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APPENDIX 14C – ANALYSIS OF SINGLE FAILURE FOR MAIN STEAM LINE BREAK

A main steam line break in state A with a break size greater than the area equivalent to a diameter of 50 mm (20 cm²), is classified as a PCC-4 event. The objective of this analysis is to determine the most conservative single failure for this event in terms of DNBR. This single failure will then be systematically used in the PCSR analysis of a main steam line break in State A performed in section 2.1 of Sub-chapter 14.5.

1. IDENTIFICATION OF CAUSES AND GENERAL DESCRIPTION

1.1. GENERAL CONCERNS

The largest main steam line break is the double-ended guillotine break of a main steam line, named 2A-SLB.

The EPR main steam lines (MSL) are designed according to the break preclusion principle, in the area extending from the SG outlet to the MSL fixed point downstream of the VIV [MSIV]. As a consequence, the "2A-SLB upstream VIV [MSIV]" break is not considered as a PCC-4 event. Thus the largest steam pipe break size in the PCC-4 category corresponds to the "2A-SLB downstream VIV [MSIV]" break.

However, despite the break preclusion principle, the 2A-SLB break located at the SG outlet is considered in the present analysis for core behaviour. This is a conservative approach which combines the largest steam line break size and the most onerous location. This sequence is addressed in sub-section 4.1 of this appendix.

The following description of the accident progression mainly considers the 2A-SLB break, either the "2A-SLB upstream of the VIV [MSIV]" or the "2A-SLB downstream of the VIV [MSIV]". Both cases result in the uncontrolled depressurisation of only one SG as failure to close the VIV [MSIV] is considered a single failure in the second case. Particular features associated with smaller breaks are indicated when necessary.

Other cases may lead to the short term depressurisation of two SG. This can occur when the initiating event is combined with the single failure of one main steam relief control valve (MSRCV) to close after its opening on another SG. These cases are presented in sub-section 4.1 of this appendix, and a description of the transient progression is provided in sub-section 5.3 of this appendix. The transient behaviour is similar to that for the 2A-SLB break, with the same mitigation actions provided. The exception is the short term depressurisation of two SG until the isolation of the faulted VDA [MSRT]. This results in the uncontrolled cooldown of two RCP [RCS] loops compared to one, in the 2A-SLB. These cases are thus not specifically addressed in the following description.

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

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

The core may become critical and return to power following reactor trip. The effects of the accident on core behaviour are more significant if it is assumed that the most conservative rod cluster control assembly (RCCA) is stuck in its fully withdrawn position after reactor trip.

1.2. SEQUENCE OF EVENTS

1.2.1. From the initiating event to the controlled state

Once the break occurs, the secondary system depressurises. The SG pressure drop or low pressure signals actuate the RT if the SLB occurs at power. These signals also close all VIV [MSIV], close all ARE [MFWS] high-load lines, if they are initially open for operation above approximately 20% FP, and isolate the ARE [MFWS] low-load lines of the affected SG. All these actions are automatic and classified F1A.

After these isolations, only the affected steam generator, which experiences a non-isolatable SLB, continues to depressurise. The break is either upstream of the VIV [MSIV] or downstream of the VIV [MSIV] with failure to close of the VIV [MSIV]. This SG is supplied by the ASG [EFWS] which is automatically actuated on a low SG level signal.

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

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

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

The operator acts to isolate the ASG [EFWS] line connected to the affected SG, and actuate the RBS [EBS], or the RCV [CVCS] if available. This initiates an RCP [RCS] boration with 7000 ppm of enriched boron. Subsequently the core rapidly returns to subcritical conditions via these F1A classified actions):

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

The controlled state is then reached with:

- the core subcritical,

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

- the RCP [RCS] coolant inventory having been stabilised.

1.2.2. From the controlled state to the safe shutdown state

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

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

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

Confirmation of the isolation of the affected steam generator

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

- prevent an unacceptable increase of containment pressure and temperature, if the steam pipe break is located inside containment,

- prevent a complete draining via the break of the associated ASG [EFWS] tank.

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

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

RCP [RCS] boration

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

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

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

RCP [RCS] cooldown

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

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

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

RCP [RCS] depressurisation

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

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

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

1.3. PRECAUTIONS LIMITING THE EVENT CONSEQUENCES

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

2. SAFETY ANALYSIS

2.1. SAFETY CRITERIA

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

The effects of a steam line break are analysed with respect to the following:

- no degradation of the fuel cladding,
- reactor coolant pressure boundary,
- reactor containment,
- amount of radioactive products released.

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

2.2. DEFINITION OF CASES STUDIED FOR EACH SAFETY CRITERION

2.2.1. No degradation of the fuel cladding

The analysis is performed to demonstrate that the decoupling criterion of no core damage is satisfied, whatever single failure is considered. This criterion is no departure from nucleate boiling (DNB) after reactor trip.

Four cases are analysed, covering the most conservative uncontrolled cooldown scenarios which could result from either steam pipe break or steam leak flow, in the framework of PCC analyses. The definition of these 4 cases is given in sub-section 4.1 of this appendix. As discussed above, the "2A-SLB at the SG outlet" is the most limiting case, being considered in the present analysis. This is an over-conservative approach because of the break preclusion principle.

The phase from the initiating event to the controlled state is analysed in detail in the following sections, relying only on F1A means.

The demonstration of the ability to reach the safe shutdown state, relying only on F1A and F1B means is based on a qualitative argument. It is argued that the steam system pipe work failure is bounded by the feedwater line break analysed in section 3 of Sub-chapter 14.5, "Feedwater line break (States A, B)".

2.2.2. Reactor coolant pressure boundary

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

2.2.3. Reactor containment

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

2.2.4. Radiological effects

The safety criteria to be met are the dose equivalent limits for release to the atmosphere as discussed in Sub-chapter 14.6.

The bounding PCC-4 transient, for the radiological releases, is the SGTR (two tubes) discussed in section 10 of Sub-chapter 14.6.

3. METHODS AND ASSUMPTIONS

3.1. METHODS OF ANALYSIS

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

- the MANTA V3.7 code (see Appendix 14A) for the overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG),modelling F1 systems operation,

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

The analysis methodology is based on the following approach:

- identification of the dominant phenomena,

- verification of the adequacy of the code to simulate those phenomena,

- Application of conservative PCC analysis rules.

a) For the system transient (MANTA code)

The dominant phenomena of this transient are:

- SG blowdown,

- asymmetric RCP [RCS] ,

- limited mixing of loop flows within the reactor pressure vessel (RPV),

- RCP [RCS] depressurisation and pressuriser emptying.

All these phenomena are within the applicability range of the MANTA code. The qualification of this code is based on:

- the use of recognised and tested correlations, e.g. the Gros D'Aillon correlation for the critical flow rate through a pipe break under saturated conditions,

- the validation of specific models by test results from small-scale simulations. For example the model of minimum mixing between loop flows within the RPV, relies on conservative data derived from a STAR-CD code calculation. This code is described in Appendix 4 of the PCSR. These assumptions are more conservative than typical data representative of current four-loop plants,

- the overall verification of the code by simulation of PWR plant transients. These include the opening of a VDA [MSRT] on the PALUEL 3 four-loop plant which was accurately simulated with MANTA. The calculated loop temperatures match well with measured values, thus validating the entire RCP [RCS] hydraulic calculation. The SG secondary side depressurisation transient calculated by MANTA closely agreed with the measured values. This therefore demonstrates the applicability of the secondary side model in these conditions [Ref-1].

The transient analysis relies on the application of the conservative PCC analysis rules discussed in Sub-chapter 14.0. Part of these rules requires the inclusion of deterministic pessimism in all relevant boundary conditions for the decoupling criteria under consideration. These pessimisms address at least:

- the characterisation of the initiating event, maximising the resulting impact,
- the plant initial conditions, control dead band limits, maximal measurement uncertainties,
- the efficiency of the protection and mitigation actions, maximum uncertainty on each I&C measurement and signal delay, and on each system response time and capacity.

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

b) For the core transient (SMART/FLICA codes)

- Qualification of the SMART code in steam line break conditions relies mainly on EPICURE experiments [Ref-2]. 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.
- Critical heat flux correlation used in FLICA was developed on an experimental basis. Experimental data consists mainly of the results of tests for AREVA NP fuel assemblies performed on the loop of the University of COLUMBIA, and OMEGA loop of the CEA 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 together in this analysis:

- for each time step, the thermal-hydraulic conditions of the RCP [RCS] and SG are calculated by the MANTA code,
- then the thermal-hydraulic conditions at the core inlet, a temperature map, flow, pressure, boron concentration, are transferred to the FLICA code which calculates the initial core thermal-hydraulics,
    - o from the core thermal-hydraulic conditions, SMART calculates the neutronic parameters and transfers them to FLICA, until convergence is achieved,
    - o from the neutronic parameters, the FLICA code calculates core thermal-hydraulics, and transfers them back to SMART
- SMART returns the power that has been generated in each one of the 241 core assemblies to MANTA. MANTA then redistributes it in the four core quadrants modelled, with one corresponding to each primary loop.

Appendix 14C - Figure 1 provides a representation of the three-codes coupling.

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

A functional validation has been performed to show that the result of each code is the same when it is used on a stand alone basis or in the coupled system.

3.2. PROTECTION AND MITIGATION ACTIONS

The following F1A I&C functions provide protection following a steam pipe break assessed against the DNBR criterion. The analyses assessing the other criteria are discussed in the relevant sub-sections of section 2.2 of this appendix:

- Reactor trip on:
  - steam generator pressure drop > MAX1,
  - steam generator pressure < MIN1,
  - pressuriser pressure < MIN2,
  - core power level > MAX3,
  - DNBR < MIN3.
- Safety injection on:
  - pressuriser pressure < MIN3.
- Closure of all main steam isolation valves on:
  - steam generator pressure drop > MAX1,
  - steam generator pressure < MIN1.
- Isolation of all main feedwater high-load lines on:
  - steam generator pressure drop > MAX1,
  - steam generator pressure < MIN1.
- Isolation of the main feedwater low-load lines to the unaffected steam generators (SG related) on:
  - steam generator level > MAX1.
- Isolation of the main feedwater low load line to the affected steam generator on:
  - steam generator pressure drop > MAX2,
  - steam generator pressure < MIN2,
  - steam generator level > MAX1.
- At low power levels, typically power below 20% FP, the high-load lines are closed on all SG, and only the low-load lines are open. Above this power level, both high-load and low-load lines are open.

In addition, automatic actuation of the ASG [EFWS] to the affected steam generator must be assumed following a steam generator water level < MIN2 signal.

The F1B functions, required to transfer the plant from the controlled state to the safe shutdown state, are described in the section 3 of Sub-chapter 14.5, related to "Feedwater line break".

4. DESCRIPTION OF CASES STUDIED (FROM THE INITIATING EVENT TO THE CONTROLLED STATE)

4.1. POTENTIAL STEAM PIPE BREAKS OR LEAKS

The Break Preclusion (BP) principle is applied to the MSL from the SG outlet to the fixed point downstream of the VIV [MSIV]. The pipe work covered includes the nozzles of the main steam safety valves (MSSV) and of the VDA [MSRT]. There is no larger pipe work connected to the MSL inside the containment.

The BP principle is not applied to the MSL downstream of the fixed point, to the VIV [MSIV] bypass line (DN 100), to the Main Feed Water System (ARE [MFWS]) line, to the Emergency Feed Water System (ASG [EFWS]) line or to the SG blowdown System (APG [SGBS]) line.

MSL break is not considered in the portion of the MSL designed under the BP principle. As a consequence, the following events resulting in an uncontrolled SG depressurisation and/or blowdown must be covered in the PCC analyses of secondary side breaks or leaks, see Appendix 14C - Figure 2:

- Event 1: 2A-SLB downstream of the anchor of the VIV [MSIV],
- Event 2: spurious opening of a VDA |MSRT],
- Event 3: spurious opening of a main steam safety valve (MSSV),
- Event 4: break of a VIV [MSIV] bypass line,
- Event 5: break of an ARE [MFWS] line downstream of the SG check valve,
- Event 6: break of an ASG [EFWS] line (downstream of the SG check valve),
- Event 7: break of an APG [SGBS] line upstream of the APG [SGBS] isolation valves.

The PCC analysis rules, discussed in Sub-chapter 14.0, require the most onerous Single Failure (SF) to be taken into account in each case. This is either:

- the most reactive rod stuck in its upper position at RT occurrence,
- the failure of an isolation or control valve to close on demand.

4.2. TRANSIENT CASES STUDIED

The first case considered is a bounding one, which is not linked to any potential PCC event, event A on Appendix 14C - Figure 2:

- Case 1: "2A-SLB at the SG outlet, with SF of one RCCA"

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

Even though this case need not be assessed due to the application of the BP principle, it is considered in the present PCC-4 analysis as a conservative case bounding all other PCC events.

In addition, the following three events are analysed, for the reasons provided:

- Case 2: "2A-SLB at the VIV [MSIV] outlet, with the SF on the VIV [MSIV]"

The double-ended guillotine rupture of the MSL at the VIV [MSIV] outlet is combined with the SF of this VIV [MSIV] to close. This 2A-SLB break is the only potential MSL break case remaining under the application of the BP principle. It provides a comparison with the previous bounding case, the difference being the assumption of the SF on the stuck rod.

- Case 3: "MSSV spurious opening, with SF on VDA [MSRT]"

The spurious opening of one MSSV, combined with the SF of one MSRCV to close after its opening in another SG. The faulted VDA [MSRT] is automatically isolated by the closure of its MSRIV at 40 bar on a "SG pressure < MIN2" signal which is classified F1A. This case is assessed because it leads to a brief depressurisation of two SGs, compared to previous cases where the depressurisation occurs on only one SG.

The spurious opening of one MSSV is assumed instead of the spurious opening of one VDA [MSRT], as it leads to a non-isolatable leak. The depressurisation via the VDA [MSRT] would be isolated at 40 bar.

The spurious opening of one MSSV causes the VDA [MSRT] to open all the unaffected SG after the automatic isolation of all the VIV [MSIV]. The MSRCV which fails to close is assumed to be fully open.

As a consequence, this case results in the following transient:

  - Initially, the uncontrolled depressurisation of two SGs, via respectively one MSSV and one fully open VDA [MSRT].

  - Followed by, the uncontrolled depressurisation of one SG, via one fully open MSSV.

- Case 4: "dilution at power, with SF on MSRT"

The accident considered is failure of one MSRCV to close after opening during a homogeneous boron dilution accident at power. The PCC-2 accident "dilution at power", considers the stuck rod as the single failure. Case 4 addresses the same PCC-2 accident, but with the single failure on the VDA [MSRT] instead of the stuck rod. This case is considered because the boron dilution results in a lower initial shutdown margin than in the previous cases.

The homogeneous dilution at power arises from a RCV [CVCS] malfunction. The RT is provided by the "Anti-dilution at power" protection channel which is F1A. classified The setpoint for the protection is set to ensure that when the core is shutdown, the core is just critical at hot zero power (HZP), with all rods in except the highest-worth rod stuck above the core and with no xenon.

In the case of a full rod insertion with the SF already assumed on the MSRCV, the core is subcritical by only the worth of the stuck rod. The uncontrolled SG depressurisation resulting from the stuck open MSRCV occurs with a very low subcriticality margin following RT.

As a conclusion, to cover all types of steam pipe break or uncontrolled steam leak-flow, the following four cases are studied in the present section:

- Case 1: double-ended guillotine break upstream of the VIV [MSIV], resulting in "2A-SLB at SG outlet, with SF on RCCA".
- Case 2: double-ended guillotine break downstream of the VIV [MSIV], resulting in "2A-SLB at VIV [MSIV] outlet, with SF on VIV [MSIV]".
- Case 3: spurious MSSV opening, resulting in "MSSV spurious opening, with SF on VDA [MSRT]".
- Case 4: dilution at power, resulting in ”dilution at power with SF on VDA [MSRT]"

Amongst these cases, the first one is run until the controlled state is reached. This demonstrates that the operator actions, isolation of ASG [EFWS] in the affected SG, and actuation of RBS [EBS], enable recovery of core subcriticality in the most limiting case. The other ones are analysed to a time after the maximum core power and minimum DNBR.

4.3. CHOICE OF SINGLE FAILURE AND PREVENTIVE MAINTENANCE

The most limiting single failures considered in each case studied are summarised as follows:

- Case 1: one rod cluster control assembly stuck in its fully withdrawn position at RT (the highest-worth RCCA).
- Case 2: failure to close the VIV [MSIV] of the affected SG on demand via the F1A signal "SG pressure drop > MAX1" or "SG pressure < MIN1".
- Case 3: failure to close one MSRCV on an unaffected SG after VDA [MSRT] opening on the F1A signal "SG pressure > MAX1" with automatic closing of MSRIV on the F1A signal "SG pressure < MIN2".
- Case 4: failure to close the MSRCV of one SG after MSRT opening on the F1A signal "SG pressure > MAX1" with automatic closing of MSRIV on F1A signal "SG pressure < MIN2".

Preventive maintenance is not assumed as it has no significant impact on the transient:

- on the primary side, boron injection via MHSI is not taken into account in the accident analysis, and there is no preventive maintenance on RBS [EBS],

- on the secondary side, the overcooling is maximised assuming a maximum ASG [EFWS] flow rate.

4.4. INITIAL STATE

A steam line break is more severe when the unit is at hot zero power.

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

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

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

A small initial nuclear power is pessimistic for the insertion of positive reactivity. A conservative value of 10^{-9} of nominal power is assumed.

The initial RCP [RCS] temperature and pressure correspond to the hot zero power state, without uncertainties. These are included in the initial shutdown margin calculation.

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

The initial conditions, identical for all four cases, and corresponding to EPR_{4250}, are presented in Appendix 14C - Table 1 [Ref-1].

4.5. SPECIFIC ASSUMPTIONS

The specific assumptions presented below correspond to a nominal power of 4250 MW_{th}.

4.5.1. Neutronic data and decay heat

As this is a Pre-Construction Safety Report, assumptions chosen may be different from those used for operating plants. The main difference comes from the number of fuel management options covered by this study. Safety analysis reports for operating plants use a specific fuel management plan as reference, whereas the present PCSR intends to cover several different fuel management options.

For each parameter, the most pessimistic value found in several fuel management plans has been selected, and used for the study. This approach is pessimistic as it combines both the lowest shutdown margin of the MOX fuel management schemes with the most negative moderator coefficient of the UO_2 fuel management schemes.

Another conservative assumption used in this study is the assumption of a zero boron concentration in the IRWST.

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

- The fuel management considered for the SMART analysis is "UO_2 - 18 months - OI" in its equilibrium cycle, which has a high moderator feedback coefficient.
- End of life (EOL) shutdown margin, at hot full power equilibrium xenon conditions. When the single failure is assumed to be a stuck rod at reactor trip, the rod cluster control assembly having the highest worth is identified.

The initial shutdown margins are evaluated at end of life and depend on the case studied:

- Case 1: 2700 pcm.

  This is a conservative assumption for the worth of N-1 rods inserted.

- Case 2: 3450 pcm.

  This is a conservative assumption for the worth of N rods inserted.

- Case 3: 3450 pcm

  Same reason as case 2.

- Case 4: 750 pcm.

  This is a conservative assumption for the worth of one rod

Core physics studies demonstrate that these margins remain even under the most onerous conditions. This corresponds to the end of an equilibrium cycle when the moderator temperature coefficient reaches its highest absolute value as discussed in Chapter 4.

In the SMART core analysis, the neutronic coefficients are adjusted by input for the initial condition using the following assumptions:

- A maximum moderator density coefficient, in magnitude, is considered.

  For the case 1, initial conditions, the moderator temperature coefficient is -97.6 pcm/°C. This value is a bounding value for the moderator temperature coefficient with all but one rods inserted in the core. This calculated value includes an allowance of -3.6 pcm/°C for uncertainties [Ref-1].

  For the case 2, 3 and 4 initial conditions, the moderator temperature coefficient is -94.5 pcm/°C. This value is a bounding value for the moderator temperature coefficient with all rods inserted in the core. This calculated value also includes an allowance of -3.6 pcm/°C for uncertainties [Ref-1].

- A minimum differential boron worth is considered.

  For the case 1 "long term study" covering reaching the controlled state, a differential boron worth of -4.9 pcm/ppm is assumed at the initial conditions. This is only significant where boron concentration is taken into account in the analysis. This value is a bounding value for the differential boron worth with all but one rods inserted in the core. It includes an allowance of -10% for uncertainties [Ref-1].

- A minimum Doppler temperature coefficient, in magnitude, is considered which maximises the power increase.

  For the case 1 initial conditions, the Doppler temperature coefficient is -2.66 pcm/°C. This value is a bounding value for the Doppler temperature coefficient with all but one rods inserted in the core. The derivation includes an allowance of 20% of uncertainties [Ref-1].

  For the case 2, 3 and 4 initial conditions, the Doppler temperature coefficient is -2.87 pcm/°C. This value is a bounding value for the Doppler temperature coefficient with all rods inserted in the core. The derivation includes an allowance of 20% of uncertainties [Ref-1].

- A minimum value is assumed for the effective delayed neutron fraction. This minimises the time to return to power.

  For the case 1 initial conditions, the effective delayed neutron fraction is 444 pcm. This value is a bounding value for the effective delayed neutron fraction with all but one rods inserted in the core. The derivation includes an allowance of 5% of uncertainties [Ref-1].

  For the case 2, 3 and 4 initial conditions, the effective delayed neutron fraction is 455 pcm. This value is a bounding value for effective delayed neutron fraction with all rods inserted in the core. The derivation includes an allowance of 5% of uncertainties [Ref-1].

- No residual heat is taken into account. Thus maximises the RCP [RCS] overcooling in the period before criticality.

  The heat transfer coefficient from pellet to clad is assumed to be at its maximum value. After returning to power, this assumption reduces the temperatures in the pellet. This in turn minimises the Doppler Effect and transfers a maximum of energy to the clad, increasing the risk of DNB.

In the SMART core analysis cold water from the affected loop is used as the inlet temperature for all the fuel assemblies of the corresponding core quadrant in all cases. No credit is taken of mixing occurring before the core inlet as shown in Appendix 14C - Figure 3. In addition, for case 3 which experiences a brief cooldown via two loops, adjacent core quadrants use the temperature of the loop where the VDA [MSRT] failed to close as their inlet temperature as shown in see Appendix 14C - Figure 4. This is a conservative assumption for the reactivity assessment, as a large part of the core is filled by cold water.

4.5.2. Assumptions related to non-F1 systems

4.5.2.1. Main Feedwater

A minimum ARE [MFWS] temperature of 120°C is assumed [Ref-1] [Ref-2].

It is assumed that a maximum feedwater flow is delivered into all steam generators.

In both cases 1 and 2, the high load ARE [MFWS] injection is modelled, although the high load ARE [MFWS] lines should have been isolated following the reactor trip. This bounding assumption covers the different initial plant states.

The ARE [MFWS] flow assumption depends on the case studied:

Case 1: Double-ended guillotine break upstream VIV [MSIV]:

- In the first part of the transient, both high-load and low-load lines are open:
  - In the affected steam generator, the ARE [MFWS] flow is 900 kg/s.
  - In the affected generator the ASG [EFWS] flow is 55.6 kg/s.
  - In the unaffected steam generators, the ARE [MFWS] flow is 600 kg/s.
- After the ARE [MFWS] full-load line isolation:
  - In the affected steam generator, the ARE [MFWS] flow is 270 kg/s.
  - In the affected generator the ASG [EFWS] flow is 55.6 kg/s.
  - In the unaffected steam generators, the ARE [MFWS] flow is 180 kg/s. (from Sub-chapter 14.1 – Table 23 the flow rate of the low-load line is equal to 30% of the nominal ARE [MFWS] flow)

Case 2: Double-ended guillotine break downstream VIV [MSIV]:

The ARE [MFWS] assumptions are identical to those of case 1.

Case 3: Spurious MSSV opening with SF on one VDA [MSRT] after RT:

Only the ARE [MFWS] low-load flow is considered in this case, as the, ARE [MFWS] high-load lines are automatically isolated at RT. If the ARE [MFWS] high-load was assumed to be operating, the SG level increase would be very rapid and the "SG level > MAX1" setpoint would quickly be reached. As a consequence, the ARE [MFWS] high-load and low-load lines would be closed early in the transient, resulting in a lower overcooling transient.

The steam relief at the stuck-open MSSV and VDA [MSRT] is shared by all the four SG for rather a long time after the valves opening. Thus, the ARE [MFWS] low-load flow rate is assumed to be the same for each of the 4 SG and equal to 180 kg/s.

Case 4: Dilution with SF on one MSRT after RT:

In this case, only the ARE [MFWS] low-load is considered, with a flow rate of 180 kg/s in all 4 SG, as discussed for case 3 above.

4.5.3. Assumptions related to F1 systems

4.5.3.1. VIV [MSIV] (F1A)

All VIV [MSIV] are closed on a "SG pressure drop > MAX1" signal with a setpoint of 2 bar/min. The setpoint of this signal is adjusted at 8.5 bar (7 + 1.5 bar) below the initial SG pressure. The delay for steam lines isolation consists of 0.9 seconds of channel delay, plus 5 seconds for the valve closure time which is modelled as a step [Ref-1].

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

The ARE [MFWS] high-load line to the affected steam generator is isolated on a "SG pressure drop > MAX1" signal with a setpoint of 2 bar/min. The setpoint of this signal is adjusted at 8.5 bar (7 + 1.5 bar) below the initial SG pressure. The delay for feedwater isolation consists of 0.9 seconds of channel delay, plus 20 seconds for the valve closure time, which is modelled as a step [Ref-1].

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

The ARE [MFWS] low-load line to the affected steam generator is isolated on a "SG pressure drop > MAX2" signal with a setpoint of 2 bar/min. The setpoint of this signal is adjusted at 18.5 bar (17 + 1.5 bar) below the initial SG pressure. The delay for feedwater isolation consists of 0.9 seconds of channel delay, plus 20 seconds for the valve closure time which is modelled as a step.

The ARE [MFWS] low-load lines to the unaffected steam generators are isolated on a "SG level > MAX1" signal with a setpoint of 69% + 2% uncertainties on the narrow range. The delay for feedwater isolation consists of 1.5 seconds of channel delay, plus 20 seconds for the valve closure time which is modelled as a step [Ref-1].

Note: For both the guillotine breaks, cases 1 and 2, the low-load ARE [MFWS] isolation on a "SG level > MAX1" signal is inhibited in the first part of the transient. This prevents this isolation as the large steam flow exiting the unaffected SG may affect the generation of this signal. This is a conservative assumption.

4.5.3.4. MHSI (F1A)

A minimum safety injection capacity is assumed, corresponding to the minimum MHSI characteristics with a minimum delivery pressure of 85 bar, as discussed in Sub-chapter 14.1.

The temperature of MHSI is the minimum IRWST temperature of 10°C [Ref-1].

The boron concentration in the in-containment refuelling water storage tank is conservatively assumed to be 0 ppm. Boration by the RIS [SIS] is not considered in any of the cases studied, as conservative assumption.

