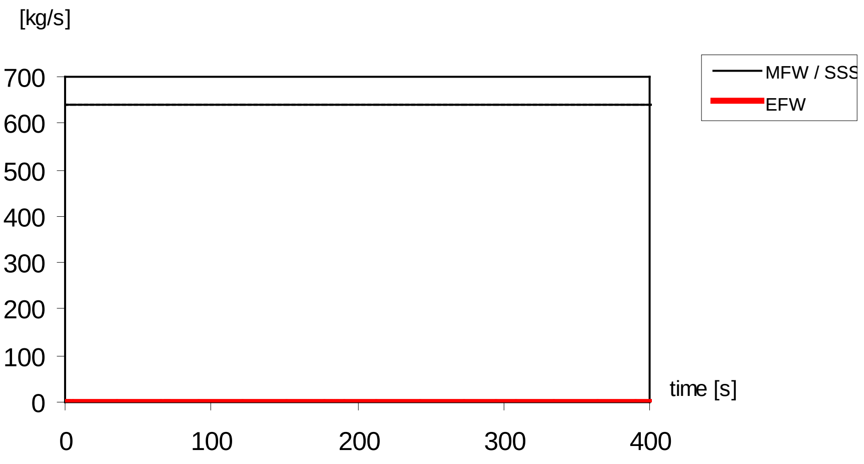
STEAM FLOWRATE

SECTION 16.1.3.2.5 - FIGURE 2 (6/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
SG Liquid Mass and Feedwater Flow Rate per SG



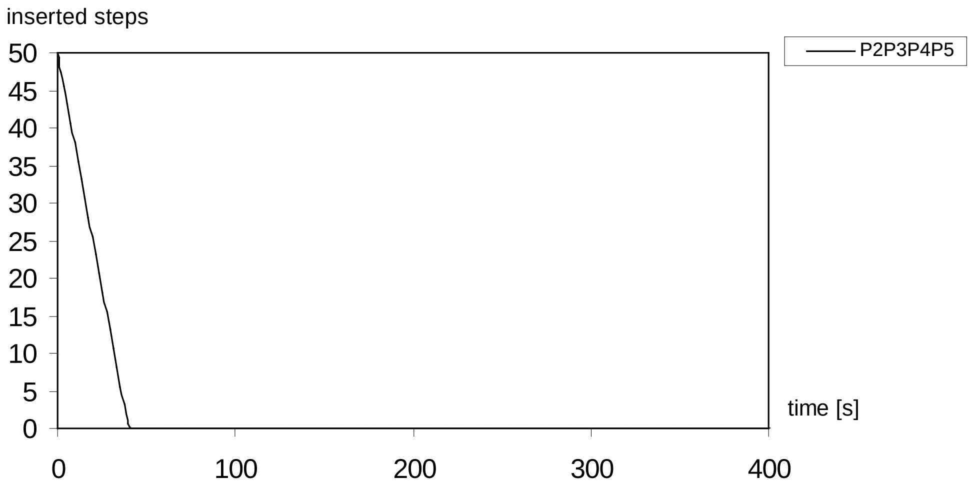
SG LIQUID MASS



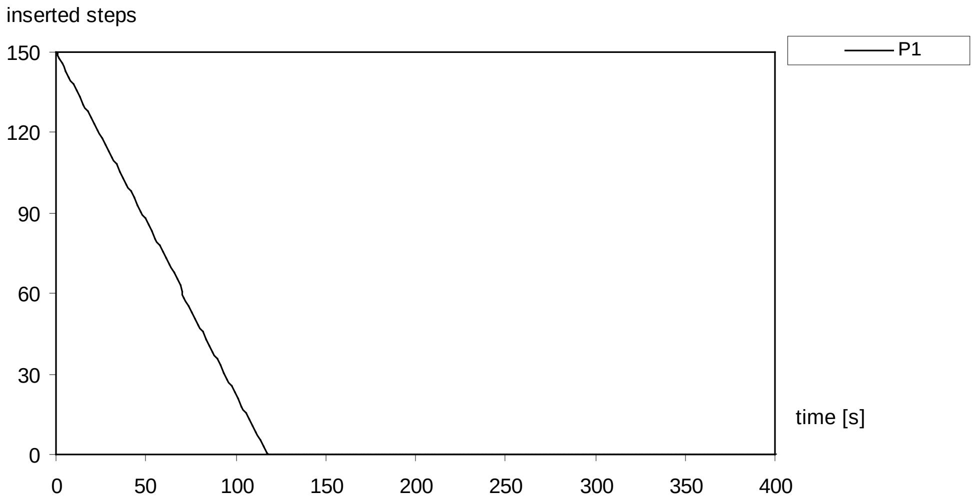
FEEDWATER FLOWRATE PER SG

SECTION 16.1.3.2.5 - FIGURE 2 (7/7)

ATWS - Uncontrolled RCCA Bank Withdrawal
P1 P2 P3 P4 P5 Bank Insertion



P2 P3 P4 P5 BANKS INSERTION



P1 BANKS INSERTION

3.3. STATION BLACKOUT (SBO), IN STATE A

The Loss of offsite power (LOOP) combined with the failure of the four emergency diesel generators leads to a total loss of power on the emergency and non-emergency 10 kV bus bars. This is referred to as Station Blackout (SBO).

3.3.1. Identification of the causes and description of the transient

The SBO event, without additional power sources, leads to the unavailability of the following main systems:

- the main feedwater system (ARE [MFWS]) and its feedwater supply, shutdown and start-up (AAD [SSS]) system and the condenser cooling system,
- the emergency feedwater supply (ASG [EFWS]) to the SG,
- the chemical and volume control system (RCV [CVCS]) and the safety injection system (RIS [SIS]),
- the component cooling water system (RRI [CCWS]) and the essential services water system (SEC [ESWS]),
- the RCP [RCS] pumps, including injection into the primary pump seals and the thermal barrier cooling,
- the ventilation system,
- the battery chargers.

In the event of SBO, the loss of the RCP [RCS] pumps initiates the automatic reactor trip (on the "low RCP [RCS] pump speed" signal).

In spite of the loss of injection at the RCP [RCS] pumps seals and the thermal barriers, the integrity of the seals and RCP [RCS] pumps and the leak-tightness of the RCP [RCS] are ensured by the standstill seal system (DEA [SSSS]) and by the closure of the RCP [RCS] pumps seals leak-off lines.

The SG are no longer supplied with water, and their level drops due to the effect of the residual heat. Steam is produced and released to atmosphere via the atmospheric steam dump valves (VDA [MSRT]) whose power is supplied from DC power batteries. In addition, the steam generator safety valves are available for the removal of the residual heat.

At 100% NP, the SG water mass is about 80 te per SG and the SG become empty approximately 1.5 hours after the start of the SBO if no alternative measures are taken. In addition, without a battery change, the battery power is exhausted in approximately 2 hours after the start of the SBO, which causes the unavailability of the entire instrumentation and control system (safety and operational).

As a result, in order to remove the residual heat and prevent core melt and to limit the long-term accident consequences, additional power supply sources are provided to bring the plant back to a safe final state in the event of SBO. Two ultimate emergency diesel generators are a source of additional power.

The ultimate emergency diesel generators have a relatively low power output and are connected to the 690 V bus bars of trains 1 and 4. They are separate from the four main diesel generators, and as a result, common mode failure of the main and ultimate emergency diesels is not taken into account (see Sub-chapter 8.3).

The ultimate emergency diesel generators mainly supply power to the:

- ASG [EFWS] pumps in trains 1 and 4,
- some parts of the ventilation system,
- main instrumentation and control systems,
- lighting of the main control room (MCR).

Operationally, one ultimate emergency diesel generator is sufficient for the SBO event, but two ultimate emergency diesel generators are required to meet the core melt frequency target.

The ultimate emergency diesel generators are manually started and connected from the control room. It is assumed that this action is performed at least 30 minutes after the occurrence of an SBO event. As mentioned above, the emptying of the SG would occur after 1.5 hours without feedwater supply. One ASG [EFWS] train is sufficient for the removal of the residual heat after 1 hour.

After starting both ultimate emergency diesel generators, manual actions are necessary to reach the final state in which the plant can remain until the offsite power supply is restored (which is assumed to occur after 24 hours). These manual actions include:

- start-up of both ASG [EFWS] pumps connected to the ultimate emergency diesel generators (in trains 1 and 4),
- opening of the ASG [EFWS] header (usually closed) for feedwater supply to the four SG,
- Initiation of RCP [RCS] cooldown via the VDA [MSRT] valves down to 60 bar to ensure the long-term protection of the RCP [RCS] pumps seals against the thermal and mechanical loads.

The final state is reached due to the above actions; thus:

- the core sub-criticality is ensured (after the automatic shutdown of the reactor, limited cooldown occurs and no boration is thus necessary),
- the removal of the residual heat is performed using the ASG [EFWS] and the VDA [MSRT] valves (both supplied with power by the ultimate emergency diesel generators),
- The activity release remains under control (due to the above-mentioned actions).

3.3.2. Methods and assumptions

For the analysis of RRC-A events, such as SBO, the initial conditions selected are principally best estimate assumptions.

All the functions and systems, except for those that are affected by the event initiation, are assumed available to mitigate the transient (i.e. no single failure or preventive maintenance is taken into account).

The following functions or systems, automatically or manually, are available and taken into account for the mitigation of an SBO:

3.3.2.1. Automatic functions:

- reactor trip on “low RCP [RCS] pump speed”,
- the pressuriser safety valves for primary pressure and residual heat removal,
- the VDA [MSRT] and the SG safety valves for secondary pressure control and residual heat removal at hot zero load conditions.

3.3.2.2. Manual functions:

- start-up of two ultimate emergency diesel generators after 30 minutes,
- start-up of the two ASG [EFWS] pumps in trains 1 and 4 after 45 minutes,
- opening of the ASG [EFWS] header after 1.5 hours,
- manual cooldown down to 60 bar via the VDA [MSRT] after 2 hours.

3.3.3. Results and conclusions [Ref-1]

The initial phase of the scenario is comparable with the loss of offsite power supply event. It is characterised by an immediate reactor trip, transition to natural circulation in the RCP [RCS] and by the residual heat removal on the secondary side via the VDA [MSRT] valves, taking credit for the large water inventory in the SG (early on, the ASG [EFWS] does not supply feedwater to the SG).

After 45 minutes, the two emergency feedwater pumps that receive their power supply from the two ultimate emergency diesel generators, are started and feed the associated SG. At this point, the water level (wide range) in all SG is approximately 50%, i.e. the SG have a heat removal capacity that is much greater than required for such conditions; at an already low residual heat (approximately 1.5% NP). The feedwater supply to both SG from the ASG [EFWS] causes the water level to return to the nominal level in approximately 10 minutes, while the levels of the other SG continue to decrease.

After 1.5 hours (i.e. 1 hour after the start-up of the ultimate emergency diesel generators), the ASG [EFWS] header is opened so that both SG emergency feedwater pumps can supply feedwater to all the SG. Later, the level of two SG that did not initially receive feedwater and whose level dropped to 35% (corresponding to approximately 30 te per SG), starts rising rapidly.

The final state is reached after manual cooldown via the VDA [MSRT] to 60 bar initiated 2 hours after the start of the event. Following the RCP [RCS] cooldown, the cold legs temperature is reduced to approximately 275°C and the primary pressure stabilises at approximately 125 bar. The temperature and pressure conditions of the final state ensure that the long-term safety requirements for the RCP [RCS] pumps seals are met. Thus, the RCP [RCS] remains leak-tight and there is no loss of primary coolant.

The characteristics of the sequence of events are:

- the primary or secondary safety valves (PSV or MSSV) are not challenged and the protection of the RCP [RCS] seals ensures a loss of primary coolant can be prevented;

- SG are not emptied and large margins are maintained.

Note that the start-up of the ultimate emergency diesel generators and the SG emergency feedwater supply must occur at latest 1.5 hours after the SBO to prevent a significant increase in the RCP [RCS] temperature. The integrity of the primary pumps' seals is maintained due to the partial cooldown.

In addition, the SG initial water inventory and the capacity of the ASG [EFWS] tanks are sufficient to cover a 24 hour time period including partial intermediate cooling.

3.3.4. System Sizing

The ultimate emergency diesel generator is sized by this event in order to supply electric power to two ASG [EFWS] pumps following Loss of offsite power (LOOP) combined with the failure of the four emergency diesel generators.

3.4. TOTAL LOSS OF FEEDWATER (STATE A)

3.4.1. Identification of Causes and Accident Description

The total loss of feedwater is defined as the spurious failure of the ARE [MFWS] pumps. The AAD [SSS] and ASG [EFWS] pumps are not available during the accident.

In the first phase of the event operator actions are not considered. Only systems and I&C functions that are automatically actuated are permitted. This period is assumed to be the initial 1800 seconds after the first signal is received. During this phase, the reactor trips, and the RCV [CVCS] (charging and letdown) attempts to compensate for Pressuriser level variations. The turbine is tripped along with the reactor trip, but the MS bypass is able to remove heat. This phase of the accident results in a reduction of the SG inventory. However, the heat transfer capability of the SG remains unchanged and the heat removal is via the MS bypass. In this phase, the following automatic actions occur:

- Partial reactor trip to 50% power, followed by reactor/turbine trip due to mismatch between reactor power and ARE [MFWS] flow. This occurs about 15 seconds after accident initiation.
- RCV [CVCS] charge/letdown to compensate for Pressuriser (PZR) level variations.
- PZR heater/spray to compensate for RCP [RCS] pressure variations.
- MS bypass for heat removal after turbine trip.
- During the second phase of the event, the PZR heaters are cut off, as soon as operator actions are possible (assumed at 1800 seconds). The SG continue to boil off, reducing their heat transfer capability. When the SG water level has reached the low level setpoint 3 m (14% WR), the RCP [RCS] pumps are cut off to reduce heat input to the RCP [RCS]. After about 1 hour, the SG completely dry out and the heat up of the primary coolant begins. The increase in RCP [RCS] temperature causes the RCP [RCS] pressure to increase to the PSV setpoint. The PSV(s) opens/closes to maintain RCP [RCS] pressure while primary coolant inventory is continuously lost, and the RPV level decreases. When the RPV level decreases below “loop low-level” setpoint, the operator initiates feed and bleed by opening the Primary Depressurisation System (PDS) and by manually starting the RIS [SIS]. The rapid depressurisation causes core voiding and clad heat-up. It also initiates SI, which arrests the clad heat-up and eventually recovers primary coolant inventory. The purpose of the analysis is to demonstrate that cladding temperatures remain within acceptable limits and that the final state (decay heat removal, core sub-critical) is reached.

In the second phase of the event, the assumed operator actions are:

- PZR heater cut off.
- The shutdown of RCP [RCS] pumps (on criteria of no ASG [EFWS] injection and low level setpoint in SG (level < 3 m; 14% WR)).
- Opening the PDS when the RPV level drops below the “loop low level” setpoint. The PDS discharge capacity is equivalent to the opening of the three PSVs.

- Start-up of both RCV [CVCS] charging pumps immediately before opening of the PDS

- Manual start-up of the RIS [SIS]

    - o It is noted that the actuation of feed and bleed by opening the PDS and start-up of RIS [SIS] represents the RRC-A feature for this scenario.

The blow down through the PDS leads to a pressure and temperature increase in the containment and IRWST (after failure of the Pressuriser relief tank). This is mitigated by the EVU [CHRS] (if needed) and the IRWST cooled via LHSI/RHR.

3.4.1.1. Final State

It must be demonstrated that the final state (required for RRC-A analyses) can be reached, i.e.

- Attainment of long-term sub-criticality

- Decay heat removal assured.

- Acceptable activity releases dictated by the integrity of the radiological barriers according to PCC-4 acceptance criteria.

For this demonstration, the following acceptance criteria are considered:

- The RCP [RCS] integrity must not be impaired (as a preliminary criterion, the pressure at the most loaded point of the RCP [RCS] should not exceed 130% of design pressure (i.e. 228.5 bar abs).

- The Peak Cladding Temperature (PCT) in the hot fuel rod should be < 1200°C.

- The containment pressure and temperature must not exceed the design limits.

3.4.2. Methods and Assumptions

The analysis does not take into account the feedwater reserves in the feedwater system (in particular the feedwater tank) which would provide an additional time before the opening of the pressuriser safety valves becomes necessary.

Calculations for the overall plant, system and core behaviour are performed using the CATHARE computer code (Version V1.3L) (See Appendix 14A). The CATHARE model includes only the average fuel rod and does not explicitly model the hot fuel rod. It is assumed that the acceptance criterion on cladding temperature (PCT Hot Rod < 1200°C) is satisfied if the peak cladding temperature in the average fuel rod remains below 600°C.

Operator actions are not considered in the first 30 minutes (1800 seconds) after receipt of the first safety-related signal.

3.4.2.1. Important events and qualification of the CATHARE models

The family of transients is a combination of loss of FW supply inducing loss of secondary side heat removal and LOCA after initiation of feed and bleed.

3.4.2.1.1. Primary Side Events

Significant events from the perspective of analytical modelling are:

- Heat up of the primary coolant and periodic opening/closing of the PSV.
- Natural circulation of nearly full primary system as well as a low RCP [RCS] inventory.
- Rapid depressurisation of the RCP [RCS] after opening of the PDS and some core uncovery and heat-up (i.e. typical LOCA condition).
- Refilling of the RCP [RCS] with the RCV [CVCS] and MHSI pumps when the primary pressure is < 85 bar. Accumulator injection. The LHSI pumps maintain the pressure at 20 bar.

3.4.2.1.2. Secondary Side Events

- Main Steam Line pressure increase due to closing of the turbine valves.
- Opening of GCT [MSB] valves to maintain a constant MS pressure at zero load.
- Loss of all SG secondary water inventory.
- Heat transfer at low water level.

3.4.2.1.3. Vessel and Core Events

- RPV level decrease due to PSV and PDS operations.
- RPV level swell caused by depressurisation.
- Core voiding and heat-up - local power and critical heat flux, degraded heat transfer

3.4.2.2. Qualification of the CATHARE models for primary side, secondary side and core events

CATHARE is validated against experimental data from a wide range of separate effects and integral experiments. The TLOFW scenario is mainly composed of loss of secondary side heat removal and LOCA phenomena. Hence, its qualification can be found in the sections for FWL break (section 3 of Sub-chapter 14.5) and LOCA (section 6 of Sub-chapter 14.5).

3.4.2.3. Initial and Boundary Conditions

Analyses of RRC-A events are conducted with realistic initial and boundary conditions. The initial values used are listed in Section 16.1.3.4 - Table 1.

3.4.3. Cases studied

The analysis follows the RRC-A guidelines (i.e. all systems/functions not affected by the initiating event are available for accident mitigation). However, to be consistent with the PSA results regarding TLOFW the analysis takes into account the following additional failures:

- for the first case, the unavailability of the first pressuriser safety valve,
- for the second case, the unavailability of an MHSI pump.

The two cases are characterised by the following conditions:

Case 1: the first PSV at 174 bar response pressure is not available both during automatic pressure limitation phase. Consequently the pressure limitation is implemented by the second and third PSV at 178 bar.

Case 2: when the RCP [RCS] pressure drops below the minimum MHSI delivery head of 85 bar (following actuation of bleed by the PDS) only three out of the four MHSI pumps are allowed to inject into the RCP [RCS].

3.4.4. Results

The sequence of events is very similar for both cases (first case with the failure of the first pressuriser safety valve, second case with the MHSI pump failure).

The results for the two cases are presented in Section 16.1.3.4 - Table 2 and Section 16.1.3.4 - Figures 1 to 4.

The accident was not recalculated for EPR 4500 MW. The capability of reaching the final state and meeting the acceptance criteria were deduced from the analysis results for EPR 4250 MW.

3.4.4.1. Results for EPR 4250 MW [Ref-1]

The plant can be without feedwater supply for 2 hours 30 minutes before feed and bleed actions are required. The primary coolant starts to heat up after approximately 1 hour and this leads to the first pressuriser safety valves being demanded (in particular due to the coast down of the primary pumps)at approximately 1 hour 50 minutes for the case 1 and 1 hour 35 minutes for case 2. The SG will eventually be empty after approximately 1 hour 15 minutes.

At t =10300 seconds for the case 1 and 9500 seconds for the case 2 the criterion for initiating the feed and bleed actions is met. Due to the opening of the PDS, the primary pressure decreases to 85 bar in 750 seconds for case 1 and 400 seconds for case 2, and the MHSI pumps start injecting. The accumulator injection starts at 11400 seconds for case 1 and 10150 seconds for case 2, after the temperature of the fuel clad has started to decrease (after the maximum core heat up).

The most important events are listed in the table, below.

Event	Time (s)	Coolant Inventory (te)	Cladding Temperature of the Average Core (°C)
100% P, initial power	0	295	365
RPV level ≤ bottom of the hot leg (criteria to start pressuriser bleed by opening the PDS)	10300 (case 1) 9500 (case 2)	145 (case 1) 130 (case 2)	350 (case 1 and 2)
Beginning of MHSI injection and minimum coolant inventory	11050 (case 1) 9900 (case 2)	50 (case 1 and 2)	420 (case 1) 370 (case 2)
Maximum Cladding Temperature for the Average Core	11200 (case 1) 10100 (case 2)	58 (case 1) 55 (case 2)	500 (case 1) 430 (case 2)

Since the maximum temperature of the average fuel rod cladding is only 500°C for the limiting case, (case 1 – with failure of the first pressuriser safety valve), the “Cladding temperature less than 1200°C” criterion for the hot rod is clearly met in both cases.

The quantity of water injected by MHSI and the accumulators until core recovery (end of accumulator injection) is 150 te for case 1 and 125 te for case 2.

The LHSI does not inject into the RCP [RCS] in the analysed time interval because the primary pressure remains too high (only the MHSI injects), and therefore it does not compensate for the decrease in the RCP [RCS] water inventory. However, the LHSI provides cooling of the IRWST water.

The main difference between the two cases is that for case 1, with the assumed loss of the first PSV, the initial PSV opening occurs a little later. In addition, when feed and bleed is initiated, the depressurisation rate is a little slower. The net result is that case 1 produces slightly higher fuel cladding temperatures.

The typical sequence of events for both cases with approximate timings is shown in Section 16.1.3.4 - Table 2. The list of plotted parameters is shown in Section 16.1.3.4 - Table 3. These are the significant parameters of interest for this event. The plots of these parameters are shown in Section 16.1.3.4 - Figures 1 and 2 for case 1, and Section 16.1.3.4 - Figures 3 and 4 for case 2.

The sequence of events shown in Section 16.1.3.4 - Table 2 indicates that the plant can remain without injection of feedwater for at least 2.5 hours before the feed and bleed actions must be initiated. Moreover, when the bleed is initiated (when RPV level reaches the bottom of the main coolant line), the peak cladding temperatures in the average rod remain below 600°C, which is within the acceptance limits.

3.4.5. Conclusion for EPR at 4500 MW

The consequences of increasing power from 4250 to 4500 MW on the most limiting case (case 1) were analysed for all phases and all events and compared with the acceptance criteria [Ref-1].

The events compared are:

- Loss of SG inventory
- The start of feed and bleed by the operator
- The start of cladding heat-up
- The speed of depressurisation, the delay of the RIS [SIS] injection and the duration of cladding heat-up
- The slope of cladding heat-up

The consequences of increasing the power are as follows:

- A decrease in the timing for the loss of inventory in SG and the time to open the first PSV valve.
- A decrease in the time required by the operator to initiate pressurised bleed (about 6% to compensate for the 6% increase in initial power). Since the operator has about 75 minutes to initiate the action, there is no effect on the accident mitigation.
- Considering a 6% decrease in the time available for operator action is conservative, since at the time of pressuriser bleed, the integral of the decay heat as a percent of initial power is lower for EPR_{4500} compared to EPR_{4250}.
- An earlier time for the start of cladding heat-up and a higher decay heat
- The time of cladding heat-up drops by about 6% from 10800 seconds to 10150 seconds for EPR_{4500} as compared to EPR_{4250}.
- The decay heat level is increased by about 2% between EPR_{4250} 10800 seconds after RT and EPR_{4500} 10100 seconds after RT.
- The decrease in the rate of depressurisation of the RCP [RCS], connected to the delay in the opening of the PSV, leads to a later RIS [SIS] injection and a lengthening of cladding heat-up time.
- The beginning of RIS [SIS] injection is connected to the RCP [RCS] pressure history which is proportional to the amount of vapour present in the primary.
- At 4250 MW, the energy removed through the opened PSV between 10300 and 11200 seconds (900 seconds) is 256 GJ. The integral of the decay heat during this time is 37 GJ; the energy used to depressurise the primary circuit during this time is 219 GJ. This energy corresponds to the vapour energy removed through the PSV, which produces the depressurisation.

- The pressure to lift the PSV and the shutoff head of the RIS [SIS] are identical at 4500 MW and at 4250 MW, the energy to depressurise the primary is the same at both powers - 219 GJ. However, the decay heat to be removed is higher, by 37 x 1.02 x 1.06 = 40 GJ. The energy removed through the PSV for the higher power EPR is 259 GJ.

- The increase in time needed to remove the additional energy is 900 x 259/256 =11 seconds.

- The timing of cladding heat-up is increased by the same amount from 400 to 411 seconds. for EPR_{4500}

- Increase in the cladding temperature for the average core.

- The maximum cladding temperature for the average core for EPR_{4500} is 500°C. For EPR_{4500} the increase in cladding heat-up is about 30°C (0.5°C/s x 411 s - 0.45°C/s x 400 seconds). The maximum cladding temperature for the average core is about 530°C. This guarantees that the criterion of cladding temperature less than 1200°C for the hot rod is met.