The delay for MHSI injection consists of:

- the time to generate the safety injection signal, which consists of the time to reach the safety injection setpoint "pressuriser pressure < MIN3" of 112 bar (115 - 3 bar), plus the channel delay of 0.9 seconds to generate the signal,
- the time to start the MHSI pumps, assumed to be 10 seconds,
- the time for the MHSI pumps to reach full flow, assumed to be 5 seconds [Ref-1].

Accumulators are modelled with a minimal injection pressure of 45 bar [Ref-1]. Accumulator injection is either not reached, case 3, or remains very low, cases 1, 2.

4.5.3.5. ASG [EFWS] (F1A)

ASG [EFWS] only supplies the affected steam generator. The ASG [EFWS] actuation setpoint "SG WR level < MIN2" is not reached in the unaffected steam generators due to the VIV [MSIV] closure.

A maximum flow rate of 200 m^3/h to the affected SG is assumed. This corresponds to the operation of one ASG [EFWS] pump at 1 bar. This assumes no credit for the active flow limitation which is a conservative assumption.

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

For both cases 1 and 2, it is conservatively assumed that the ASG [EFWS] is actuated to the affected steam generator at the start of the transient.

For case 3, "spurious MSSV opening with SF on one VDA [MSRT]", the ASG [EFWS] is not modelled. In the SG with the VDA [MSRT] stuck open, the water loss is stopped by the MSRIV closure following a "SG pressure < MIN2" signal. In the SG with the stuck-open MSSV, the water loss is not sufficient for the "SG WR level < MIN2" setpoint to be reached before operator intervention.

For case 4, "dilution with SF on one VDA [MSRT]", the ASG [EFWS] is assumed to be activated to the affected SG coincident with the ARE [MFWS] low-load isolation. If this conservative assumption is not made, the "SG WR level < MIN2" signal could occur before the operator intervention.

4.5.4. Other assumptions

4.5.4.1. Assumptions related to reactor coolant pumps trip

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

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

The worst time for reactor coolant pumps trip is a few seconds before the power peak is reached if offsite power is assumed to remain available.

SLB transient analyses performed in BDR-99 for the EPR at 4900 MW_{th}, included in Appendix 14B, have shown that the resulting decrease in DNBR is not significant compared to the existing margin to the DNBR limit.

As a result, all cases studied in the present section are analysed without reactor coolant pump trip, i.e. with continued forced circulation.

4.5.4.2. Break flow (or stuck-open valve flow) correlation:

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

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

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

4.5.4.3. Other assumptions maximising core cooling:

- A maximum SG tube heat transfer coefficient is used in the analyses.
- It is assumed in the MANTA code that for core inlet mixing a maximum of 86% of the flow entering through an inlet nozzle remains in the associated core quadrant. This minimum loop flow mixing within the reactor pressure vessel is conservative for the core power transient [Ref-1].

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

5. RESULTS

These cases are calculated with EPR_{4250} characteristics.

5.1. DOUBLE-ENDED GUILLOTINE BREAK UPSTREAM OF THE VIV [MSIV] (WITH SF ON STUCK ROD)

The transient calculation and description are split into 2 phases, the short term phase lasting until the power peak and minimum DNBR have been reached, and the long term phase lasting until the controlled state has been reached with the core sub-critical.

5.1.1. Short term phase

The double-ended guillotine break of the main steam line leads to a rapid depressurisation of the secondary side [Ref-1].

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

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

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

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

The Doppler feedback limits the return to power. During the power increase, boiling occurs in the upper part of the fuel assembly with the stuck rod. This also limits the power increase. After 100 seconds the thermal power has stabilised. Following accumulator actuation at approximately 200 seconds, the nuclear power begins to rise again. At this time, the power distribution in the hot assembly has a very high peaking factor. The nuclear power increase causes a very high local Doppler feedback effect in the hot assembly. This acts to limit the peaking factor in that assembly. As a consequence, boiling in the hot assembly is reduced with a further resultant increase in the nuclear power. A few seconds later, the pressure begins to increase, further reducing boiling in the hot assembly, and allowing the nuclear and thermal power to increase.

The maximum thermal power of 17.3% NP is reached as the affected steam generator begins to empty at approximately 350 seconds.

The minimum value of DNBR is reached at 255 seconds and is equal to 1.42. The minimum DNBR occurs between the times of minimum RCP [RCS] pressure, 208 seconds, and maximum thermal power, 346 seconds. This shows that the decoupling criterion of no core DNB is fulfilled as the criterion of DNBR > 1.12 discussed in Sub-chapter 14.1 is met.

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

Appendix 14C - Table 2 gives the sequence of events.

Appendix 14C - Figures 5 to 11 show the evolution of the main parameters during the transient.

Appendix 14C - Table 3 gives the main thermal-hydraulic parameters at the time of minimum DNBR.

5.1.2. Long term phase

The affected SG having been emptied, the nuclear power stabilises at a power level which corresponds to the steam discharge of the ASG [EFWS] flow rate in the affected SG. All the thermal-hydraulic parameters stabilise at this state point: [Ref-1]

- the core remains critical with reactivity equal to zero,
- the core power is removed via the break. With ASG [EFWS] steam discharge in the affected steam generator the core power stabilises at approximately 3% of full power,
- the coolant inventory is stable.

30 minutes after the steam line break occurs, the operator isolates the ASG [EFWS] to the affected SG and actuates the RCP [RCS] boration via the RBS [EBS]. This is achieved with a flow rate of 1.4 kg/s per loop with 7000 ppm of enriched boron injected upstream of the SIS-line with an assumed volume of 0.555 m^3 at 0 ppm [Ref-2].

Following the ASG [EFWS] isolation, the temperature of the cold leg associated with the affected SG increases. Consequently the nuclear power decreases, through the moderator feedback effect.

At 2160 seconds, the boron injected via the RBS [EBS] reaches the core, 1 ppm boron in the core. This leads to a decrease of the reactivity below 0 and the reactor shuts down.

At the end of the transient, 700 seconds after the operator intervention, the nuclear power has fallen to 0 and the reactivity is equal to -275 pcm.

Appendix 14C - Figures 12 to 19 show the evolution of the main parameters during the transient.

The controlled state is reached:

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

The controlled state has been reached using only F1A means:

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

5.2. DOUBLE-ENDED GUILLOTINE BREAK DOWNSTREAM OF THE VIV [MSIV] (WITH SF OF A VIV [MSIV])

The double-ended guillotine break of the main steam line leads to a quick depressurisation of the secondary side [Ref-1].

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

Only the affected steam generator continues to depressurise after VIV [MSIV] closure.

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

Although the pressure level reached in this transient is almost the same as that reached in case 1, the behaviour of the core is very different in this case. The power distribution has a smaller peaking factor, because there is no stuck rod, and there is no boiling in any part of the core. Thus, only the Doppler feedback limits the power increase, in this case. This results in a higher power level at 150 seconds, despite the higher shutdown margin and the delayed return to power. After 150 seconds, the high power level and the low peaked power distribution in the core, lead to stabilisation of the cold leg temperature in three of the four loops.

Following accumulator actuation, the nuclear power increases, as seen in case 1. After 250 seconds, the thermal power is sufficient to halt the cold leg temperature decrease. The **maximum nuclear power of 18.8% NP is reached at 262 seconds**. As the cold leg temperatures begin to rise, the reactivity increase is halted. The Nuclear and thermal power both begin slowly to decrease. Once the affected SG empties, the nuclear and thermal powers rapidly fall as the RCP [RCS] heats up.

The minimum value of DNBR is reached at 192 seconds and is equal to 2.79, well above the decoupling criterion of 1.12. The minimum DNBR occurs at the time of minimum RCP [RCS] pressure.

Appendix 14C - Table 4 gives the sequence of events.

Appendix 14C - Figures 20 to 26 show the evolution of the main parameters during the transient.

Appendix 14C - Table 5 gives the main thermal-hydraulic parameters at the time of minimum DNBR.

5.3. SPURIOUS MSSV OPENING (WITH SF ON VDA [MSRT])

The spurious MSSV opening in SG2, combined with the stuck-open VDA [MSRT] in SG1, leads to a rapid depressurisation of the secondary side. Both valves are assumed to open at time zero, which maximises the resulting overcooling [Ref-1].

The "SG pressure drop > MAX1" signal setpoint is reached 66 seconds after the opening of the two valves. This signal leads to the isolation of the steam lines at 71 seconds.

After VIV [MSIV] closure, only the two affected steam generators continue to depressurise.

The "SG pressure drop > MAX2" signal setpoint is reached at 90 seconds in SG1, and at 102 seconds in the SG2. This signal leads to the complete isolation of the ARE [MFWS] in these two SG 20 seconds after the individual signal is generated. These SG are then no longer fed.

This case shows a later return to criticality, at 307 seconds. This is due to the smaller size of the leak and to the high shutdown margin with no stuck rod. At the time the core returns critical, only the MSSV is still open as the VDA [MSRT] closes at 199 seconds on a "SG pressure < MIN2" signal Reactivity insertion occurs at a low rate, and at no point does the core becomes prompt critical.

Soon after returning critical, the **power rises to a level of 4.7% NP**. This is high enough to halt the cold leg temperatures decrease. As the cold leg temperatures begin to rise, the reactivity increase is stopped. Subsequently the nuclear and thermal powers begin to slowly decrease.

Due to the low power level reached and to the lack of stuck rod, the **DNBR stays above 10** throughout the accident.

Appendix 14C - Table 6 gives the sequence of events.

Appendix 14C - Figures 27 through 33 show the evolution of the main parameters during the transient.

5.4. BORON DILUTION WITH SF ON VDA [MSRT]

The opening of the VDA [MSRT] on SG1 leads to a rapid depressurisation of the secondary side [Ref-1].

This case returns to power much earlier than case 3, due to the low shutdown margin following the dilution prior to RT. As in Case 3, when the core returns critical the reactivity insertion rate is too small to cause a prompt criticality.

The return to power takes place before steam lines isolation. At this point, all the cold legs, except the one with pressuriser, have the same temperature. Power distribution is quite symmetric across the core with no difference between quadrants. Between 85 and 100 seconds, the core thermal power halts the temperature decrease in all the loops.

At 100 seconds, steam lines isolation occurs. The power transferred to loops 2, 3 and 4 continues to raise the cold leg temperature of these loops. The power transferred to loop 1 is not sufficient to match the energy removal via the leak. Therefore, the cold leg temperature in loop 1 decreases again, causing a temperature difference at the core inlet. The power level increases again, this time with a non symmetric inlet temperature and consequently power distribution.

At 158 seconds, the ARE [MFWS] flow is isolated to the affected SG and the ASG [EFWS] is actuated. As the ASG [EFWS] flow is smaller than the previous ARE [MFWS] flow, the power transferred to loop 1 becomes sufficient to stop the temperature decreasing further in the loop 1 cold leg. Cold leg temperatures and power stabilise until about 230 seconds, when two further events take place:

- opening of the VDA [MSRT] in SG 2 and 4. This a short cooldown transient, during the time period for the MSRCV to close to its equilibrium opening, starting from its fully open position when the VDA [MSRT] opens,

- isolation of ARE [MFWS] in SG 2, SG 3 and SG 4.

The VDA [MSRT] opening on loops 2 and 4 causes a cold leg temperature decrease in these loops. The core reactivity and nuclear power temporarily increase. As ARE [MFWS] is isolated on all loops, compensation for this cold legs temperature decrease by the power increase is very rapid. The maximum power level of 18.4% NP is reached at approximately 240 seconds.

With all ARE [MFWS] now isolated, energy generated by core quadrants 2, 3 and 4 raises the cold leg temperature of the associated loops. At 317 seconds the VDA [MSRT] of SG3 opens, reducing the temperature in cold leg of loop 3. Therefore, the temperature in cold leg 3 stays below that in cold legs 2 and 4.

The temperature in cold leg 1 stays almost constant as energy produced by quadrant 1 matches the energy removed by the leak.

The affected steam generator is emptied after 1000 seconds, and the nuclear and thermal powers rapidly reduce.

The primary pressure never decreases below 130 bar in the whole transient and thus the MHSI is not activated. This occurs because the nuclear power increases shortly after the VDA [MSRT] opening and the SG cooling is not large. The early increase in the nuclear power is caused by the reduced initial shutdown margin resulting from the boron dilution prior to RT.

With the low pressure decrease and to the absence of a stuck rod, DNBR stays above 10 throughout the accident.

Appendix 14C - Table 7 gives the sequence of events.

Appendix 14C - Figures 34 through 40 show the evolution of the main parameters during the transient.

6. CONSEQUENCES OF THE MODIFICATION TO THE PARTIAL COOLDOWN RATE AND ASSOCIATED SETPOINTS

Following a 2A steam line break or a spurious opening of a MSSV, a partial cooldown is automatically actuated following a low pressuriser pressure signal. This partial cooldown has no effect on the transient during the first minutes of the accident. In practice the secondary side pressure decreases at a faster rate than the partial cooldown could achieve. Later in the transient the pressure in the non-affected SG follows the VDA [MSRT] setpoint until the end of the partial cooldown as shown on Appendix 14C – Figure 17.

For cases 1 and 2 the minimum DBNR occurs at about 200 seconds and for case 3 at about 400 seconds when the partial cooldown does not control the secondary pressure. Therefore the cooldown rate increase from -100°C/h to -250°C/h will have no impact on the DNBR calculation.

The increase of the SG pressure drop setpoints associated with the partial cooldown rate modification can have a small impact on the transient. The ARE low-load isolation might not be actuated on a “SG pressure drop > MAX2” signal but a little time later on a “SG level > MAX1” signal. This delay in ARE low-load isolation will lead to a slightly higher cooldown of the non-affected primary loop. As the flow mixing between the primary loops is low, the evolution of the temperature of the affected loop will almost be the same. Therefore the increase in power due to moderator feedback, in the core area where the rod is stuck out, will also be similar and the DNBR will not be impacted. Therefore, it can be concluded that the decoupling criterion of no core DNB will be fulfilled with a higher partial cooldown rate and associated SG pressure drop setpoints.

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

The safe shutdown state is defined as a state where the core is subcritical, the RIS/RRA [SIS/RHRS] operating conditions have been reached and the affected steam generator is isolated.

The transition from the controlled state to the safe shutdown state is covered by that following a feedwater line break discussed in section 3 of Sub-chapter 14.5.

Following isolation of the affected steam generator, the RCP [RCS] and SG status in the steam line break case are quite similar to those for the feedwater line break accident.

In addition, the capabilities of F1B systems needed in both cases are identical, the VDA [MSRT], ASG [EFWS], RBS [EBS], and PSV.

8. EXTENSION TO 4500 MW_{TH}

In this section, it is demonstrated that the results obtained for a nominal power of 4250 MW_{th} can be extended to a nominal power of 4500 MW_{th}. In particular, case 1 of the study analysed for 4250 MW_{th} is re-calculated with assumptions more representative of operation at 4500 MW_{th}.

8.1. ASSUMPTIONS FOR 4500 MW_{TH} POWER OPERATION

For 4500 MW_{th} power operation, the changes to the EPR 4250 analysis are associated with:

- The initial state of the plant,
- The specific assumptions

8.1.1. Initial state

The initial plant state is identical to the one described in sub-section 4.4 of this appendix. The initial conditions taken into account for the analysis are presented in Appendix 14C – Table 8.

8.1.2. Neutronic data and decay heat

Neutronic data taken into account for the EPR_{4500} are pessimistic, and are identical to the data presented in sub-section 4.5 of this appendix. The only data changed is the initial shutdown margin, which is increased for the EPR_{4500} fuel management schemes.

- Case 1: 3400 pcm.
- Case 2: 4500 pcm.
- Case 3: 4500 pcm.
- Case 4: 1100 pcm.

Further, for the case 1 calculation, the fuel management considered is "UO_2 - 18 months –IO", and the stuck rod is at location M02 [Ref-1].

8.1.3. Assumptions related to non-F1 systems

For the ARE [MFWS] flow, the increase in the flow for 4500 MW_{th} power operation is modelled. Thus, the ARE [MFWS] flows modelled for case 1 analysed with EPR_{4500} assumptions are:

- In the first part of the transient, both high-load and low-load lines are opened:
  - In the affected steam generator, the ARE [MFWS] flow is 967.5 kg/s.
  - In the affected generator the ASG [EFWS] flow is 55.6 kg/s.

- In the unaffected steam generators, the ARE [MFWS] flow is 645 kg/s.

- After the ARE [MFWS] full load isolation:
  - In the affected steam generator, the ARE [MFWS] flow is 290.25 kg/s.
  - In the affected generator the ASG [EFWS] flow is 55.6 kg/s.
  - In the unaffected steam generators, the ARE [MFWS] flow is 193.5 kg/s.

8.1.4. Assumptions related to F1 systems

The assumptions related to F1 systems are identical to those described in sub-section 4.5.3 of this appendix.

8.1.5. Other assumptions

Other assumptions are identical to those described in sub-section 4.5.4 of this appendix.

8.2. RESULTS

8.2.1. Double-ended guillotine break upstream of the VIV [MSIV] (with SF on stuck rod)

8.2.1.1. Short term phase

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

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

Following VIV [MSIV] closure, only the affected steam generator continues to depressurise.

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

The reactor becomes critical and the thermal power increases.

The Doppler feedback limits the return to power. During the power increase, boiling occurs in the upper part of the fuel assembly containing the stuck rod, M02. This phenomenon, combined with the reduction in the rate of temperature decrease in the affected cold leg, limits the power increase. After accumulator injection occurs at approximately 150 seconds, the nuclear power begins to rise again. From 200 seconds, the power generated in the core matches the power removed by the affected SG. The cold leg temperature stops falling and the primary pressure increases. The primary pressure increase causes the level at which boiling occurs in the M02 assembly and the core power continues to increase.

The maximal thermal power of 15% NP is reached at approximately 405 seconds when the affected steam generator begins to empty.

The minimum value of DNBR is reached at 425 seconds and is equal to 2.12. The minimum DNBR occurs just after the maximum neutronic power. It has therefore been demonstrated that the decoupling criterion of no core DNB, DNBR above 1.12 as discussed in Sub-chapter 14.1 is met.

During the latter stages of SG draining, the thermal power decreases and the fast secondary transient ends. At this point, the RCP [RCS] overcooling is ended.

Appendix 14C - Table 9 gives the sequence of events.

Appendix 14C - Figures 41 through 47 show the evolution of the main parameters during the transient.

Appendix 14C - Table 10 gives the main thermal-hydraulic parameters at the time of minimum DNBR.

8.2.1.2. Long term phase

Calculations for the long term phase have not been performed at a nominal power of 4500 MW_{th}. However, the plant behaviour can be deduced from that presented in sub-section 5.1.2 of this appendix.

The relevant parameters for the long-term phase, ASG [EFWS] flow rate, moderator coefficient and boron worth, are identical for both calculations. Thus, the main events in the long-term phase, plant stabilisation at the power level corresponding to the steam discharge of the ASG [EFWS] and a sustainable reactivity decrease, will be similar.

8.2.2. Double-ended guillotine break downstream of the VIV [MSIV] (with SF on VIV [MSIV])

This transient has not been calculated for a nominal power of 4500 $MW_{th.}$ Results presented in sub-sections 5.1 and 5.2 of this appendix show that for similar excessive cooling, the case without a stuck rod is less onerous for the challenge to the fuel de-coupling criteria than the case with one stuck rod. This occurs because less peaked power distribution leads to higher DNBR values.

Therefore, the double-ended guillotine rupture downstream of the VIV [MSIV], which benefits from a higher initial shutdown margin and from the absence of a rod stuck out, will result in higher DNBR values than the case presented in sub-section 8.2.1.1 of this appendix.

8.2.3. Spurious MSSV opening (with SF on VDA [MSRT])

This transient has not been calculated for a nominal power of 4500 MW_{th}. The plant behaviour can be deduced from that presented in sub-section 5.3 of this appendix.

In the transient described in sub-section 5.3 of this appendix, the initial shutdown margin of 3450 pcm is only completely lost after 300 seconds. Once the shutdown margin has been lost, the power arising in the core is sufficient to match the temperature decrease in the cold legs. Subsequently, the power decreases due to the moderator feedback effect.

The differences between the cases studied at 4250 MW_{th} and 4500 MW_{th} are:

- the initial SG water inventory is higher by 2.1 te for the case at 4500 MW_{th},

- the initial shutdown margin is higher by approximately 1000 pcm for the case at 4500 MW_{th} [Ref-1]

The initial shutdown margin, larger in the case at 4500 MW_{th}, will lead to a later return to power. This will occur approximately roughly 200 seconds based on the slope of the negative reactivity increase. This assumes the rate of increase remains the same during the period between 300 seconds and 500 seconds. Due the continued break flow, the SG water mass at the time of the return to criticality, will therefore be lower than that in the case studied at 4250 MW_{th}. Once the initial shutdown margin has been eroded, the power generated in the core will be sufficient, as in the case presented in sub-section 5.3, to compensate for the temperature decrease in the cold legs. Subsequently the power will decrease due to the moderator feedback effect.

The transient calculated for a nominal power of 4500 MW_{th} will therefore be similar to that calculated for a 4250 MW_{th} nominal power. The most significant difference between the two cases will be a later return to criticality in the case at a nominal power of 4500 MW_{th}. As the safety criteria are met with comfortable margins for a nominal power of 4250 MW_{th} as shown in sub-section 5.1.3 of this appendix, they will also be met for a nominal power of 4500 MW_{th}.

8.2.4. Boron dilution with SF on VDA [MSRT]

This transient has not been calculated for a nominal power of 4500 MW_{th}. The plant behaviour can be deduced from that presented in sub-section 5.4 of this appendix.

The main characteristics of the boron dilution transient studied in sub-section 5.4 of this appendix for a nominal power of 4250 MW_{th} are the following:

- a return to power occurs before steam lines isolation. This arises from the significantly reduced initial shutdown margin. Consequently, power generation occurs in the whole core early in the transient.

- High power peaks occur, triggered by sudden cool downs in the different core quadrants. The cooldown of quadrant 1 occurs following steam line isolation. A short-duration cooldown of the remaining quadrants is triggered by the opening of the VDA [MSRT] isolation valves before closure of the control valves. This is then followed by a power stabilisation at around 8% of nominal power.

- A high primary pressure, as the power generation is sufficient to compensate for the RCP [RCS] cooldown by the SG.

- A decrease of the nuclear power, caused by the depletion of the water inventory in the affected SG.

- A DNBR which remains above 10 throughout the transient.

The differences between the cases studied at 4250 MW_{th} and 4500 MW_{th} are the following:

- the SG initial water inventory is higher by 2.1 te in the case at 4500 MW_{th},

- the initial shutdown margin is larger by 350 pcm in the case at 4500 MW_{th}.

The differences between the two transients will concern the time of return to criticality. If the reactivity increase is similar to the case at 4250 MW_{th}, criticality should occur approximately 20 seconds later than in the case studied in sub-section 5.4 of this appendix. The return to power should then occur slightly before steam line isolation.

The slightly higher SG water inventory, with a nominal power of 4500 MW_{th}, will lead to a later termination of the transient. This is linked to the depletion of the water inventories in the SG. This has no impact on the safety criteria as the maximum power and minimum pressure, which lead to the lowest DNBR values, occur early in the transient.

The main characteristics of the boron dilution transient presented in sub-section 5.4 of this appendix, a slow cooldown with the primary pressure remaining high, will therefore be similar for a transient with a nominal power of 4500 MW_{th}. Thus, the DNBR values will remain high throughout the transient, and the associated safety criteria will be met.

9. CONCLUSION

The most conservative single failure for a main steam line break in State A in terms of DNBR is on a RCCA (i.e. a rod is stuck). This single failure is systematically used in the PCSR analysis of main steam line break in State A performed in section 2.1 of Sub-chapter 14.5

APPENDIX 14C - TABLE 1 [REF-1]

Main Steam Line Break

Main Assumptions – 4250 MWth

Parameters		Values used
Initial conditions		
- Reactor power	(FP)	10^{-9}
- Shutdown margin (cases 1 / 2 / 3 / 4)	(pcm)	-2700 / -3450 / -3450/-750
- RCP [RCS] boron concentration	(ppm)	0
- RCP [RCS] flow rate		T/H design flow rate
- Average RCP [RCS] temperature	(°C)	303.3
- Pressuriser pressure	(bara)	155
- Pressuriser level	(% MR)	34 – 5 = 29
- Flow limiter cross-section (per SG)	(m²)	0.13
- MSIV cross-section	(m²)	0.32
- Initial ARE [MFWS] flow rate in the affected SG	(kg/s)	900
- Initial ARE [MFWS] flow rate in the unaffected SG	(kg/s)	600
- ASG [EFWS] flow rate in the affected SG	(kg/s)	55.6
Safety injection:		
- Time to open valves and start pumps	(s)	10
- Time to reach full flow	(s)	5
- Concentration of borated water		
. in IRWST	(ppm)	0
. in the safety injection lines downstream of the connection of the pump mini-flow line returning into the IRWST	(ppm)	0
. in accumulators	(ppm)	0
"SG pressure drop > MAX1" setpoint	(bar/min)	2 Setpoint adjusted 8.5 bar below the initial value (7 + 1.5 bar)
"SG pressure drop > MAX2" setpoint	(bar/min)	2 Setpoint adjusted 18.5 bar below the initial value (17 + 1.5 bar)
"Pressuriser pressure < MIN3" setpoint	(bara)	115 - 3 = 112
Steam line isolation delay	(s)	0.9 + 5 = 5.9
Main feedwater low-load line isolation delay	(s)	20

APPENDIX 14C - TABLE 2 [REF-1]

Sequence of Events

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) - 4250 MWth

EVENT	TIME (seconds)
Main steam line break	0
"SG pressure drop > MAX1" setpoint is reached (RT/VIV [MSIV] isolation[1])	1
"SG pressure drop > MAX2" setpoint is reached in affected SG1	1
Steam lines are isolated	6
Pressuriser is empty (Pressuriser level = 0)	17
Reactor becomes critical	18.4
Main feedwater high-load line isolation in all the SG	21
Main feedwater low-load line isolation in the affected SG	21
"Pressuriser pressure < MIN3" setpoint is reached (SIS actuation)	25
MHSI begins to inject	40
Opening of the VDA [MSRT] 3 isolation valve	73
Opening of the VDA [MSRT] 2 and 4 isolation valves	92
"SG pressure drop > MAX2" setpoint is reached in SG 2 and SG 4	119
ARE [MFWS] low-load line isolation in SG 2 and SG 4	139
"SG level > MAX1" setpoint is reached in SG 3	172
Accumulators injection (opening / closing)	176 / 209
ARE [MFWS] low-load line isolation in SG 3	192
Minimum pressuriser pressure is reached (38.0 bar)	208
Minimum DNBR is reached (1.42)	255
Nuclear power peak is reached (17.3% NP)	346

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

APPENDIX 14C - TABLE 3 [REF-1]

Conditions at the Time of Minimum DNBR

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF of a stuck rod)

Time	255 s
Thermal power	15.1%
Concentration of boron in the core	0 ppm
Average core pressure	42.0 bara.
Cold leg temperature in affected loop	203.7°C
Cold leg temperature in unaffected loops	236.6°C
Core flow rate (fraction of nominal)	100
Minimum DNBR	**1.42**

APPENDIX 14C - TABLE 4 [REF-1]

Sequence of Events

Double-Ended Guillotine Break Downstream of the VIV [MSIV] (SF on the VIV [MSIV])

EVENT	TIME (seconds)
Main steam line break	0
"SG pressure drop > MAX1" setpoint is reached (RT/VIV [MSIV] isolation[2])	1
"SG pressure drop > MAX2" setpoint is reached in affected SG	1
Steam lines are isolated	6
Pressuriser is empty (Pressuriser level = 0)	16
Main feedwater high-load line isolation in all the SG	21
Main feedwater low-load line isolation in the affected SG	21
"Pressuriser pressure < MIN3" setpoint is reached (RIS [SIS] actuation)	23
Reactor becomes critical	36
MHSI begins to inject	38
Opening of the VDA [MSRT] 3 isolation valve	83
Opening of the VDA [MSRT] 2 and 4 isolation valves	104
"SG pressure drop > MAX2" setpoint is reached in SG 3	117
"SG pressure drop > MAX2" setpoint is reached in SG 2 and SG 4	128
ARE [MFWS] low-load line isolation in the unaffected SG 3	137
ARE [MFWS] low-load line isolation in the unaffected SG 2 and SG 4	148
Accumulators injection (opening / closing)	178 / 192
Minimum pressuriser pressure is reached (40.5 bar)	192
Minimum DNBR is reached (2.79)	192
Nuclear power peak is reached (18.8%NP)	262

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

APPENDIX 14C - TABLE 5 [REF-1]

Conditions at the Time of Minimum DNBR

Double-Ended Guillotine Break Downstream of the VIV [MSIV] (SF on the VIV [MSIV])

Time	192 s
Thermal power	16.8%
Concentration of boron in the core	0 ppm
Average core pressure	41.1 bara.
Cold leg temperature in affected loop	207.2°C
Cold leg temperature in unaffected loops	245.0°C
Core flow rate (fraction of nominal)	100
Minimum DNBR	**2.79**

APPENDIX 14C - TABLE 6 [REF-1]

Sequence of Events Spurious MSSV Opening (SF on VDA [MSRT])

EVENT	TIME (seconds)
Spurious opening of MSSV 2 0pening of the VDA [MSRT] 1 (MSRCV stuck open)	0
"SG pressure drop > MAX1" setpoint is reached (RT/VIV [MSIV] isolation[3])	66
Pressuriser is empty (Pressuriser level = 0%)	68
Steam lines are isolated	71
"SG pressure drop > MAX2" setpoint is reached in SG 1	90
"SG pressure drop > MAX2" setpoint is reached in SG 2	102
"Pressuriser pressure < MIN3" setpoint is reached (RIS [SIS] actuation)	103
Main feedwater low-load line isolation in SG 1	110
Main feedwater low-load line isolation in SG 2	122
MHSI begins to inject	126
Opening of the VDA [MSRT] 4 isolation valve	191
Opening of the VDA [MSRT] 3 isolation valve	197
Closing of the VDA [MSRT] 1 isolation valve	199
"SG pressure drop > MAX2" setpoint is reached in SG 4	208
"SG pressure drop > MAX" setpoint is reached in SG 3	215
ARE [MFWS] low-load line isolation in the unaffected SG 4	228
ARE [MFWS] low-load line isolation in the unaffected SG 3	235
Minimum pressuriser pressure is reached (50.2 bar)	249
Reactor becomes critical	307
Nuclear power peak is reached (4.7%NP)	431

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

APPENDIX 14C - TABLE 7 [REF-1]

Sequence of Events

Nominal Power 4250 MWth – Boron Dilution (SF on VDA [MSRT])

EVENT	TIME (seconds)
Opening of the VDA [MSRT] 1 (MSRCV stuck open)	0
Reactor becomes critical	47
Pressuriser is empty (Pressuriser level = 0)	79
"SG pressure drop > MAX1" setpoint is reached (RT/ VIV [MSIV] isolation[5])	95
Steam lines are isolated	100
"SG pressure drop > MAX2" setpoint is reached in affected SG	138
Main feedwater low-load line isolation in the affected SG	158
Emergency feedwater actuated in the affected SG	158
"SG level > MAX1" setpoint is reached in SG 2 and SG 4	211
"SG level > MAX1" setpoint is reached in SG 3	216
Opening of the VDA [MSRT] 2 and 4 isolation valves	223
ARE [MFWS] low-load line isolation in the unaffected SG 2 and SG 4	231
ARE [MFWS] low-load line isolation in the unaffected SG 3	236
Nuclear power peak is reached (18.4%NP)	243
Minimum pressuriser pressure is reached (130.3 bar)	245
Opening of the VDA [MSRT] 3 isolation valve	317

APPENDIX 14C - TABLE 8 [REF-1]

Main Steam Line Break

Main Assumptions – 4500 MWth

Parameters		Values used
Initial conditions		
- Reactor power	(FP)	10^{-9}
- Shutdown margin (cases 1 / 2 / 3 / 4)	(pcm)	-3400 / -4500 / -4500/-1100
- RCP [RCS] boron concentration	(ppm)	0
- RCP [RCS] flow rate		T/H design flow rate
- Average RCP [RCS] temperature	(°C)	303.3
- Pressuriser pressure	(bara)	155
- Pressuriser level	(% MR)	31 – 5 = 26
- Flow limiter cross-section (per SG)	(m²)	0.13
- VIV [MSIV] cross-section	(m²)	0.32
- Initial ARE [MFWS] flow rate in the affected SG	(kg/s)	967.5
- Initial ARE [MFWS] flow rate in the unaffected SG	(kg/s)	645
- ASG [EFWS] flow rate in the affected SG	(kg/s)	55.6
Safety injection:		
- Time to open valves and start pumps	(s)	10
- Time to reach full flow	(s)	5
- Concentration of borated water		
. in IRWST	(ppm)	0
. in the safety injection lines downstream of the connection of the pump mini-flow line returning into the IRWST	(ppm)	0
. in accumulators	(ppm)	0
"SG pressure drop > MAX1" setpoint	(bar/min)	2 Setpoint adjusted 8.5 bar below the initial value (7 + 1.5 bar)
"SG pressure drop > MAX2" setpoint	(bar/min)	2 Setpoint adjusted 18.5 bar below the initial value (17 + 1.5 bar)
"Pressuriser pressure < MIN3" setpoint	(bara)	115 - 3 = 112
Steam line isolation delay	(s)	0.9 + 5 = 5.9
Main feedwater low-load line isolation delay	(s)	20

APPENDIX 14C - TABLE 9 [REF-1]

Sequence of Events

Double-Ended Guillotine Break Downstream of the VIV [MSIV] (SF on the VIV [MSIV]) – 4500 MWth

EVENT	TIME (seconds)
Main steam line break	0
"SG pressure drop > MAX1" setpoint is reached (RT/VIV [MSIV] isolation[4])	1
"SG pressure drop > MAX2" setpoint is reached in affected SG	1
Steam lines are isolated	6
Pressuriser is empty (Pressuriser level = 0)	16
Main feedwater high-load line isolation in all the SG	21
Main feedwater low-load line isolation in the affected SG	21
"Pressuriser pressure < MIN3" setpoint is reached (RIS [SIS] actuation)	21.6
Reactor becomes critical	27
MHSI begins to inject	36.6
Opening of the VDA [MSRT] 3 isolation valve	65
Opening of the VDA [MSRT] 2 and 4 isolation valves	83
"SG pressure drop > MAX2" setpoint is reached in SG 2 and SG 4	111
ARE [MFWS] low-load line isolation in the unaffected SG 2 and SG 4	131
"SG level > MAX1" setpoint is reached in SG 3	149
Accumulators injection (opening / closing)	153 / 201
ARE [MFWS] low-load line isolation in SG 3	169
Minimum pressuriser pressure is reached (36.4 bara)	201
Nuclear power peak is reached (15%NP)	405
Minimum DNBR is reached (2.12)	425

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

APPENDIX 14C - TABLE 10 [REF-1]

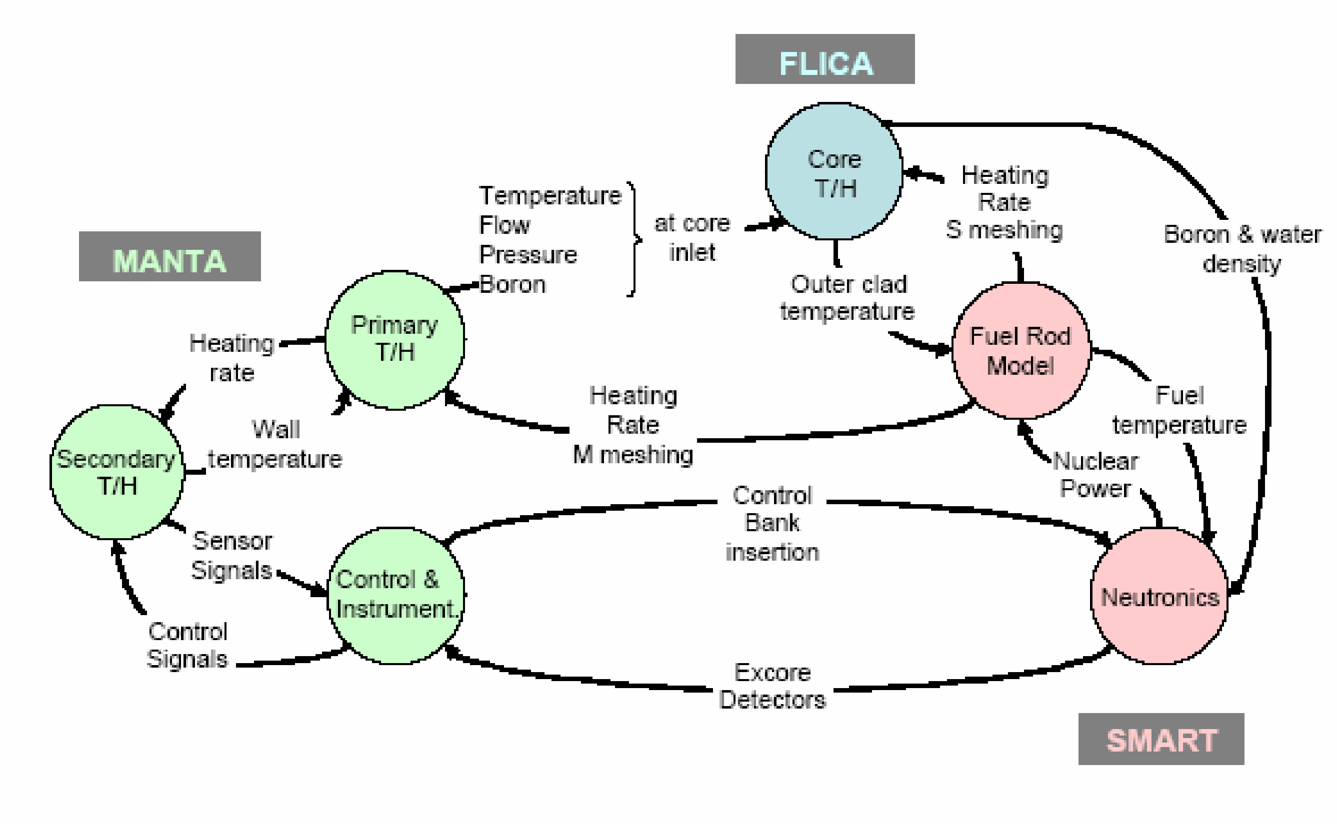
Conditions at the Time of Minimum DNBR

Double-Ended Guillotine Break Downstream of the VIV [MSIV] (SF on VIV [MSIV]) – 4500 MWth

Time	425 s
Thermal power	14.72%
Concentration of boron in the core	0 ppm
Average core pressure	56.2 bara.
Cold leg temperature in affected loop	201.4°C
Cold leg temperature in unaffected loops	232.4°C
Core flow rate (fraction of nominal)	1
Minimum DNBR	**2.12**

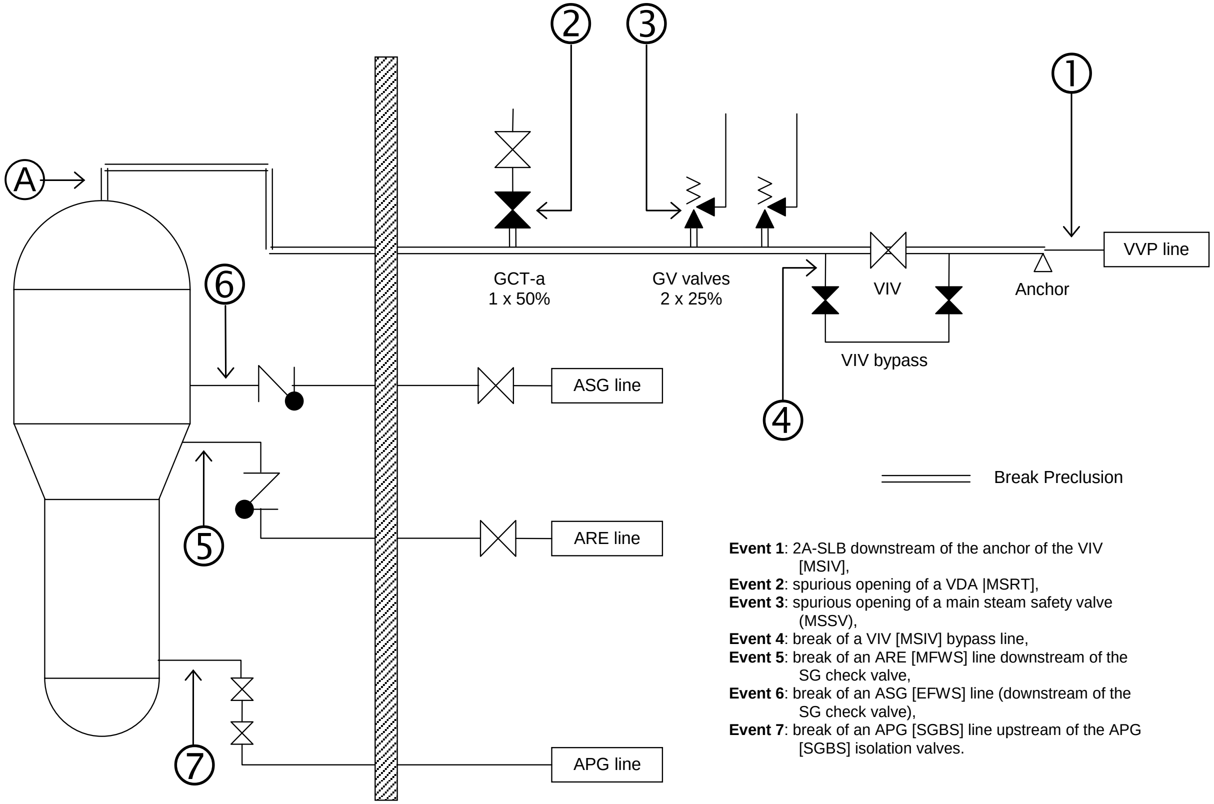
APPENDIX 14C - FIGURE 1 [REF-1]

Main Steam Line Break - MANTA / SMART / FLICA codes coupling



APPENDIX 14C - FIGURE 2

Events covered in the PCC studies



Event 1: 2A-SLB downstream of the anchor of the VIV [MSIV],
Event 2: spurious opening of a VDA [MSRT],
Event 3: spurious opening of a main steam safety valve (MSSV),
Event 4: break of a VIV [MSIV] bypass line,
Event 5: break of an ARE [MFWS] line downstream of the SG check valve,
Event 6: break of an ASG [EFWS] line (downstream of the SG check valve),
Event 7: break of an APG [SGBS] line upstream of the APG [SGBS] isolation valves.

APPENDIX 14C - FIGURE 3 [REF-1]

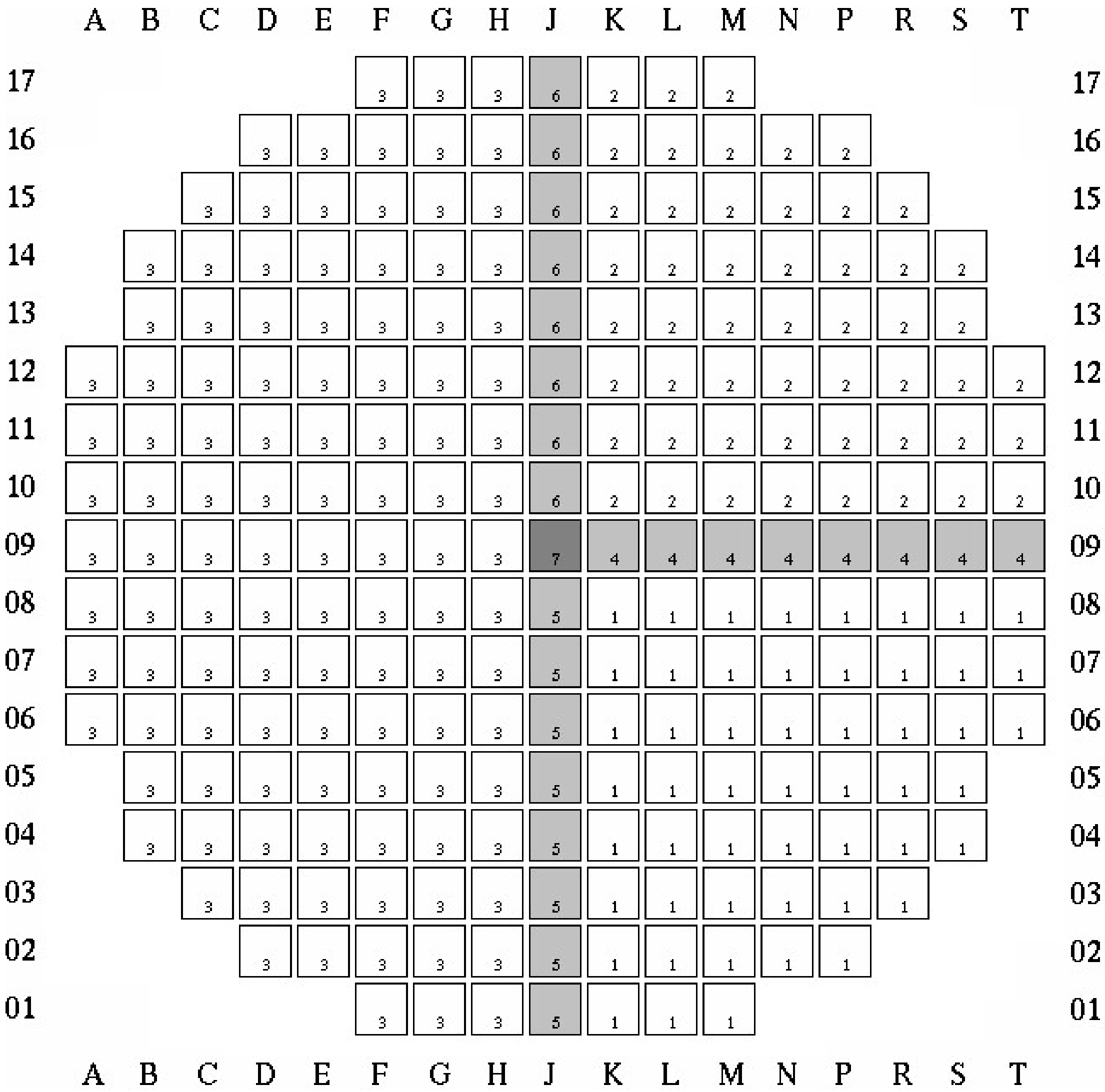
Core inlet temperature for cases 1 and 2

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T	
17						3	3	3	3	3	3	3						17
16				3	3	3	3	3	3	3	3	3	3	3				16
15			3	3	3	3	3	3	3	3	3	3	3	3	3			15
14		3	3	3	3	3	3	3	3	3	3	3	3	3	3	3		14
13		3	3	3	3	3	3	3	3	3	3	3	3	3	3	3		13
12	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	12
11	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	11
10	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	10
09	3	3	3	3	3	3	3	3	2	2	2	2	2	2	2	2	2	09
08	3	3	3	3	3	3	3	3	2	1	1	1	1	1	1	1	1	08
07	3	3	3	3	3	3	3	3	2	1	1	1	1	1	1	1	1	07
06	3	3	3	3	3	3	3	3	2	1	1	1	1	1	1	1	1	06
05		3	3	3	3	3	3	3	2	1	1	1	1	1	1	1		05
04		3	3	3	3	3	3	3	2	1	1	1	1	1	1	1		04
03			3	3	3	3	3	3	2	1	1	1	1	1	1			03
02				3	3	3	3	3	2	1	1	1	1	1				02
01						3	3	3	2	1	1	1						01
	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T	

1 = affected loop cold leg temperature
2 = average between 1 and 3 temperatures
3 = average of the three non affected loops cold legs temperatures

APPENDIX 14C - FIGURE 4 [REF-1]

Core inlet temperature for case 3

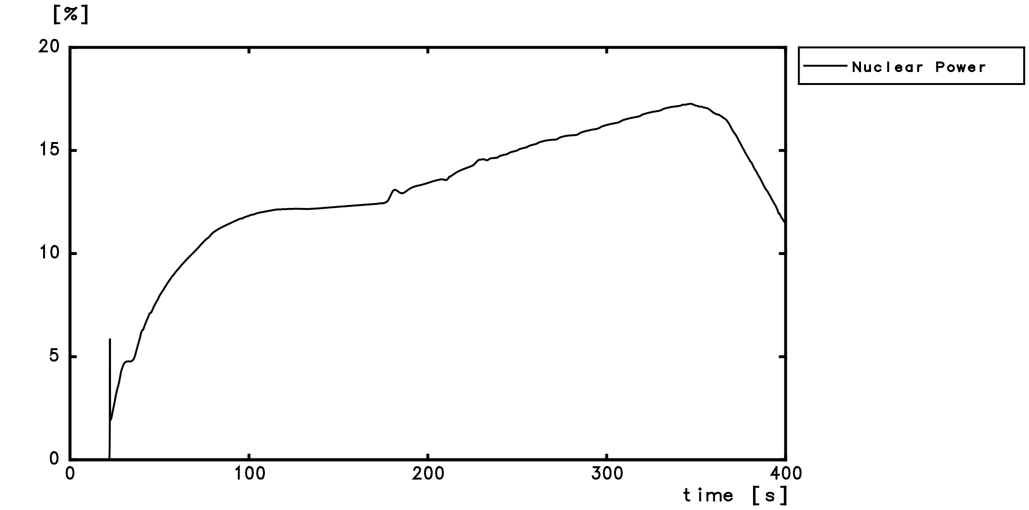


1 = affected loop cold leg temperature
2 = loop where VDA [MSRT] failed cold leg temperature
3 = average of the two non affected loops cold legs temperatures
4 = average between 1 and 2 temperatures
5 = average between 1 and 3 temperatures
6 = average between 2 and 3 temperatures
7 = average between 1, 2 and 3 temperatures

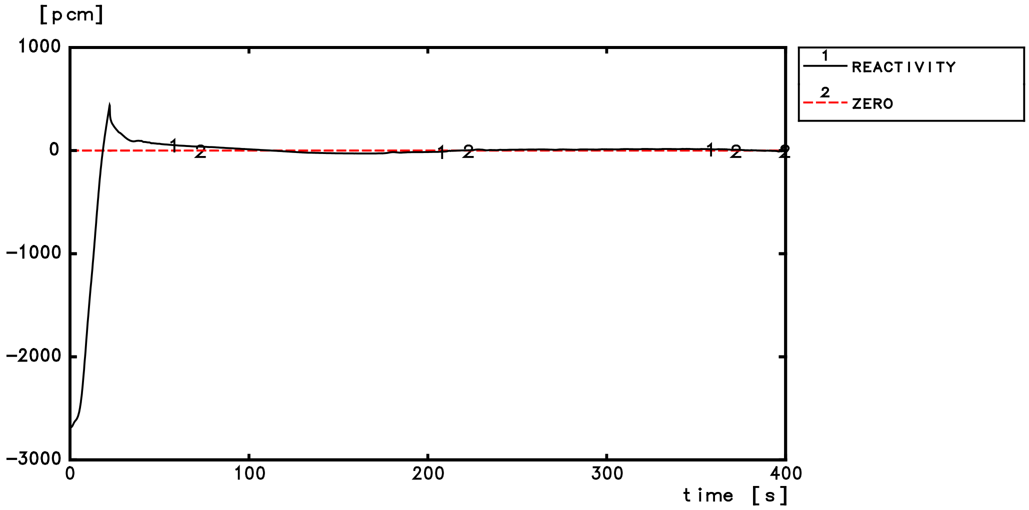
APPENDIX 14C - FIGURE 5 [REF-1]

Nuclear Power – Reactivity

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) – EPR 4250 MWth



NUCLEAR POWER

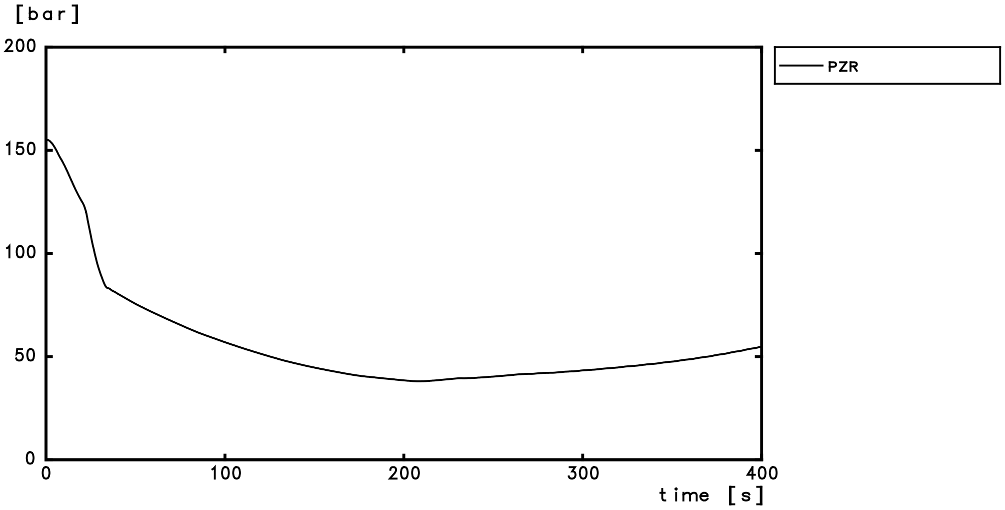


REACTIVITY

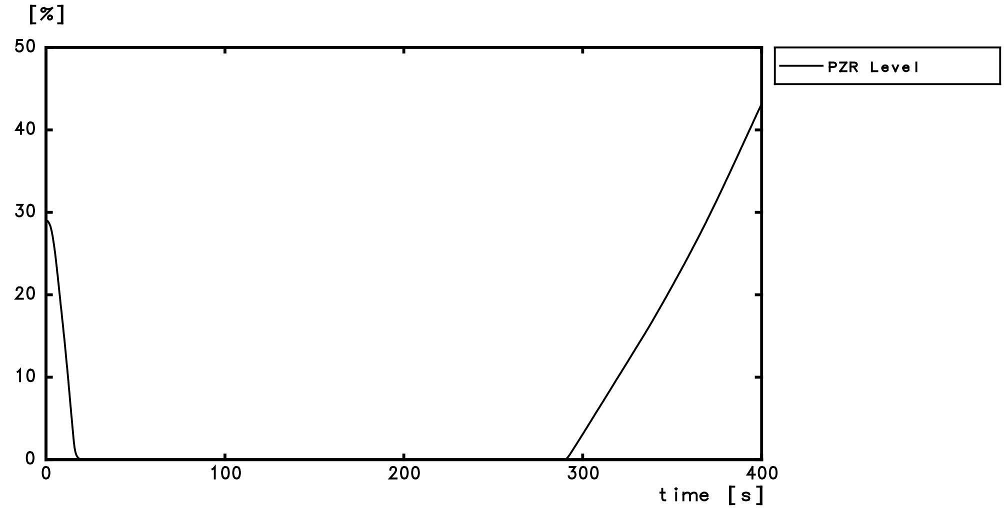
APPENDIX 14C - FIGURE 6 [REF-1]