- The arguments presented above show that all the LOCA acceptance criteria are met for EPR_{4500}, provided the operator starts feed and bleed by starting RIS [SIS] and opening the PDS in the pressuriser when the level in the primary system reaches the bottom of the hot leg.

Impact of the new design of pressuriser safety valves

- The impact of the change to the characteristics of the PSV type will change the transient dynamics because of the higher hysteresis of 28 bar compared to 10 bar. The increased hysteresis will lead to the discharge of more water from the PSV and consequently the RCP [RCS] loop level criterion for feed and bleed will be reached earlier in the transient. However, based on the results presented in Section 16.1.3.4 – Figure 2, the modification will not significantly impact the calculated cladding temperature.

3.4.6. Conclusions

The final state is reached with the following conclusions:

- Sub-criticality is initially reached shortly after reactor trip from "mismatch reactor power / ARE [MFWS]-flow‘’. In the long term, sub-criticality is ensured by boration with RCV [CVCS] and RBS [EBS], which can be started after 30 minutes. Over the additional 2 hours available, prior to feed and bleed initiation, this boration is sufficient to avoid the return to criticality (RBS [EBS] alone can inject about 35 te of high borated water before bleed is initiated). Subsequent to feed and bleed initiation and depressurisation, MHSI, accumulator injection, and LHSI (if it injects) provide significant additional contribution to maintaining sub-criticality.

- The RCP [RCS] pressure does not exceed the response level of the PZR safety valves and thus remains far below acceptance limit of 130% of design pressure.

- The core heat up is far below the acceptance criteria of peak-clad temperature (hot rod) of 1200°C. The feed and bleed action at RPV level corresponding to the bottom of the main coolant loop guarantees sufficient heat removal so that the final state with the balance between bleed (PDS) and feed (MHSI, RCV [CVCS] pumps) and heat removal with LHSI/RRI [LHSI/CCWS] via IRWST is reached.

- The mass and energy release into the containment from PSV and PDS operations is significantly lower than the design basis heat loads for the containment. Hence, containment pressure build-up resulting from the PSV and PDS opening will be limited to values well below the design value of 5.5 bar.

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

The calculation results show, that for the RRC-A transient "Total Loss of Feedwater", the acceptance criteria are met and the required final state can be reached by means of the RRC-A feature “PZR bleed” even with consideration of additional failure of one MHSI pump.

3.4.7. System Sizing

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

SECTION 16.1.3.4 - TABLE 1

Initial Conditions: Total Loss of Feedwater in State A
(Failure of first PSV or 1 MHSI Pump)
EPR 4250 MW

Parameters	**Initial Values**
Reactor coolant system	
Reactor power Average RCP [RCS] temperature Reactor coolant pressure RPV coolant flow PZR water volume / level	100% of nominal power 313.6°C (nominal) 155 bar (nominal) 22135 kg/s (TH) 40 m^3/ 6.89 m (nominal)
Steam generators	
Steam pressure Initial SG level	78.0 bar (nominal) 15.69 m (nominal)
Feedwater	
Main feedwater flow ARE [MFWS] temperature	100% of nominal flow 230°C (nominal)

SECTION 16.1.3.4 - TABLE 2

Typical Sequence of Events: Total Loss of Feedwater in State A (Failure of first PSV or 1 MHSI pump)

Case 1: TLOFW with failure of first PSV
Case 2: TLOFW with failure of 1 MHSI pump

TIME (s)	EVENT
0	Loss of all operational FW (ARE[MFWS] and AAD [SSS])
5	Partial trip to 50% power from mismatch reactor power / FW flow
15	Reactor and turbine trip from lasting mismatch reactor power / FW flow
1300	SG water level drops below ASG [EFWS] actuation limit (40% of WR), but no ASG [EFWS] injection
2600	SG water level drops below 3 m (14% of WR), RCP [RCS] pumps are cut off and first slight increase of RCP [RCS] temperatures
4400	SG are completely emptied
6700 (case 1) 5700 (case 2)	First response of PZR safety valve
10300 (case 1) 9500 (case 2)	RPV level at bottom of main coolant line Opening of the PDS by operator (bleed)
11200 (case 1) 10100 (case 2)	Max. cladding temperature in average fuel rod of ≈ 500°C (case 1) and ≈ 430°C (case 2)
11400 (case 1) 10150 (case 2)	Beginning of accumulator injection (primary pressure < 45 bar)
11800 (case 1) 10600(case 2)	End of accumulators injection, the final state is reached, the flow through the PSVs is compensated only by MHSI

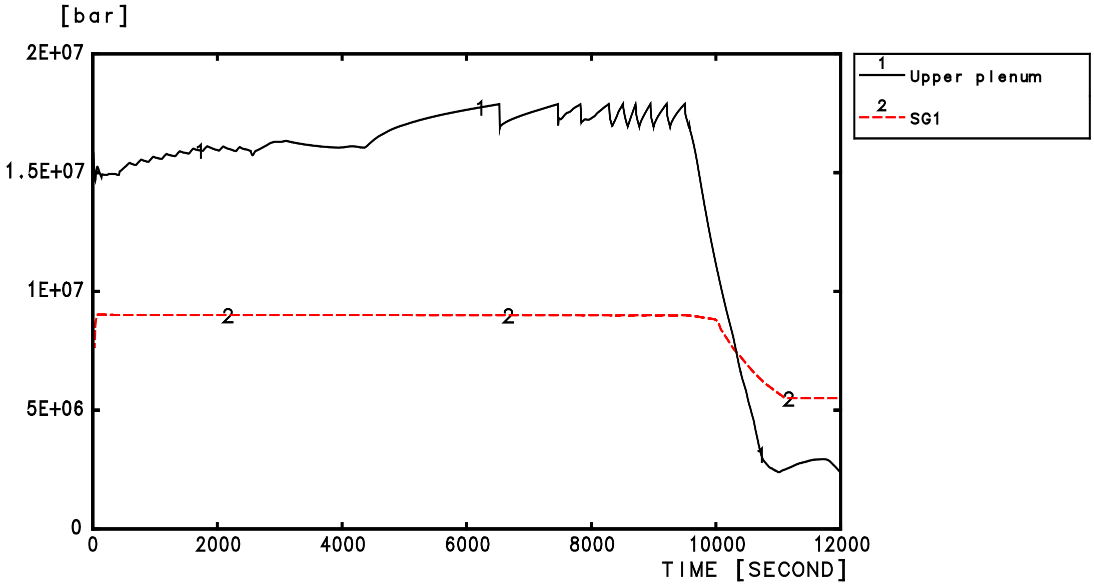
SECTION 16.1.3.4 - TABLE 3

List of Figures: Total Loss of Feedwater in State A (Failure of first PSV or 1 MHSI Pump)

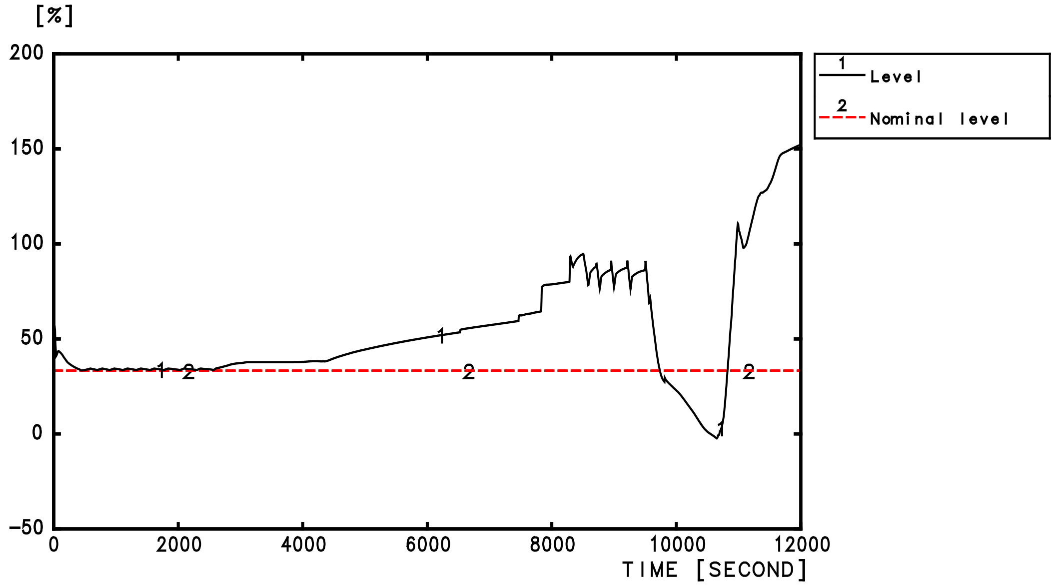
Figures 1 and 2 refer to case 1 with failure of first PSV
Figures 3 and 4 refer to case 2 with failure of 1 MHSI pump

SECTION 16.1.3.4 - FIGURE 1

Total Loss of Feedwater – Case 1: Failure of first PSV
Typical Primary and Secondary System Pressure
Pressuriser Level



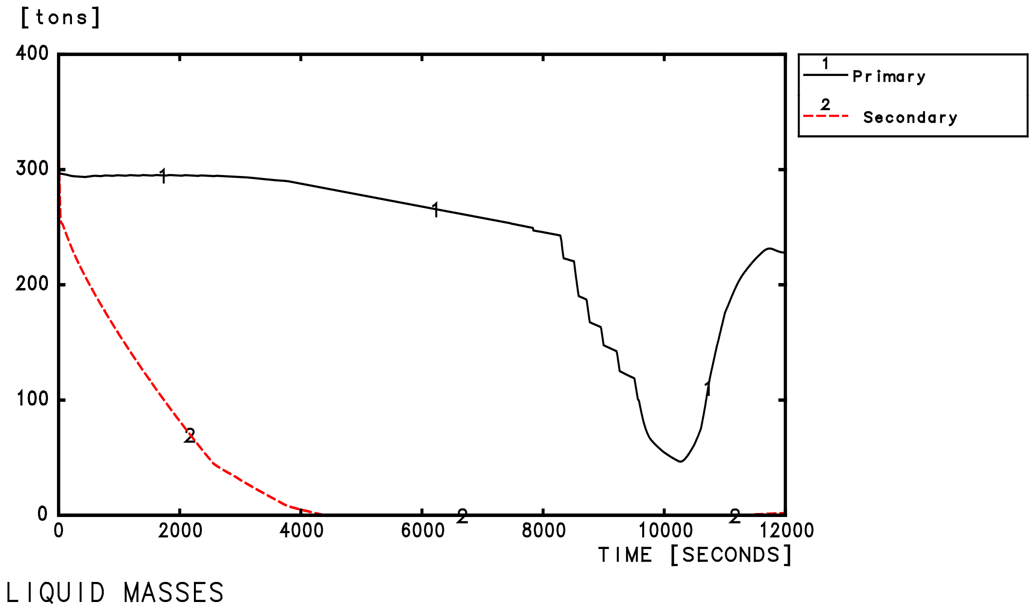
PRIMARY AND SECONDARY PRESSURES

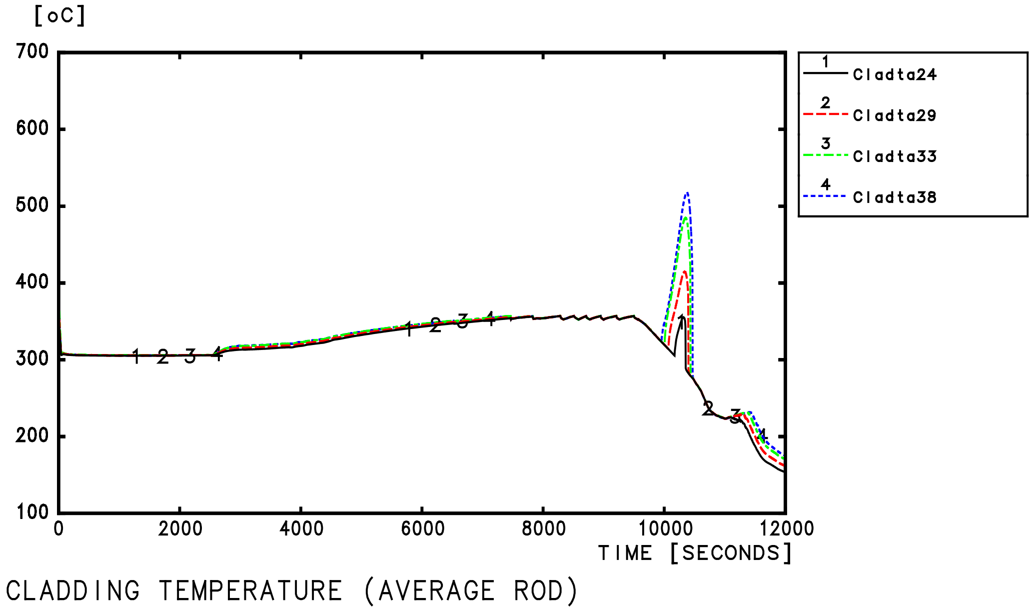


PRESSURIZER LEVEL

SECTION 16.1.3.4 - FIGURE 2

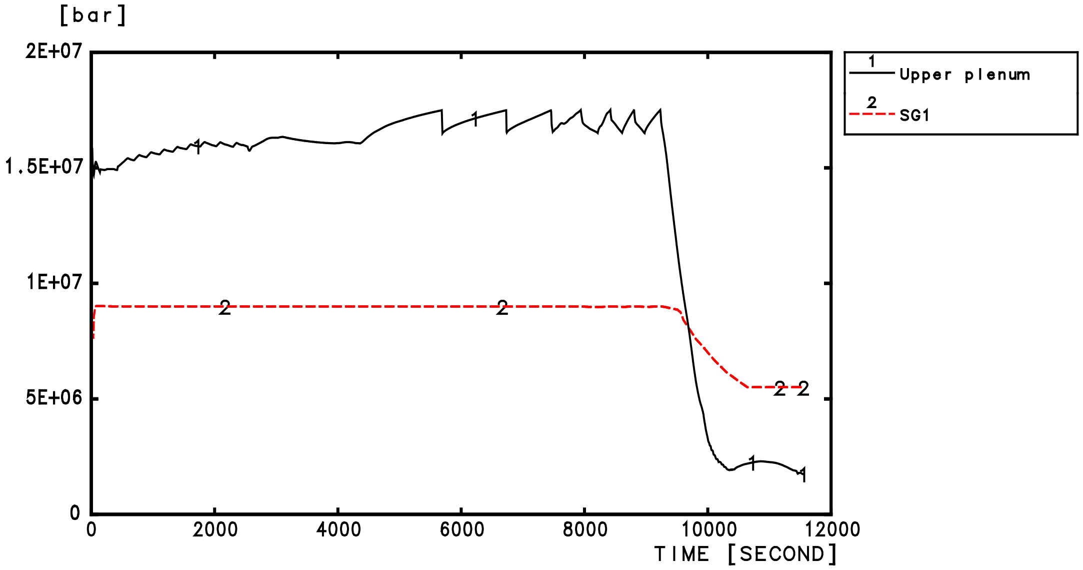
Total Loss of Feedwater – Case 1 EPR 4250: Failure of first PSV
Typical Primary and Secondary Mass Inventory
Cladding Temperatures in Upper Core



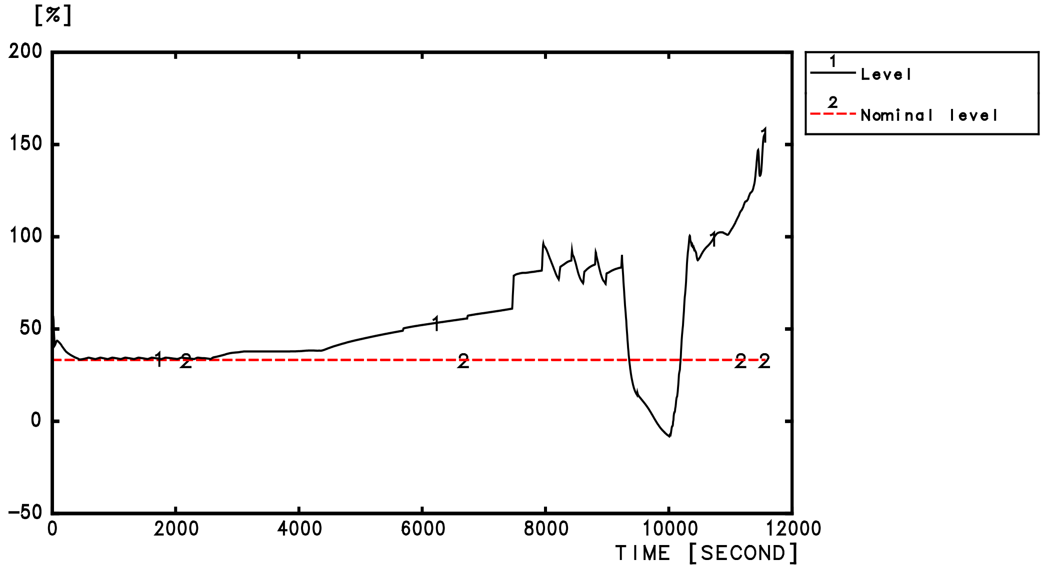


SECTION 16.1.3.4 - FIGURE 3

Total Loss of Feedwater – Case 2 EPR 4250: Failure of 1 MHSI
Typical Primary and Secondary System Pressure
Pressuriser Level



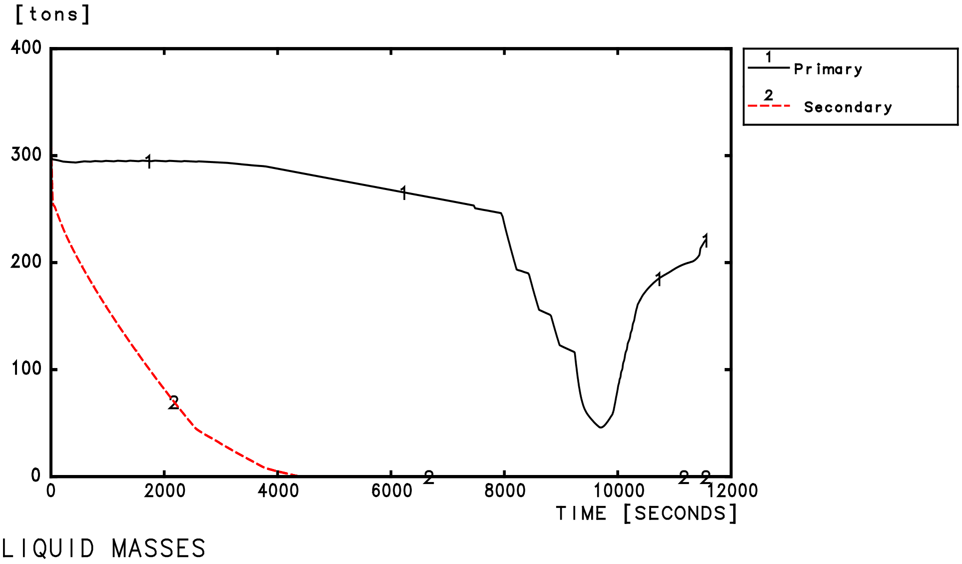
PRIMARY AND SECONDARY PRESSURES

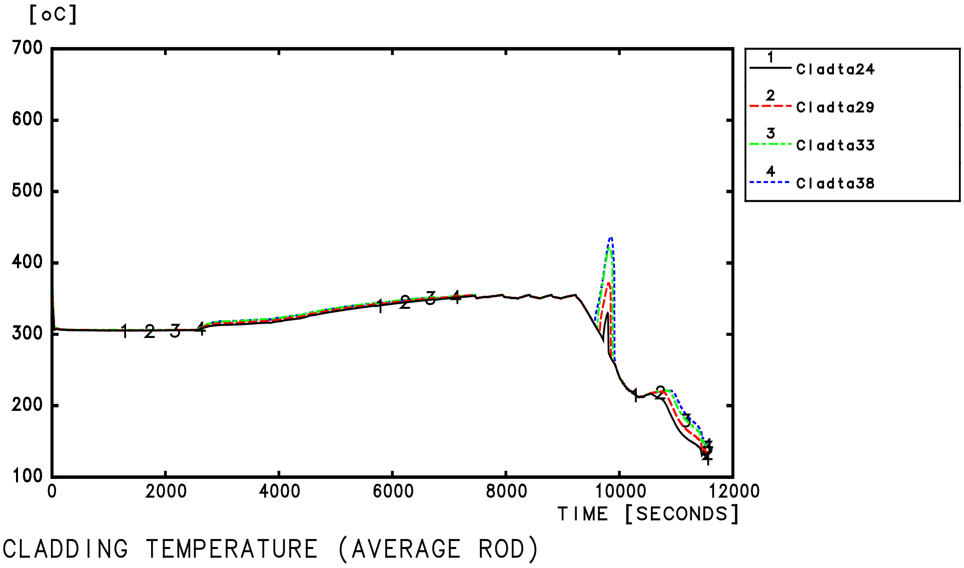


PRESSURIZER LEVEL

SECTION 16.1.3.4 - FIGURE 4

Total Loss of Feedwater – Case 2: Failure of 1 MHSI
Typical Primary and Secondary Mass Inventory
Cladding Temperatures in Upper Core





3.5. TOTAL LOSS OF COOLING CHAIN LEADING TO A LEAKAGE ON RCP [RCS] PUMPS SEALS (STATE A)

3.5.1. INTRODUCTION

This RRC-A event is equivalent to the total loss of the RRI/SEC [CCWS/ESWS] cooling chains (TLOCC). As a result, the following systems are lost:

- MHSI, RHR/LHSI (partly), PTR [FPCS] (partly), RCP [RCS] pumps thermal barriers, RCV [CVCS] and RCP [RCS] pumps seal injection.

3.5.2. Identification of Causes and Accident Description

Secondary side heat removal systems such as start-up/shutdown system AAD [SSS] or ASG [EFWS] and the VDA [MSRT] are not affected in this event. Consequently, the core heat removal is ensured by the secondary side as long as the RCP [RCS] inventory is sufficiently high that the primary coolant flow is maintained. Conservation of RCP [RCS] inventory can only be jeopardised if a loss of coolant occurs, in this case (TLOCC) due to the failure of RCP [RCS] pumps seals. The accident scenario and its mitigation are described below.

Following TLOCC, the accident is detected by the reactor protection system (RPR [PS]) via the failure of the RCP [RCS] pumps (due to loss of seal injection and thermal barrier) and the corresponding RT due to RCP [RCS] pump speed signal (less than 91%). The coast down of the RCP [RCS] pumps occurs within about 5 minutes and then natural circulation ensures decay heat removal. On the secondary side, the heat is removed through the GCT [MSB] or VDA [MSRT] at pressure levels above 90 bar and temperatures greater than 300°C. As a consequence of reaching a new steady-state condition at hot zero power, the primary temperature also remains greater than 300°C. It is possible that the RCP [RCS] pumps shaft seals as well as the standstill seals cannot resist the combination of high RCP [RCS] pressure (> 155 bar) and temperature (> 300°C).

Using conservative assumptions, the failure of all RCP [RCS] pumps seals with no closure of leak-off lines leads to a maximum break flow of about 28 kg/s per RCP [RCS] pump at the initial operating pressure level. Therefore, considering the failure of all four pumps, a total initial break flow of 112 kg/s is possible. This break flow is about half the value considered for bounding PCC-3 SB-LOCA analyses based on the bounding break size of 20 cm^2. The accident is initially comparable with another RRC-A scenario, i.e. SB-LOCA without MHSI (Chapter 14). Based on this scenario (practically double the break size compared to TLOCC) the following sequence of events is expected:

After about 5 minutes the occurrence of the RIS [SIS] signal (pressuriser pressure < 113.5 bar) which actuates secondary side partial cooldown to about 55 bar. Both the MHSI and LHSI pumps start operating but, with the exception of two LHSI pumps which have a diverse independent cooling chain (section 4.5.2 below), they will fail with time because of TLOCC.

Core heat-up occurs after about 2 to 2.5 hours and operator action is required to finally mitigate the accident and avoid core uncovery.

3.5.3. Accident Mitigation

The definitive mitigation consists of three actions [Ref-1]:

a) Manual initiation of fast cooldown via the four VDA [MSRT], i.e. full opening of all valves and depressurisation of the secondary side such that the RCP [RCS] pressure drops below the zero delivery head of the LHSI pumps of 20 bar. This manual fast cooling is required when there is an RIS [SIS] signal but no MHSI flow.

b) During the RCP [RCS] depressurisation, the accumulators automatically inject at a RCP [RCS] pressure of < 45 bar and thus the core heat-up, which is very limited, will stop. Because of the accumulator injection, sufficient boron is also injected into RCP [RCS] so that any potential for re-criticality is avoided.

c) Below a RCP [RCS] pressure of 20 bar, the long-term coolant make-up for the break flow is ensured by the operation of two LHSI pumps which are cooled via a diverse independent cooling chain, i.e. independent from the RRI [CCWS]. To compensate the break flow from all the four RCP [RCS] pumps at this low pressure level, only one LHSI pump is required.

The long-term heat removal is ensured by the SG via the main feedwater flow and GCT [MSB] which are not affected by TLOCC.

3.5.4. Conclusions

The two following design features safely mitigate the TLOCC accident in state A:

- Manual depressurisation of the RCP [RCS] by fast secondary side cooldown to < 20 bar. This operator action is already claimed for the mitigation of the SB-LOCA accident scenario following failure of the MHSI pumps.

- Diverse independent cooling chain for two LHSI pumps (in safety divisions 1 and 4) so that these two pumps cannot fail following TLOCC and are available for long-term break flow compensation and heat removal.

These two measures (in addition to the remaining automatic devices such as RT, accumulator injection, etc, as well as the complete availability of the secondary side heat removal capabilities) ensure attainment of the final state without any inadmissible core cooling degradation. With initiation of the secondary side fast cooldown at the latest possible time (2 hours after RT occurrence), there is no core heat-up. Furthermore, the positive contribution of a limited operation of the MHSI pumps despite TLOCC, which can last for about 15 minutes, is not taken into account for this assessment.