Primary Pressure – Pressuriser Level

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) – EPR 4250 MWth



PRIMARY PRESSURE

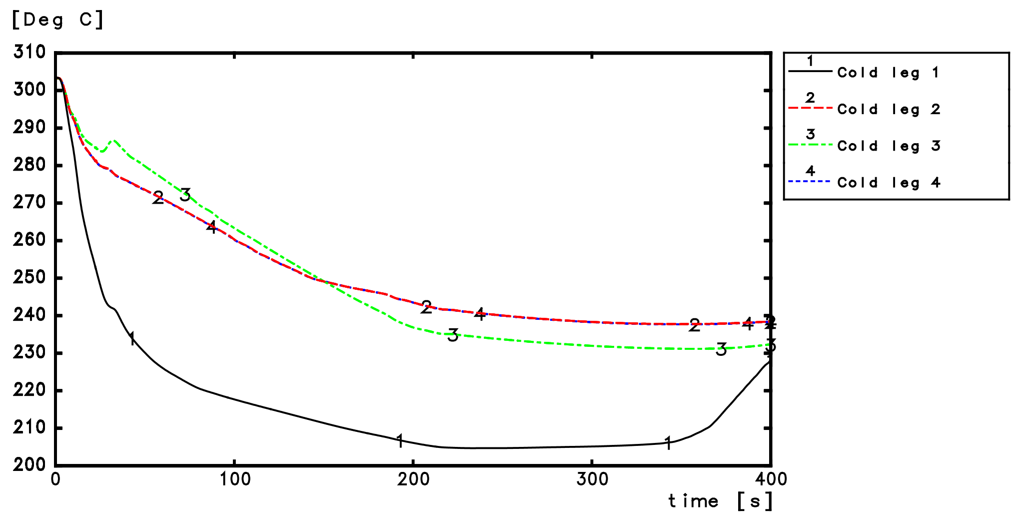


PRESSURISER LEVEL

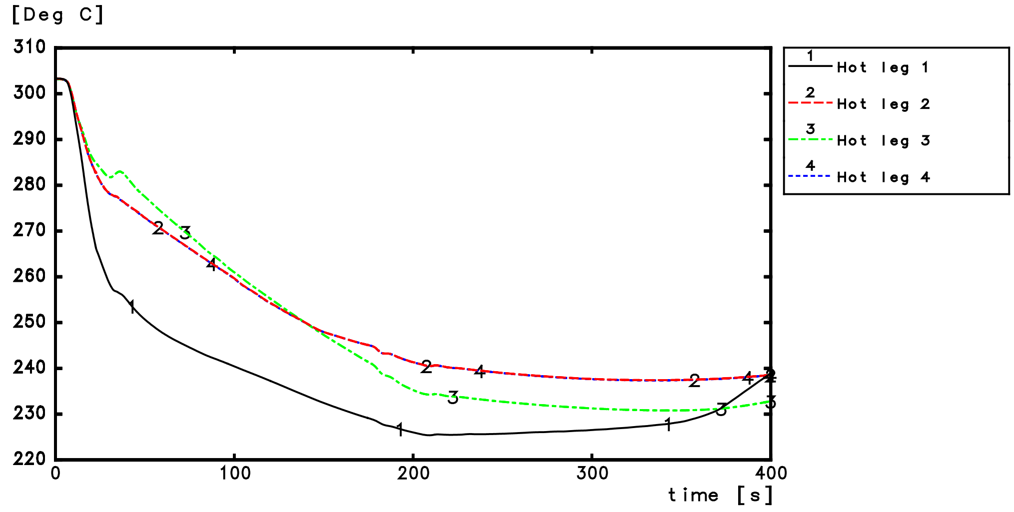
APPENDIX 14C - FIGURE 7 [REF-1]

Cold Leg Temperatures – Hot Leg Temperatures

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) – EPR 4250 MWth



COLD LEGS TEMPERATURES

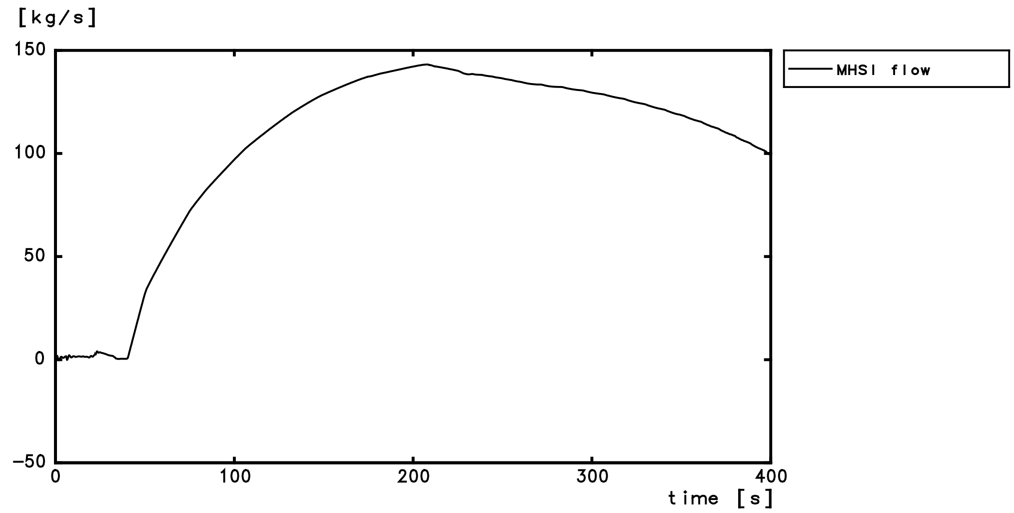


HOT LEGS TEMPERATURES

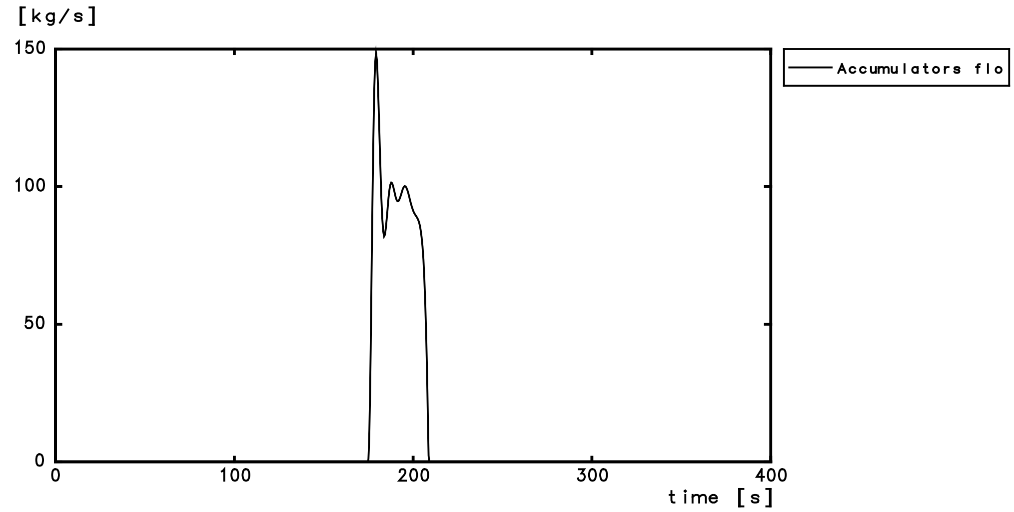
APPENDIX 14C - FIGURE 8 [REF-1]

Total MHSI Flow – Total Accumulators Flow

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) – EPR 4250 MWth



TOTAL MHSI FLOW

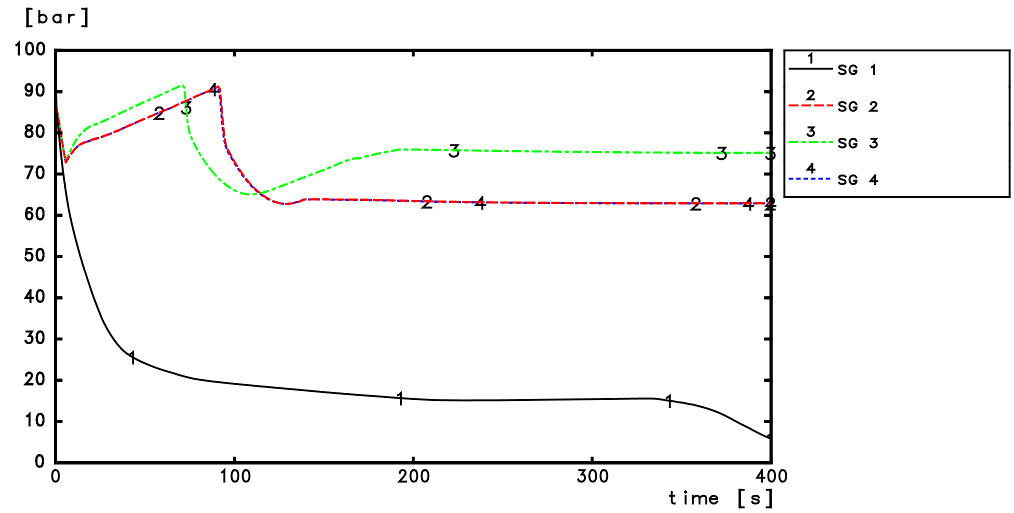


TOTAL ACCUMULATORS FLOW

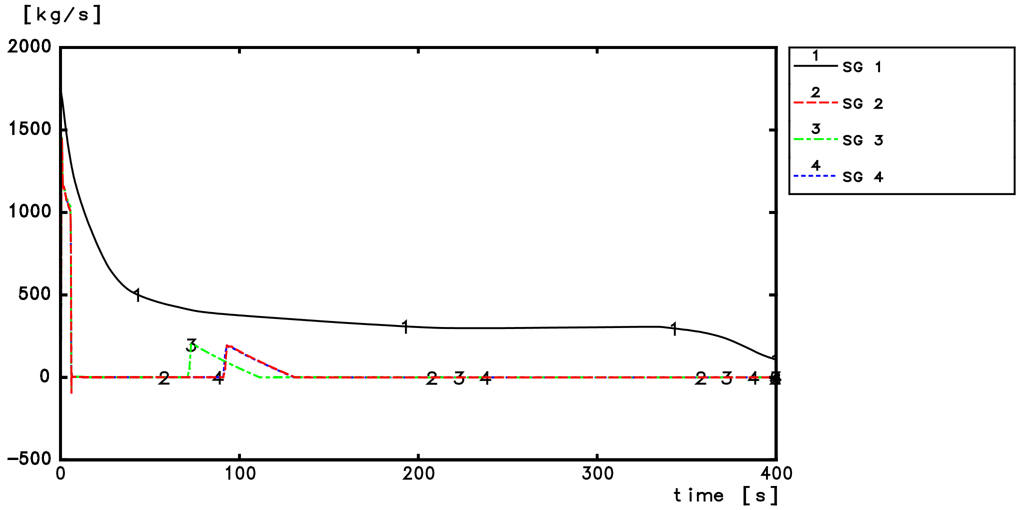
APPENDIX 14C - FIGURE 9 [REF-1]

Steam Generator Pressure – Steam Generator Steam Flow

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) – EPR 4250 MWth



STEAM GENERATOR PRESSURE

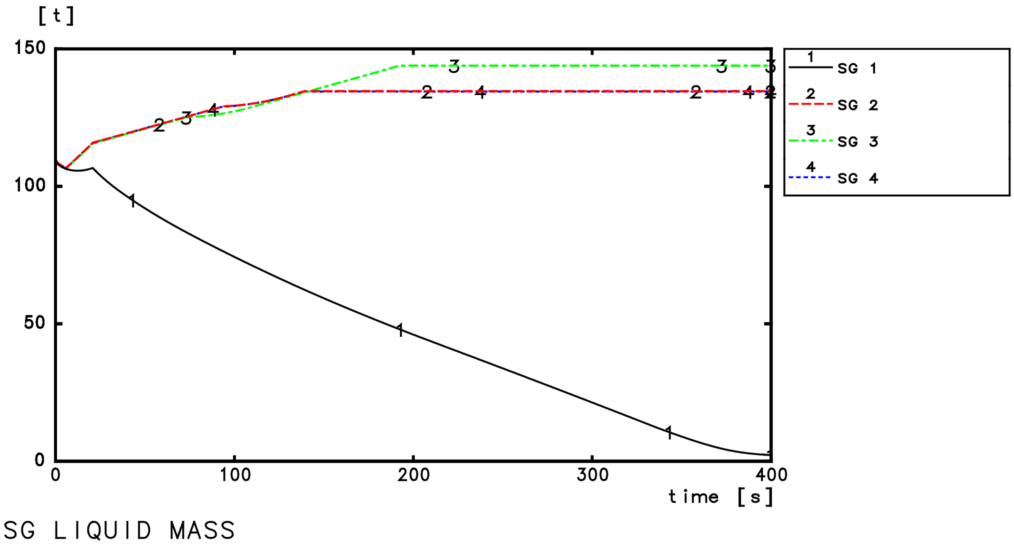


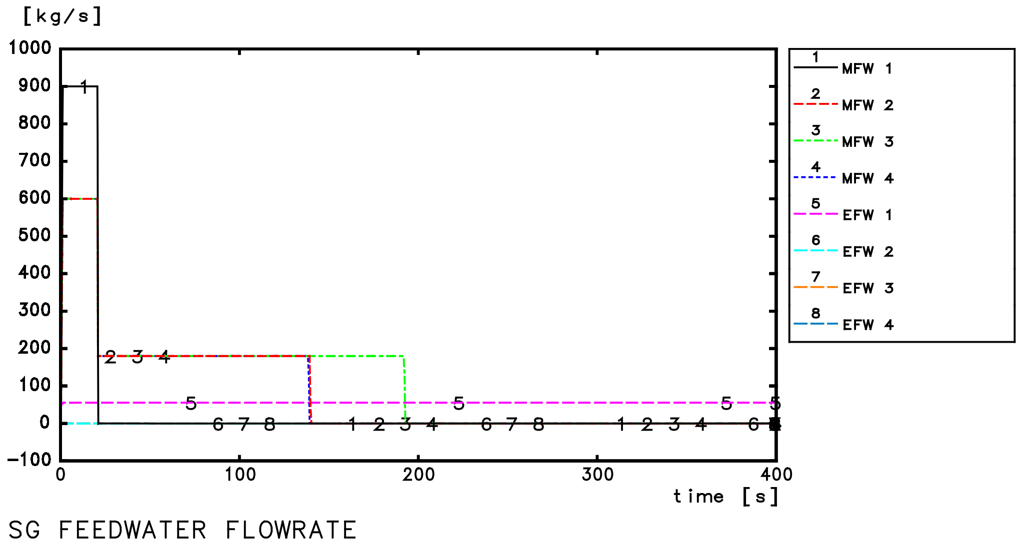
STEAM GENERATOR STEAM FLOW

APPENDIX 14C - FIGURE 10 [REF-1]

SG Liquid Mass – SG Feedwater Flow Rate

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on Stuck Rod) – EPR 4250 MWth

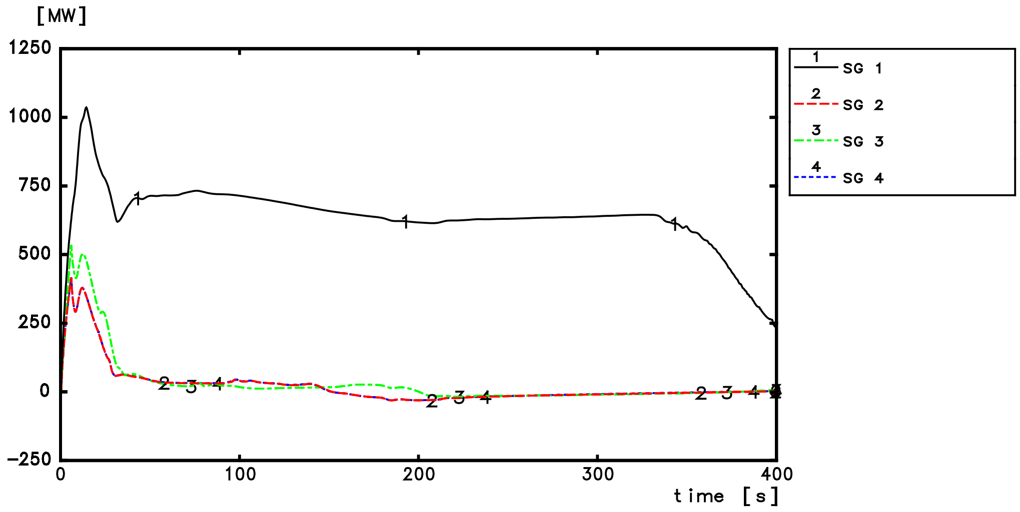




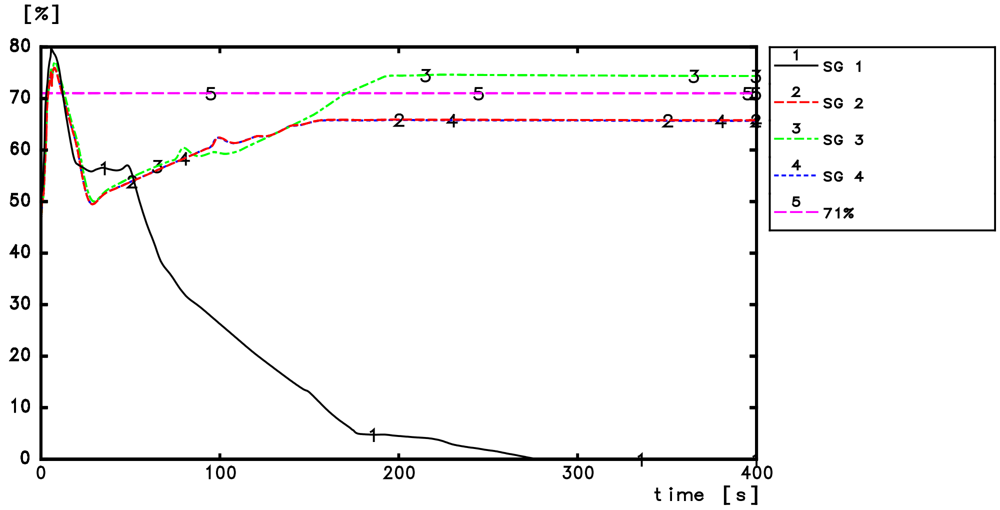
APPENDIX 14C - FIGURE 11 [REF-1]

SG Power Exchange – SG Narrow Range (NR) Level

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on Stuck Rod) – EPR 4250 MWth



SG POWER EXCHANGE

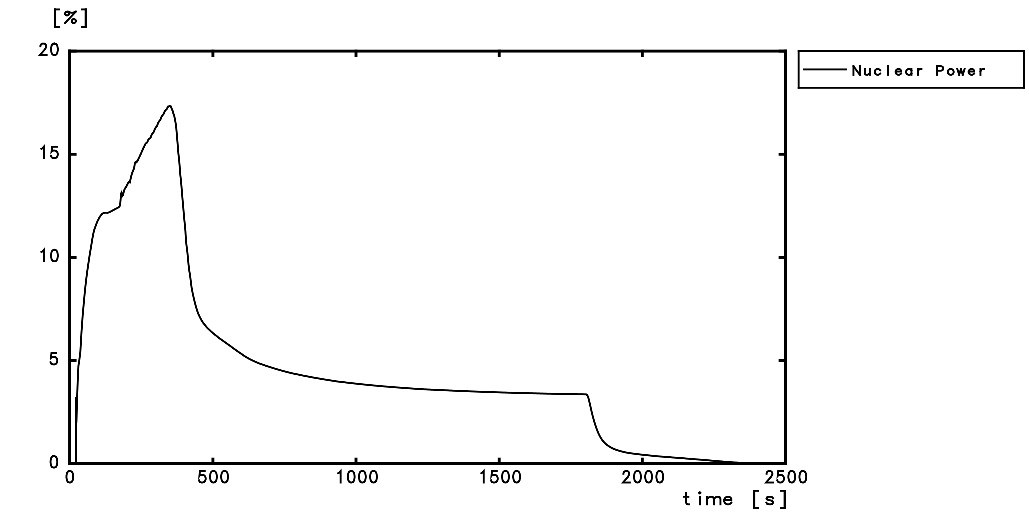


SG NR LEVEL

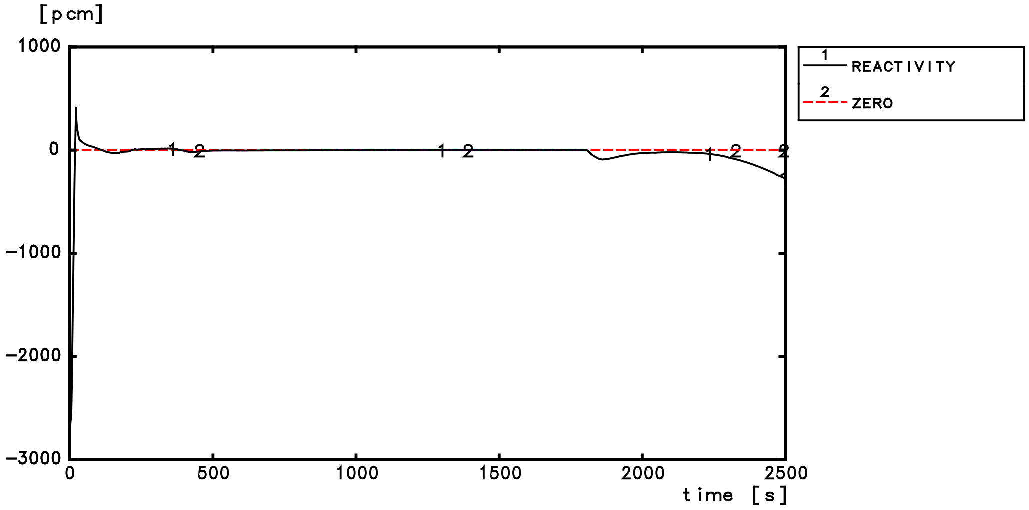
APPENDIX 14C - FIGURE 12 [REF-1]

Nuclear Power – Reactivity

Double-Ended Guillotine Break Upstream of the VIV [MSIV]- Long Term Phase – EPR 4250 MWth



NUCLEAR POWER

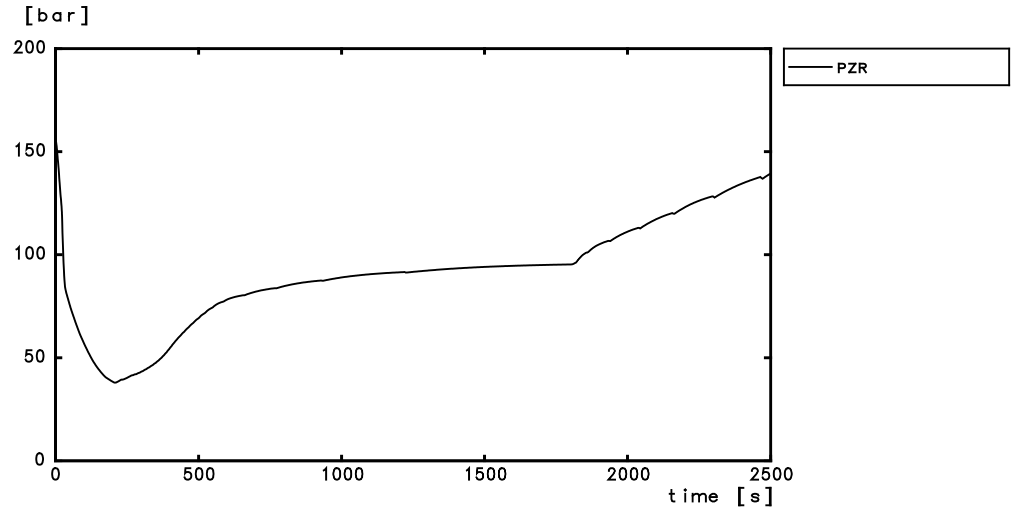


REACTIVITY

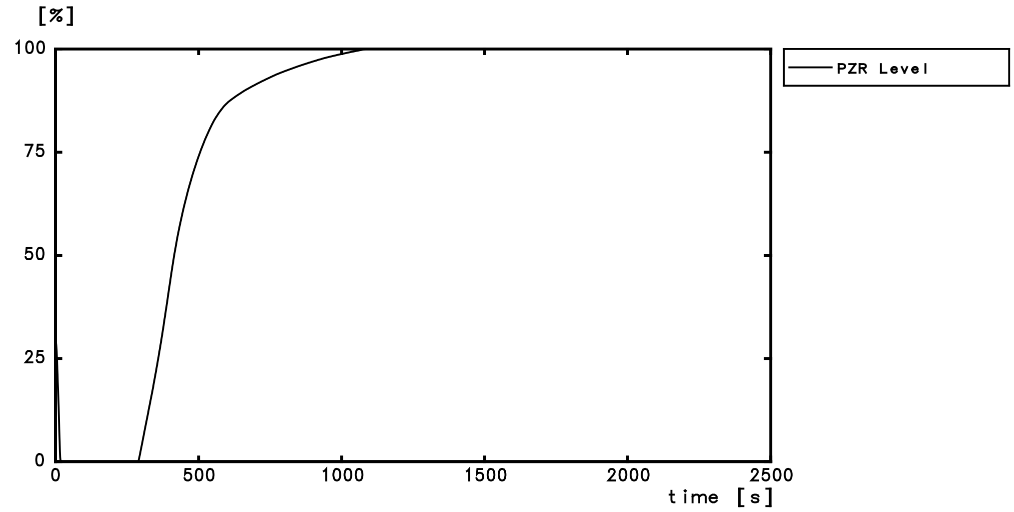
APPENDIX 14C - FIGURE 13 [REF-1]

Primary Pressure – Pressuriser Level

Double-Ended Guillotine Break Upstream of the VIV [MSIV]- Long Term Phase – EPR 4250 MWth



PRIMARY PRESSURE

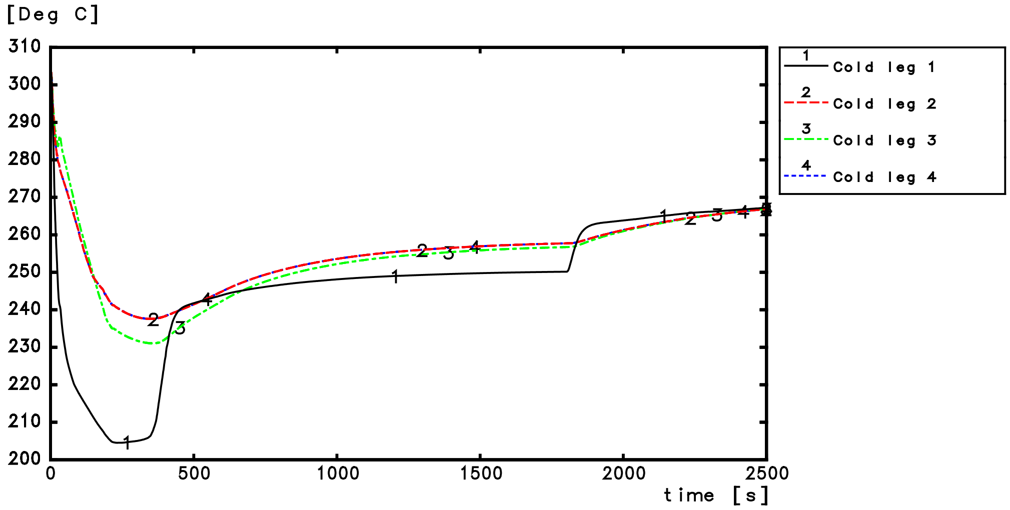


PRESSURISER LEVEL

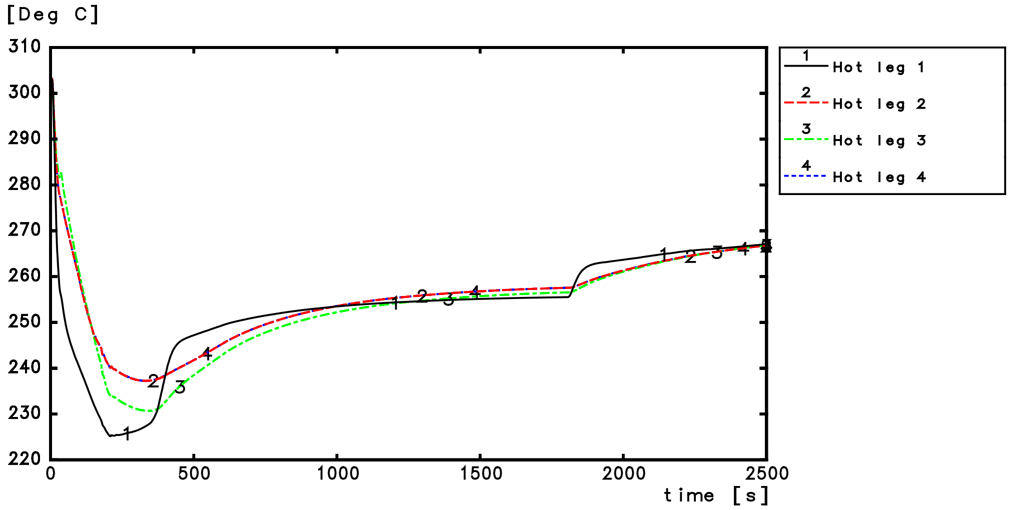
APPENDIX 14C - FIGURE 14 [REF-1]

Cold Leg Temperatures – Hot Leg Temperatures

Double-Ended Guillotine Break Upstream of the VIV [MSIV]- Long Term Phase – EPR 4250 MWth



COLD LEGS TEMPERATURES

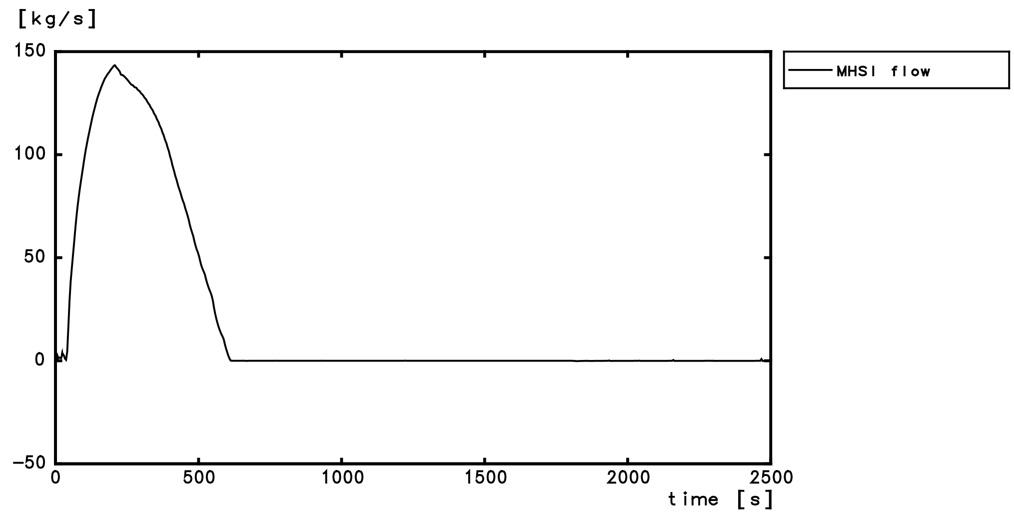


HOT LEGS TEMPERATURES

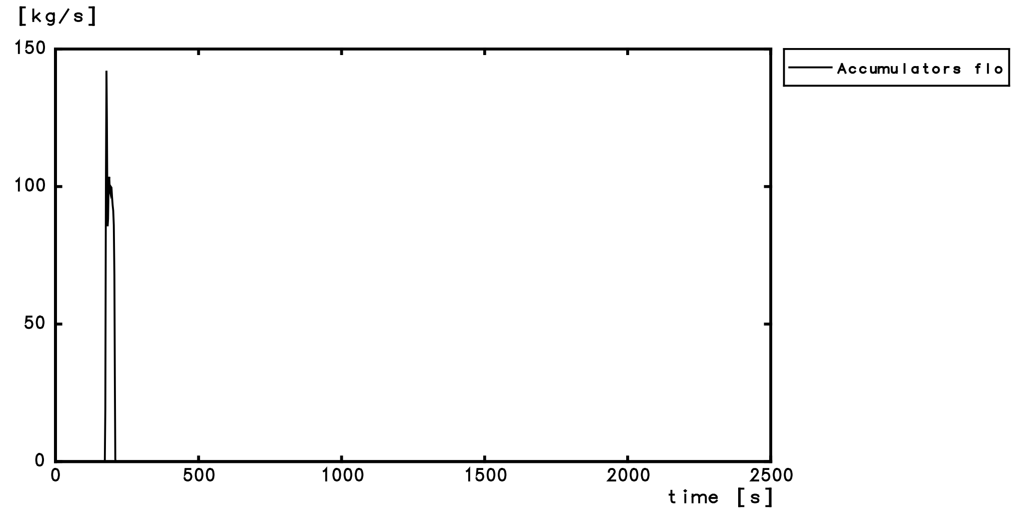
APPENDIX 14C - FIGURE 15 [REF-1]

Total MHSI Flow – Total Accumulators Flow

Double-Ended Guillotine Break Upstream of the VIV [MSIV]– Long Term Phase – EPR 4250 MWth



TOTAL MHSI FLOW

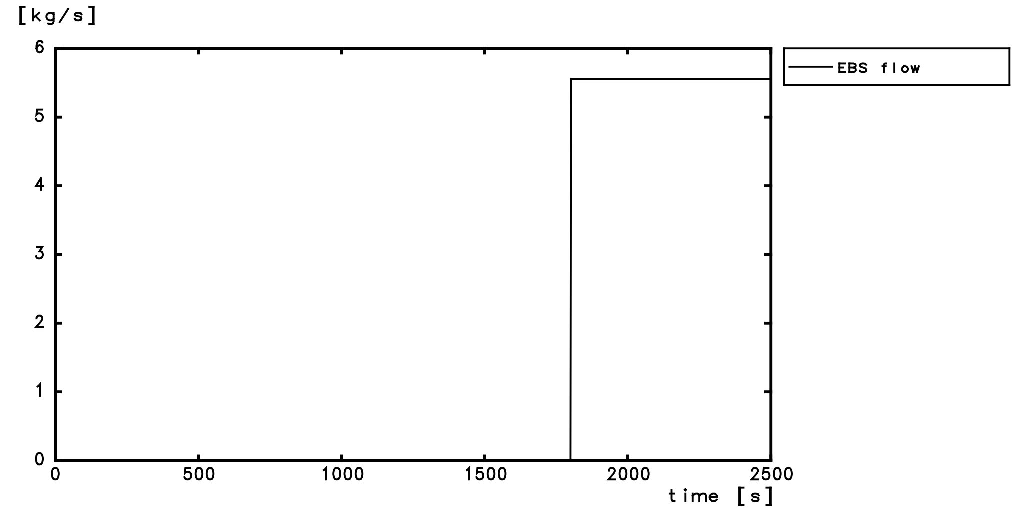


TOTAL ACCUMULATORS FLOW

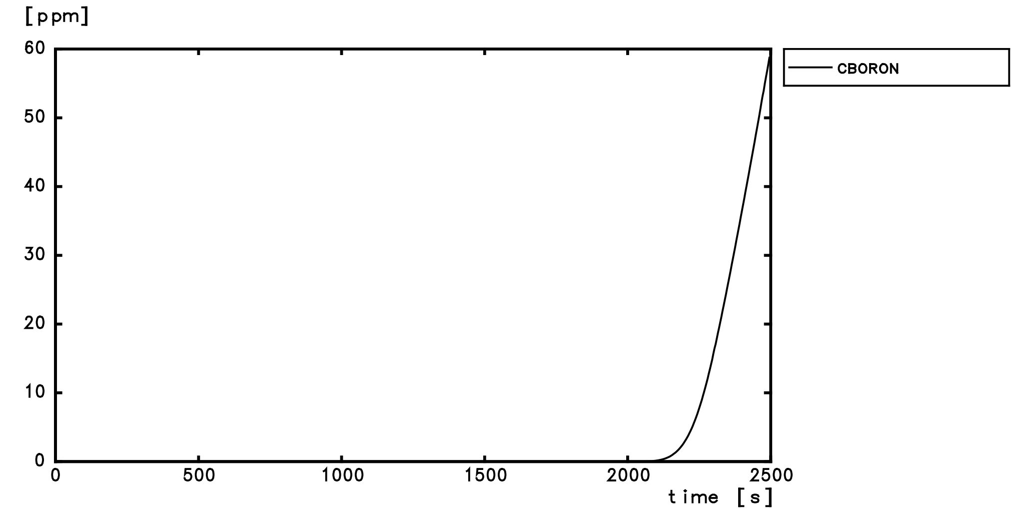
APPENDIX 14C - FIGURE 16 [REF-1]

Total EBS [RBS] Flow – Core Boron Concentration

Double-Ended Guillotine Break Upstream of the VIV [MSIV]- Long Term Phase – EPR 4250 MWth



TOTAL EBS FLOW

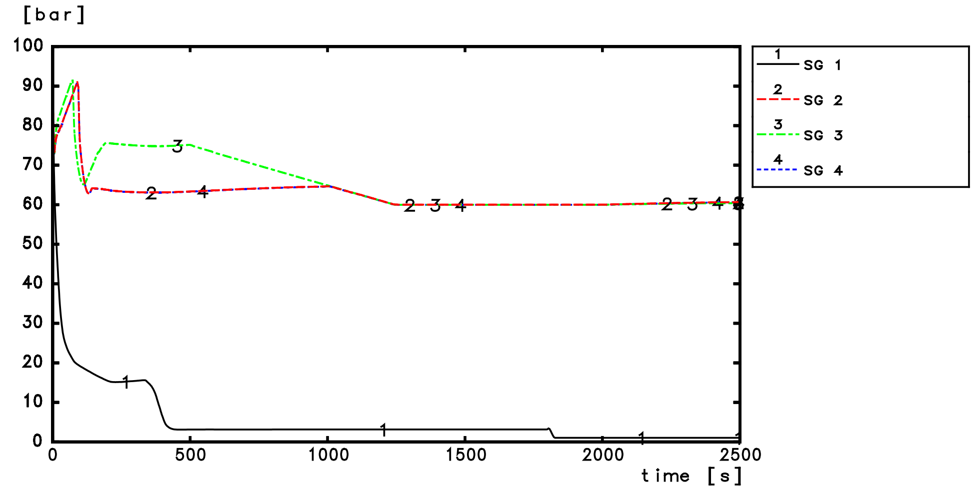


BORON CONCENTRATION

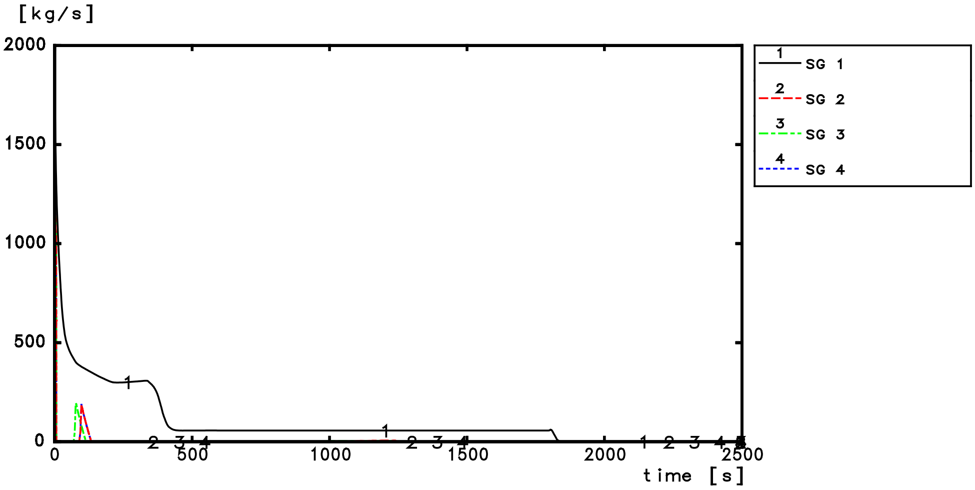
APPENDIX 14C - FIGURE 17 [REF-1]

Steam Generator Pressure – Steam Generator Steam Flow

Double-Ended Guillotine Break Upstream of the VIV [MSIV]- Long Term Phase – EPR 4250 MWth



STEAM GENERATOR PRESSURE

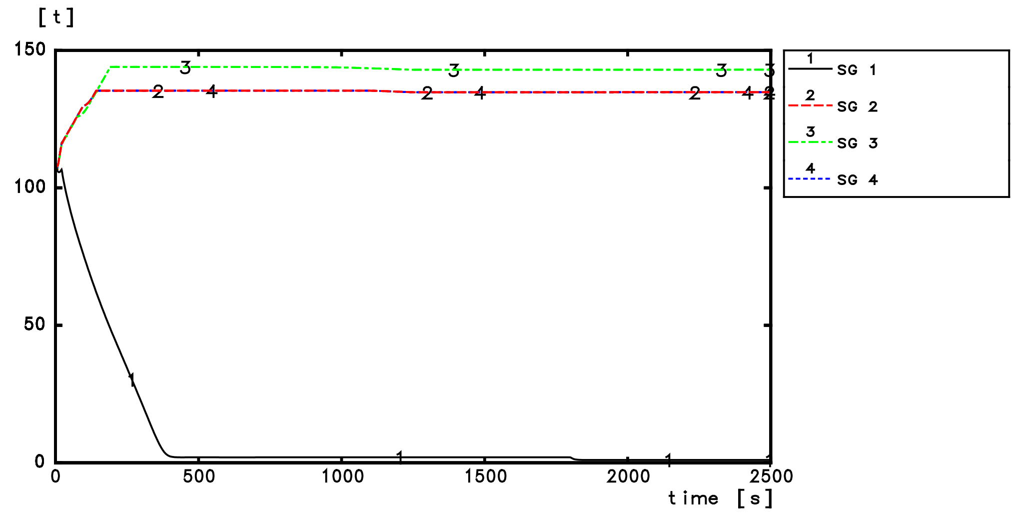


STEAM GENERATOR STEAM FLOW

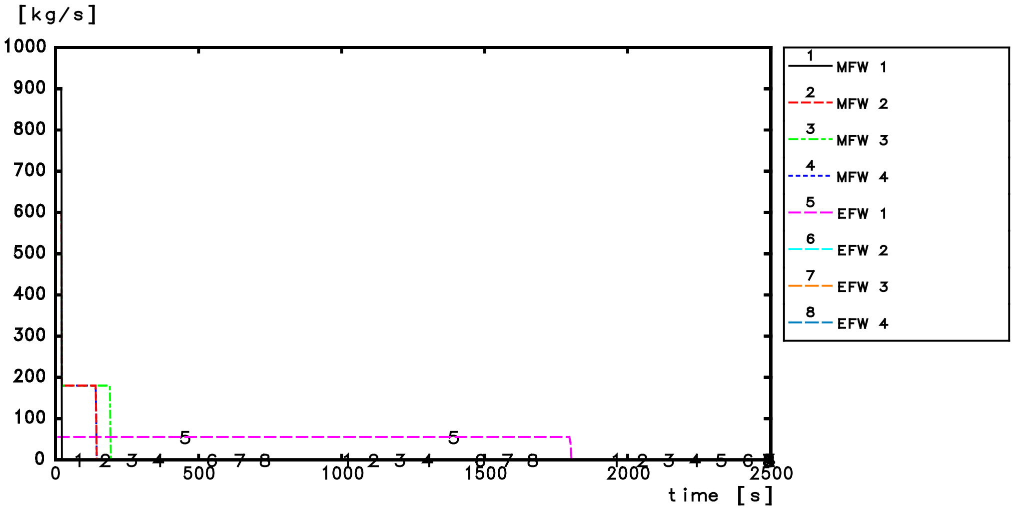
APPENDIX 14C - FIGURE 18 [REF-1]

SG Liquid Mass – SG Feedwater Flow Rate

Double-Ended Guillotine Break Upstream of the VIV [MSIV]- Long Term Phase – EPR 4250 MWth



SG LIQUID MASS

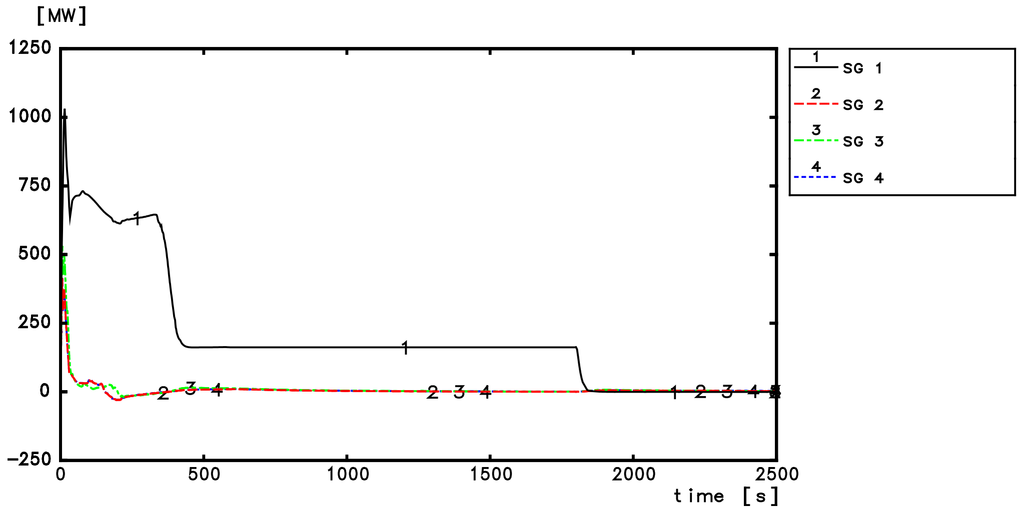


SG FEEDWATER FLOWRATE

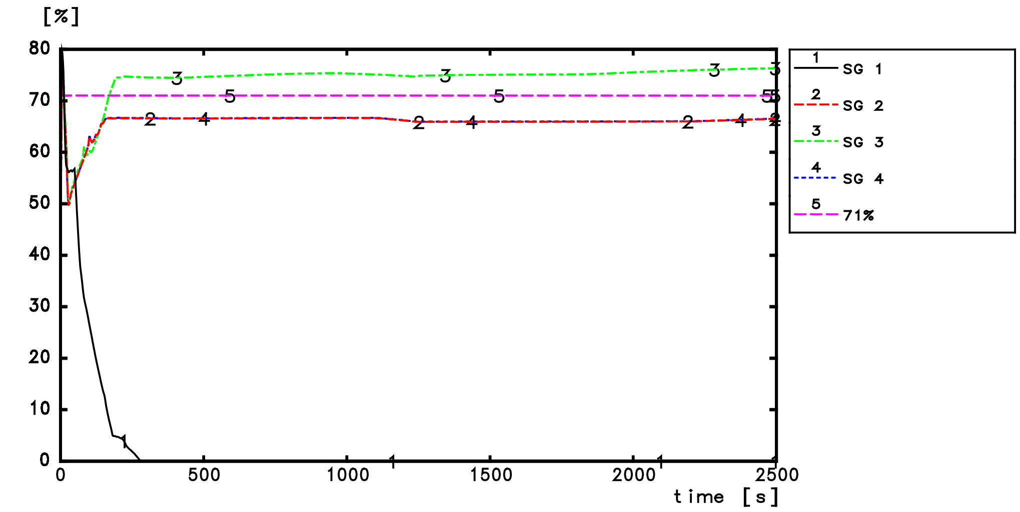
APPENDIX 14C - FIGURE 19 [REF-1]

SG Power Exchange – SG NR Level

Double-Ended Guillotine Break Upstream of the VIV [MSIV]- Long Term Phase – EPR 4250 MWth



SG POWER EXCHANGE

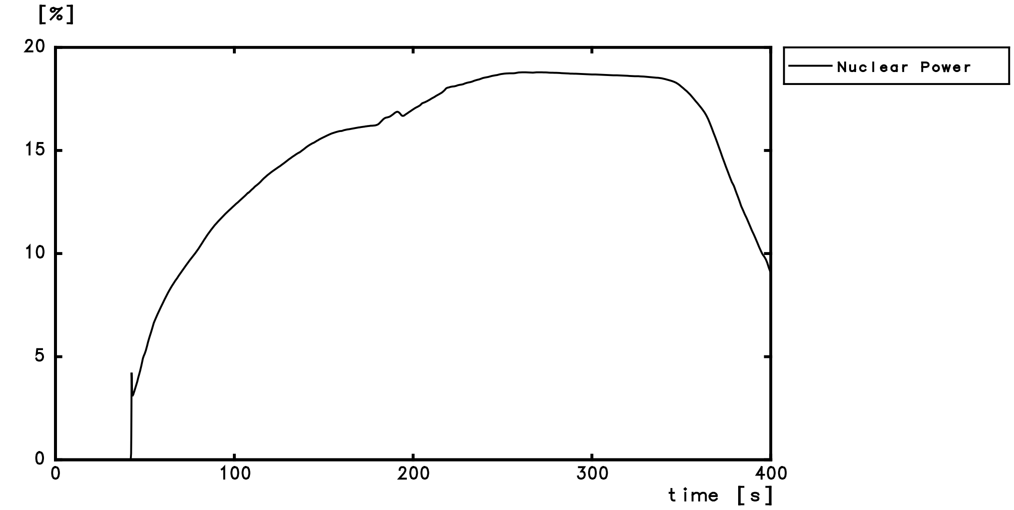


SG NR LEVEL

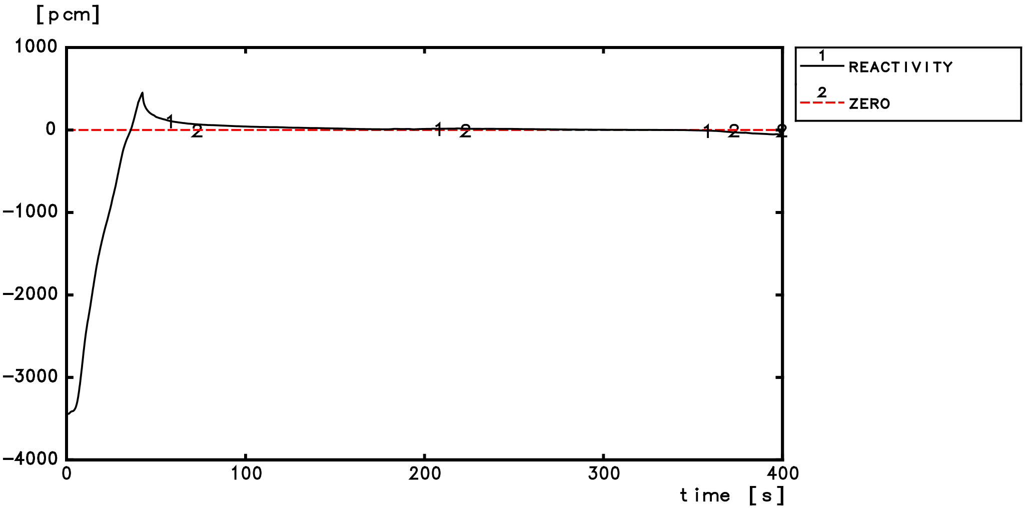
APPENDIX 14C - FIGURE 20 [REF-1]

Nuclear Power – Reactivity

Double-Ended Guillotine Break Downstream of the VIV [MSIV] (SF on VIV [MSIV]) – EPR 4250 MWth