The containment pressure and IRWST temperature build-up during this accident scenario are not significant - the parameters will remain well below the relevant design or safety limits.

For informational purposes, this scenario may be compared to the RRC-A scenario SB-LOCA with failure of all the LHSI pumps. This latter scenario is bounding with respect to the break size/containment pressure build-up (20 cm^2, or about double that resulting from bounding failure of the RCP [RCS] pumps seals) and heat load for the IRWST (assuming no cooling in the first hours until actuation of the EVU [CHRS]). It shows that the peak containment pressures and the maximum increase of IRWST water temperature are approximately 2 bar and 90°C, respectively.

3.5.5. System Sizing

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

3.6. LOCA (BREAK SIZE UP TO 20 CM^2) WITH FAILURE OF THE PARTIAL COOLDOWN SIGNAL (STATE A)

3.6.1. Identification of causes and accident description

3.6.1.1. General

A small break LOCA with failure of the partial cooldown (PC) signal is classified as a RRC-A event. The break size is limited to 20 cm^2 (∅ 50 mm) for this category.

The main consequences of this accident are:

- Degradation of the core heat removal due to loss of reactor coolant through the break.
- Containment overpressure due to mass and energy release.
- Release of fission products into the environment due to post LOCA containment leaks.

This section deals only with the core heat transfer degradation due to a SB-LOCA with failure of PC. The consequences on the containment are dealt with in the section related to containment design (section 1 of Sub-chapter 6.2).

3.6.1.2. Typical sequence of events

The sequence of events for the first part of the transient is identical to a SB-LOCA with PC. The sequence of events for the second part of this transient is specific to this event, involving operator action which is necessary to mitigate the loss of PC.

First part of the sequence is identical to a typical SB-LOCA:

- The break results in a loss of reactor coolant inventory which is beyond the capability of the RCV [CVCS]. The loss of primary coolant results in a decrease in primary system pressure and pressuriser level.
- A reactor trip occurs on low PZR pressure (< MIN2). The RT signal automatically trips the turbine and closes the ARE [MFWS] full load control valves.
- As the secondary side pressure increases, the Main Steam Bypass (GCT [MSB]) valves open initiating steam dump to condenser.
- The SG are fed by the ARE [MFWS] system through the low load control valves. In case of unavailability of the ARE [MFWS] system, the start-up and shutdown feed pumps start and feed the SG through the low load control valves.
- The safety Injection signal is actuated on very low PZR pressure (< MIN3). The RIS [SIS] signal automatically starts the MHSI and LHSI pumps and initiates a partial cooldown of the secondary system. For SB-LOCA with PC, partial cooldown cools the primary system and lowers the RCP [RCS] pressure.

The second part of the sequence of events is specific to the failure of the PC signal:

- The SI signal which initiates the PC is assumed to fail. In this case the MHSI pumps are operable, i.e. they are ready to start once the SI signal has been actuated, but they cannot inject as RCP [RCS] pressure is too high.
- The failure of the PC signal results in neither GCT [MSB] valves nor the VDA [MSRT] performing the secondary side pressure reduction, although both are operable.

Therefore, the following sequence of events is required to limit the accident consequences:

- Given that the PC is not effective, there is no MHSI flow to compensate for the break flow and prevent continuous RCP [RCS] leakage. Therefore, the mitigation of the accident relies on the operator actions.
- The operator has more than 30 minutes grace time after RT to perform a manual cooldown without impairing core cooling. The preferred operator action consists in decreasing the setpoint of the four VDA [MSRT] to 60 bar in one step. The 60 bar target represents the normal pressure level after PC. The GCT [MSB] would also be available for this action, but following a too rapid secondary side depressurisation the header might be isolated automatically from the dP/dt signal.
- During the manual cooldown to 60 bar the RCP [RCS] pressure decreases sufficiently to allow MHSI injection into the cold legs.
- If MHSI is effective in recovering the core level, the further event sequence experiences the same phenomena and long-term actions as in PCC-3 SB-LOCA.

3.6.2. Safety criteria

With respect to core behaviour, the acceptance criteria of PCC-4 LOCA are used:

- The peak cladding temperature shall remain below 1200°C.
- The maximum cladding oxidation shall remain below 17% of the total cladding thickness.
- The maximum hydrogen generation shall remain below 1% of the amount generated if all the active part of the cladding were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- The long term cooling is ensured: the calculated core temperature is maintained at an acceptably low value and residual and decay heat is removed.

3.6.3. Methods and assumptions

3.6.3.1. Method

The CATHARE computer code (Version V1.3L) is used (Appendix 14A).

The transient "SB-LOCA with failure of the PC signal" does not introduce new events compared to those considered in the CATHARE code qualification basis for standard LOCA. This qualification basis includes the following "LOCA without MHSI" mock-up tests:

- BETHSY 9.1b ISP 27.
- BETHSY 6.2 TC.
- LSTF SB-CL-21.
- LOBI BL-34.

The CATHARE code, including the physical models and the RCP [RCS] modelling, is used as defined for the PCC LOCA analyses (refer to Sub-chapters 14.4 and 14.5) (i.e. the same level of conservatism is maintained with respect to the physics). For simplicity, however, a more realistic approach is also acceptable.

The CATHARE code provides a detailed representation of the primary and secondary systems. The simulation is based on a 4-loop model (which corresponds exactly to the real plant). The broken loop is in loop 4 with the break conservatively located in the lower part of the cold leg and the PZR is connected to the hot leg of loop 2.

The transient analysis is performed according to the RRC-A analysis rules (section 2 above).

3.6.3.2. Main assumptions

3.6.3.2.1. Accident definition

The case studied in this section is a 20 cm² (∅ 50 mm) break in a cold leg pump discharge pipe with failure of the PC signal.

The present analysis aims at assessing the effectiveness of a "late stepwise cooldown to 60 bar" in meeting the core cooling acceptance criteria, in conjunction with the actuation criterion defined in emergency procedures.

3.6.3.2.2. Protection and mitigation actions

The final state for RRC-A analysis is defined in section 2, for LOCA analysis as follows:

- The core is sub-critical.
- The core is reflooded.
- The decay and residual heat is removed.
- The break flow is compensated for by the RIS/RRA [SIS/RHRS] flow.
- The activity release is limited by the integrity of the barriers, in accordance with the acceptance criteria.

According to the rules defined for safety analyses (Sub-chapter 14.0), the following assumptions are considered:

- The final state is reached taking into account all available systems (safety classified and non-safety classified) provided they are not impaired by the event.
- The final state is reached without consideration of additional failures.
- The final state is reached without taking into account equipment maintenance.
- The Loss of Offsite Power is not coincident to the event.
- Best estimate assumptions are applied to the accident analysis. However, some conservative data and assumptions, for example system characteristics, are used to simplify the analysis or are used due to difficulties in defining best estimate data;
- No operator action is considered before 30 minutes after reactor trip.

3.6.3.2.3. Specific assumptions related to F1 systems

Reactor Trip (F1A):

RT signal is actuated on low PZR pressure (< MIN2: 133.5 bar).

The specific assumptions related to the resulting actions, considering conventional delays, are listed below:

- Beginning of rod insertion 1.2 seconds after RT signal.
- 3.5 seconds delay for complete rod insertion.
- Best estimate decay heat curve (Sub-chapter 14.1).
- Total main feed water flow reduction to low load head capacity 1.2 seconds after RT signal.
- Turbine Trip 1.2 seconds after RT signal.

Safety Injection (F1A):

RIS [SIS] signal is actuated on very low PZR pressure (< MIN3:113.5 bar).

The specific assumptions, considering penalising delays, are listed below:

- 15.9 seconds delay for RIS/RRA [SIS/RHRS] pump start-up, this delay includes the pump starting time.
- Minimal characteristic for RIS/RRA [SIS/RHRS] pumps (Sub-chapter 14.1).
- 50°C for injection flow temperature corresponding to an initial IRWST temperature of 50°C. This temperature is considered constant during the transient because of the slow increase of IRWST temperature. Otherwise, this temperature increase is limited by IRWST cooling ensured by the four LHSI trains in operation on pump mini-flow configuration.

The specific assumptions related to the accumulators are as follows:

- 47 m^3 total volume.
- 35 m^3 water volume.
- 45 bar abs initial pressure.
- 2500 m^{-4} discharge line resistance.
- 50°C water temperature.

The following RIS/RRA [SIS/RHRS] modelling is adopted:

- Both the three LHSI/RHR trains and the three accumulators inject into the cold legs of the intact loops 1, 2 and 3.
- The LHSI/RHR train related to the broken loop is considered to fully discharge into the containment, with no contribution to RCP [RCS] injection.

<u>RBS [EBS] (F1A):</u>

The extra borating system is not credited for boration.

<u>VDA [MSRT] (F1A):</u>

The VDA [MSRT] capacity is the minimum value (see Sub-chapter 14.1) (50% nominal SG steam flow). VDA [MSRT] setpoints are nominal values.

The automatic setpoints are as follows:

- 95.5 bar before beginning of partial cooldown.
- 60 bar after the end of partial cooldown.

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

The GCT [MSB] is credited (see below), with the consequence that there is no response of the VDA [MSRT] until actuation of "stepwise cooldown to 60 bar" which is performed by the four VDA [MSRT].

<u>ASG [EFWS] (F1A):</u>

ASG [EFWS] is not actuated because of ARE [MFWS] system operation (see specific assumptions on other systems).

<u>RCP [RCS] pumps TRIP (F1A):</u>

Normally an automatic RCP [RCS] pump trip follows a LOCA, occurring on low-pressure drop across the RCP [RCS] pumps (< 80% nominal RCP [RCS] pumps pressure drop).

In the present analysis, however, it is assumed that the RCP [RCS] pumps are tripped on reactor trip, which is the most conservative assumption (compared to delayed RCP [RCS] pump trip between reactor trip and RCP [RCS] pump trip signal).

3.6.3.2.4. Specific assumptions for other systems

GCT [MSB]:

The GCT [MSB] capacity is the minimum value (see Sub-chapter 14.1) (50% nominal SG steam flow). GCT [MSB] setpoints are nominal values.

The automatic setpoints are as follows:

- 90 bar before beginning of partial cooldown.
- 55 bar after the end of partial cooldown.

The GCT [MSB] is available since the main steam header is open.

ARE [MFWS]:

ARE [MFWS] supply is available with the following assumptions:

- After reactor trip flow rate reduction to maximum of 20% in 10 seconds.
- Control of nominal SG level by the low load train.
- Decrease of ARE [MFWS] temperature at SG inlet from 230°C to 190°C.

PZR pressure control:

The PZR pressure control system (PZR heaters) is modelled because its operation delays the RT signal. A total heating power of 2592 kW is considered until the PZR is completely empty.

RCV [CVCS]:

The potentially positive contribution of the RCV [CVCS] in terms of leak compensation is not considered.

3.6.3.3. Operator action

The operator action consists in the actuation of the "stepwise cooldown to 60 bar", modelled in the calculation by the reduction of VDA [MSRT] setpoints from 95.5 bar to 60 bar.

The decoupled conservative criterion considered in the CATHARE calculation is:

- Core outlet temperature (referring to the average channel) > 350°C and no implementation of PC.

3.6.3.4. Reactivity balance

No reactivity calculation is performed in the CATHARE calculation. The core sub-criticality is ensured all over the transient after RT occurrence, supported by the following:

- At the time of "stepwise cooldown to 60 bar" actuation, the void fraction in the core is high.

- Following actuation of the cooldown to 60 bar, the void fraction of the coolant increases due to the fast RCP [RCS] depressurisation rate.
- However even without voiding or any boration, the core design excludes criticality at RCP [RCS] temperatures corresponding to a secondary pressure of 60 bar even at End of Cycle conditions.

3.6.3.5. Thermal-hydraulic Analysis

3.6.3.5.1. Initial conditions

The initial state conditions, given in Section 16.1.3.6 - Table 1 are related to best estimate conditions at 100% at nominal power.

The axial power shape for the average rod in the average assembly is given in Section 16.1.3.6 - Figure 1. This power shape, skewed at the top of the core, has the following characteristics:

- An initial linear power of 154.3 W/cm at 100% power
- An enthalpy rise factor of 1.
- A peaking factor of 1.57 at 3.5 m of active core height.
- An axial offset of 15%.

This power shape is chosen as it provides a reasonably conservative distribution of power versus core height with the power distribution skewed to the upper part of the core. It is limiting for SB-LOCA analysis because of core uncovery process: as the core uncovers, the cladding in the upper part of the core heats up and 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 pellet temperature is the same as for PCC analysis. This temperature is averaged over a section of fuel and is given as a boundary condition limit. This initial temperature has a negligible impact on the maximum cladding temperature reached because the core heats up long after RT in a typical SB-LOCA sequence and depends on decay heat only

3.6.3.5.2. Results [Ref-1]

The sequence of events is given in Section 16.1.3.6 - Table 2.

The most representative parameters are presented in the following figures:

- Figure 2: Reactor and SG power. RCP [RCS] and SG secondary side pressures.
- Figure 3: Break mass flow rate (steam, liquid and total). Break flow velocities (steam and liquid).
- Figure 4: Swell levels in RPV. Core void fraction.
- Figure 5: Coolant liquid and steam temperatures in RPV and core. Cladding temperatures.

- Figure 6: Integrated flow rates of safety injection. Coolant inventory on primary and secondary side.

- Figure 7: Total safety injection and break flow rate. Integrated safety injection and break flow rates.

The detailed explanation of the parameters is given in Section 16.1.3.6 - Table 3.

The accident was not recalculated for EPR_{4500}. Achieving the final state and satisfying the associated acceptance criteria is deduced from the results obtained from the analysis of this transient for EPR_{4250}. It can be shown that the acceptance criteria are met with enough margins to accommodate the increase in power without additional analyses [Ref-2].

3.6.3.5.2.1. Accident analysis for EPR at 4250MW

- The operator action, consisting of a secondary side "stepwise cooldown to 60 bar" via the four VDA [MSRT], at about 3400 seconds (≈ 1 hour), is sufficient to reach the final state:

- The operator action is initiated when core outlet temperature (average channel) reaches 350°C, with no implementation of PC.

- Almost immediately after the start of cooldown, MHSI becomes effective.

- The core heat-up stops some 50 seconds later (all time delays counted from the time of operator action).

- The peak cladding temperature in the average fuel rod is less than 400°C, reached at time of operator action.

- Thus without any dedicated hot fuel rod analysis it is obvious that high margins exist with respect to the decoupling criterion of 1200°C also in the hot rod.

Neither the accumulators nor the LHSI are necessary to balance the break flow.

3.6.3.5.2.2. Accident analysis for EPR at 4500 MW

The break flow depends on the primary pressure and the break quality. For this break size (20 cm^2), the SG are needed for decay heat removal. After the initial depressurisation (≈500 seconds), the primary pressure stabilises slightly below the secondary pressure. Until about 2800 seconds, the break flow is mostly liquid and the break quality depends mainly on the initial primary inventory. This phase of the transient is almost identical for EPR_{4250} and EPR_{4500}.

From 2800 seconds and before operator action, the break flow is mostly steam. Because the initial power is greater (by 5.9%), the steam flow from the core is slightly more important for EPR_{4500} than EPR_{4250}. The beginning of core heat-up occurs earlier as does exceedence of the 350°C core exit temperature criteria. The delay between the steam break flow (2800 seconds) and the exceedence of the 350°C core exit temperature criteria (3400 seconds) is 600 seconds for EPR_{4250}. For EPR_{4500} this time is reduced. The reduction is proportional to the steam production rate, which depends on the initial power and the residual power. The initial power is higher by 5.9%. The residual power is lower by 0.8% (because of lower initial temperature in the RCS for the higher power reactor). The point at which the 350°C exit core temperature criteria is reached is about 30 seconds earlier (≈ 600 x 5.1%) for EPR_{4500} than the EPR_{4250}.

Due to a higher initial power (5.9%) and an Fq higher by 3.2% for EPR 4500, the cladding temperature is higher compared to EPR_{4250}. The increase in cladding temperature for EPR_{4250} is 72°C (Section 16.1.3.6 - Figure 5). For EPR_{4500} the increase in cladding temperature will be higher by 8.4% (5.9 + 3.2 - 0.8 = 8.4%). The cladding temperature will reach a maximum of 387°C (≈380 + 72 x 8.4%) for EPR_{4500}.

After operator action the MHSI becomes immediately effective, arresting the core heat-up and restoring the primary inventory as for EPR_{4250}.

The arguments presented above assure that all LOCA acceptance criteria are met as well as reaching the final state for EPR_{4500}. The operator action, which consists of performing a secondary side "stepwise cooldown to 60 bar" via the four VDA [MSRT] is initiated when the core outlet temperature reaches 350°C and when there is a signal that the partial cooldown is not effective. The core outlet temperature of 350°C is reached at 3370 seconds (≈ 57 minutes), which gives enough time for operator action.

3.6.4. Conclusions

The following conclusions can be drawn concerning SB-LOCA with failure of PC:

- The peak cladding temperature remains below the acceptance criterion (1200°C).
- The maximum percentage of total cladding thickness oxidised at the hot spot is less than the limit value (17%)[6].
- There is no cladding rupture.
- Integrity of the core geometry is maintained.
- Long term cooling is ensured.

All the LOCA acceptance criteria are met.

The residual and decay heat are effectively removed. A stable final state is reached. The water inventory is maintained with at least one LHSI pump, while the RCP [RCS] cooldown is ensured either by the SG or by at least one other LHSI pump in RHR mode.

After MHSI has become effective, the sequence of events to RHR conditions is similar to PCC-3 LOCA and involves similar mitigation procedures.

3.6.5. System Sizing

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

[6] The calculation value is less than 1%.

SECTION 16.1.3.6 - TABLE 1

Initial Conditions
20 cm² (∅ 50 mm) CL Break with Failure of PC, State A at 4250 MW

Parameters	**Values**
Reactor coolant system	
Initial reactor power Initial average RCP [RCS] temperature Initial reactor coolant pressure RPV coolant flow PZR water volume / level	100% of nominal power 313.6°C (nominal) 155 bar (nominal) 22135 kg/s (thermo hydraulic flow rate) 40 m^3 / 6.89 m (nominal)
Steam generators	
Initial steam pressure Initial SG level	78.0 bar (nominal) 15.69 m (nominal)
Feedwater	
Main feedwater flow Initial ARE [MFWS] temperature	100% of nominal flow 230°C (nominal)

SECTION 16.1.3.6 - TABLE 2

Sequence of Events (Global Time Values)
Typical Results for 20 cm² (∅ 50 mm) Cold Leg Break
with Failure of PC, State A at 4250 MW

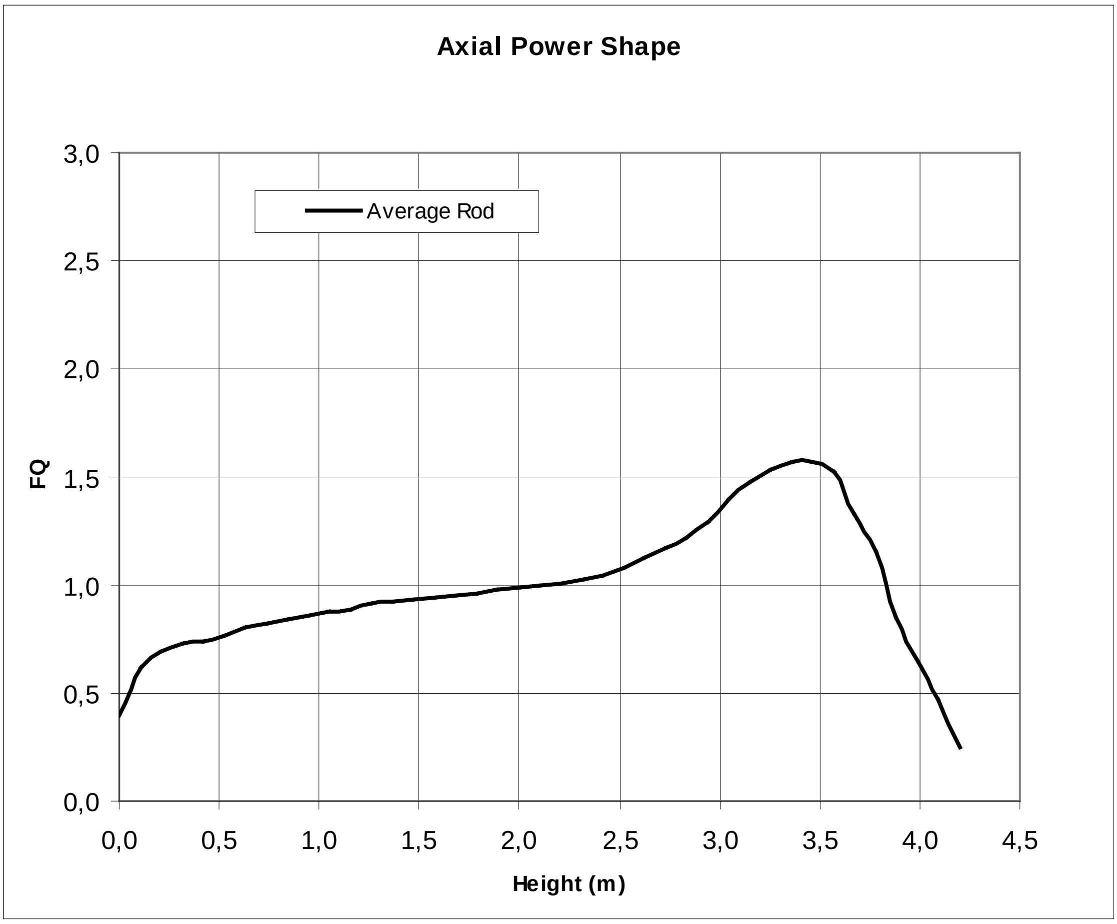
TIME (s)	EVENT
0	Break opening
100	Reactor, turbine and RCP [RCS] pumps trip; ARE [MFWS] reduction to low load on PZR pressure < MIN2 (133.5 bar), sec. pressure is controlled at 90 bar by GCT [MSB]
150	Safety Injection signal on PZR pressure < MIN3 (113.5 bar), but failure of Partial Cooldown, 15 s later starting of MHSI and LHSI pumps
500	RCP [RCS] pressure stabilises somewhat above secondary pressure of 90 bar
3250	Beginning of core heat-up
3400	Operator action: beginning of "fast cooldown" via the 4 VDA [MSRT] to 60 bar core outlet temperature referring to average channel = 350°C peak cladding temperature in average rod < 400°C
3400	Beginning of MHSI injection
3450	End of core heat-up, RCP [RCS] minimum inventory
3600	Secondary pressure at 60 bar, RCP [RCS] pressure falls below secondary pressure
4700	Beginning of accumulator injection (RCP [RCS] pressure < 45 bar)
5000	End of calculation

SECTION 16.1.3.6 - TABLE 3

List of Figures
Typical Results for 20 cm^2 (Ø 50 mm) Cold Leg Break
with Failure of PC, State A