NUCLEAR POWER

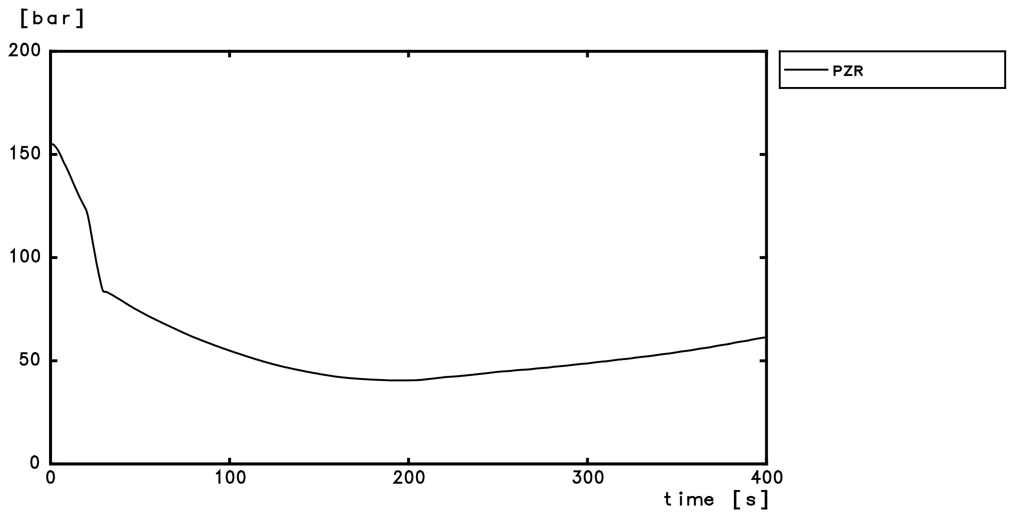


REACTIVITY

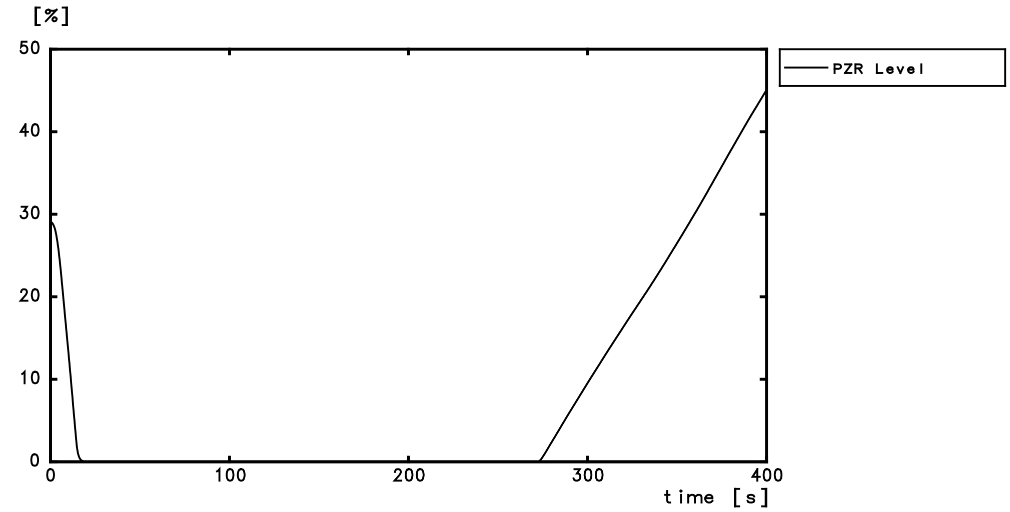
APPENDIX 14C - FIGURE 21 [REF-1]

Primary Pressure – Pressuriser Level

Double-Ended Guillotine Break Downstream of the VIV [MSIV] (SF on VIV [MSIV]) – EPR 4250 MWth



PRIMARY PRESSURE

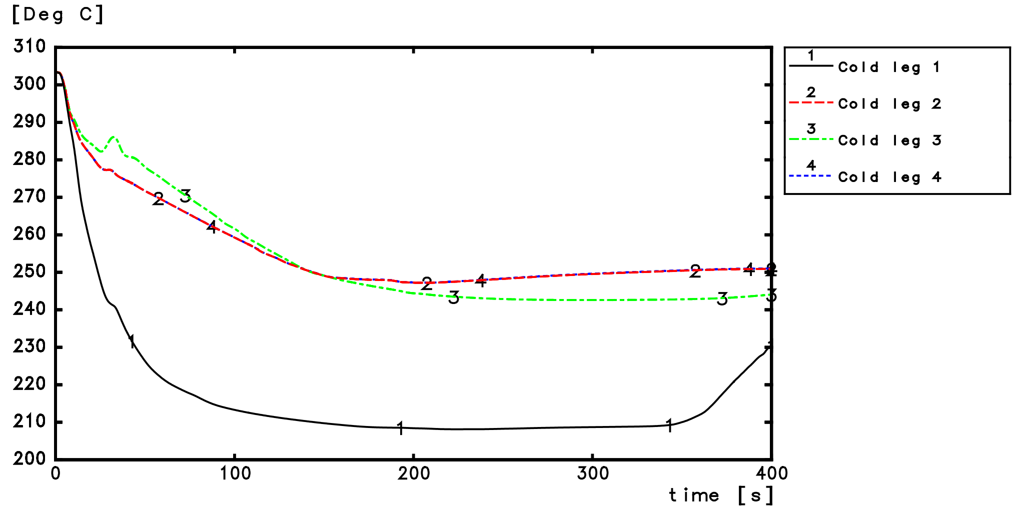


PRESSURISER LEVEL

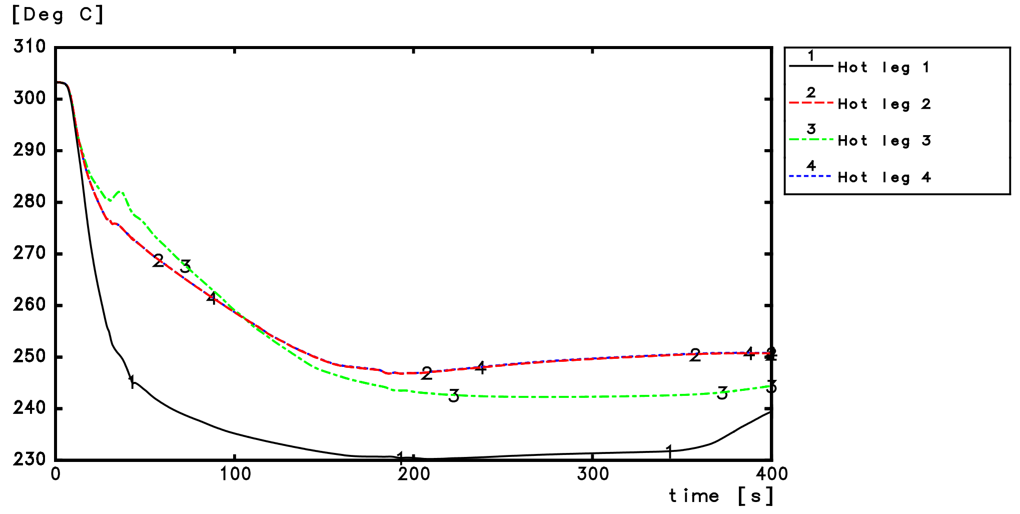
APPENDIX 14C - FIGURE 22 [REF-1]

Cold Leg Temperatures – Hot Leg Temperatures

Double-Ended Guillotine Break Downstream of the VIV [MSIV] (SF on VIV [MSIV]) – EPR 4250 MWth



COLD LEGS TEMPERATURES

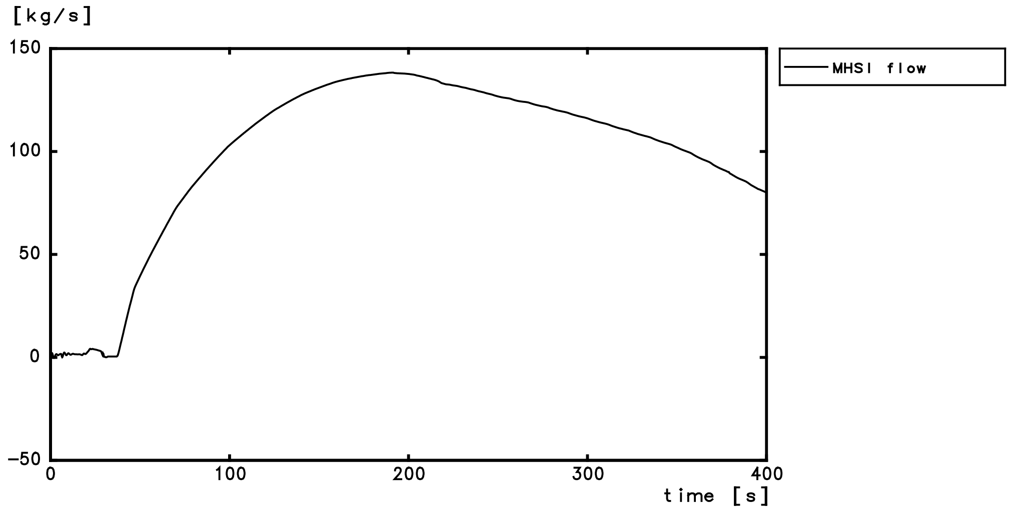


HOT LEGS TEMPERATURES

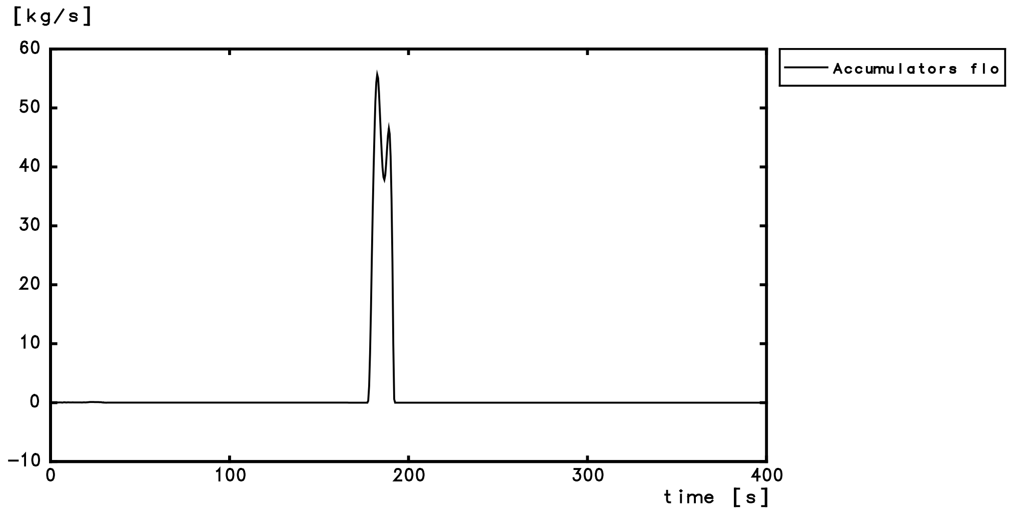
APPENDIX 14C - FIGURE 23 [REF-1]

Total MHSI Flow – Total Accumulators Flow

Double-Ended Guillotine Break Downstream of the VIV [MSIV] (SF on VIV [MSIV]) – EPR 4250 MWth



TOTAL MHSI FLOW

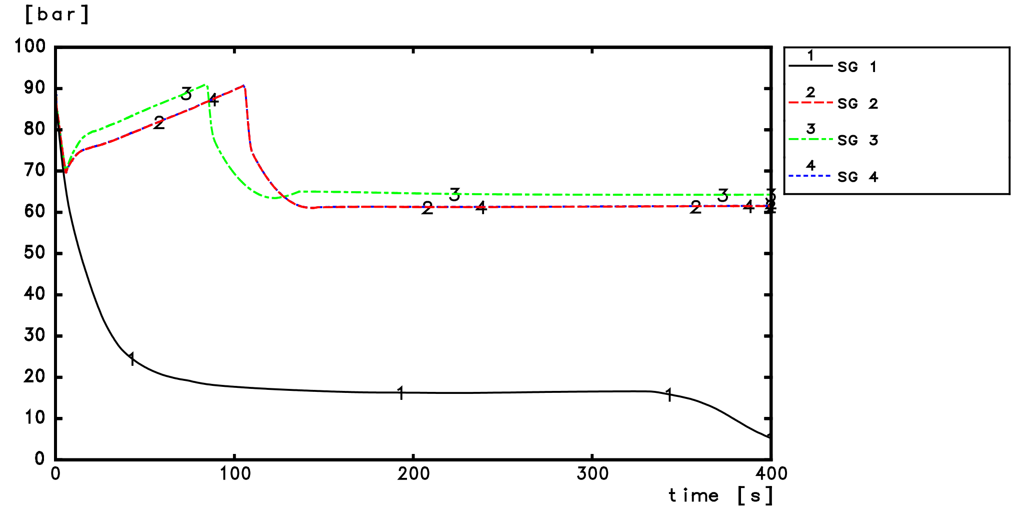


TOTAL ACCUMULATORS FLOW

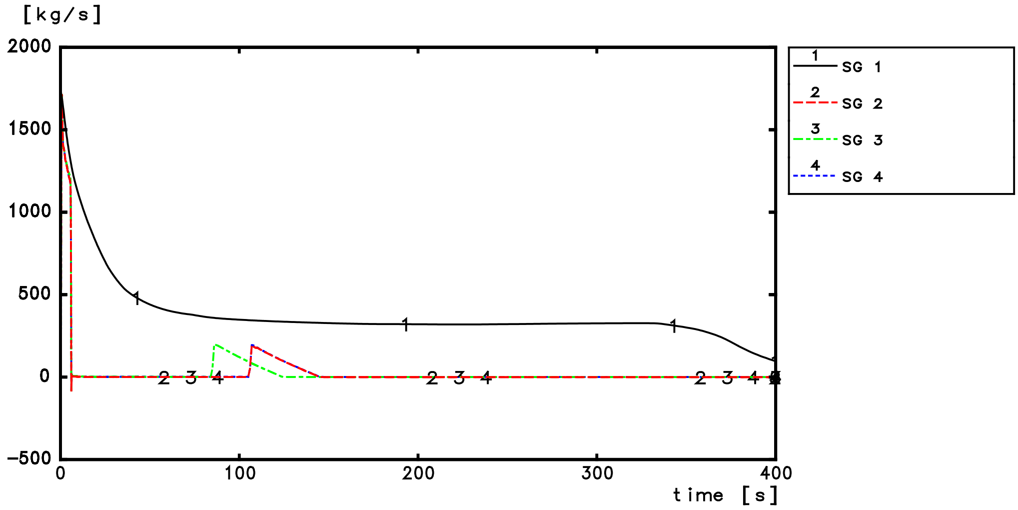
APPENDIX 14C - FIGURE 24 [REF-1]

Steam Generator Pressure – Steam Generator Steam Flow

Double-Ended Guillotine Break Downstream of the VIV [MSIV](SF on VIV [MSIV]) – EPR 4250 MWth



STEAM GENERATOR PRESSURE

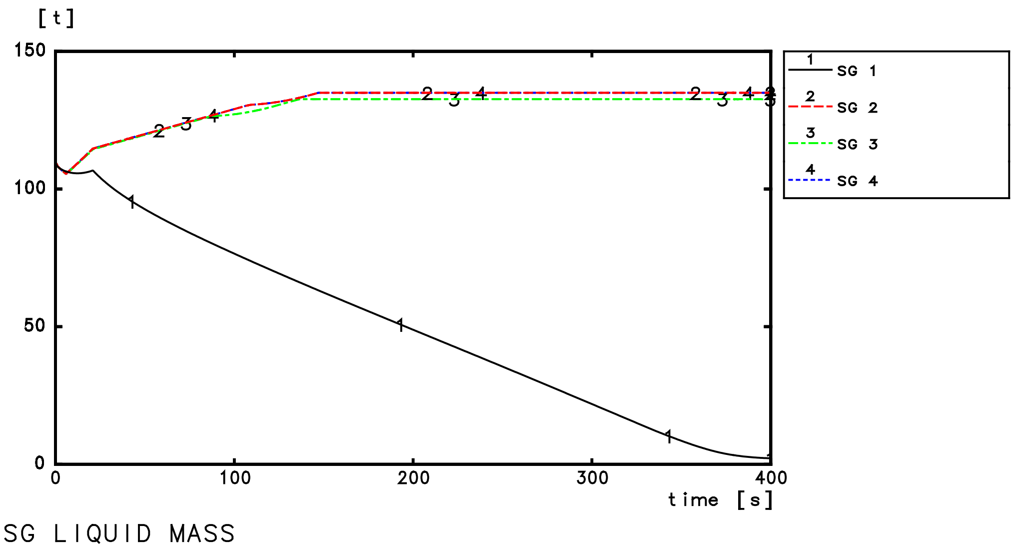


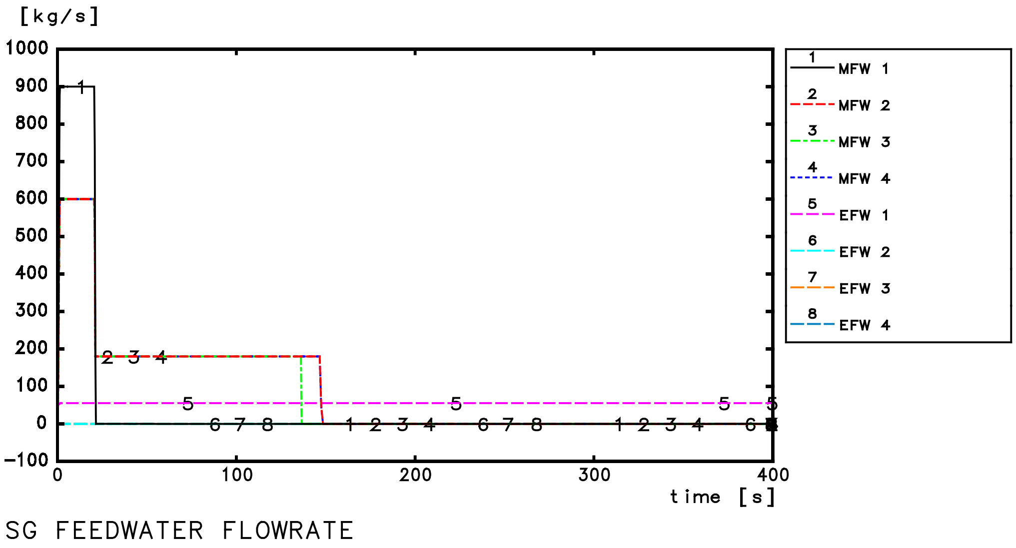
STEAM GENERATOR STEAM FLOW

APPENDIX 14C - FIGURE 25 [REF-1]

SG Liquid Mass – SG Feedwater Flow Rate

Double-Ended Guillotine Break Downstream of the VIV [MSIV] (SF on VIV [MSIV]) – EPR 4250 MWth

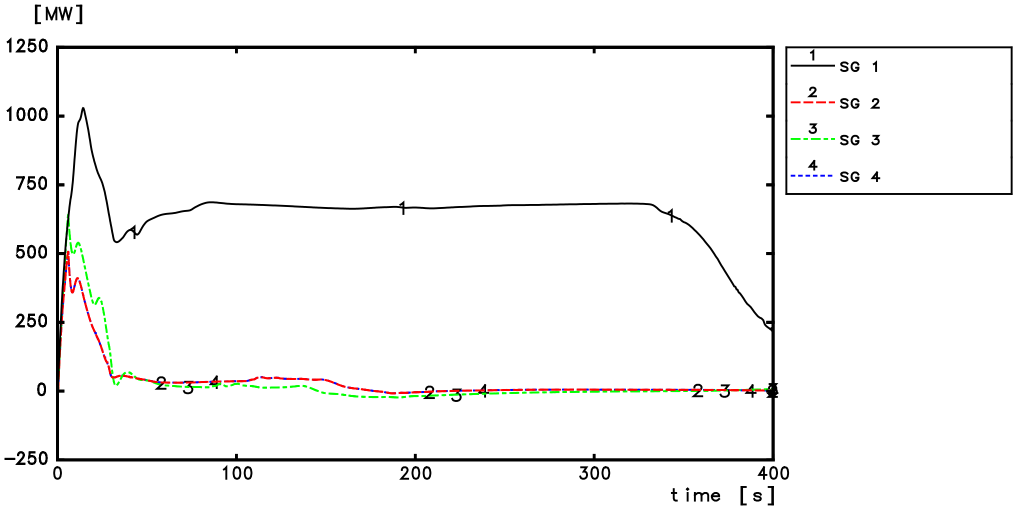




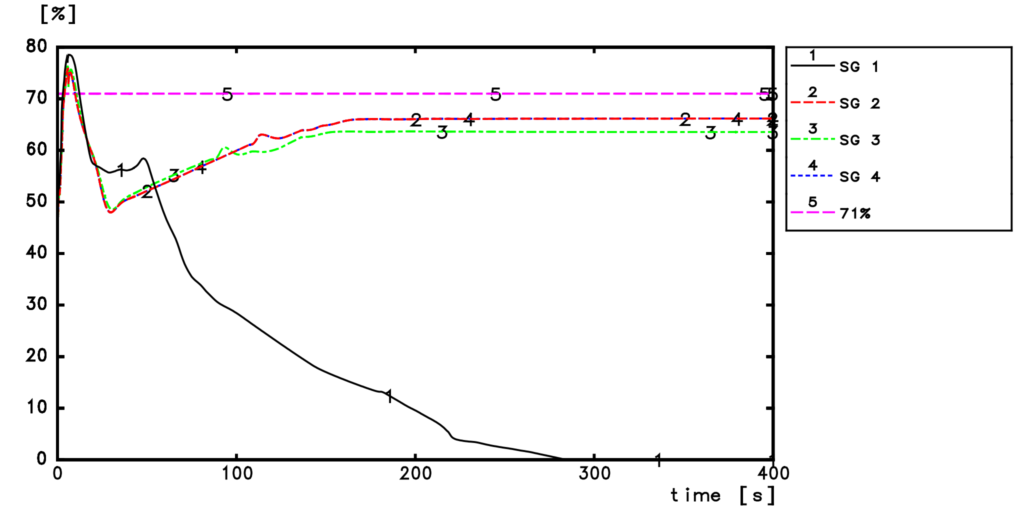
APPENDIX 14C - FIGURE 26 [REF-1]

SG Power Exchange – SG NR Level

Double-Ended Guillotine Break Downstream of the VIV [MSIV] (SF on VIV [MSIV]) – EPR 4250 MWth



SG POWER EXCHANGE

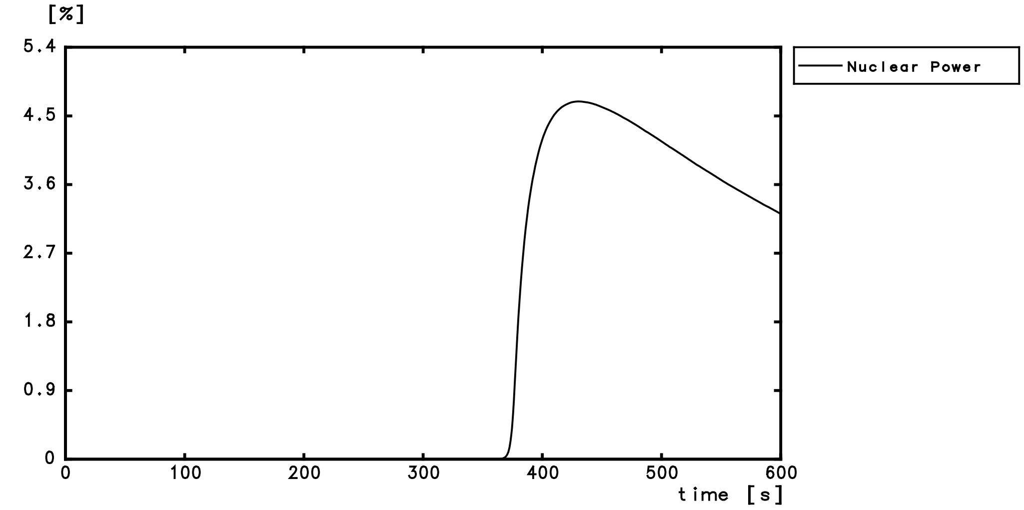


SG NR LEVEL

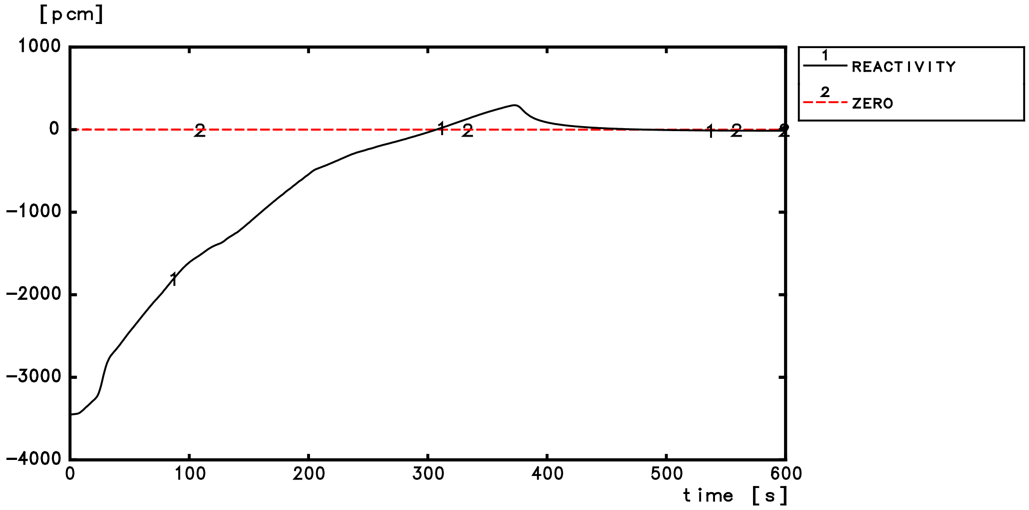
APPENDIX 14C - FIGURE 27 [REF-1]

Nuclear Power – Reactivity

Spurious MSSV Opening (SF on VDA [MSRT]) – EPR 4250 MWth



NUCLEAR POWER

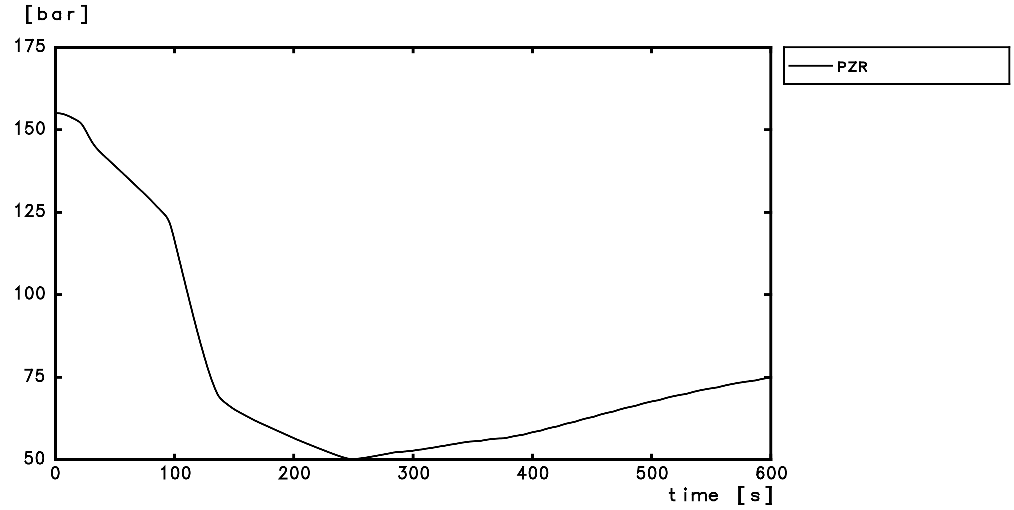


REACTIVITY

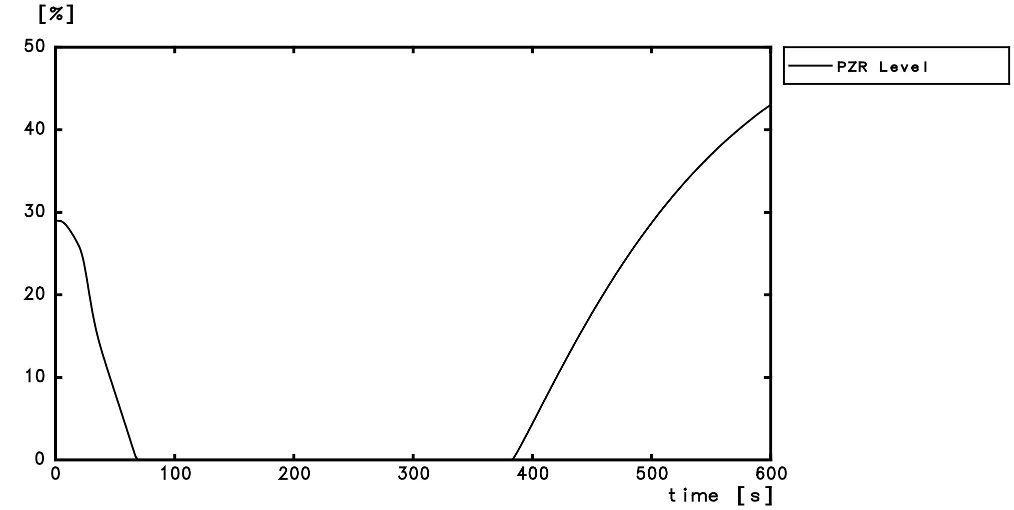
APPENDIX 14C - FIGURE 28 [REF-1]

Primary Pressure – Pressuriser Level

Spurious MSSV Opening (SF on VDA [MSRT]) – EPR 4250 MWth



PRIMARY PRESSURE

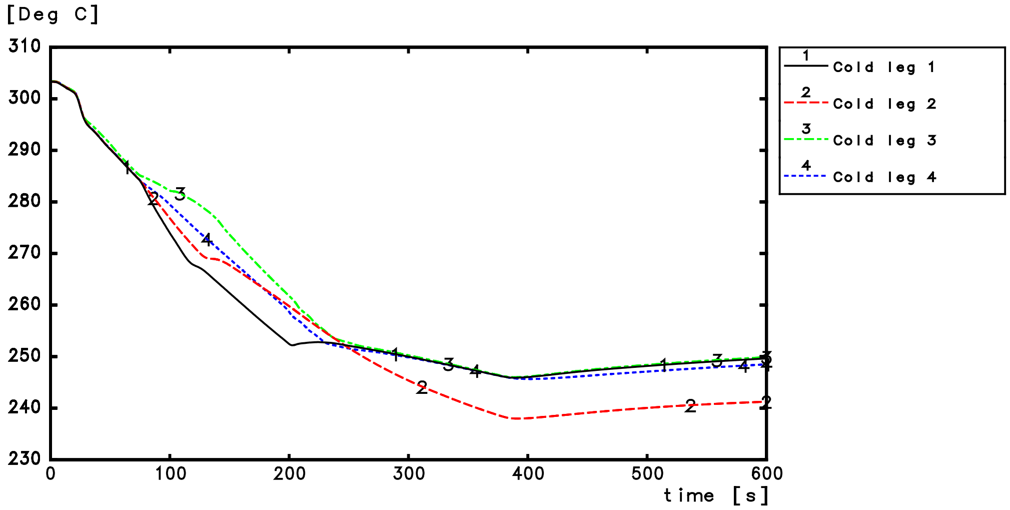


PRESSURISER LEVEL

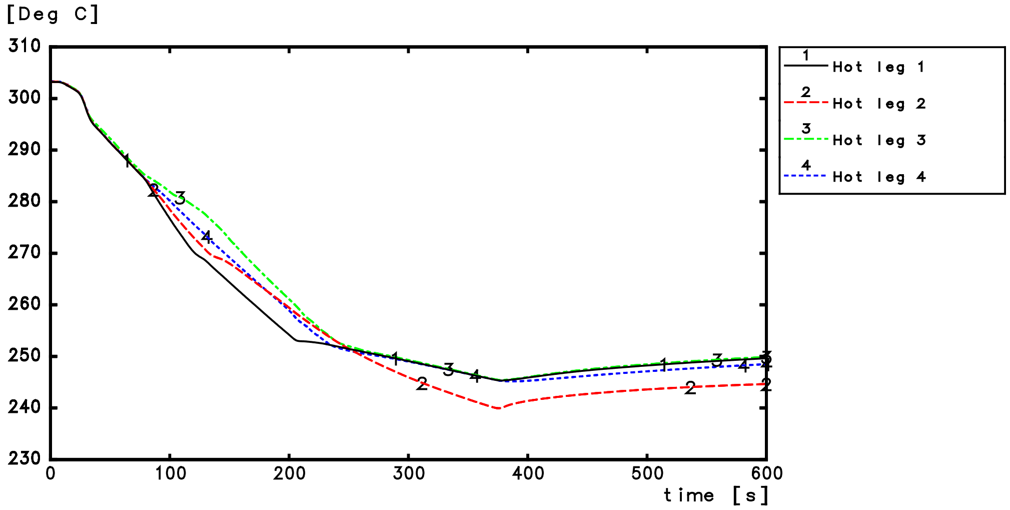
APPENDIX 14C - FIGURE 29 [REF-1]

Cold Leg Temperatures – Hot Leg Temperatures

Spurious MSSV Opening (SF on VDA [MSRT]) – EPR 4250 MWth



COLD LEGS TEMPERATURES

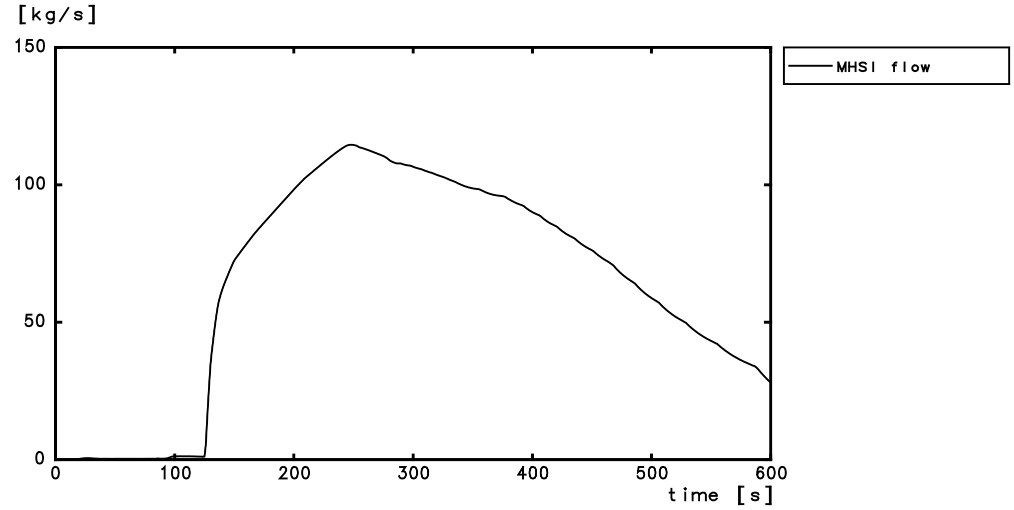


HOT LEGS TEMPERATURES

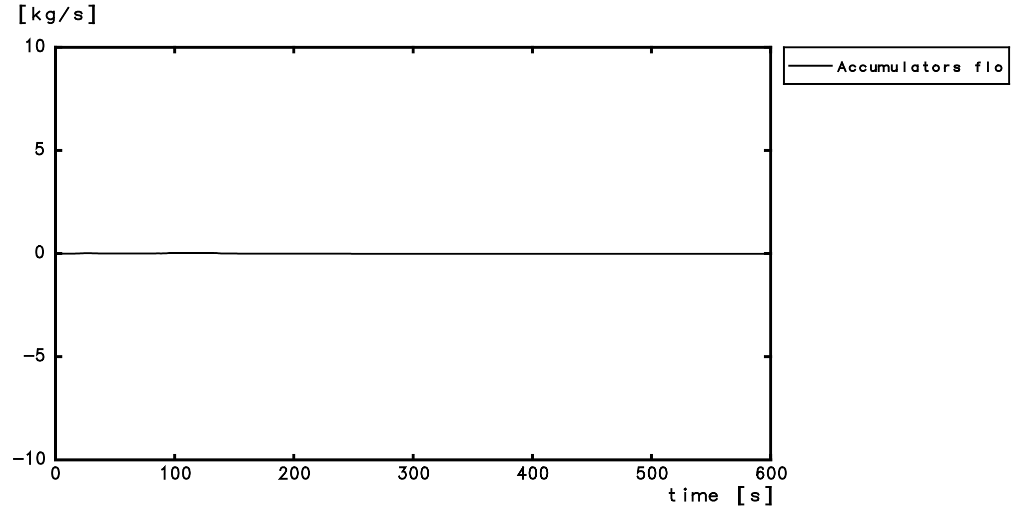
APPENDIX 14C - FIGURE 30 [REF-1]

Total MHSI Flow – Total Accumulators Flow

Spurious MSSV Opening (SF on VDA [MSRT]) – EPR 4250 MWth



TOTAL MHSI FLOW

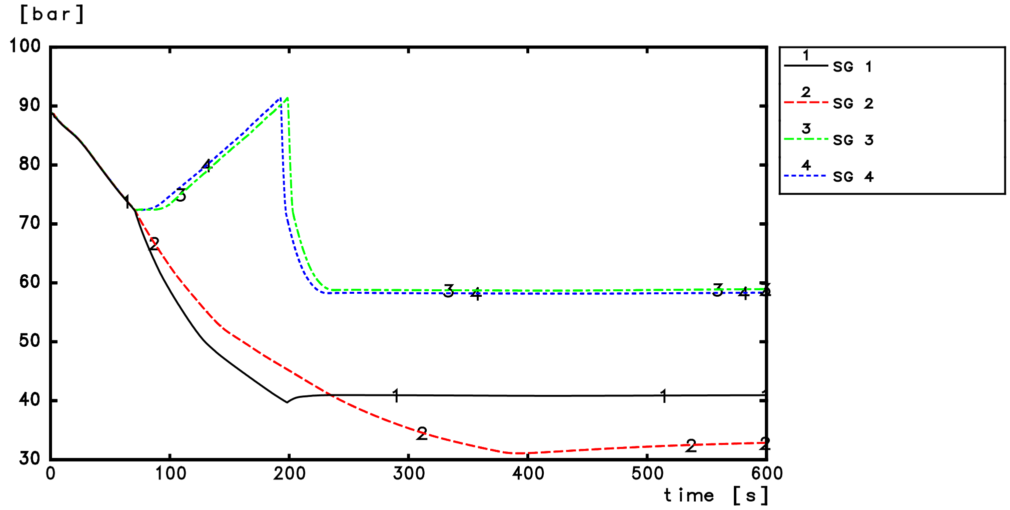


TOTAL ACCUMULATORS FLOW

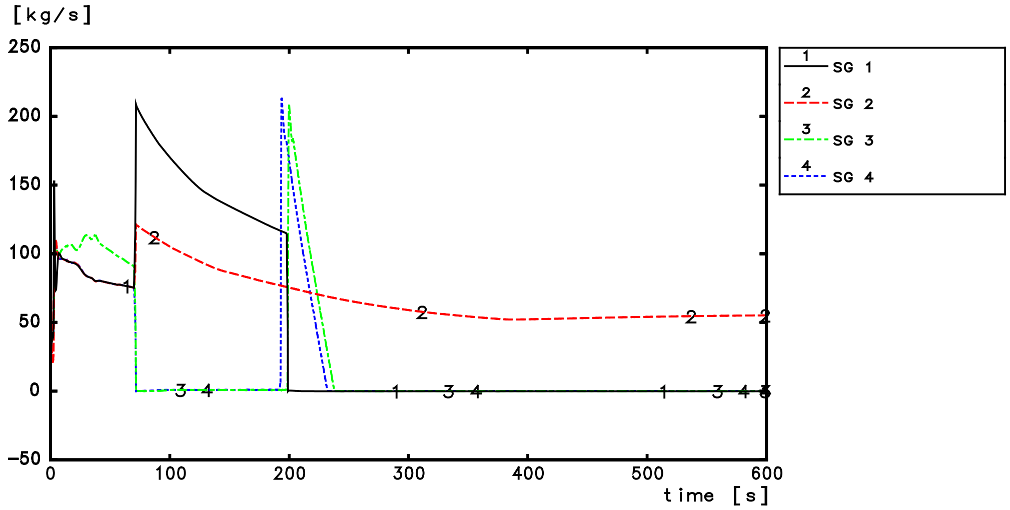
APPENDIX 14C - FIGURE 31 [REF-1]

Steam Generator Pressure – Steam Generator Steam Flow

Spurious MSSV Opening (SF on VDA [MSRT]) – EPR 4250 MWth



STEAM GENERATOR PRESSURE

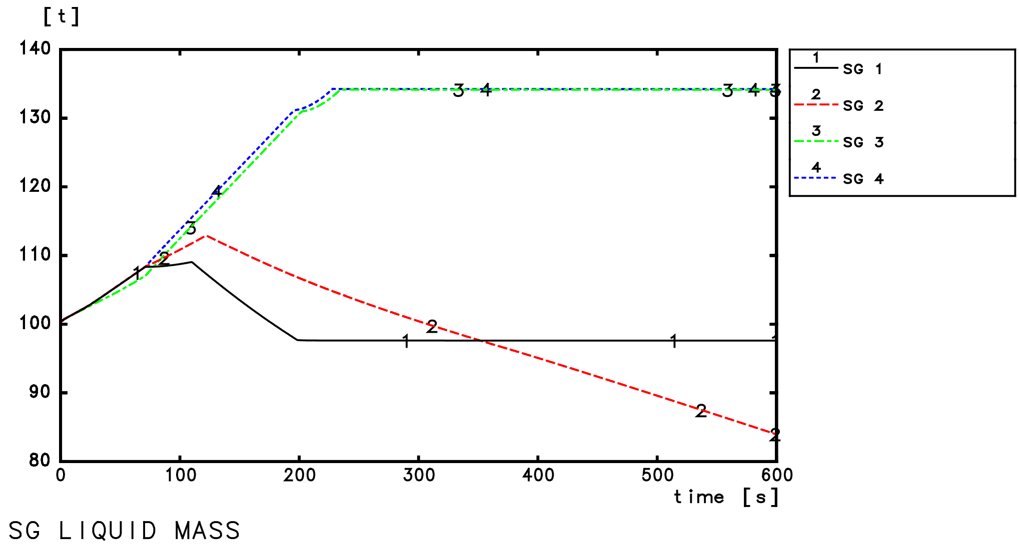


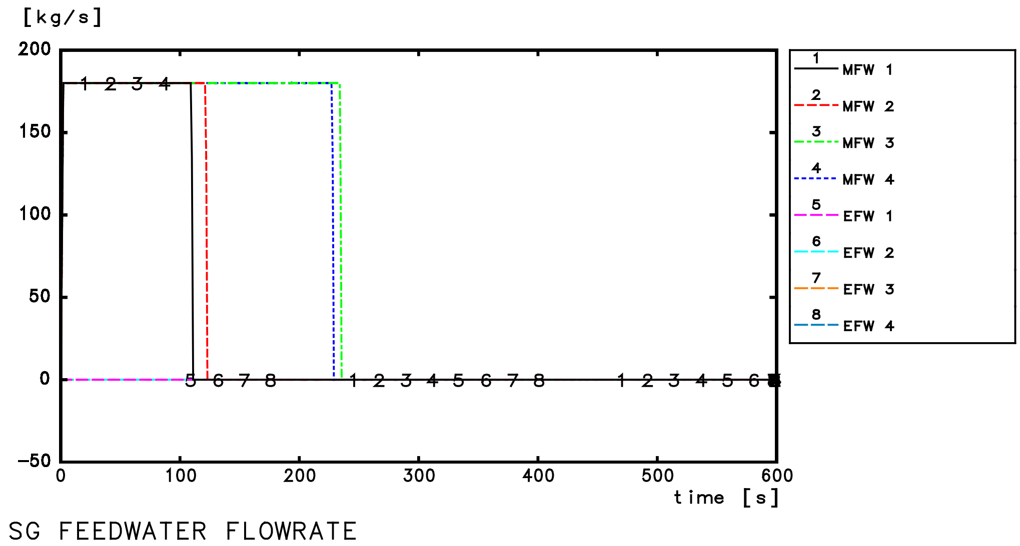
STEAM GENERATOR STEAM FLOW

APPENDIX 14C - FIGURE 32 [REF-1]

SG Liquid Mass – SG Feedwater Flow Rate

Spurious MSSV Opening (SF on VDA [MSRT]) – EPR 4250 MWth

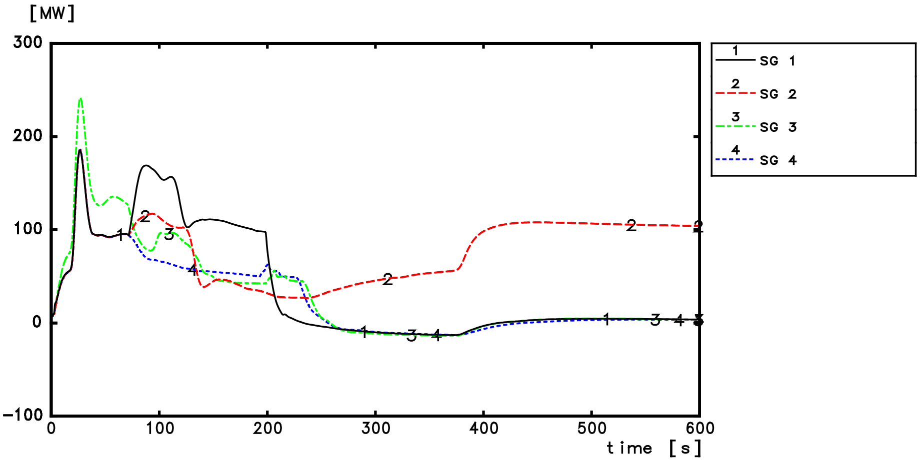




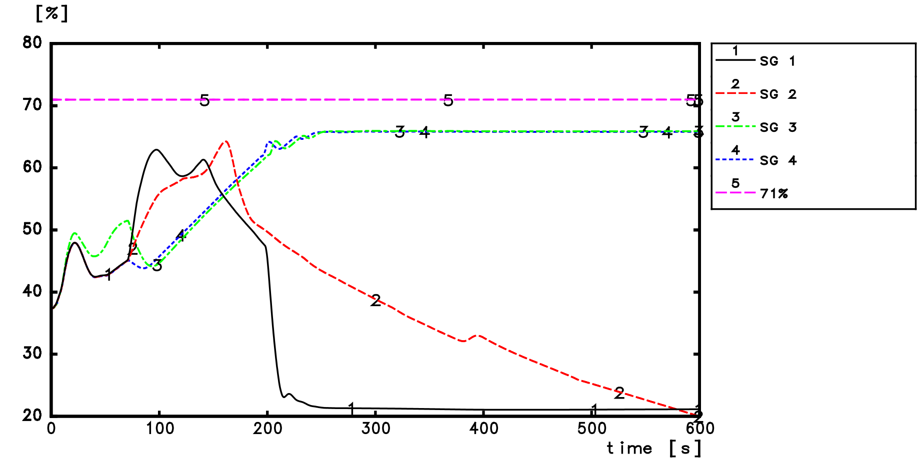
APPENDIX 14C - FIGURE 33 [REF-1]

SG Power Exchange – SG NR Level

Spurious MSSV Opening (SF on VDA [MSRT]) – EPR 4250 MWth



SG POWER EXCHANGE

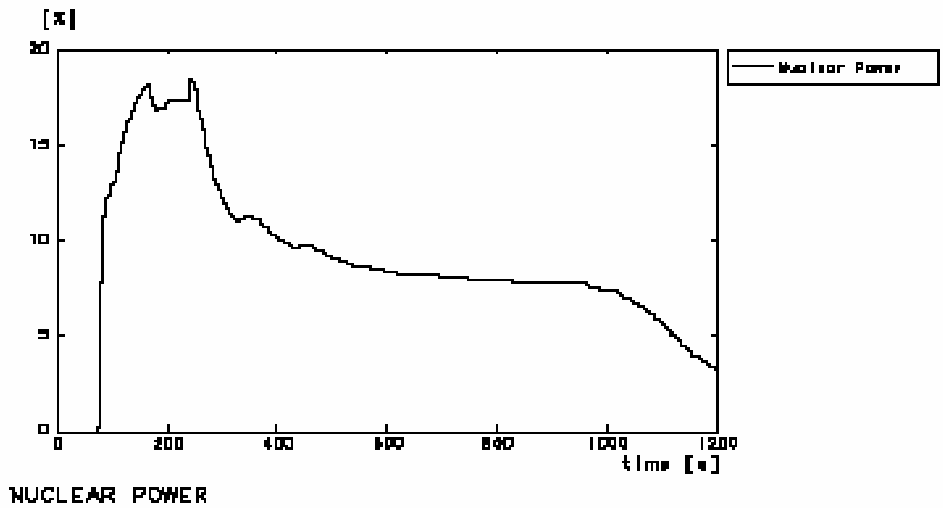


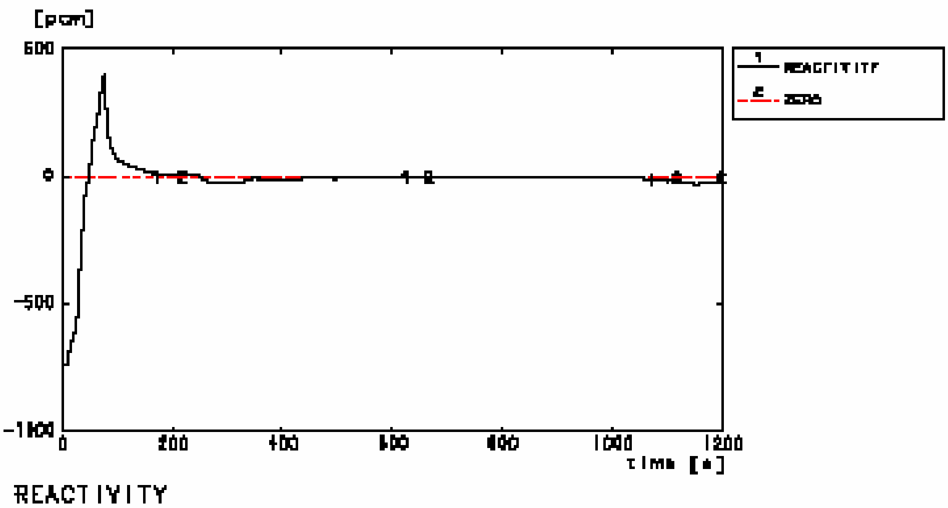
SG NR LEVEL

APPENDIX 14C - FIGURE 34 [REF-1]

Nuclear Power – Reactivity

Dilution (SF on VDA [MSRT]) – EPR 4250 MWth

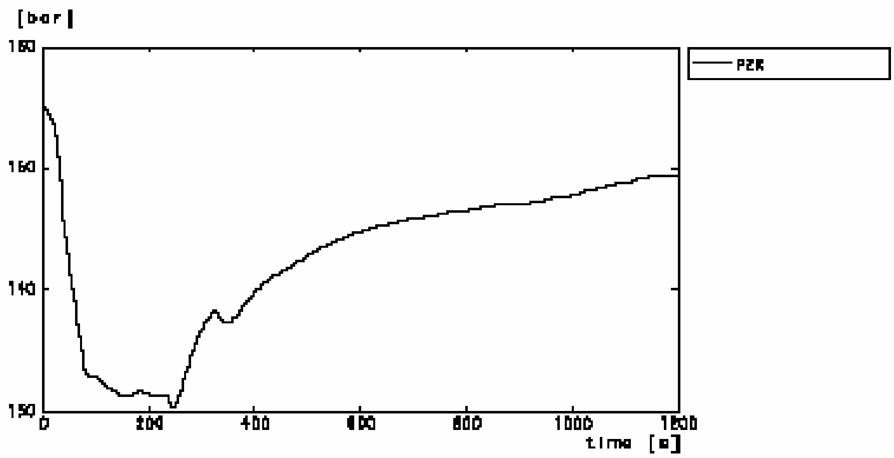




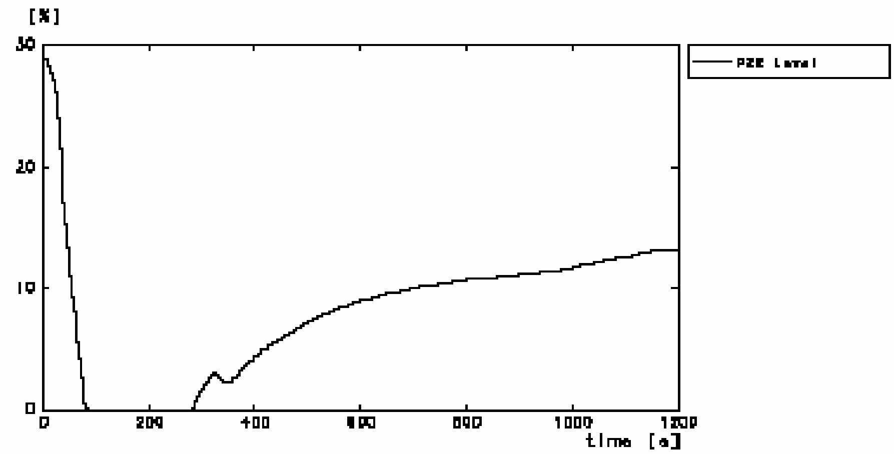
APPENDIX 14C - FIGURE 35 [REF-1]

Primary Pressure – Pressuriser Level

Dilution (SF on VDA [MSRT]) – EPR 4250 MWth



PRIMARY PRESSURE

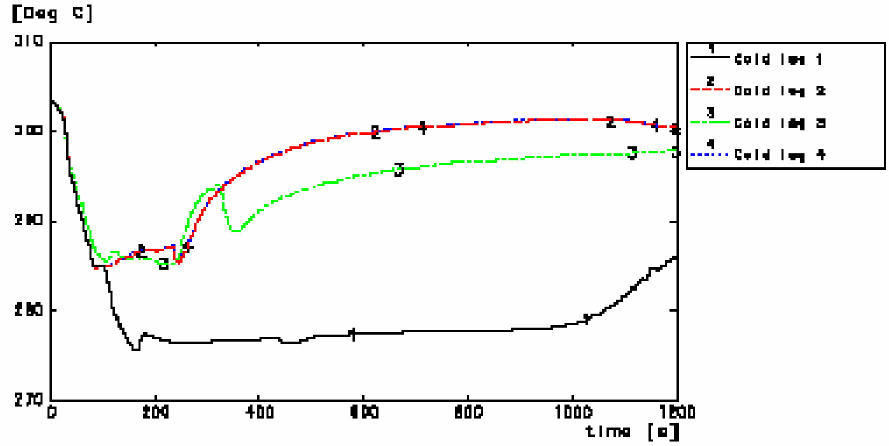


PRESSURISER LEVEL

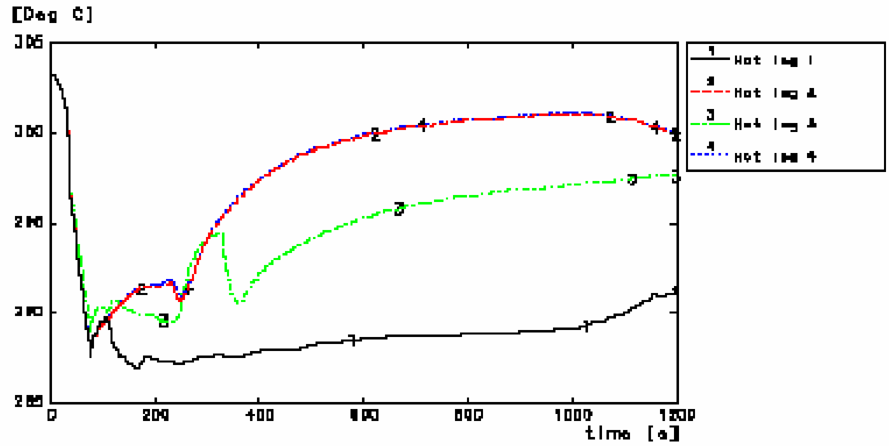
APPENDIX 14C - FIGURE 36 [REF-1]