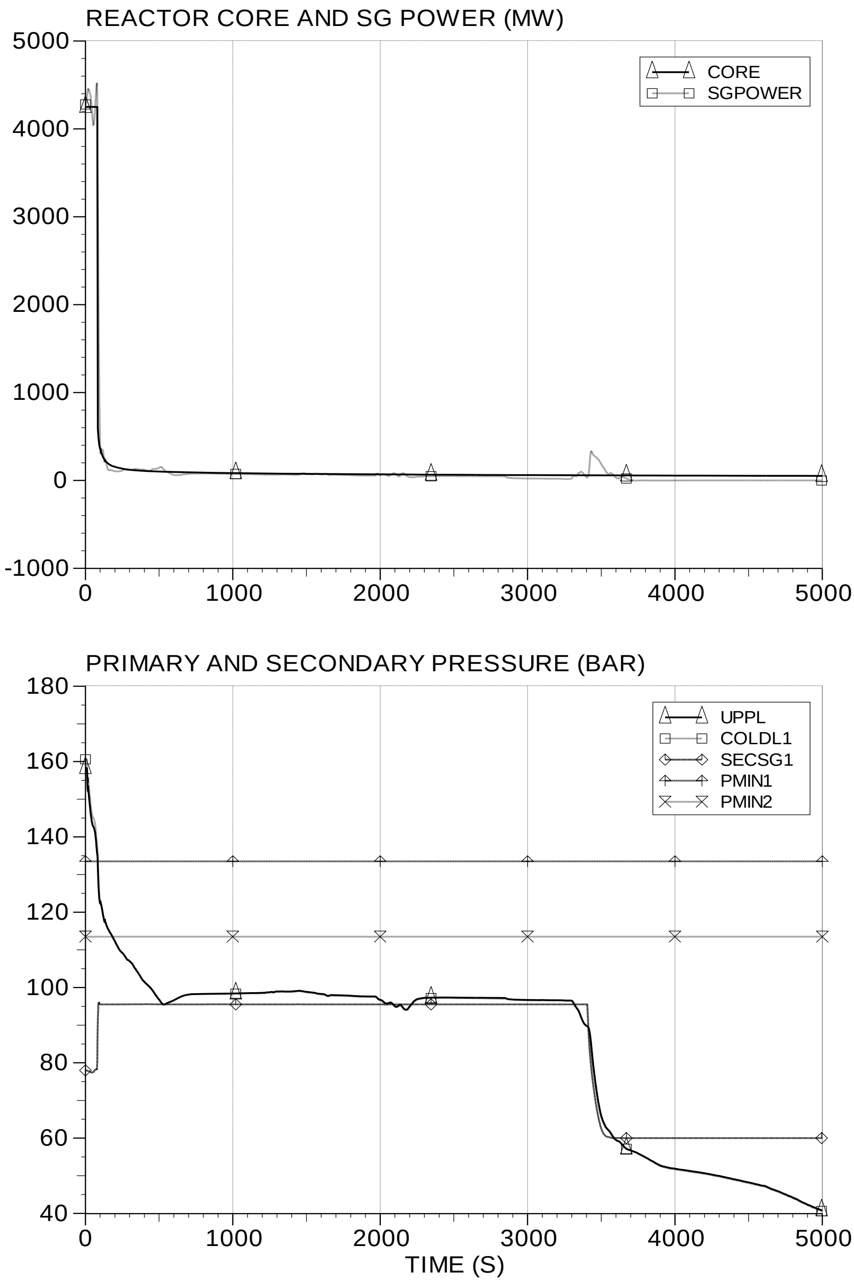
SECTION 16.1.3.6 - FIGURE 1

RRC-A Analysis
Typical Axial Power Shape of Average Rod



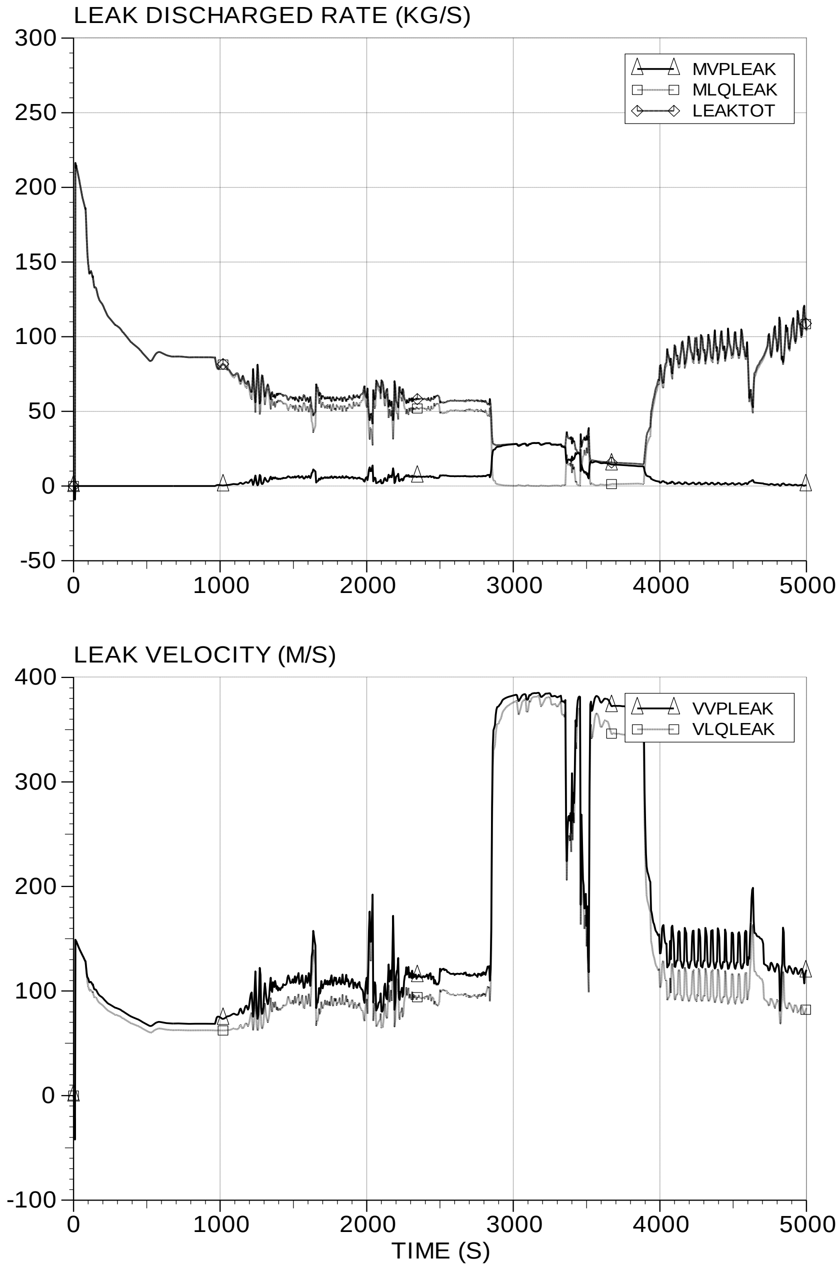
SECTION 16.1.3.6 - FIGURE 2

Typical Results for 20 cm^2 (Ø 50 mm) Cold Leg Break with Loss of PC, State A Core Power and Total Heat Exchange in Steam Generator Primary and Secondary System Pressure



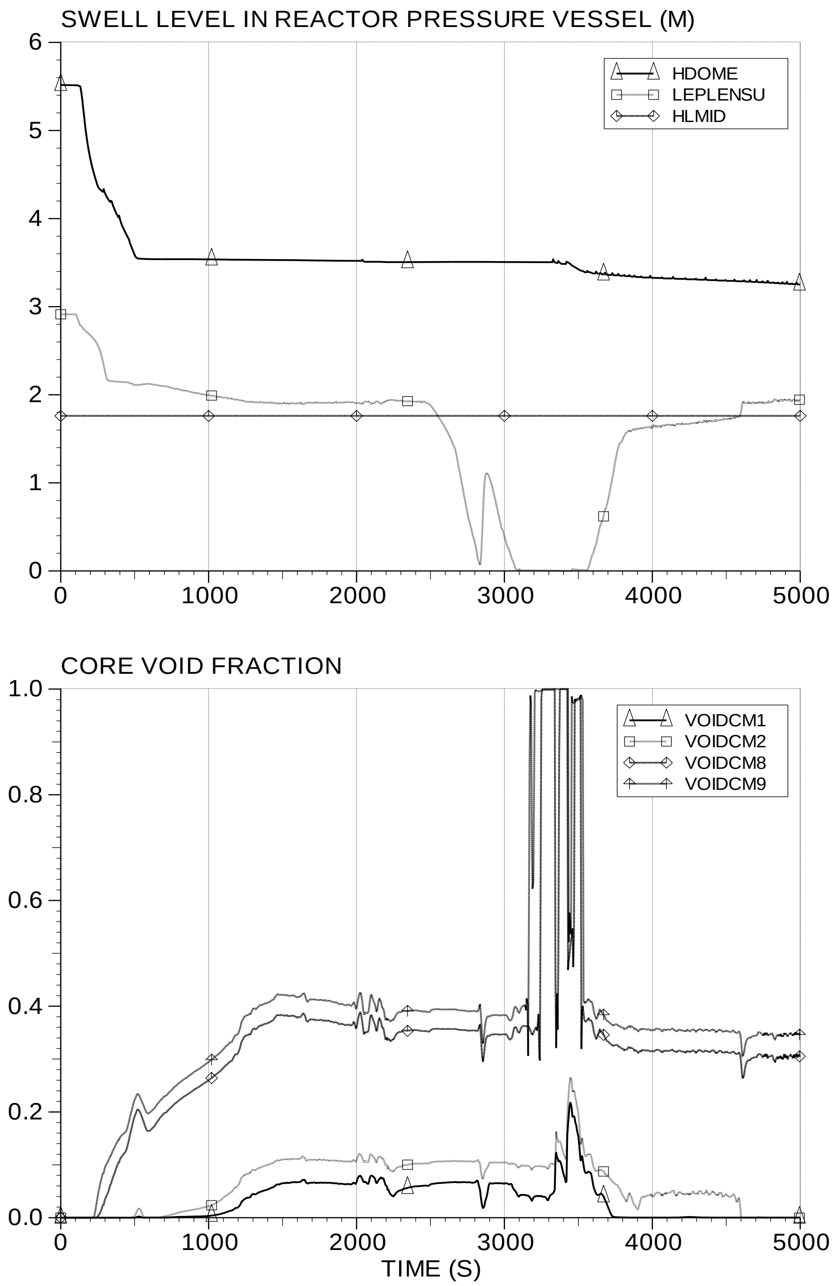
SECTION 16.1.3.6 - FIGURE 3

Typical Results for 20 cm^2 (Ø 50 mm) Cold Leg Break with Loss of PC, State A
Vapour and Liquid Mass Flow at the Leak
Vapour and Liquid Velocity at the Leak



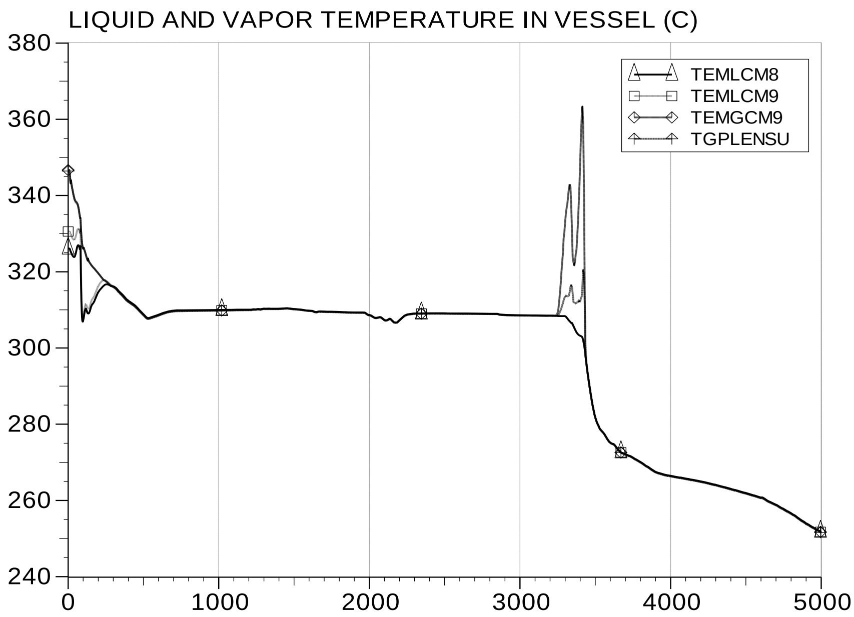
SECTION 16.1.3.6 - FIGURE 4

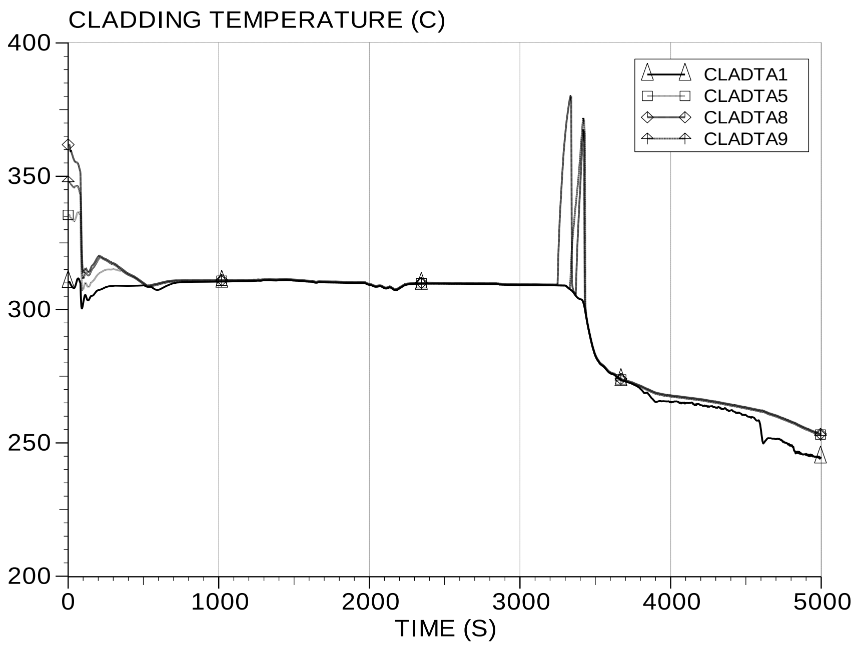
Typical Results for 20 cm² (Ø 50 mm) Cold Leg Break with Loss of PC, State A Swell Level in Reactor Pressure Vessel - Void Fraction in the Core



SECTION 16.1.3.6 - FIGURE 5

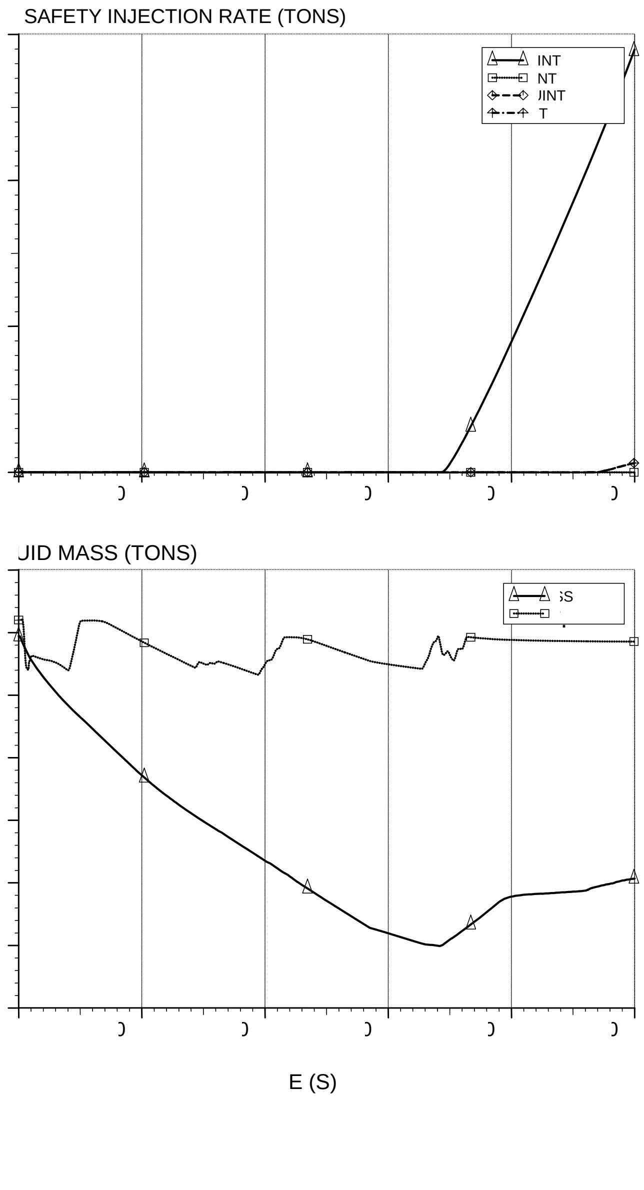
Typical Results for 20 cm^2 (Ø 50 mm) Cold Leg Break with Loss of PC, State A
Liquid and Vapour Temperature in Reactor Pressure Vessel
Cladding Temperature of the Average Rods





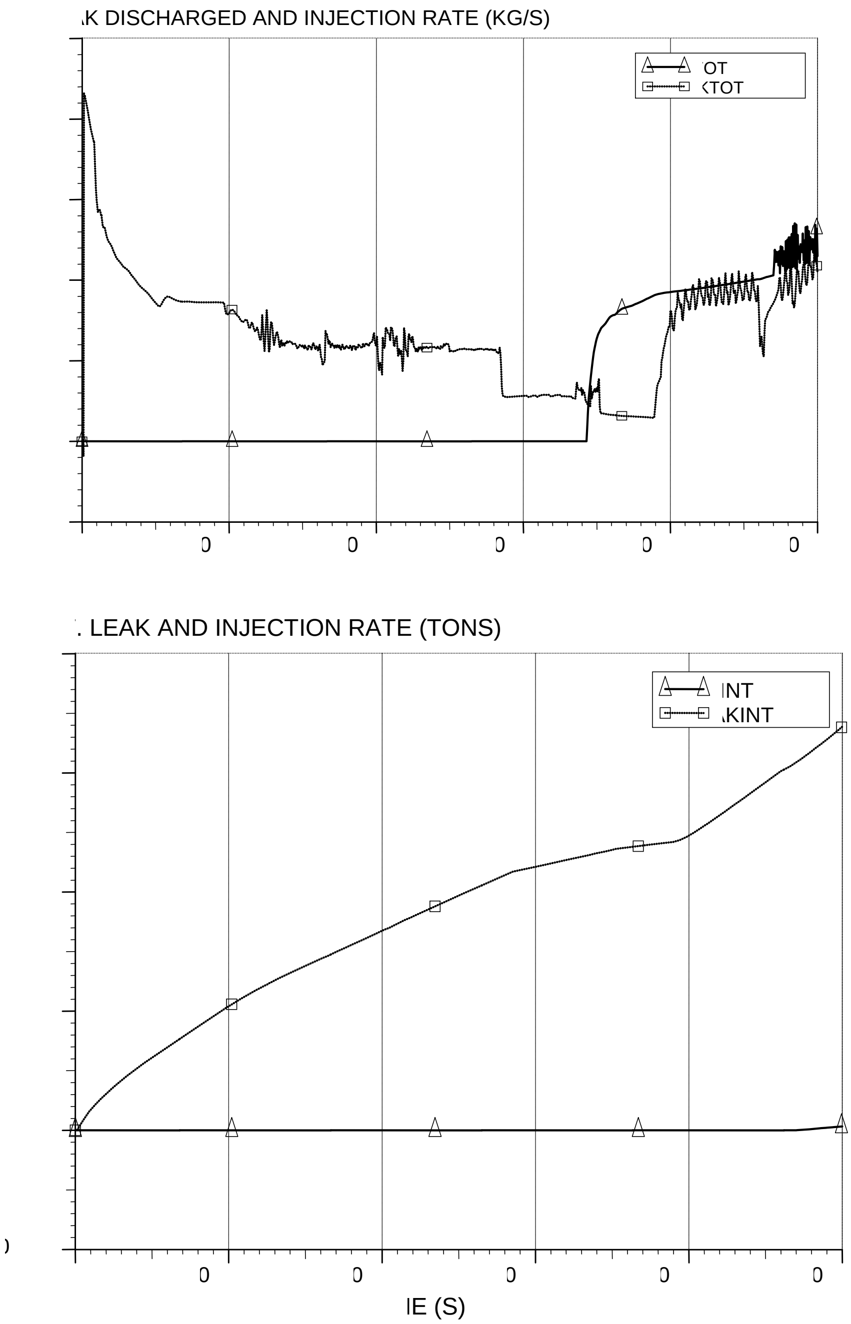
SECTION 16.1.3.6 - FIGURE 6

Typical Results for 20 cm² (Ø 50 mm) Cold Leg Break with Loss of PC, State A
Integral Safety Injection Rate
Water Inventory in the Primary and Secondary System



SECTION 16.1.3.6 - FIGURE 7

Typical Results for 20 cm² (Ø 50 mm) Cold Leg Break with Loss of PC, State A
Safety Injection and Leak Discharge Rate
Integral Safety Injection and Leak Discharge Rate



3.7. LOCA (BREAK SIZE UP TO 20 CM^2) WITHOUT MHSI (STATE A)

3.7.1. Identification of causes and accident description

3.7.1.1. General

A small break LOCA with loss of MHSI is classified as a RRC-A event. The break size is limited to 20 cm^2 (∅ 50 mm) for this category.

The main consequences of this accident are:

- Core heat transfer degradation due to loss of reactor coolant through the break.
- Containment overpressure due to mass and energy release.
- Radioactivity releases of fission products into the containment and subsequently into the environment by leaks in containment.

This section describes only the core heat-up aspect of a SB-LOCA with loss of MHSI. The consequences on the containment are described in Chapter 6 related to containment design (section 1 of Sub-chapter 6.2).

3.7.1.2. Typical sequence of events

The first part of the sequence is identical to the SB-LOCA. The second part of the sequence is specific to this event, involving the operator action required to mitigate the accident.

The first part of the sequence, identical to a typical SB-LOCA, is:

- 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 PZR pressure (< MIN2). The RT signal automatically trips the turbine and closes the ARE [MFWS] full load control valves.
- As the secondary side pressure increases, the Main Steam Bypass (GCT [MSB]) valves open, allowing steam dump to the condenser.
- The SG are fed by the ARE [MFWS] system through the low load control valves. Following unavailability of the ARE [MFWS] system, the start-up and shutdown pump starts and feeds the SG through the low load control valves.
- Safety Injection signal is actuated on very low PZR pressure (< MIN3). The RIS [SIS] 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.

The second part of the sequence is specific to the loss of MHSI:

There is a common cause failure of all MHSI pumps to operate.

- At the end of the partial cooldown, the RCP [RCS] pressure is about equal to the secondary side (about 60 bar because of RCP [RCS] saturation) which is too high to enable injection by accumulator or LHSI (MHSI not being available). Therefore, the mitigation of the accident relies on operator action.

- In this case, the operator strategy consists of decreasing the secondary side pressure to reach the accumulator injection pressure and then the LHSI shutoff head by means of the VDA [MSRT]. The GCT [MSB] is not used because of main steam header isolation on SG pressure drop signal or SG pressure < MIN1.

- The criterion for initiating fast cooldown of the secondary side is defined conservatively as:

    - Core outlet temperature Tco > 350°C and no MHSI.

- Since the RCP [RCS] water inventory can be low when the criterion is reached, the cooldown is fast to allow the rapid injection by the accumulators: this operation is called "fast cooldown".

- The fast cooldown consists of adjusting the setpoint of the VDA [MSRT] to a value lower than the LHSI delivery pressure, leading to full opening of the four VDA [MSRT].

- As soon as the fast cooldown is actuated by the operator, the RCP [RCS] pressure decreases quickly which allows injection by the accumulators and then by LHSI. The accumulator injection quickly re-floods the core and limits the fuel clad temperature increase. The long-term core cooling is ensured by LHSI.

- As the break cannot remove the decay heat, the RCP [RCS] pressure is governed by the secondary side at a level compatible with LHSI injection and core cooling.

- The final state is ensured by control of RCP [RCS] inventory via at least one LHSI pump and control of RCP [RCS] temperature via the SG (VDA [MSRT], or GCT [MSB] if available, and ASG [EFWS]) or via at least one other LHSI/RHR pump. The transfer to the LHSI/RHR operating mode is possible as soon as the conditions inside the RCP [RCS] loops allow the operation of the LHSI pumps in RIS/RRA [SIS/RHRS] mode:

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

    - RCP [RCS] hot leg temperature < 180°C,

    - ΔTsat and RPVL consistent with LHSI/RHR suction from the hot leg.

3.7.2. Safety criteria

For this sequence, it must be demonstrated that the acceptance criteria for the LOCA in PCC-4 (see Sub-chapter 14.2) are met:

- The peak cladding temperature shall remain lower than 1200°C.

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

- the maximum hydrogen generation, produced by the chemical reaction of the cladding with water or steam, must not exceed 1% of the hydrogen that would be produced if all the cladding materials had reacted (with exclusion of the expansion volume cladding),
- The core geometry shall remain coolable: calculated changes in core geometry shall be such that the core remains coolable.
- The long term cooling is ensured: the calculated core temperature is maintained at an acceptably low value and decay heat is removed.

3.7.3. Methods and assumptions

3.7.3.1. Method

The CATHARE computer code (Version V1.3L) is used (Appendix 14A).

The transient "small break LOCA without MHSI" does not introduce new dominant events compared to those considered in the CATHARE code qualification basis for standard LOCA. This qualification basis includes the following "LOCA without MHSI" mock-up tests:

- BETHSY 9.1b ISP 27.
- BETHSY 6.2 TC.
- LSTF SB-CL-21.
- LOBI BL-34.

The CATHARE code, including the physical models and the RCP [RCS] modelling, is used as defined for the PCC LOCA analyses (see Sub-chapters 14.4 and 14.5) i.e. the same level of conservatism is maintained. This modelling is used for simplicity; a more realistic approach is also acceptable.

The CATHARE code provides a detailed representation of the primary and secondary systems. The simulation is based on a 4-loop model, and total similarity between actual plant and the plant model. The broken loop is in loop 4 with the break conservatively located in the lower part of the cold leg and the PZR is connected to the hot leg of loop 2.

The transient analysis is performed according to the RRC-A analysis rules (section 2).

3.7.3.2. Main assumptions

3.7.3.2.1. Accident definition

The case studied in this section is a 20 cm² (∅ 50 mm) break in a cold leg pump discharge pipe with total loss of MHSI.

The analysis aims at assessing the efficiency of the "fast cooldown" to meet the core cooling acceptance criteria. The “fast cooldown” is actuated based on the criterion described above and anticipated in the emergency guidelines.

3.7.3.2.2. Protection and mitigation actions

The final state for RRC-A analysis is defined in section 2, for LOCA analysis as follows:

- The core is sub-critical.
- The core is re-flooded.
- The decay heat is removed.
- The break flow is compensated for by the RRA [RHRS] flow.
- The activity release is controlled by maintaining the integrity of the barriers in accordance with the acceptance criteria.

According to the rules defined for safety analyses (Sub-chapter 14.0), the following assumptions are considered:

- The final state is reached assuming availability of all the systems (safety classified and non-safety classified) provided they are not affected by the event.
- The final state is reached without taking into account additional failures.
- The final state is reached without taking into account maintenance on equipment.
- The loss of offsite power supply is not coincident with the event.
- Best estimate assumptions are used for the accident analysis; however, some conservative data or assumptions, for example system characteristics, are used to simplify the analysis or because it is difficult to define best estimate data,
- No operator action is considered before 30 minutes after reactor trip. The operator action is initiated on information of "core outlet temperature above 350°C" and "no MHSI".

3.7.3.2.3. Specific assumptions related to F1 systems

<u>Reactor Trip (F1A):</u>

RT signal is actuated on low PZR pressure (< MIN2: 133.5 bar).

The specific assumptions considering conservative delays are listed below:

- Beginning of rod insertion 1.2 seconds after RT signal.
- 3.5 seconds delay for complete rod insertion.
- Best estimate decay heat curve (Sub-chapter 14.1).
- Total main feed water flow reduction to low load head capacity 1.2 seconds after RT signal.
- Turbine Trip 1.2 seconds after RT signal

Safety Injection (F1A):

RIS [SIS] signal is actuated on very low PZR pressure (< MIN3: 113.5 bar).

The specific assumptions, considering onerous delays, are listed below:

- 2.9 seconds delay to actuate the VDA [MSRT] setpoint decrease at a rate of 100°C/h.
- 15.9 seconds delay for RRA [RHRS] pump start-up, (this delay includes the pump starting time).
- Minimum flow characteristic for RRA [RHRS] pumps (Sub-chapter 14.1).
- 50°C injection flow temperature corresponding to an initial IRWST temperature of 50°C. This temperature is considered constant during the transient because of the slow increase of IRWST temperature. Otherwise, this temperature increase is limited by IRWST cooling performed by the four LHSI trains in operation on pump mini-flow configuration.

The specific assumptions related to the accumulators are as follows:

- 47 m^3 total volume.
- 35 m^3 water volume.
- 45 bar abs initial pressure.
- 2500 m^{-4} discharge line resistance.
- 50°C water temperature.

The following RRA [RHRS] modelling is adopted:

- Three RRA [RHRS] trains and three accumulators inject into the cold legs of the intact loops 1, 2 and 3.
- The RRA [RHRS] train injecting into the broken loop is considered to discharge into the containment, with no contribution to RCP [RCS] injection.

VDA [MSRT] (F1A):

The VDA [MSRT] capacity is the minimum value (Sub-chapter 14.1) (50% nominal SG steam flow). VDA [MSRT] setpoints are nominal values.

The automatic setpoints are as follows:

- 95.5 bar before beginning of partial cooldown.
- 60 bar after the end of partial cooldown.

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

The GCT [MSB] is credited (see below), The consequence of GCT [MSB] operation is that VDA [MSRT] are not actuated until "fast cooldown". The "fast cooldown" is performed by the four VDA [MSRT].

ASG [EFWS] (F1A):

ASG [EFWS] is not actuated because of ARE [MFWS] system operation (see specific assumptions on other systems).

RCP [RCS] pumps TRIP (F1A):

Normally an automatic RCP [RCS] pump trip following the LOCA occurs on low pressure drop across the RCP [RCS] pumps (< 80% nominal RCP [RCS] pump pressure drop).

In the present analysis, it is postulated that the RCP [RCS] pumps are tripped with reactor trip, which is assumed to be the most conservative assumption (compared to delayed RCP [RCS] pumps trip between reactor trip and RCP [RCS] pumps trip signal).

RBS [EBS] (F1B):

The RBS [EBS] is manually actuated by the operator following partial cooldown and no MHSI injection. Both RBS [EBS] pumps are available and inject when the "fast cooldown" is manually actuated. The total injected flow rate into the RCP [RCS] is 5.6 kg/s. The initial boron concentration of each RBS [EBS] tank is equal to 7000 ppm in enriched boron. In the CATHARE calculation, the RBS [EBS] pump operation is not simulated, taking only credit for the LHSI and accumulator injections.

3.7.3.2.4. Specific assumptions for other systems

GCT [MSB]:

The GCT [MSB] capacity is the minimum value (Sub-chapter 14.1) (50% nominal SG steam flow). GCT [MSB] setpoints are nominal values.

The automatic setpoints are as follows:

- 90 bar before beginning of partial cooldown.
- 55 bar after the end of partial cooldown.

The GCT [MSB] is available since the main steam header is open.

ARE [MFWS]:

ARE [MFWS] supply is available and the following assumptions are made:

- After reactor trip, flow rate reduced to 20% of initial value in 10 seconds.
- Control of nominal SG level by the low load train.
- Decrease of ARE [MFWS] temperature at SG inlet from 230°C to 190°C.

PZR pressure control:

The PZR pressure control system (PZR heaters) is modelled because its operation delays the RT signal. A total heating power of 2592 kW is considered until the PZR is completely empty.

RCV [CVCS]:

This is not taken into account for break flow compensation or boration.

3.7.3.3. Operator action

The operator action is the actuation of the "fast cooldown", modelled in the calculation as the full opening of the four VDA [MSRT].

The criterion considered in the CATHARE calculation is:

- Core outlet temperature (average channel) > 350°C and no MHSI.

3.7.3.4. Reactivity balance

No reactivity calculation is performed in the CATHARE calculation. The core sub-criticality is assumed for the entire transient after RT occurrence. This assumption is supported by the following arguments:

- At the time of "fast cooldown" actuation, the void fraction in the core is high.
- At the actuation of the partial cooldown, the void fraction of the coolant increases due to the fast RCP [RCS] depressurisation rate.
- The accumulators inject borated water once the RCP [RCS] pressure reaches 45 bar, corresponding to a saturation temperature of 257°C. At this temperature level, the core remains sub-critical with a high shutdown margin as all the rods are inserted.
- The accumulator borated water enters the core at a fast rate, due to the fast RCP [RCS] depressurisation rate inducing high accumulator injection flow rate, and due to the low RPV water inventory at the time of accumulator injection.
- The LHSI injection provides additional borated water when the RCP [RCS] pressure reaches 20 bar, corresponding to a saturation temperature of 210°C. In this temperature range, core re-criticality could occur in the absence of boron injection. However, at the time of LHSI injection, the core boron concentration is already high, since about 40 te of water from the accumulators have already entered the RPV.

3.7.4. Thermal-hydraulic Analysis

3.7.4.1. Initial conditions

The initial state conditions, given in Section 16.1.3.7 - Table 1 relate to 100% of nominal power with an assumed thermal-hydraulic flow rate.

The axial power shape for the average rod in the average assembly is given in Section 16.1.3.7 - Figure 1. This power shape, skewed at the top of the core, has the following characteristics:

- An enthalpy rise factor of 1
- A peaking factor of 1.57 at 3.5 m of active core height
- An axial offset of 15%.

This power shape is chosen because it provides a reasonably limiting distribution of power versus core height. There is more power distributed in the upper part of the core. This power distribution is limiting for SB-LOCA analysis because of core uncovery: as the core uncovers, the cladding in the upper part of the core heats up and is sensitive to the linear power at that elevation while the cladding temperature in the lower part of the core remains close to the saturation temperature.

The initial fuel pellet temperature is the same as for PCC analysis. This temperature is averaged over a section of fuel and is given as a boundary condition limit. This initial temperature has a negligible impact on the maximum cladding temperature reached because the core heats up a long time after RT in a typical SB-LOCA sequence. The heat-up depends on decay heat only, not on the stored energy of the fuel. The initial fuel stored energy is removed long before heat-up occurs.

3.7.4.2. Results

Typical sequence of events is given in Section 16.1.3.7 - Table 2.

The most representative parameters are presented in the following figures:

- Figure 2: Reactor and SG power, RCP [RCS] and SG secondary side pressures.
- Figure 3: Break mass flow rate (steam, liquid and total), Break flow velocities (steam and liquid).
- Figure 4: Swell levels in RPV, Core void fraction.
- Figure 5: Coolant liquid and vapour temperatures in RPV and core, Cladding temperatures.
- Figure 6: Integrated flow rates of RIS [SIS], Coolant inventory on primary and secondary side.
- Figure 7: Total of RIS [SIS] and break flow rates, Integrated RIS [SIS] and break flow rates.

The detailed explanation for the parameters is given in Section 16.1.3.7 - Table 3.

The accident is not recalculated for EPR_{4500}. Achieving the final state and satisfying the acceptance criteria are deduced from the results obtained from the analysis of EPR_{4250}, which shows that the acceptance criteria are satisfied by a large margin, which can accommodate the increase in power.

3.7.4.2.1. Accident analysis for EPR at 4250 MW [Ref-1]

The operator action, consisting of performing a secondary side "fast cooldown" via the full opening of the four VDA [MSRT], at about 5000 seconds (≈ 1.4 hour), is sufficient to reach the safe state:

- The operator action is initiated when core outlet temperature (referring to the average channel) reaches 350°C, with no MHSI injection.
- 30 seconds later the accumulator injection starts.
- The core heat-up stops immediately.
- 320 seconds later, the LHSI begins (all time delays counted from the time of operator action).
- The peak cladding temperature in the average rod is less than 450°C, reached at time of operator action. Thus without any dedicated hot rod analysis it is obvious that a high margin exists with respect to the acceptance criterion of 1200°C in the hot rod.

3.7.4.2.2. Accident analysis for EPR at 4500 MW [Ref-1]

The break flow depends on the primary pressure and the break quality. For this break size (20 cm^2), the SG are needed for residual power removal. After the initial depressurisation (≈ 500 seconds), the primary pressure stabilises slightly below the secondary pressure. At about 4200 seconds, the break flow is mostly liquid and the break quality depends mainly on the initial primary inventory. This phase of the transient is almost identical for EPR_{4250} and EPR_{4500}.

From 4200 seconds and before operator action, the break flow is mostly steam. Because the initial power is larger (by 5.9%), the steam flow from the core is slightly more important for EPR_{4500} than EPR_{4250}. Also more importantly for the higher power reactor is the condensate returning from the SG in the broken loop and the break flow itself. The beginning of core heat-up would occur earlier as does the point at which the 350°C core exit temperature criteria is reached. The delay between the steam break flow (4200 seconds) and the criteria of reaching 350°C at the core exit (5000 seconds) is 800 seconds for EPR_{4250}. For EPR_{4500} this time is reduced. The reduction is proportional to the vapour production, which depends on the initial power and the residual power. The initial power is higher by 5.9%. The residual power is lower by 1.4%. The point at which the 350°C core exit temperature criterion is reached is about 30 seconds earlier (≈ 800 x 4.4%) for EPR_{4500} than the EPR_{4250}.

Due to a higher initial power (5.9%) and an FQ higher by 3.2% for EPR_{4500}, the cladding temperature is higher compared to EPR_{4250}. The increase in cladding temperature for EPR_{4250} is 145°C (Section 16.1.3.7 - Figure 5). For EPR_{4500} the increase in cladding temperature will be higher by 7.8% (5.9 + 3.2 -1.4 = 7.8%). The cladding temperature will reach a maximum of 435°C (≈ 420 + 145 x7.8%) for EPR_{4500}.

After operator action, the delay in the injection of accumulator water and RIS [SIS] is less affected by the increase in power.

The arguments presented above assure that all LOCA acceptance criteria are met as well as reaching the final state for EPR_{4500}. The operator action, consisting of a secondary side "fast cooldown" via the four VDA [MSRT] is initiated when the core outlet temperature reaches 350°C and when the signal that the MHSI is not injecting is present. The core outlet temperature of 350°C is reached at 4970 seconds (≈ 1 hour 23 minutes) giving sufficient time for operator action.

3.7.4.3. Consequences of the change to the partial cooldown rate

Following the SI signal actuated on a pressuriser pressure < MIN3 signal, the partial cooldown is automatically initiated to remove the energy from the primary side via the secondary side. Consequently the primary pressure is reduced which allows the MHSI to inject borated water.

As a consequence of increasing the cooldown rate, the MHSI delivery pressure of 85 bar will be reached earlier due to a more rapid reduction in primary pressure. Therefore, the cooldown rate increase from -100°C/h to -250°C/h will have a beneficial effect on the mitigation of the small break LOCA accident.

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

Therefore, it can then be concluded that all the decoupling criteria will be met.

3.7.4.4. Conclusions

The following conclusions can be drawn concerning SB-LOCA with loss of MHSI:

- The peak cladding temperature remains below the 1200°C acceptance criteria.
- the maximum oxidation of the cladding does not exceed 17% of the total thickness of the cladding at the hot spot,
- there is no cladding failure[7],
- the integrity of the core geometry is maintained;
- long-term cooling is ensured.

All the LOCA acceptance criteria are met.

The residual power is effectively removed. The final state is reached. The water inventory is maintained with at least one LHSI pump, while the SG or at least one other LHSI pump operating in residual heat removal mode are sufficient to perform cooldown of the RCP [RCS].

After accumulator injection and injection from the LHSI pump into the RCP [RCS], core sub-criticality is assured. In this case, no additional boration is required, but additional boration may be needed for breaks smaller than 5 cm².

3.7.5. System Sizing

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

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[7] Considering an initial internal fuel rod pressure of 90 bar. This result can be changed with the increase of this pressure without major modification of the cladding maximum temperature.

SECTION 16.1.3.7 - TABLE 1

Initial Conditions
20 cm² (∅ 50 mm) Cold Leg Break with Loss of MHSI, in State A at 4250 MW

Parameters	Values
Reactor coolant system	
Initial reactor power Initial average RCP [RCS] temperature Initial reactor coolant pressure RPV coolant flow PZR water volume / level	100% of nominal power 313.6°C (nominal) 155 bar (nominal) 22135 kg/s (T/H design flow rate) 40 m^3 / 6.89 m (nominal)
Steam generators	
Initial steam pressure Initial SG level	78.0 bar (nominal) 15.7 m (nominal)
Feedwater	
Main feedwater flow Initial ARE [MFWS] temperature	100% of nominal flow 230°C (nominal)

SECTION 16.1.3.7 - TABLE 2

Sequence of Events (Global Time Values)
Typical Results for 20 cm² (∅ 50 mm) Cold Leg Break
with Loss of MHSI, in State A at 4250 MW

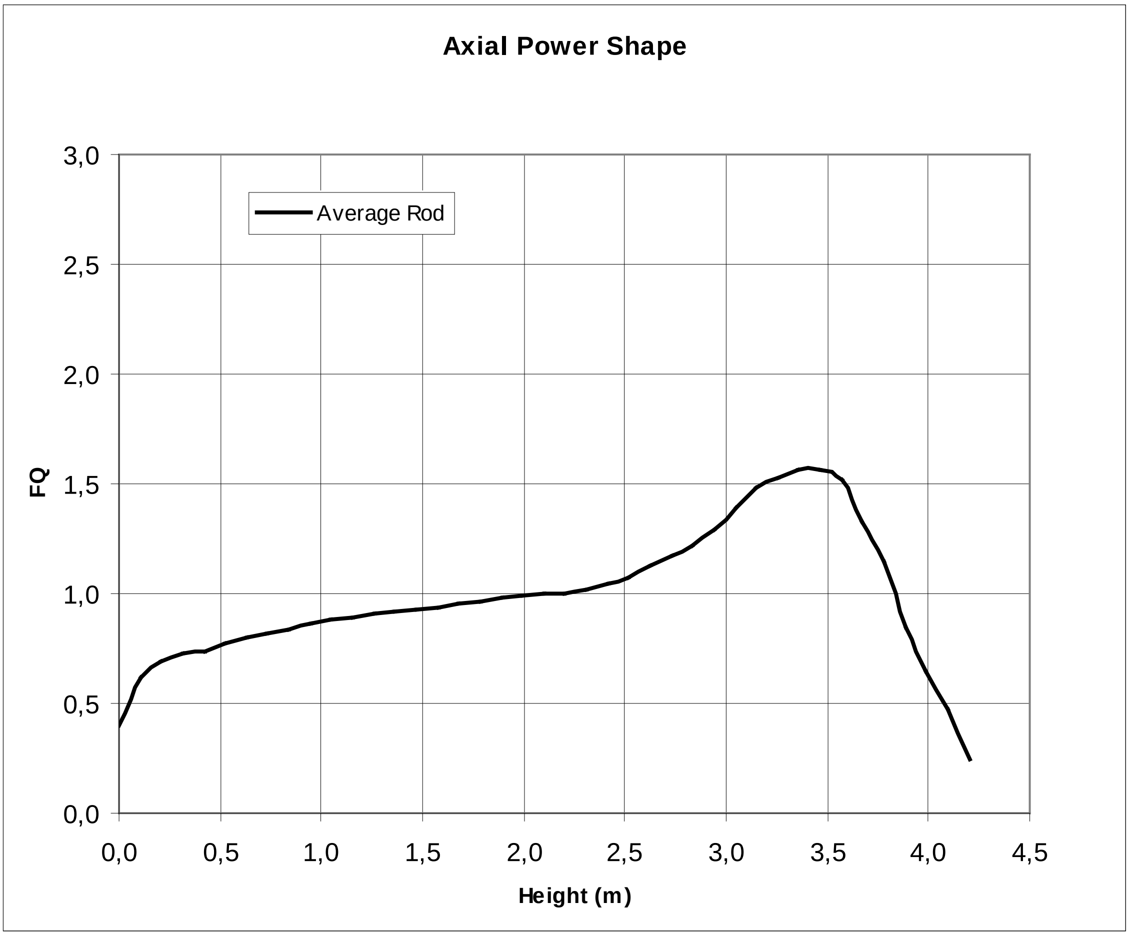
TIME (s)	EVENT
0	Break opening
100	Reactor, turbine and RCP [RCS] pumps trip; ARE [MFWS] reduction to low load on PZR pressure < MIN2 (133.5 bar)
150	Safety Injection signal on PZR pressure < MIN3 (113.5 bar) and beginning of Partial Cooldown via GCT [MSB] 15 s later starting of LHSI pumps (failure of MHSI pumps)
1150	End of PC, secondary pressure at 55 bar, RCP [RCS] pressure slightly above
4750	Beginning of core heat-up
5000	Operator action: beginning of "fast cooldown" via the 4 VDA [MSRT] Core outlet temperature referring to average channel = 350°C Peak cladding temperature in average Rod < 450°C
5030	Beginning of accumulator injection (RCP [RCS] pressure < 45 bar) and end of core heat-up, minimum of RCP [RCS] inventory
5350	Beginning of LHSI injection (RCP [RCS] pressure < 20 bar)
6000	New steady-state in RCP [RCS] with pressure at > 15 bar, continuous make-up of RCP [RCS]
8000	End of calculation

SECTION 16.1.3.7 - TABLE 3

List of Figures: Typical results for 20 cm2 (Ø 50 mm) cold leg break with loss of MHSI, in state A

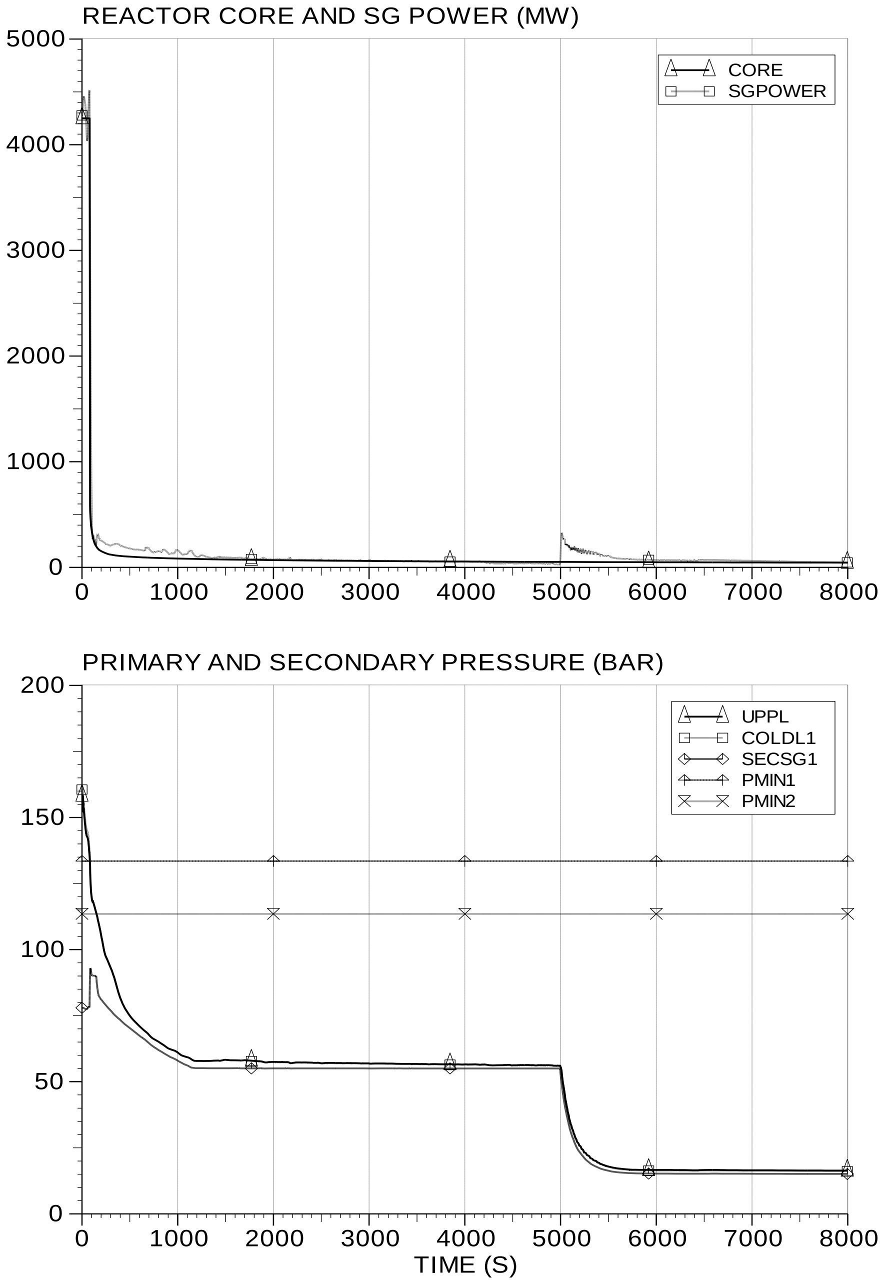
SECTION 16.1.3.7 - FIGURE 1

RRC-A Analysis
Typical Axial Power Shape of Average Rod



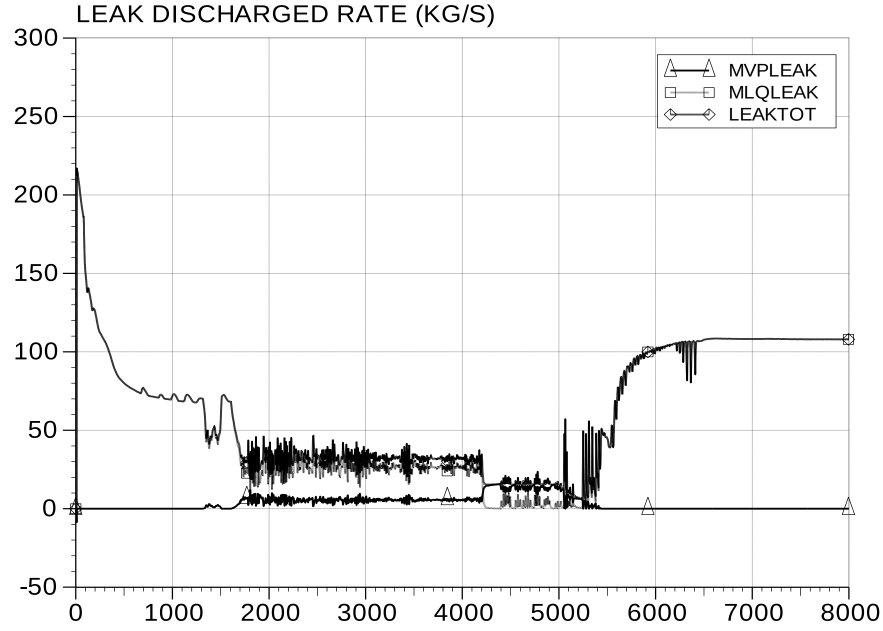
SECTION 16.1.3.7 - FIGURE 2

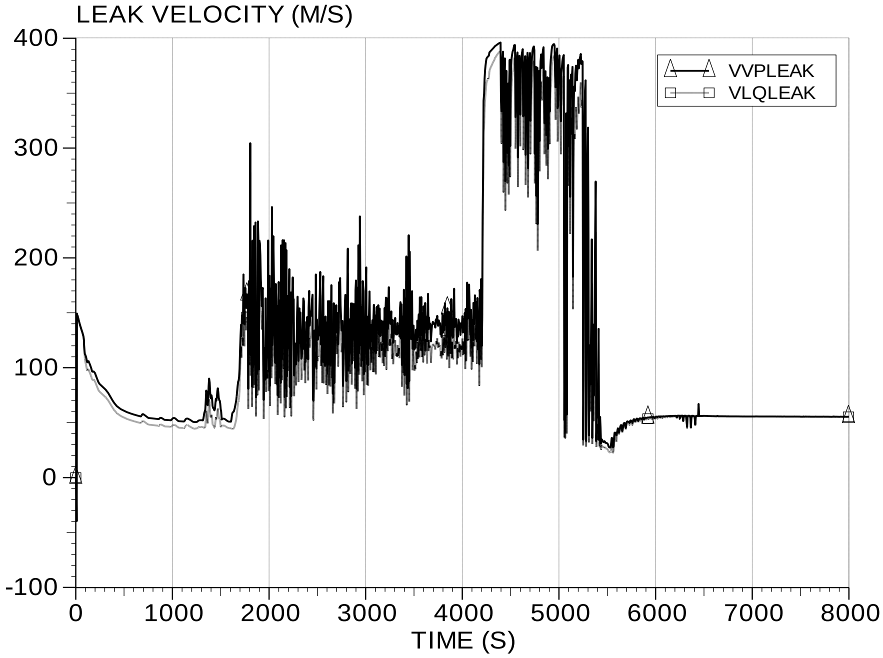
Typical Results for 20 cm^2 (Ø 50 mm) Cold Leg Break with Loss of MHSI, in State A
Core Power and Total Heat Exchange in Steam Generator
Primary and Secondary System Pressure



SECTION 16.1.3.7 - FIGURE 3

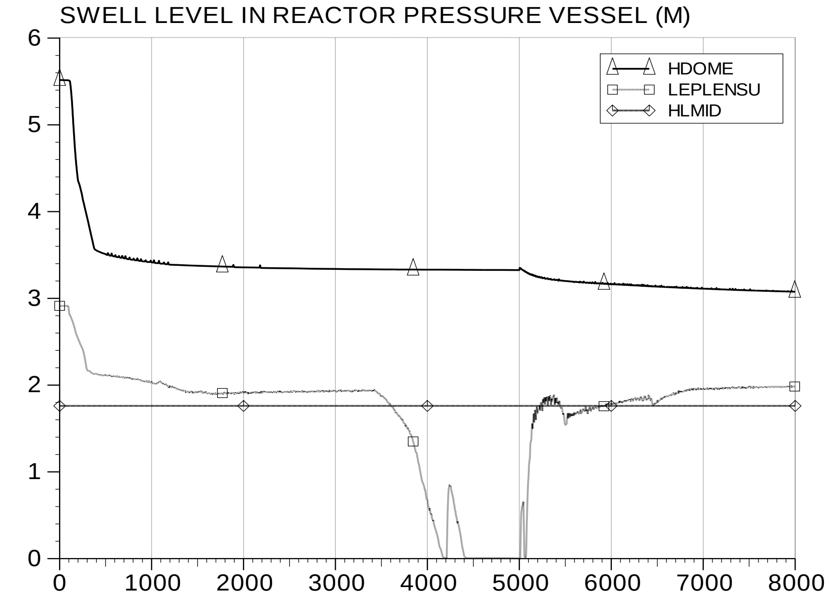
Typical Results for 20 cm² (Ø 50 mm) Cold Leg Break with Loss of MHSI, in State A
Vapour and Liquid Mass Flow at the Leak
Vapour and Liquid Velocity at the Leak

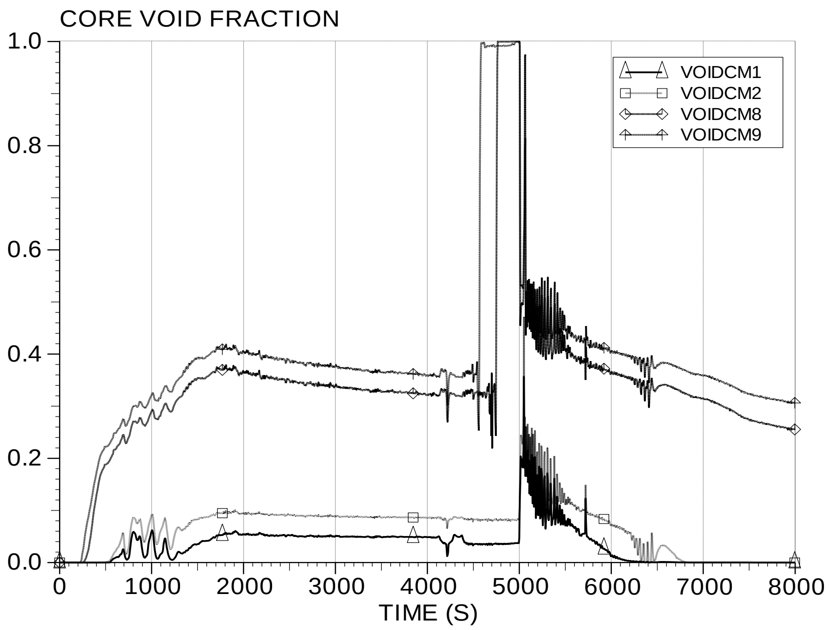




SECTION 16.1.3.7 - FIGURE 4

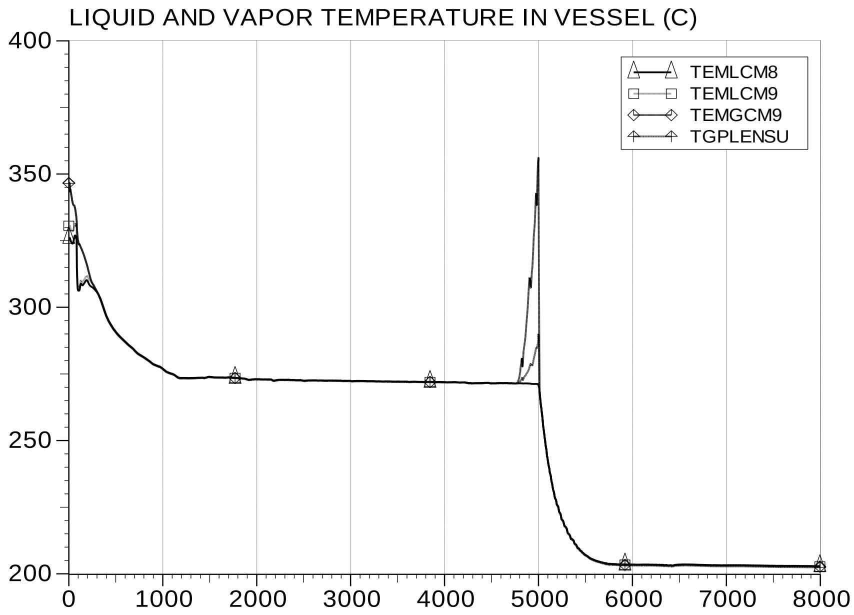
**Typical Results for 20 cm^2 (Ø 50 mm) Cold Leg Break with Loss of MHSI, in State A
Swell Level in Reactor Pressure Vessel
Void Fraction in the Core**

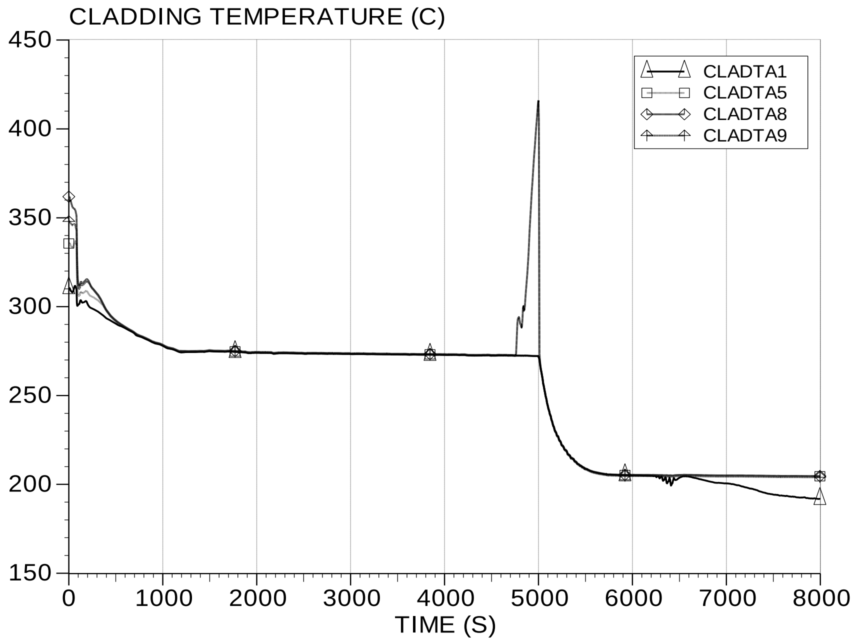




SECTION 16.1.3.7 - FIGURE 5

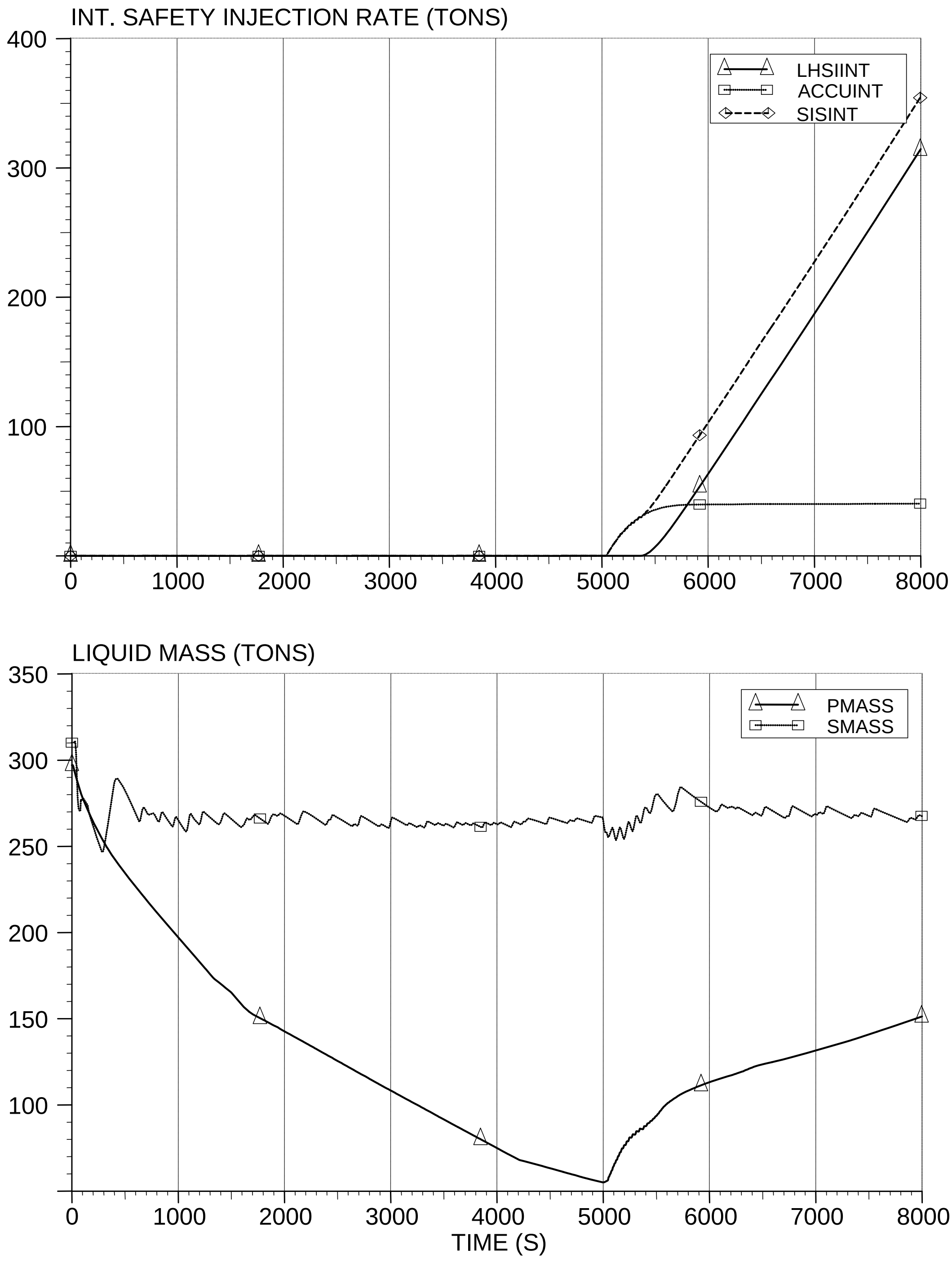
Typical Results for 20 cm^2 (Ø 50 mm) Cold Leg Break with Loss of MHSI, in State A
Liquid and Vapour Temperature in Reactor Pressure Vessel
Cladding Temperature of the Average Rods





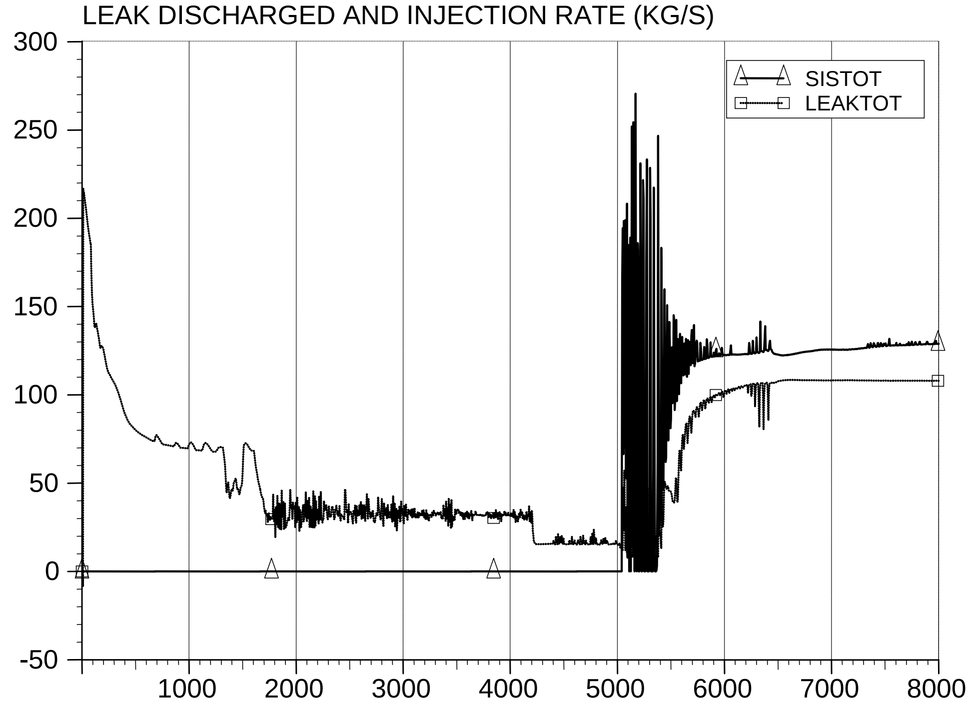
SECTION 16.1.3.7 - FIGURE 6

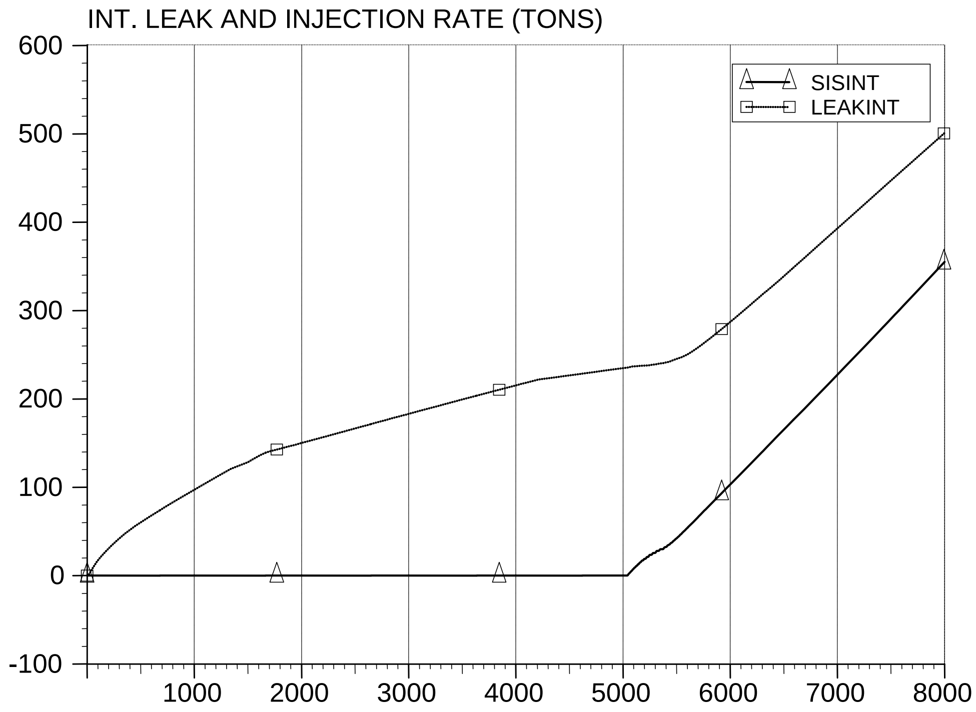
Typical Results for 20 cm² (Ø 50 mm) Cold Leg Break with Loss of MHSI, in State A Integral Safety Injection Rate Water Inventory in the Primary and Secondary System



SECTION 16.1.3.7 - FIGURE 7

Typical Results for 20 cm² (Ø 50 mm) Cold Leg Break with Loss of MHSI, in State A
Safety Injection and Leak Discharge Rate
Integral Safety Injection and Leak Discharge Rate





3.8. LOCA (BREAK SIZE UP TO 20 CM^2) WITHOUT LHSI (STATE A)

3.8.1. Identification of Cause and Accident Description

The initiating event is a postulated small break located at the cold leg of the reactor coolant pipe. A small break is defined as a leak with an equivalent diameter of less than 50 mm or a cross sectional area of less than 20 cm^2.

The RRC-A sequence is identified by a combination of the initiating event and the total loss of a relevant safety system. The total loss of the low head safety injection system is assumed to be caused by a common cause failure. In accordance with the RRC-A rules (section 2), no additional failures (e.g. single failure or emergency power mode) are assumed in the systems required to reach the final state of the transient.

After event initiation, the break mass flow rate increases rapidly to its maximum in a fraction of a second, but decreases as the primary system pressure falls and the flow changes from single-phase sub-cooled liquid to saturated two-phase mixture, with increasing steam quality. The rate of depressurisation changes when flashing and boiling starts in the core.

After reaching the reactor trip criterion "pressuriser pressure < MIN2", reactor and turbine are tripped automatically within a few seconds. The subsequent depressurisation rate of the primary and secondary systems depends on the partial cooldown started at the same time as the RIS [SIS].

The total mass flow rate of the medium head safety injection system and the accumulators is able to compensate for the loss of coolant through the break and to fill the primary coolant system.

Given that the LHSI/RHR system is assumed to be completely unavailable, the final state (see below) can only be reached by manual initiation of the secondary side cooldown via the GCT [MSB] station at a rate of 50°C/h down to 2 bar. Without this cooldown via the secondary side, the decay heat would largely be dissipated to the IRWST (via break) and in the long term this could lead to loss of the MHSI. Finally, after cooldown via the secondary side, the heat removal from reactor coolant system and IRWST is ensured by the cooling chain EVU/RRI/SEC [CHRS/CCWS/ESWS] and the SG.

The primary coolant inventory and the sub-criticality is ensured by the MHSI.

<u>Final State:</u>

It must be demonstrated that the final state (required for RRC-A sequences) can be reached, i.e.:

- Attainment of long-term core sub-criticality.
- Decay heat removal ensured.
- Control of activity releases and the integrity of radiological barriers.

For this demonstration, the following essential acceptance criteria are required:

- Limited peak cladding temperature (< 1200°C).

- Peak containment pressure below design pressure (< 5.3 bar).
- Peak IRWST temperature below design temperature (target: < 100°C).

The following systems are available to achieve these criteria:

- Four MHSI trains are available to compensate for the break flow in the short and long term and for ensuring long-term sub-criticality (however, one complete RIS [SIS] train is assumed to flow into the containment due to the break).
- In addition to the MHSI, two RBS [EBS] trains are available to assure core sub-criticality.
- The MHSI injection is effective due to partial cooldown at 100°C/h via the condenser steam dump valve (GCT [MSB]) or via the main steam relief trains (VDA [MSRT]) actuated by the RIS [SIS] signal.
- The main steam bypass (GCT [MSB]) station or the VDA [MSRT] and the AAD [SSS] (or the 4-train ASG [EFWS]) for the heat removal during the partial cooldown and the normal plant cooldown phase.
- The 2-train EVU [CHRS] for heat removal in the long term from the IRWST to the SEC [ESWS].
- The four accumulators for RCP [RCS] filling (however, one complete RIS [SIS] train is assumed to be lost due to the break).
- The containment isolation.

3.8.2. Methods and Assumptions

3.8.2.1. Methods

The thermal hydraulic response of the primary side, the secondary side and the connected containment system (inclusive IRWST) is simultaneously calculated with the coupled program system S-RELAP5 (Version V1.4)/COCO (Version 2V10.3) (Appendix 14A).

Section 16.1.3.8 - Table 1 lists the important SB-LOCA events, the code models used and the corresponding test cases for qualification of these models.

3.8.2.2. Assumptions

3.8.2.2.1. Accident definition

The case studied in this section is a 20 cm² (∅ 50 mm) break in a cold leg pump discharge pipe with total loss of LHSI occurring at 100% power level.

The present analysis aims at assessing the efficiency of the "manual cooldown" to meet the core cooling decoupling criteria, for consistency with the actuation criterion given in the emergency guidelines.

3.8.2.2.2. Protection and mitigation actions

The final state for RRC-A analysis is defined in section 2, expressed in LOCA analysis as follows:

- The core is sub-critical.
- The core is re-flooded.
- The decay heat is removed.
- The break flow is compensated by the RIS/RRA [SIS/RHRS] flow.
- The activity release is under control with the integrity of the barriers in accordance with the acceptance criteria.

In accordance with the rules defined for safety analyses in Sub-chapter 14.0, the following assumptions are considered:

- The final state is reached with consideration of all the systems available (safety classified and non-safety classified) provided they are not impaired by the event.
- The final state is reached without additional failures.
- The final state is reached without equipment maintenance.
- The Loss of Offsite Power is not coincident with the event.
- The assumptions for the accident analyses are largely realistic, however, some conservative data or hypotheses (e.g. in terms of system characteristics) are considered mainly for simplicity or because of the difficulty in defining realistic data.
- No operator action is considered before 30 minutes after reactor trip. The operator action, secondary side cooldown via the MS bypass system at 50°C/h down to 2 bar, is initiated on “response of RIS [SIS]” and “no LHSI” 30 minutes after RIS [SIS] signal actuation.

3.8.2.2.3. Specific assumptions related to safety systems

<u>Reactor Trip:</u>

RT signal is actuated on low PZR pressure (< MIN2: 133.5 bar).

The specific assumptions, considering normal delays, are listed below:

- Beginning of rod insertion 1.2 seconds after RT signal.
- 3.5 seconds delay for complete rod insertion.
- Best estimate decay heat curve (Sub-chapter 14.1).
- Total main feed water flow reduction to low load head capacity 1.2 seconds after RT signal.

Safety Injection:

RIS [SIS] signal is actuated on very low PZR pressure (< MIN3: 113.5 bar, accounting for degraded containment conditions).

The specific assumptions, considering penalising delays, are listed below:

- 2.9 seconds delay to actuate the GCT [MSB] setpoint decrease at a rate of 100°C/h.
- 15.9 seconds delay for RIS/RRA [SIS/RHRS] pump start-up, this delay includes the pump starting time.
- Minimum characteristic for RIS/RRA [SIS/RHRS] pumps (Sub-chapter 14.1).
- 42°C for injection flow initial temperature corresponding to an initial IRWST temperature of 42°C. This temperature increases during the event, as there is no IRWST cooling by LHSI.

The specific assumptions related to the accumulators are as follows:

- 47 m^3 total volume.
- 35 m^3 water volume.
- 45 bar abs of initial pressure.
- 2500 m^{-4} discharge line resistance.
- 50°C water temperature.

The following RIS [SIS]/RRA [RHRS] modelling is adopted:

- The three RIS/RRA [SIS/RHRS] trains and the three accumulators inject into the cold legs of the intact loops.
- The SIS/RHRS train and the accumulator injecting into the broken loop is considered to discharge into the containment, with no contribution to RCP [RCS] injection.

VDA [MSRT]:

They are not actuated because of GCT [MSB] availability.

ASG [EFWS]:

ASG [EFWS] is not actuated because of ARE [MFWS] system operation (see specific assumptions to other systems).

RCP [RCS] pumps TRIP:

The automatic RCP [RCS] pumps trip following a LOCA occurs on low pressure drop across the RCP [RCS] pumps (< 80% nominal RCP [RCS] pumps pressure drop).

3.8.2.2.4. Specific assumptions related to other systems

GCT [MSB]:

The GCT [MSB] capacity is the minimum value defined (Sub-chapter 14.1) (about 50% of nominal plant steam flow). GCT [MSB] setpoints are nominal values.

The automatic setpoints are as follows:

- 90 bar before beginning of automatic partial cooldown.
- 55 bar at end of automatic partial cooldown.
- 2 bar after the end of cooldown at 50°C/h.

ARE [MFWS]:

ARE [MFWS] supply is available, with the following assumptions:

- After reactor trip, flow rate reduced to 20% of initial value in 10 seconds.
- Control of nominal SG level by the low load train.

PZR pressure control:

The PZR pressure control system (PZR heaters) is modelled because its operation delays the RT signal. A total heating power of 2592 kW is considered until the PZR is completely empty.

RBS [EBS]:

The RBS [EBS] is manually actuated, however, in the S-RELAP calculation, the RBS [EBS] pump operation is not simulated, taking only credit for the MHSI and accumulator injections which provide sufficient boration.

3.8.2.2.5. Operator action

The operator action consists in the actuation of the "cooldown at 50°C/h" by means of the GCT [MSB].

The criterion considered in the S-RELAP calculation is:

- “Response of RIS [SIS] signal” and “no LHSI 30 minutes after RIS [SIS] signal occurrence”.

3.8.2.2.6. Reactivity balance

There is no specific reactivity calculation. Core sub-criticality is assured through the entire transient after RT occurrence without use of the RBS [EBS]. This is supported by the following:

- At the time of "manual cooldown", the void fraction in the core is high and a considerable amount of MHSI water is injected into the system.

- The accumulators inject borated water once the RCP [RCS] pressure reaches 45 bar, corresponding to a saturation temperature of 257°C. At this temperature, the core remains sub-critical with a high shutdown margin assuming all the rods are inserted.

- With ongoing cooldown of the plant to a lower pressure and temperature a large amount of highly borated water from both MHSI and accumulators enters the RCP [RCS].

3.8.3. Results

There is no recalculation of the RRC-A transient in the EPR Preliminary Safety Analysis Report. The capability to fulfil the safety and decoupling criteria is derived from the results of the RRC-A event analysis performed for EPR_{4900} (Appendix 16B) and from the comparison of the relevant characteristics between EPR_{4500} and EPR_{4900}.

3.8.3.1. Accident analysis for EPR 4900

The main initial and boundary conditions used for the analysis are contained in Section 16.1.3.8 - Table 2.

Approximately 1 minute after initiation of the break, the reactor and turbine trips are automatically actuated by the signal “pressuriser pressure < MIN2” (132 bar, conservative value accounting for degraded containment conditions and uncertainty). The RIS [SIS] signal “pressuriser pressure < MIN3” (112 bar, conservative value accounting for degraded containment conditions and uncertainty) which starts the MHSI pumps and actuates the secondary side partial cooldown is reached 133 seconds after the start of the event. The injection rates of the Main Feedwater or the Start-up/Shutdown system are able to maintain the SG water level at normal condition.

Five minutes after the start of the event, the fluid at the outlet of the average core channel reaches saturated conditions and, approximately two minutes later, the pressure difference across the reactor coolant pumps decreases below 80% of its rated value and the pumps are automatically tripped.

At 424 seconds the RCP [RCS] pressure decreases below the shutoff head of the medium head safety injection system, the three MHSI trains start to inject into the primary coolant system and the loss of primary coolant is partially compensated.

Thirty minutes after receiving the RIS [SIS] signal, the plant cooldown via the Main Steam Bypass (GCT [MSB]) station is initiated by operator action. Within the next 1 hour 52 minutes, the primary system pressure follows the resulting secondary system pressure gradient down to a minimum of 11 bar at 8680 seconds.

After refilling of the primary system, at approximately 2.5 hours, single-phase natural circulation restarts and the fluid mixing process leads to balanced fluid temperature and boron concentration in the reactor coolant system.

The accumulators inject their entire inventory within 1 hour 40 minutes and the primary coolant system refills to its initial coolant mass inventory. The accumulator injection stops the steam break flow and the second containment pressure and temperature peaks are limited to values well below the containment design limits.

After accumulator depletion, the primary pressure increases above 25 bar until MHSI injection balances the break flow. The subsequent RCP [RCS] pressure transient depends on the number of MHSI pumps in operation. The increasing primary pressure and break mass flow rate result in a third containment pressure and temperature peak, 3 hours 15 minutes after initiation of the break.

An energy balance at 4 hours shows the following distribution: part of the total heat from reactor core and structural materials (≈ 45.7 MW)][b] is removed via the steam generators secondary side (≈ 32.7 MW) and the rest is released through the break (≈ 13.0 MW) to the containment atmosphere and to the IRWST water.

The capacity of one EVU [CHRS] heat exchanger is 13 MW at an inlet temperature of 94°C. The analysis shows a temperature increase of the IRWST water up to 92.5°C during the 4h time period. To maintain operation of the MHSI in the long term, this increase can be stopped (or even prevented) by initiating one or two EVU [CHRS] trains which limits the IRWST water temperature to a value below the RIS [SIS] design temperature (120°C).

During the transient no core uncovery occurs (i.e. the fuel cladding temperatures remain at saturated conditions with no cladding rupture).

The primary and secondary coolant inventory is stabilised and the residual heat removal from the RCP [RCS] is performed in the long term by the SG and the EVU [CHRS] cooldown train. The borated water from the accumulators and from the IRWST, via the MHSI, ensures sub-criticality.

The final state characteristics of the “small break LOCA with loss of the LHSI” scenario are:

- core sub-criticality ensured by boration via the MHSI and the accumulators;
- residual heat removal by the SG and the EVU [CHRS] trains,
- activity releases controlled since the barriers (fuel cladding, primary circuit and containment) maintain their integrity.

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

Apart from the main initial conditions of the plant at nominal 100% power level, the following systems or functions are of primary importance for mitigation of the scenario:

- Reactor protection and operational system setpoints (RT, RIS [SIS] signal, VDA [MSRT], GCT [MSB]).
- Characteristics of MHSI and accumulators.
- Containment features (IRWST water volume and initial temperature, free volume, heat structures, heat removal via EVU [CHRS]).

The relevant data for the initial plant parameters and safety functions described in previous paragraphs are given for both EPR_{4900} and EPR_{4500} in Section 16.1.3.8 - Table 3.

The comparison of the data shows that there are only minor changes between EPR_{4900} and EPR_{4500}. Some EPR_{4500} conditions are slightly more favourable compared to EPR_{4900}: the power level is lower in EPR_{4500} and the MHSI capacity is relatively higher in EPR_{4500}. The higher MHSI shutoff head for EPR_{4500} also compensates for the slightly higher RCP [RCS] temperature and secondary side pressure level. The slightly lower SG heat transfer surface is not relevant since the heat to be removed after RT is only in the order of a few percent of nominal conditions.

3.8.3.3. Impact of variation of data and characteristics

The event consists of three phases before the final state is reached, i.e.:

- The initial automatic phase with RT and response of RIS [SIS] signal which induces partial cooldown and finally MHS injection.
- The manual cooldown of the plant at 50°C/h via the GCT [MSB] or VDA [MSRT] which leads to accumulator injection as well as higher MHSI injection rate and consequently complete refilling of the RCP [RCS] at low RCP [RCS] pressure.
- The manual actuation of the EVU [CHRS] after four hours, to remove the heat from the IRWST and limit the IRWST water temperature.

The lower power level of EPR_{4500} has a beneficial effect on all three phases of the transient.

- Phase 1 will end up at slightly more favourable conditions because the RCP [RCS] conditions (temperature and pressure) at the end of PC are identical to the EPR 4900 conditions because of identical secondary side pressure. However, at end of PC the core power for EPR_{4500} is somewhat lower and the MHSI shutoff heat is slightly higher. Furthermore, the lower the power level, the faster the depressurisation rate, hence the MHSI injection is reached sooner. Due to the more favourable conditions at the end of Phase 1 and because the EPR 4900 analysis showed that there is also no core uncovery, no core uncovery is expected for EPR 4500 conditions.
- Phase 2: there is no difference in cooldown efficiency via GCT [MSB] or VDA [MSRT] between EPR_{4500} and EPR_{4900} because of practically identical capacities. Similarly, the MHSI and accumulator efficiency is comparable so that make-up of the RCP [RCS] will occur in the same way. However, due to the lower core power, it is expected that the heat supply to the IRWST via the break flow is somewhat lower because the heat removal via the SG should be the same in EPR_{4900} and EPR_{4500}.
- Phase 3 should be more favourable for EPR_{4500} conditions because of the reduced heat supply to the IRWST, i.e. slower IRWST water temperature increase compared to EPR_{4900}. Therefore, the potential need to start the EVU [CHRS] to limit the IRWST temperature increase (avoiding steam discharge) can occur later than 4 hours after event initiation as indicated in the EPR_{4900}. EVU [CHRS] would be started on reaching a certain IRWST high temperature limit (exact value to be defined as part of the detailed design). Both EVU [CHRS] pumps start at EVU [CHRS] actuation.

However, the cooling mode for the IRWST water is different from EPR_{4900}. For EPR_{4500} the cooling occurs via the containment spray whereas in EPR_{4900} a direct cooling circuit is available. It is possible that the actual spray-cooling mode is not immediately as efficient as the former direct cooling mode because the spray inherently removes heat not only from the IRWST water but also from the upper containment structures.

This possible slight drawback can be balanced by an earlier manual actuation of the EVU [CHRS], e.g. after detection of the LHSI failure or simultaneously with actuation of the secondary side cooldown, so that the heat-up of the IRWST is further delayed. Furthermore, for an even stronger reduction of the heat supply to the IRWST water, the secondary side cooldown at a gradient of 50°C/h could be replaced by a fast cooldown (complete opening of the VDA [MSRT]). Any risk of re-criticality due to this fast cooldown is excluded since:

- 100 te of highly borated MHSI are injected 30 minutes after RT (i.e. before cooldown).
- With (fast) cooldown further efficient boration occurs due to accumulator injection.

Finally, it is noted that the EPR_{4900} analysis was performed under a very conservative assumption by using an initial IRWST water inventory of only 1300 m^3 instead of about 1900 m^3 normally available.

3.8.4. Consequences of the change to the partial cooldown rate

Following the SI signal actuated on a pressuriser pressure < MIN3 signal, the partial cooldown is automatically initiated to remove the energy from the primary side via the secondary side. Consequently the primary pressure is reduced which allows the MHSI to inject borated water.

As a consequence of increasing the cooldown rate, the MHSI delivery pressure of 85 bar will be reached earlier due to a more rapid reduction in primary pressure. Therefore, the cooldown rate increase from -100°C/h to -250°C/h will have a beneficial; effect on the mitigation of the small break LOCA accident.

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

Thus, it can then be concluded that all the decoupling criteria will be met.

3.8.5. Conclusion

The EPR_{4900} results for the event “SB-LOCA without LHSI” also cover the conditions applicable for the EPR_{4500}. This conclusion is justified by two facts:

- Very similar plant conditions with slight advantage for the EPR_{4500} plant status (power level, relatively higher RIS [SIS] efficiency).
- Very conservative assumptions for IRWST heat supply and removal in EPR_{4900} analysis.

It can be concluded that the final state; i.e.

- achieving long term core sub-criticality,
- ensuring that the residual heat is removed,
- controlled radioactive releases due to the integrity of radiological barriers,

is reached and the corresponding acceptance criteria (maximum temperature of the cladding < 1200°C, pressure of the containment < 5.5 bar and temperature of the IRWST < 100°C) are met.

3.8.6. System Sizing

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

SECTION 16.1.3.8 - TABLE 1

Major Phenomena and Qualification of the Models used in the Computer Codes S-RELAP5 and COCO

No.	SBLOCA Phenomena	Code Models	Test Facilities
1	Critical Break Flow	subcooled choking and two-phase choking model	Marviken Test 22 and 24 [Ref-1]
2	Forced Loop Flow	centrifugal pump performance model	1-½ Loop Model Semiscale [Ref-1]
3	Coolant Pump Coast down	centrifugal pump performance model	EPRI Pump Two-phase Degradation Test [Ref-1]
4	Natural Circulation	constitutive models	PKL IIIB 3.2A [Ref-1]
5	Pump Loop Seal Clearance	constitutive models	UPTF TRAM A5 [Ref-1]
6	Flashing and Boiling	fluid field equations and heat transfer models	ORNL THTF Mixture Level Swell [Ref-1]
7	Emergency Core Cooling System Injection	constitutive model	UPTF TRAM A6 [Ref-1]
8	Steam Generator Heat Transfer	fluid field equations and heat transfer models	PKL IIIB 3.2A [Ref-1]
9	Phase Separation and Reflux Condenser	fluid field equations and heat transfer models	PKL IIIB 3.2A
10	Pressure and Temperature Evaluation in the Containment Atmosphere and Sump	lumped parameter model with mass, volume and energy balance	Review of Heat Transfer Coefficients for Condensing Steam in a Containment Building Following a Loss-of-Coolant Accident [Ref-1]

SECTION 16.1.3.8 - TABLE 2

Initial Conditions, 20 cm² (∅ 50 mm) CL break with loss of LHSI (At Power) for EPR 4900 calculation

Parameters	Values used
Reactor coolant system	
Initial reactor power Initial average RCP [RCS] temperature Initial reactor coolant pressure RPV coolant flow PZR water level	100% of nominal power 311.2°C (nominal) 155 bar (nominal) 22240 kg/s (thermo-hydraulic flow rate) 7.0 m (nominal)
Steam generator	
Initial steam pressure Initial SG level	74.2 bar (nominal) 16.0 m (nominal)
Feedwater	
Main feedwater flow Initial ARE [MFWS] temperature	100% of nominal flow 230°C (nominal)
Containment	
Free volume IRWST water volume Initial IRWST water temperature EVU [CHRS] capacity per train	80000 m^3 1300 m^3 42°C 13 MW

SECTION 16.1.3.8 - TABLE 3

Relevant Differences between EPR 4500 and EPR 4900

Parameter	EPR 4900	EPR 4500	Comment
Initial plant data			
Thermal reactor power (MW)	4900	4500	
Thermal SG power (MW)	1235	1131	
Total RCP [RCS] mass flow (kg/s)	22240	22225	
Avg. RCP [RCS] temperature (°C)	311.2	312.8	
PZR pressure (bar)	155	same	
PZR level (m)	7.0	6.9	
ARE [MFWS]/MS flow rate / SG (kg/s)	695	639	
ARE [MFWS] temperature (°C)	230	same	
SG water level (m)	16.0	15.7	
SG water mass (ton)	87.7	80.3	
SG heat transfer area (m^2)	8171	7960	
Pressure in SG (bar)	74.2	78	
Important setpoints			
Reactor trip (bar) from low PZR pressure	135	same	Nominal value
RIS [SIS] signal (bar) from low PZR pressure	115	same	Nominal value
Accumulator injection at RCP [RCS] CL pressure (bar)	45	same	minimum pressure
VDA [MSRT] opening at …..bar	93	95.5	PC always to 60 bar
GCT [MSB] opening at …..bar	87	90	PC always to 55 bar
System characteristics			
MHSI head (bar) & capacity	80 (min)	85 (min)	But same capacity
Accumulator water (m^3)	32.5	same	Nominal value
VDA [MSRT] capacity (% of FP)	about 50%	same	
GCT [MSB] capacity (% of FP)	about 50%	same	
AAD [SSS] & ARE [MFWS] capacities			Same power-related capacities
IRWST water volume (m^3)	1900	same	Nominal value
Containment free volume (m^3)	80000	same	Nominal value
EVU [CHRS] capacity (MW) and cooling mode	about 13.0 direct cooling	Same by containment spray	Per train at 90°C in IRWST
Containment structural masses & heat transfer areas			No relevant differences

3.9. UNCONTROLLED DROP IN THE PRIMARY LEVEL WITHOUT SI SIGNAL FROM THE REACTOR PROTECTION SYSTEM (IN STATE C3 OR D)

3.9.1. Introduction

The event sequence considered is the uncontrolled level drop during RRA [RHRS] operation at a low RCP [RCS] level (¾ loop operations) in state C3 or D.

Mid-loop (¾ loop) operation is required to reduce the inventory in the upper plenum of the RCP [RCS], and especially in the U-tubes of the steam generators during plant start-up. It is also used to drain the pressuriser and to purge the head of the RPV with nitrogen before RPV head removal during plant shutdown (state C3 and D), or to maintain water level below a maximum level in the RCP [RCS] during maintenance of the Steam Generators or RCP [RCS] pumps.

This uncontrolled RCP [RCS] level drop may be caused by problems related to the level measurements and monitoring.

3.9.2. Description of the Event Sequence

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

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

For these cases, a conservative draining flow rate of approximately 25 l/s is credited.

Compared to section 16 of Sub-chapter 14.3, this section deals with possible actions following failure of the loop level measurements or the protection system signals and consequently the failure of the “RIS [SIS] actuation on RCP [RCS] loop level” mitigation.

3.9.3. Description of the Initial State

State C3 is defined as cold shutdown with three RIS/RRA [SIS/RHRS] trains in operation (RCP [RCS] temperature < 55°C, RCP [RCS] Pressure = 1 bar). The RCP [RCS] is partially open and can be rapidly re-closed so that the SG can be used again for residual heat removal. The RCP [RCS] is at ¾ loop operation level. The automatic protection function of RCP [RCS] loop level is activated (< MIN 1). RIS [SIS] is only performed by the MHSI trains.

State D is defined as cold shutdown with three RIS/RRA [SIS/RHRS] trains in operation (RCP [RCS] temperature < 55°C, RCP [RCS] Pressure = 1 bar). The RCP [RCS] is open to containment so that the SG cannot be used for decay heat removal. The RCP [RCS] level can be at ¾ loop or above.

Also in State D, the automatic protection function regarding RIS/RRA [SIS/RHRS] is the RIS [SIS] actuation on RCP [RCS] loop level < MIN 1. In this state, RIS [SIS] is only performed by MHSI. No automatic actuation of the LHSI trains for RIS [SIS] is considered.

The decay heat removal is performed by three RIS/RRA [SIS/RHRS] trains, with the fourth train in stand-by for manual actuated safety injection operation.

The letdown flow is routed via the low-pressure reduction station, which is connected to the two discharge points downstream of the RIS/RRA [SIS/RHRS] heat exchangers (train 3 and 4). In this plant shutdown state, the RCV [CVCS] pumps are normally stopped and the purification flow re-injection into the RCP [RCS] is ensured by the [RIS/RRA] SIS/RHRS train 3 or 4 via the bypass line of the RCV [CVCS] pumps.

The MHSI pumps (train 1 to 4) are in stand-by for safety injection. Their delivery head is reduced to 40 bar before start up of RHR operation by opening of the dedicated large minimum flow line.

Preventive maintenance is not considered during this operation phase.

3.9.4. MHSI Make-up Actuation by the Protection System (F1A)

In states C3 and D, the actuation of the MHSI make-up is generally performed by the signal "Loop Level < MIN1". This signal initiates the RCPB isolation: the letdown line isolation valves are closed by the protection system signal "Loop Level < MIN1" signal. This performs the RCV [CVCS] letdown line isolation, which stops the draining before reaching RIS/RRA [SIS/RHRS] pump stop conditions. Additionally, the MHSI pumps are actuated and help to recover the RCP [RCS] inventory.

3.9.5. Additional Means of Automatic MHSI Make-up Actuation

In the PSA assessment for shutdown accidents, the failure of the loop level measurements is identified as a significant contributor to the frequency of occurrence of uncontrolled RCP [RCS] level drop.

Therefore, a different criterion for automatic RCPB isolation and MHSI make-up was required. In addition, the operators need additional information about the RCP [RCS] inventory – independent of the loop measurement – in order to enable them to initiate any RCP [RCS] make-up.

3.9.6. Substitution Criterion for Automatic MHSI Make-up Actuation

If the chemical and volume control system RCV [CVCS] fails to maintain a water level high enough in the reactor loops, the low reactor coolant system loop level threshold will eventually be reached but the safety injection signal is assumed not to be effective. However, a back-up signal actuating safety injection on a low loop level diverse signal, which doesn't rely on the same loop level measurements of the protection system, exists to cope with such events and it initiates the following actions:

- Isolation of the reactor coolant pressure boundary including closure of the RCV [CVCS] letdown lines isolation valves.
- Starting of the medium head safety injection pumps, their large and small mini-flow lines being open which limits the delivery head of the pumps;

3.9.7. Reaching Safe Shutdown state

Actuation of the safety injection system is triggered early enough to prevent any vortex formation at the RRA [RHRS] suction line. Indeed, as soon as the MHSI inject, the primary inventory increases again, this guarantees that the loop level is high enough for RHR operation. The RHR operation is therefore not interrupted.

In state C3, the operation of the safety injection system will induce a maximum pressure rise up to the delivery head of the pumps.

In state D, the reactor coolant system being open and not pressurised, the reactor pool water level will increase because of medium head safety injection. Decay heat is removed via residual heat removal system operation.

One MHSI train is sufficient to recover the primary inventory.

Further into the transient, the safe shutdown state is reached: heat removal is maintained via the operation of residual heat removal system, the core is sub-critical and core coolant inventory is stabilised or increasing as a result of medium head safety injection.

3.9.8. Conclusion

The consequences of an uncontrolled level drop are mitigated by the RCPB isolation and MHSI make-up.

A back-up signal actuating safety injection on a low loop level diverse signal, which doesn’t rely on the same loop level measurements of the protection system provides another improvement in accident mitigation in shutdown states.

3.9.9. System Sizing

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

3.10. NON RCV [CVCS] HOMOGENEOUS DILUTION WITH FAILURE OF DILUTION SOURCE ISOLATION BY THE OPERATOR (STATES C_B AND D)

3.10.1. Identification of causes

The RRC-A event to be considered is a non-VCT (Volume Control Tank) isolatable homogeneous dilution, of 20 te/h, and failure of dilution source isolation by the operator, in states C (with depressurised RCP [RCS]) and D. This RRC-A event is equivalent to the PCC-4 event: "Boron dilution due to a non-isolatable rupture of a heat exchanger tube (States C to E)" (see section 12 of Sub-chapter 14.5) for which no operator error is taken into account.

The main cause of a non-VCT isolatable homogeneous dilution is a reverse leak through the damaged tubes of a heat exchanger in the Component and Cooling Water System (RRI [CCWS]). The maximum continuous dilution flow retained for the analysis is 20 te/h (rounded-off value).

3.10.2. Description of the initial state

State C with depressurised RCP [RCS] is defined as cold shutdown with three RIS/RRA [SIS/RHRS] trains in operation (RCP [RCS] temperature ≈ 55°C and Primary pressure of 1 bar). The RCP [RCS] is partly open but it can be rapidly re-closed so that the SG can be used for residual heat removal again. The RCP [RCS] is at ¾ loop operation level (also called mid-loop operation).

State D is defined as cold shutdown with three RIS/RRA [SIS/RHRS] trains in operation (RCP [RCS] temperature at 55°C and Primary pressure at 1 bar). The RCP [RCS] is open so that the SG cannot be used for decay heat removal. The RCP [RCS] level can be at mid-loop or above.

The decay heat removal is performed by three RIS/RRA [SIS/RHRS] trains: one train is in standby for manual actuated safety injection/RHR operation.

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

Preventive maintenance is not considered during this operation phase.

The initial boron concentration is the initial and minimal IRWST (In-containment Refuelling Water Storage Tank) boron concentration (see Sub-chapter 14.1): 2405 ppm for UO_2 fuel management schemes and 2600 ppm for MOX fuel management scheme. This data refers to natural boron. The bounding UO_2 management scheme is the "UO_2 In Out 18 months" scheme.

3.10.3. Description of the event sequence

The homogeneous dilution leads to a decrease in the boron concentration in the whole reactor coolant system.

Alarms due to abnormal or frequent make up to RRI [CCWS] or low RRI [CCWS] tank level can inform the operator but are not taken into account for the analysis.

The "Anti-dilution in shutdown conditions with RCP not in operation" protection channel, which uses boron meter information (see section 13 in Sub-chapter 14.3), ensures both an automatic F1A RCV [CVCS] isolation downstream of the VCT and an automatic F2 opening of the IRWST (In-containment Refuelling Water Storage Tank) suction valves of the RCV [CVCS] pumps. As no RCV [CVCS] pump is running, there is no boration of the reactor coolant system. However, following this action, the operator must be informed and emergency operating procedures provide necessary information to isolate the dilution source. In order to take account of system configurations where the "bypass line of the RCV [CVCS] pumps and the volume control tank" may not be in operation, no credit is taken from this information.

Before reaching criticality, the shutdown high neutron flux alarm (F1A) is actuated. The core remains sub-critical for at least 30 minutes after the alarm actuation. The operator, acting according to the instructions of the emergency procedures, is supposed to initiate boration by starting the RBS [EBS] to ensure that the core remains sub-critical. The source of dilution is then assumed to be isolated either with a manual action from the main control room, at the earliest 30 minutes after the alarm actuation, or by a local manual action outside the main control room, at the earliest one hour after the alarm actuation. For this RRC-A event, the failure of these operator actions is assumed.

The decrease in the boron concentration of the reactor coolant continues until the critical boron concentration value in cold shutdown state is reached. The flux increase is stopped by the actuation of the specific RRC-A signal: RBS [EBS] actuation on fixed high neutron flux initiated by the Source Range Detectors (see Section 16.1.3.10 - Figure 1).

This boration is sufficient to recover sub-critical conditions rapidly, to compensate the dilution flow and to increase the boron concentration in the entire reactor coolant system. After the draining of the RBS [EBS] tank, the dilution again leads to a decrease in the boron concentration in the entire reactor coolant system.

Finally, four hours after the high neutron flux alarm (F1A) actuation, the core sub-critical is assured. It is further assumed that the operator isolates the dilution source as no operator errors are postulated when the action take place more than 4 hours after the event occurrence.

3.10.4. Main assumptions

The maximum continuous dilution flow retained for the analyses is 5.5 kg/s (20 te/h).

The required volume of borated water in the reactor corresponds to 120 m^3 (bounding case) at mid-loop level.

The decrease in boron concentration in the reactor coolant is monitored by the RCV [CVCS] boron meters when the reactor coolant flow is through the bypass line of the RCV [CVCS] pumps and the volume control tank. These boron meters are located on the RCV [CVCS] charging line downstream of the connection with the RCV [CVCS] bypass line.

The High neutron flux (alarm) setpoint is assumed to be a flux value equal to three times the current flux in the initial shutdown conditions.

The high neutron flux setpoint of the RRC-A signal is assumed to be fixed, defining a critical condition, which ensures that there is no significant power excursion i.e. no temperature and pressure increase. For the present analysis, a decrease in boron concentration of 50 ppm is assumed between the criticality and the RRC-A signal actuation.

Conservatively, only one RBS [EBS] train is taken into account. The boration flow rate is 10 m^3/h and the volume of one RBS [EBS] tank is 27 m^3 (see Chapter 6). The boron concentration is 11200 ppm natural boron. The delay taken into account between RBS [EBS] actuation and RBS [EBS] effective boration is 2 minutes.

No operator errors are postulated when the action takes place more than 4 hours after the event occurrence

3.10.5. Results and conclusions

The bounding fuel management scheme for this analysis is the "UO2 In Out 18 months" scheme at Beginning Of life. For this fuel management scheme, the critical boron concentration is 1290 ppm (see section 12 of Sub-chapter 14.5).

The boron concentration during the transient is obtained using the formula:

$$BC(t) = BC_{inj} - \left(BC_{inj} - BC_{t=0}\right) \times EXP\left(-\frac{D}{M} \times t\right)$$

Where M is the primary water mass, D is either dilution flow or dilution flow plus the boration flow and BC_{inj} is its boron concentration (BC_{inj} is equal to zero for a dilution).

The progression of the transient is as follows, where the time is in minutes after actuation of the shutdown high neutron flux alarm (F1A):

Time (min)	BC (ppm)	Status
- 125	2405	Beginning of dilution
- 104	2255	VCT isolation
0	1662	F1A Alarm
86	1290	Core just critical
101	1233	RBS[EBS] effective boration
106	1290	Core just sub-critical
240	2378	Isolation of the dilution

Note: The maximum time to isolate the dilution after the F1A alarm actuation is 489 minutes (8 hours and 9 minutes). The RBS [EBS] tank will be empty after 4 hours and 23 minutes.

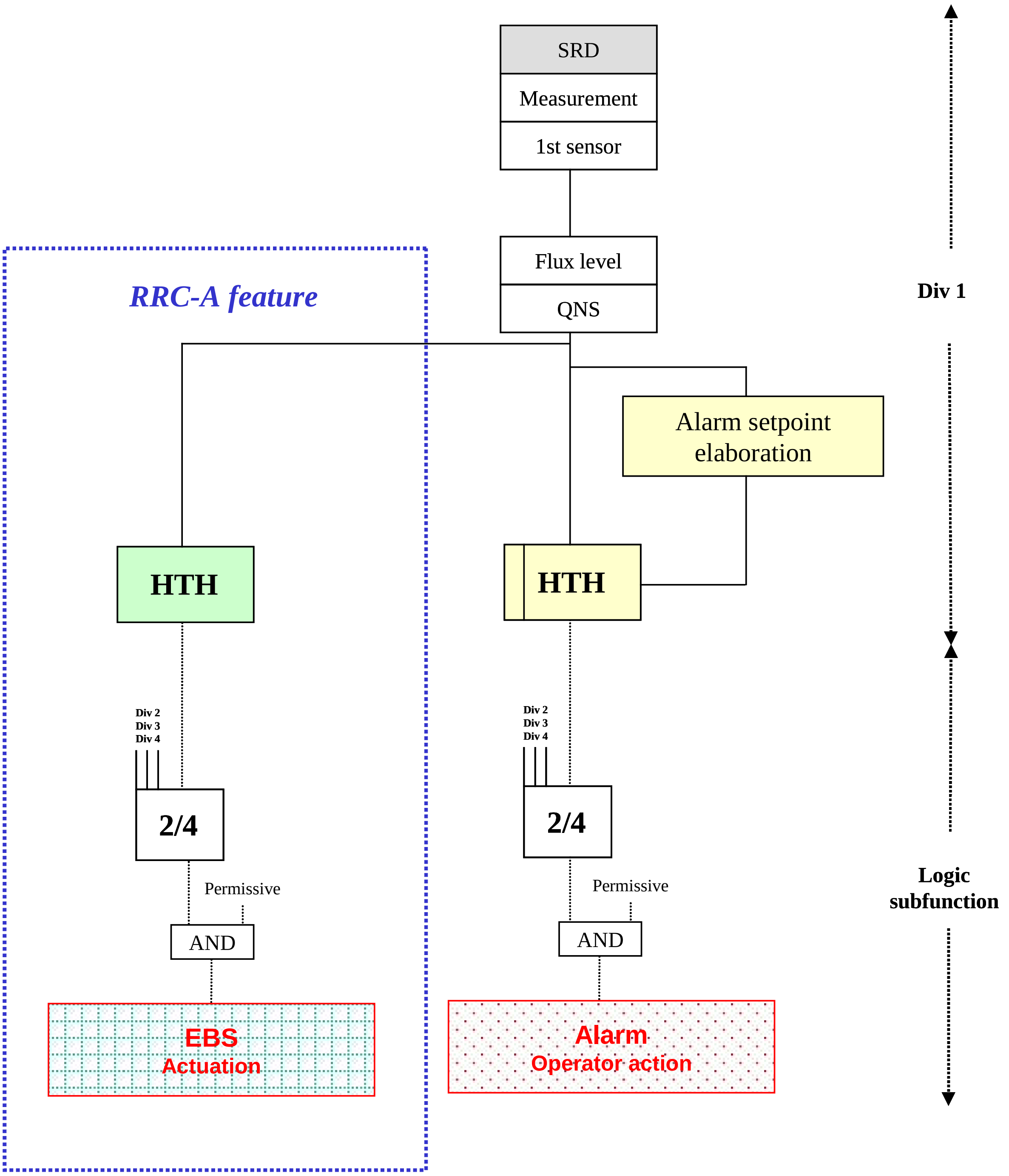
Following a "non VCT isolatable homogeneous dilution and failure of dilution source isolation by operator in states C_B and D", the RBS [EBS] boration is initiated by the "high neutron flux (Source range)". The specific RRC-A feature - F2 signal using F1A Source Range Detectors and F1A RBS [EBS] – returns the core to sub-critical conditions and maintains sub-criticality until the isolation of the dilution source by the operator. This reduces the frequency of global core melt.

3.10.6. System sizing

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

SECTION 16.1.3.10 - FIGURE 1

High Neutron Flux Alarm (Source Range) Channel and RRC-A RBS [EBS] Actuation on High Neutron Flux



3.11. TOTAL LOSS OF COOLING CHAIN (IN STATE D)

3.11.1. Introduction

This RRC-A event is equivalent to the total loss of the RRI/SEC [CCWS/ESWS] cooling water system. The main consequence of this event is the loss of the following systems used to cool the RCP [RCS]:

MHSI, RIS/RRA [SIS/RHRS] (partially), fuel pool cooling system (partially), and RCV [CVCS]

3.11.2. Total loss of component cooling water systems (state D)

3.11.2.1. Identification of the causes and description of the accident

The total loss of the RRI/SEC [CCWS/ESWS] is assumed to occur during the cold shutdown in state D, during which:

- generally, three out of four LHSI/RHR trains are operating in residual heat removal mode,
- the RCP [RCS] is at atmospheric pressure and is open to containment atmosphere,
- the temperature of the primary coolant is 55°C,
- the RCP [RCS] water inventory corresponds to 3/4 loop operation,
- the decay power after 24h is 27 MW (about 0.6%)

In these conditions it is not possible to restore decay heat removal via the secondary side. The core heat removal must therefore be by evaporation and makeup of the primary coolant.

The final state of the reactor is reached after start-up of both LHSI pumps with diverse cooling to provide the required makeup.

The containment, which acts as a heat sink during this event, can be cooled separately if required by the EVU [CHRS] with its independent cooling system.

3.11.2.2. Accident Mitigation

3.11.2.2.1. Detection

The accident is detected by the operator by several indications and measurements.

They include:

- The Loss of the RRI [CCWS] function is indicated by the automatic shutdown of the pumps.

- Shutdown of the LHSI pumps is automatically activated by the operational instrumentation and control systems on detection of high oil, seal or motor temperatures.
- In addition to alarms preceding automatic pump shutdown, the main operating parameters show abnormal primary temperature and loop level measurements to the operators in the control room.

3.11.2.2.2. Sequence of events

After loss of the cooling chain, the removal of the residual heat from the primary system is interrupted and the primary coolant temperature increases until the saturation temperature is reached at approximately 30 minutes.

Following the RCP [RCS] reaching the saturation temperature, the removal of decay heat occurs via evaporation of the coolant, which causes a drop in the loop level to below the actuation threshold of the safety injection system (MHSI) in about 15 minutes. This start-up of the MHSI pumps cannot be prevented and they operate for approximately 15 minutes until they stop due to insufficient cooling.

Without any other means of mitigation, the RCP [RCS] inventory would decrease progressively and cause the core to uncover. Containment will act as a heat sink for the decay heat power and its pressure gradually starts to increase.

3.11.2.2.3. Main actions

The required operator action involves starting up two LHSI/RHR pumps provided with diverse cooling to perform long-term makeup and to ensure the core is always covered with at least a two-phase mixture. In order to achieve this, the operator must ensure that the LHSI pumps take suction from the IRWST and that the hot leg suction line is closed. Due to the availability of both LHSI/RHR trains, long-term heat removal (up to 24 hours) from the RCP [RCS] is achieved with a large margin because the LHSI/RHR capacity is much greater than the capacity required to remove residual heat. The water inventory of the IRWST is also more than enough to provide cooling water for 24 hours.

Assuming that the operators reactivate the two LHSI/RHR trains with diverse cooling 30 minutes after the occurrence of the initiating event, start-up of the MHSI pumps on low loop level may even be prevented.

The operation of the two LHSI pumps (with capacity 450 m^3/h each) leads to a fast filling of the RCP [RCS]. As a result, the normal inventory of approximately 350 m^3 is reached within approximately 30 minutes, i.e. practically about 1 hour after the event initiation. Complete filling, including the reactor pool (approximately 1600 m^3), would only take approximately 1.6 hours. Once the desired RCP [RCS] inventory has been reached, the LHSI pumps are stopped.

The operation of the LHSI pumps for a maximum 1.6 hours is not compromised by the increase in IRWST water temperature. Assuming an average IRWST inventory of 1100 m^3 and no heat storage for any structural masses, the IRWST water temperature would increase by about 35°C (i.e. the LHSI/RHR pump operation is not jeopardised).

The continuous heat addition to the containment through evaporation is acceptable in terms of the pressure increase over 24 hours, since the pressure remains far below the containment design pressure. The EVU [CHRS], which also has a diverse cooling supply, is available to be activated for limiting the pressure increase and additional heat removal system.

As soon as the component cooling water system is re-established (which is assumed to occur after 24 hours of total loss of the RRI/SEC [CCWS/ESWS]), the normal residual heat removal mode can be restored, resulting in a rapid return to normal cooling conditions in the RCP [RCS] and an end to steam release into the containment.

3.11.2.3. Conclusions

Accidental loss of RRI/SEC [CCWS/ESWS] in state D is mitigated by the following dedicated requirements:

- Two trains of the LHSI/RHR, equipped with diverse cooling systems, are available for activation by the operator based on unambiguous signals.
- The capacities of the two LHSI/RHRS trains, including the IRWST water inventory, are sufficient to keep the core covered over the 24h even with RRI/SEC [CCWS/ESWS] system unavailable.
- Containment integrity is assured over the 24-hour accident period, even without taking into account possible heat removal by systems such as EVU [CHRS].

Note that the operator action of starting the two LHSI/RHRS trains with diverse cooling which is assumed to occur approximately 30 minutes after the event might be delayed for 3 to 4 hours after the start of the event while still preventing the core uncovery. This large additional margin arises from the following assumptions:

- Water supply by two MHSI pumps only during a period of about 8 minutes (actuation from low loop level).
- Reduction of the RCP [RCS] inventory to a level where core cooling is ensured by at least two-phase mixture.

The plant is kept in a safe state during and after a postulated total loss of cooling chain (TLOCC) in state D since:

- Sub-criticality is already ensured prior to the accident.
- Heat removal from the core is maintained with large margins without SG.
- Activity release is within the PCC-4 limits since the core remains intact and the containment integrity is never compromised.

3.11.3. System Sizing

EVU [CHRS] Sizing

The EVU [CHRS], which has a diverse cooling supply, is available to be activated for limiting the containment pressure increase and as an additional heat removal system. The sizing of this system should be such that, if activated, it should meet the acceptance criteria for this RRC-A event.

3.12. TOTAL LOSS OF COOLING CHAIN OR ULTIMATE HEAT SINK FOR 100 HOURS (STATE A TO C)

3.12.1. Accident description

This sequence combines the total loss of the RRI/SEC [CCWS/ESWS] or the total loss of the ultimate heat sink in states A to C without leakage from the primary pump seals, coupled with a total loss of feedwater from the steam generators (normal supply) over 100 hours.

The related RRC-A feature is the re-supply to the ASG [EFWS] tanks (see Sub-chapter 6.6).

In the event of loss of the heat sink in states A to C, the residual heat is removed automatically after reactor trip by the main steam relief trains (VDA [MSRT]) (or the SG safety valves) and by feeding of the SG assured by the ASG [EFWS]. In order for the ASG [EFWS] system storage capacity to be sufficient for the removal of the residual heat over 100 hours, it is necessary to re-supply the ASG [EFWS] tanks.

3.12.2. System sizing

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

3.13. LOSS OF THE TWO MAIN TRAINS OF THE FUEL POOL COOLING SYSTEM DURING SHUTDOWN FOR REFUELLING (STATE F) / STATION BLACKOUT

3.13.1. Loss of the two main trains of the fuel pool cooling system during shutdown for refuelling (state F)

3.13.1.1. Introduction

In the RRC-A event “Loss of two trains of the fuel pool cooling system during shutdown for refuelling", the two main trains are considered lost from the beginning of the transient (initiating event). In this situation, it is possible to cool the fuel pool by using the additional train (third PTR [FPCS] train), that has been designed for this case.

The principal functional flow diagram of the PTR [FPCS] and general information on the transients associated with the pool are presented in Sub-chapter 14.3.

3.13.1.2. Situation before the accident (normal operation – PCC-1)

In normal operation in state F, two main PTR [FPCS] trains (with one pump in service per train) are used to cool the fuel pool. Each PTR [FPCS] heat exchanger is cooled by an RRI [CCWS] train.

The thermal load that must be removed from the fuel pool is at its maximum just after the last fuel element has been discharged from the reactor and placed in the fuel pool.

3.13.1.3. Boundary conditions

As for any other RRC-A event, the transient is analysed with best-estimate assumptions.

Loss of offsite power (LOOP), single failure, preventive maintenance and earthquakes are not considered in analysing RRC-A events.

Following the best-estimate approach, residual heat is calculated without including an uncertainty margin.

3.13.1.4. Transient

At the beginning of the transient, both the main PTR [FPCS] trains are assumed to be simultaneously lost.

The PTR [FPCS] water temperature starts to rise. The third PTR [FPCS] train is started up manually, at the latest when the fuel pool water temperature reaches 95°C.

Since the third PTR [FPCS] train and its cooling system are designed for a fuel pool outlet temperature of 95°C (see section 3 of Sub-chapter 9.1), the PTR [FPCS] water temperature will thus not exceed this value. At this temperature, cooling of the fuel pool may be continued indefinitely. In the long term, the temperature will be stabilised below 80°C (see section 3.2 of Sub-chapter 9.1).

In reality, in order to limit the temperature increase in the fuel pool and to achieve a safety margin, the third PTR [FPCS] train would be started up manually from the Control Room shortly after the loss of both the main trains. Work to recover at least one of the two main trains would be started at the same time.

3.13.1.5. Conclusion

Given that the fuel pool water temperature does not exceed 95°C and is stabilised in the long term below 80°C, the decoupling criteria are met for this RRC-A event.

3.13.2. Station blackout (SBO) (PTR [FPCS] aspects)

3.13.2.1. Introduction

In the event of an SBO, the power supply to both PTR [FPCS] main trains is lost as they are supplied from the normal electric distribution and the main diesel generators. As a result, only the third PTR [FPCS] train can be used for cooling the fuel pool water as this train is backed up by an ultimate emergency diesel generator in states D, E and F.

The principal functional flow diagram of the PTR [FPCS] and general information on the transients associated with the pool are presented in section 3 of Sub-chapter 9.1.

3.13.2.2. Situation before the accident (normal operation – PCC-1)

States E and F correspond to a shutdown for refuelling. In normal operation, two main PTR [FPCS] trains (with one pump per train) are used to cool the fuel pool. Each PTR [FPCS] heat exchanger is cooled by an RRI [CCWS] train.

The thermal load that must be removed from the fuel pool is at its maximum just after the last fuel element has been discharged from the reactor and placed in the fuel pool.

3.13.2.3. Boundary conditions

As for any other RRC-A event, the transient is analysed using best-estimate assumptions.

LOOP, single failure, preventive maintenance and earthquakes are not considered in analysing RRC-A events.

Following the best-estimate approach, residual heat is calculated without including an uncertainty margin.

3.13.2.4. Transient

At the beginning of the transient, both the main PTR [FPCS] trains are simultaneously lost following the station blackout.

Given that all cooling is lost, the PTR [FPCS] water temperature starts rising.

The third PTR [FPCS] train and its cooling system are started up manually from the control room, at the latest when the fuel pool temperature reaches 95°C.

Since the third PTR [FPCS] train, its cooling system and the required support systems, are designed for a fuel pool outlet temperature of 95°C (see Sub-chapter 9.1, section 3), the PTR [FPCS] water temperature will not exceed this value. Fuel pool cooling is maintained at this temperature, for the entire duration of the station blackout (SBO). In the long term, the temperature of the PTR [FPCS] will be stabilised below 80°C (see section 3.2 of Sub-chapter 9.1).

When the power supply is restored (via the electrical network or the emergency diesel generators), assumed at the latest 24 hours after the station blackout (SBO), fuel cooling may be continued using the two main PTR [FPCS] trains.

3.13.2.5. Conclusion

Given that the fuel pool water temperature does not exceed 95°C and, in the long term, is stabilised below 80°C, the acceptance criteria are met for the station blackout condition (SBO), i.e. for a loss of the electrical network combined with the loss of four emergency diesel generators.

4. RADIOLOGICAL CONSEQUENCES OF RRC-A SITUATIONS

4.1. SAFETY REQUIREMENTS

4.1.1. Safety Objectives

The objective of the radiological consequence studies of RRC-A operating conditions is to demonstrate that, following these transients, the release of radionuclides outside the installation will have very low consequences for members of the public.

4.1.2. Radiological objectives

The same radiological objectives as the PCC-4 type operating conditions are associated with RRC-A operating conditions. These objectives, as well as the doses that are associated with them, are presented in Chapter 14.

4.1.3. Requirements linked to the design

The study of the radiological consequences of RRC-A accidents must demonstrate that the criteria mentioned in the previous paragraph are met. It thus contributes to verification of the design of the plant.

4.2. ANALYSIS OF THE RADIOLOGICAL CONSEQUENCES

In the analysis of the RRC-A transients presented in the previous sections of this Sub-chapter 16.1, except for the section on the loss of spent fuel pool cooling, it is demonstrated that the PCC-4 accident decoupling criteria are met. The radiological consequences of such transients are therefore implicitly limited by the radiological limits associated with the PCC-4 conditions; a detailed evaluation of the radiological consequences is therefore not necessary.

For the RRC-A sequence associated with the loss of spent fuel pool cooling (loss of the two main cooling trains of the PTR [FPCS] in state F), it is demonstrated that the water in the pool does not reach the boiling point. The radiological consequences of this transient are negligible; no calculation of specific radiological consequences is required.

SUB-CHAPTER 16.1 – REFERENCES

External references are identified within this sub-chapter by the text [Ref-1], [Ref-2], etc at the appropriate point within the sub-chapter. These references are listed here under the heading of the section or sub-section in which they are quoted.

3. RRC-A SITUATION ANALYSIS

3.1. ATWS SIGNAL (RBS [EBS] ACTUATION) FOLLOWING ROD FAILURE

3.1.1. Excessive increase in secondary steam (opening of the GCT [MSB])

3.1.1.5. Methods of analysis

a) MANTA

[Ref-1] MANTA Code synthetic qualification assessment. NFPSD DC 85 Revision D. AREVA. September 2008. (E)

b) SMART and FLICA

[Ref-2] SCIENCE V2 Nuclear Code Package - Qualification Report. NFPSD DC 89 Revision A. AREVA. March 2004. (E)

[Ref-3] Qualification Report FLICA IIIF - Version 3. NFPSD DC 188 Revision A. AREVA. (E)

c) Coupling between the thermal-hydraulic and the neutronic codes

[Ref-4] Functional Validation of MANTA-SMART-FLICA Coupling. NEPD-F DC 10157 Revision A. AREVA. October 2008. (E)

3.1.1.6. Specific assumptions

3.1.1.6.5. Neutronic data

[Ref-1] S Laurent. Fuel Management – Neutronic Design Report (SCIENCE Calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

3.1.1.7. Results and conclusions

3.1.1.7.1. Results for EPR at 4250 MW

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSa “ATWS by Rods Failure [State A]”. Edition 2003. AREVA. (E)

3.1.2. ATWS "LOSS OF MAIN FEEDWATER"

3.1.2.6. Specific assumptions

3.1.2.6.1. Transients analyses / initial power states

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSa "ATWS by Rods Failure [State A]". Edition 2003. AREVA. (E)

[Ref-2] S Laurent. Fuel Management – Neutronic Design Report (SCIENCE Calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

3.1.2.6.4. Neutronic data

[Ref-1] S Laurent. Fuel Management – Neutronic Design Report (SCIENCE Calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

3.1.2.7. Results and Conclusions

3.1.2.7.1. Results for EPR at 4250 MWth

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSa "ATWS by Rods Failure [State A]". Edition 2003. AREVA. (E)

3.1.3. ATWS - Loss of Offsite Power (LOOP)

3.1.3.6. Specific assumptions

3.1.3.6.4. Neutronic data

[Ref-1] S Laurent. Fuel Management – Neutronic Design Report (SCIENCE Calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

3.1.3.7. Results and Conclusions

3.1.3.7.1. Results for EPR 4250 MWth

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSa "ATWS by Rods Failure [State A]". Edition 2003. AREVA. (E)

3.1.4. ATWS – RCV [CVCS] malfunction that leads to a decrease in the boron concentration of the primary coolant

3.1.4.6. Specific assumptions

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSb "ATWS by PS Failure [State A]". Edition 2003. AREVA. (E)

[Ref-2] EPR Preliminary Safety Analysis Report, Sub-chapter 15.1 “Plant characteristics assumed in the accident analyses”. Edition 2003. EPRR DC 1693. AREVA. (E)

3.1.4.7. Results and Conclusions

3.1.4.7.1. Results for EPR 4250 MWth

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSa “ATWS by Rods Failure [State A]”. Edition 2003. AREVA. (E)

3.1.5. ATWS – Uncontrolled RCCA bank withdrawal

3.1.5.2. Sequence of events

[Ref-1] EPR Preliminary Safety Analysis Report, Sub-chapter 15.1 “Plant characteristics assumed in the accident analyses”. Edition 2003. EPRR DC 1693. AREVA. (E)

3.1.5.6. Specific assumptions

3.1.5.6.4. Neutronic data

[Ref-1] S Laurent. Fuel Management – Neutronic Design Report (SCIENCE Calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

3.1.5.7. Results and conclusion

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSa “ATWS through Mechanical Locking of the Rods (State A)”. Edition 2006. AREVA. (E)

3.2. ATWS BY RPR [PS] FAILURE (STATE A)

3.2.4. ATWS – RCV [CVCS] malfunction leading to a decrease in the boron concentration of the primary coolant

3.2.4.5. Specific assumptions

3.2.4.5.1. Neutronic data

[Ref-1] S Laurent. Fuel Management – Neutronic Design Report (SCIENCE Calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

3.2.4.5.2. Assumptions concerning the dilution and boration calculations

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSb “ATWS by PS Failure [State A]”. Edition 2003. AREVA. (E)

[Ref-2] EPR Preliminary Safety Analysis Report, Sub-chapter 15.1 “Plant characteristics assumed in the accident analyses”. Edition 2003. EPRR DC 1693. AREVA. (E)

3.2.4.6. Results and conclusion

3.2.4.6.1. Results for EPR 4250 MW

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSb “ATWS by PS Failure [State A]”. Edition 2003. AREVA. (E)

3.2.5. ATWS – Uncontrolled RCCA bank withdrawal

3.2.5.6. Specific assumptions

3.2.5.6.4. Neutronic data

[Ref-1] S Laurent. Fuel Management – Neutronic Design Report (SCIENCE Calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

3.2.5.7. Results and conclusion

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSb “ATWS through Protection System Failure (State A)”. Edition 2006. AREVA. (E)

3.3. STATION BLACKOUT (SBO), IN STATE A

3.3.3. Results and conclusions

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSc “Total Loss of Offsite and Onsite Power Supply (State A)”. Edition 2006. AREVA. (E)

3.4. TOTAL LOSS OF FEEDWATER (STATE A)

3.4.4. Results

3.4.4.1. Results for EPR 4250 MW

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSd “Total Loss of Feedwater in State A (Failure of 1st PSV or 1 MHSI Pump)”. Edition 2003. AREVA. (E)

3.4.5. Conclusion for EPR at 4500 MW

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSd “Total Loss of Feedwater Supply to the Steam Generators (State A)”. Edition 2006. AREVA. (E)

3.5. TOTAL LOSS OF COOLING CHAIN LEADING TO A LEAKAGE ON RCP [RCS] PUMPS SEALS (STATE A)

3.5.3. Accident Mitigation

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSe “Total Loss of the Cooling Chain leading to an opening at the joints of the Primary Coolant Pumps (State A)”. Edition 2006. AREVA. (E)

3.6. LOCA (BREAK SIZE UP TO 20 CM^2) WITH FAILURE OF THE PARTIAL COOL-DOWN SIGNAL (STATE A)

3.6.3. Methods and assumptions

3.6.3.5. Thermal-hydraulic Analysis

3.6.3.5.2. Results

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSf “LOCA (Break Size up to 20 cm^2) with Failure of Partial Cooldown Signal, in State A”. Edition 2003. AREVA. (E)

[Ref-2] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSf “APRP (Opening smaller than 20 cm^2) with Failure of the Partial Cooling Signal (State A)”. Edition 2006. AREVA. (E)

3.7. LOCA (BREAK SIZE UP TO 20 CM^2) WITHOUT MHSI (STATE A)

3.7.4. Thermal-hydraulic Analysis

3.7.4.2. Results

3.7.4.2.1. Accident analysis for EPR at 4250 MW

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSg “LOCA (Break Size up to 20cm^2) without MHSI, in State A”. Edition 2003. AREVA. (E)

3.7.4.2.2. Accident analysis for EPR at 4500 MW

[Ref-1] EPR Preliminary Safety Analysis Report, Section 19.1.2 FSg “APRP (Opening smaller than 20 cm^2) without ISMP (State A)”. Edition 2006. AREVA. (E)

3.8. LOCA (BREAK SIZE UP TO 20 CM^2) WITHOUT LHSI (STATE A)

SECTION 16.1.3.8 - TABLE 1

[Ref-1] Topical report – Comparison SRELAP/CATHARE
Topical report – COCO containment code – model description and validation. NGPS1/2004/en/0507 Revision A. AREVA. October 2004. (E)