Cold Leg Temperatures – Hot Leg Temperatures

Dilution (SF on VDA [MSRT]) – EPR 4250 MWth



COLD LEGS TEMPERATURES

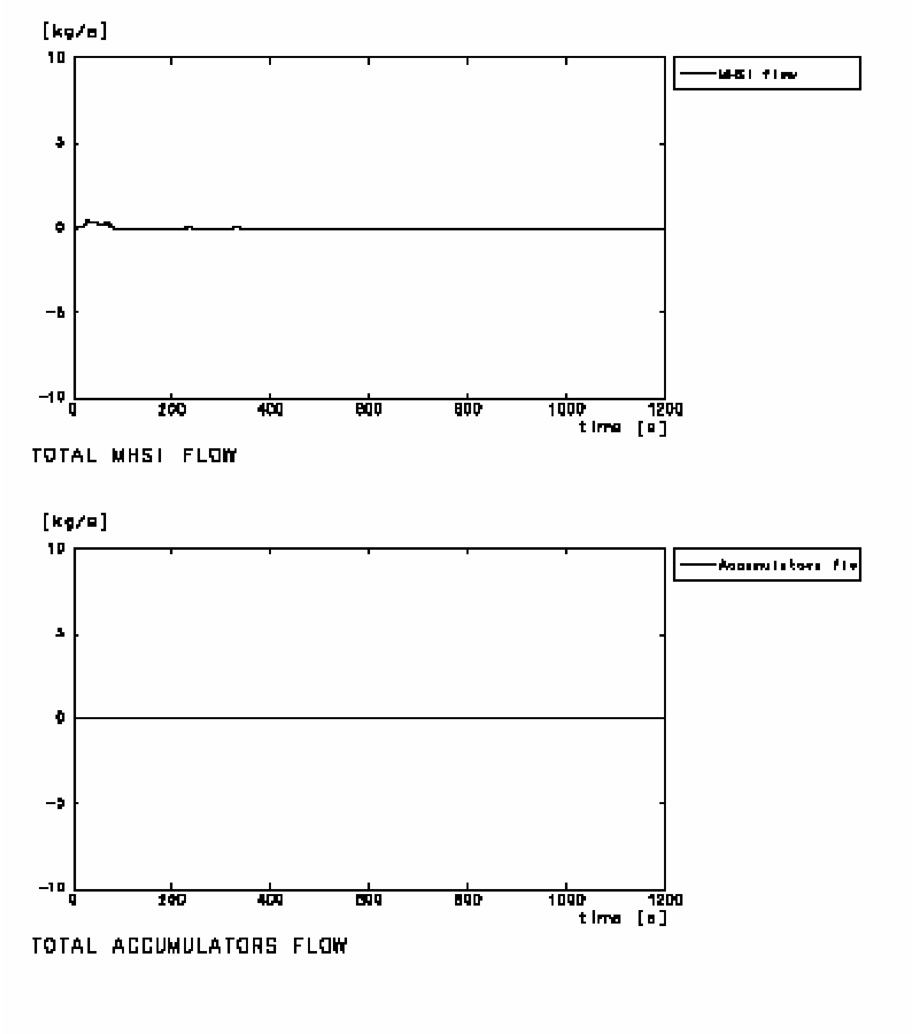


HOT LEGS TEMPERATURES

APPENDIX 14C - FIGURE 37 [REF-1]

Total MHSI Flow – Total Accumulators Flow

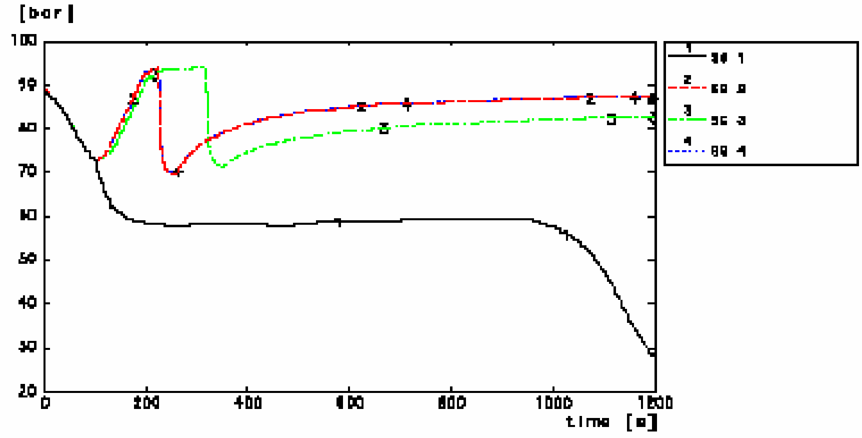
Dilution (SF on VDA [MSRT]) – EPR 4250 MWth



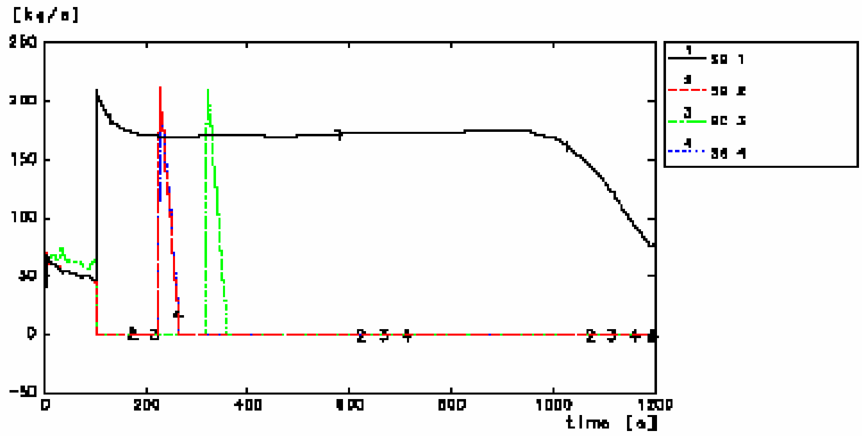
APPENDIX 14C - FIGURE 38 [REF-1]

Steam Generator Pressure – Steam Generator Steam Flow

Dilution (SF on VDA [MSRT]) – EPR 4250 MWth



STEAM GENERATOR PRESSURE

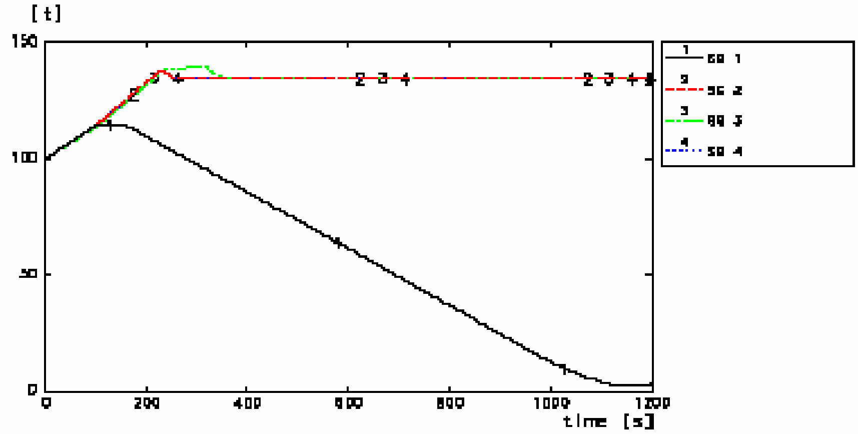


STEAM GENERATOR STEAM FLOW

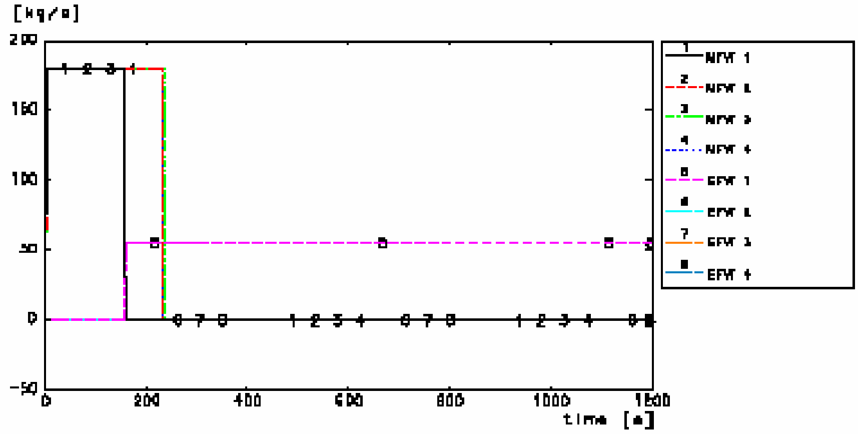
APPENDIX 14C - FIGURE 39 [REF-1]

SG Liquid Mass – SG Feedwater Flow Rate

Dilution (SF on VDA [MSRT]) – EPR 4250 MWth



SG LIQUID MASS

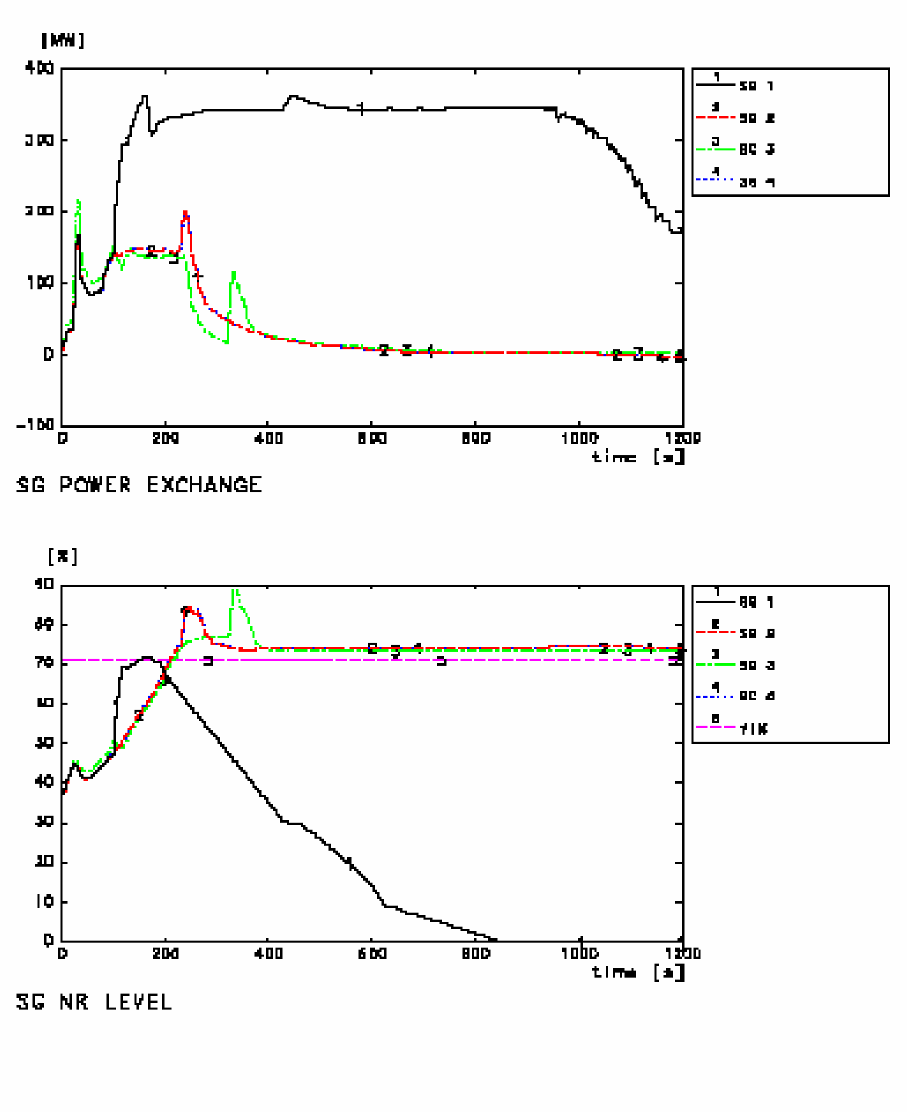


SG FEEDWATER FLOWRATE

APPENDIX 14C - FIGURE 40 [REF-1]

SG Power Exchange – SG NR Level

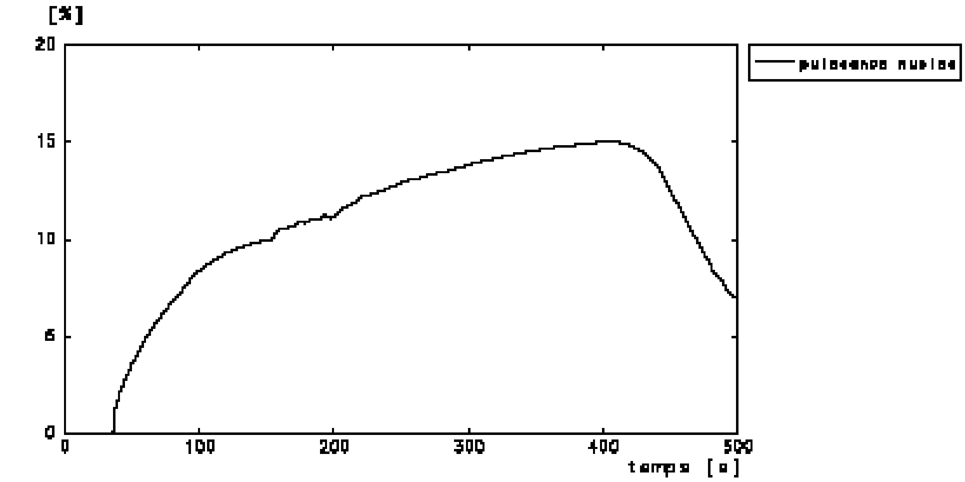
Dilution (SF on VDA [MSRT]) – EPR 4250 MWth



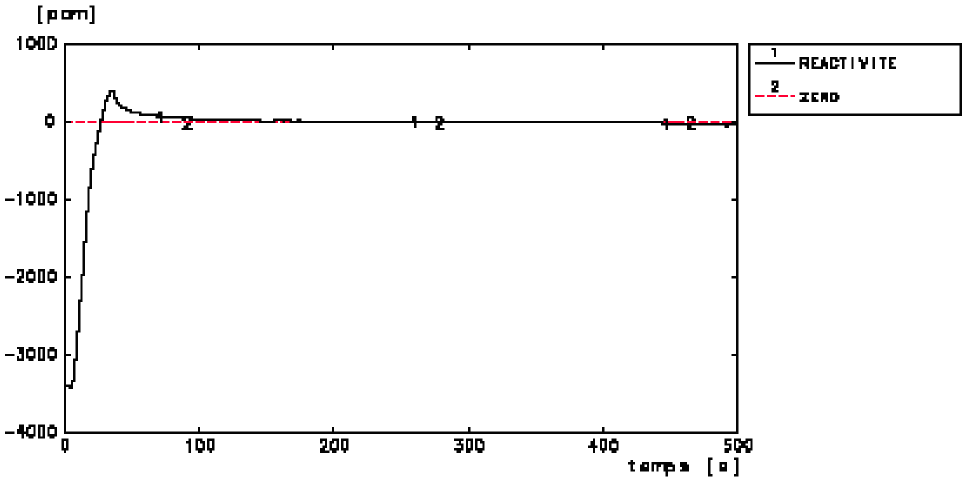
APPENDIX 14C - FIGURE 41 [REF-1]

Nuclear Power – Reactivity

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) – EPR 4500 MWth



Nuclear Power

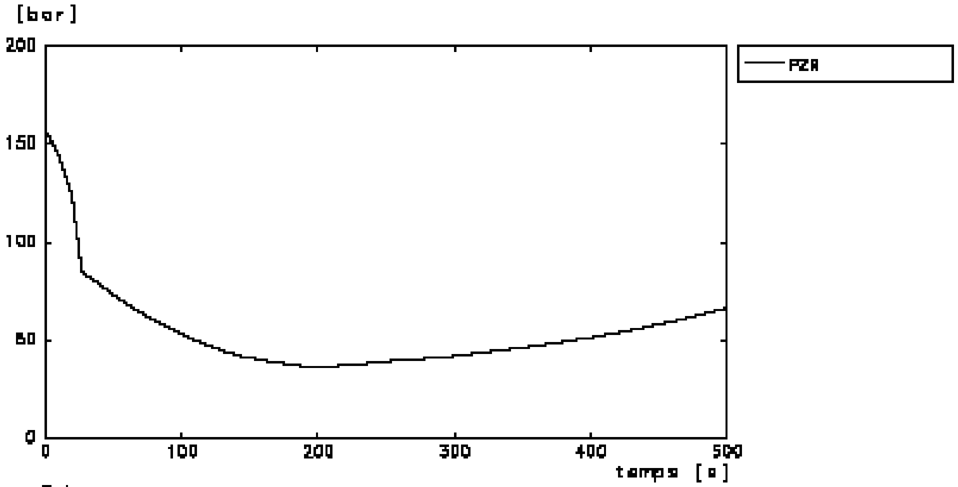


Reactivity

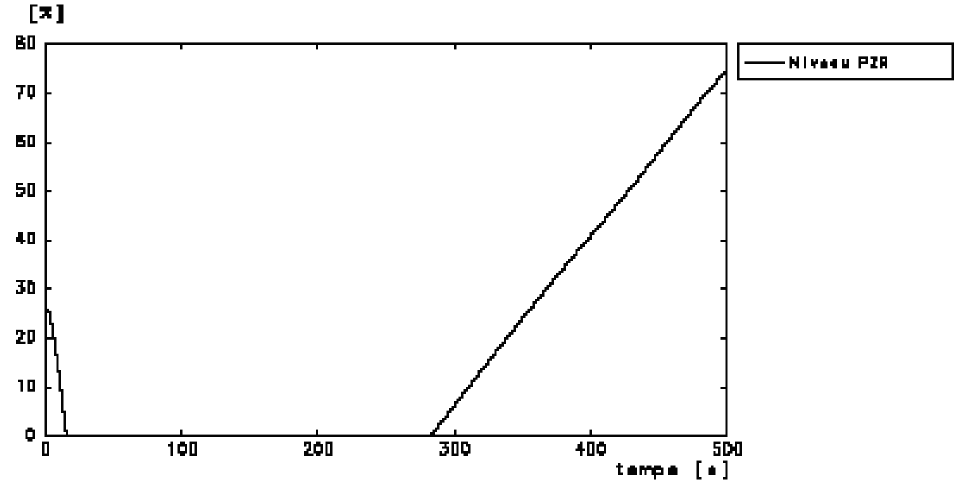
APPENDIX 14C - FIGURE 42 [REF-1]

Pressuriser Level and Pressure

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) – EPR 4500 MWth



Primary pressure

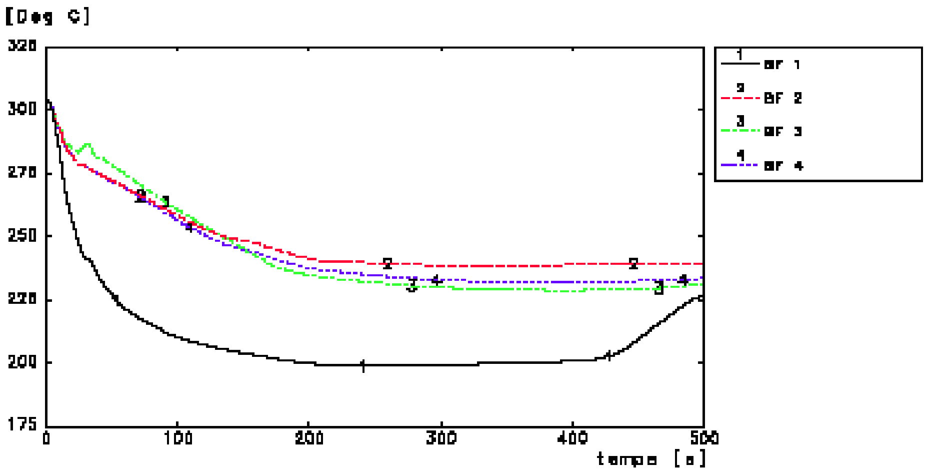


Pressurizer level

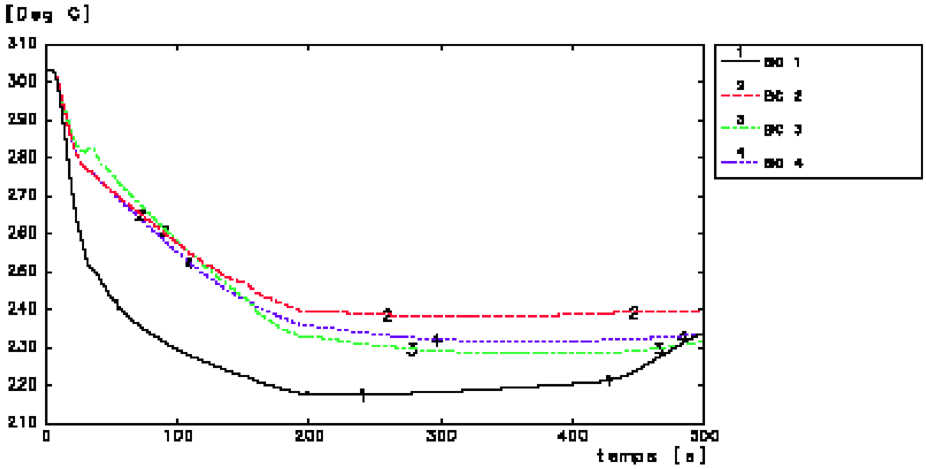
APPENDIX 14C - FIGURE 43 [REF-1]

Cold and Hot Leg Temperatures

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) – EPR 4500 MWth



Cold legs temperature

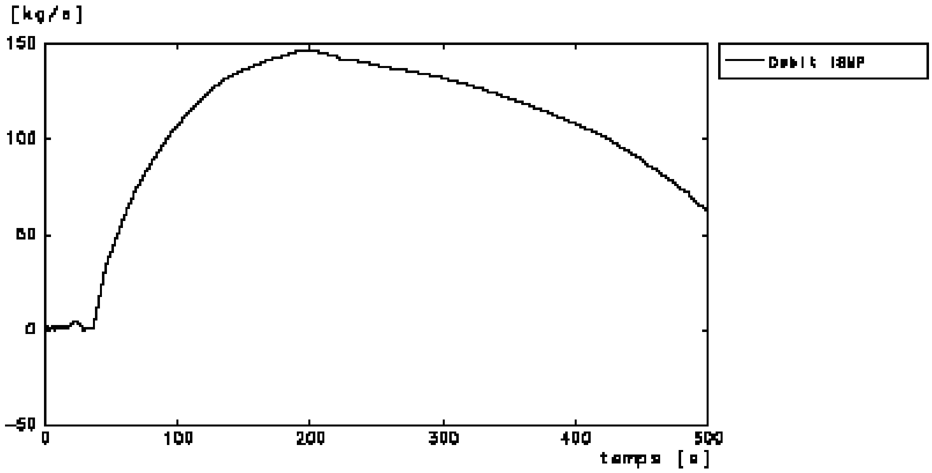


Hot legs temperature

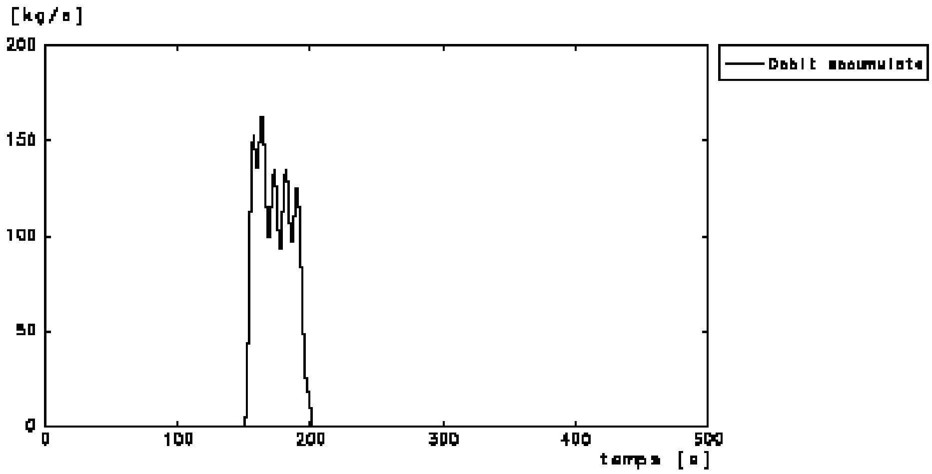
APPENDIX 14C - FIGURE 44 [REF-1]

SI and Accumulators Flow Rates

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) – EPR 4500 MWth



MHSI total flowrate

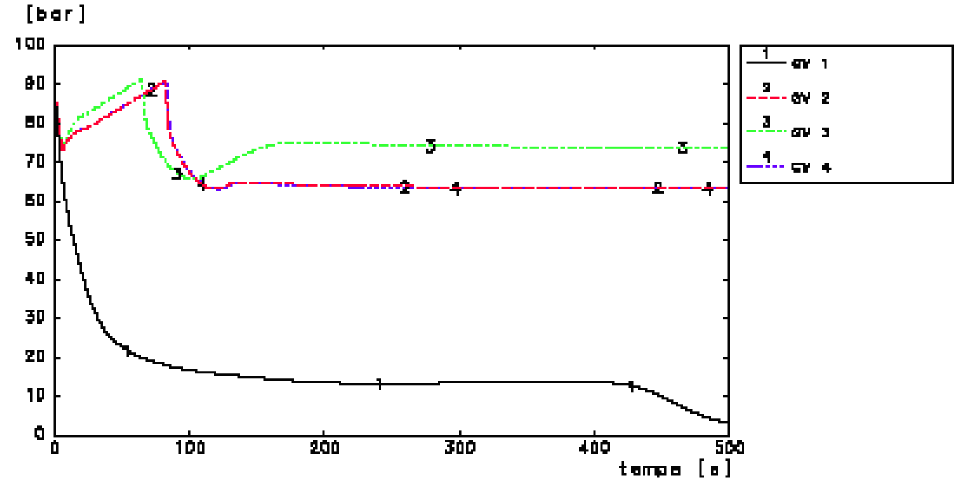


Accumulators total flowrate

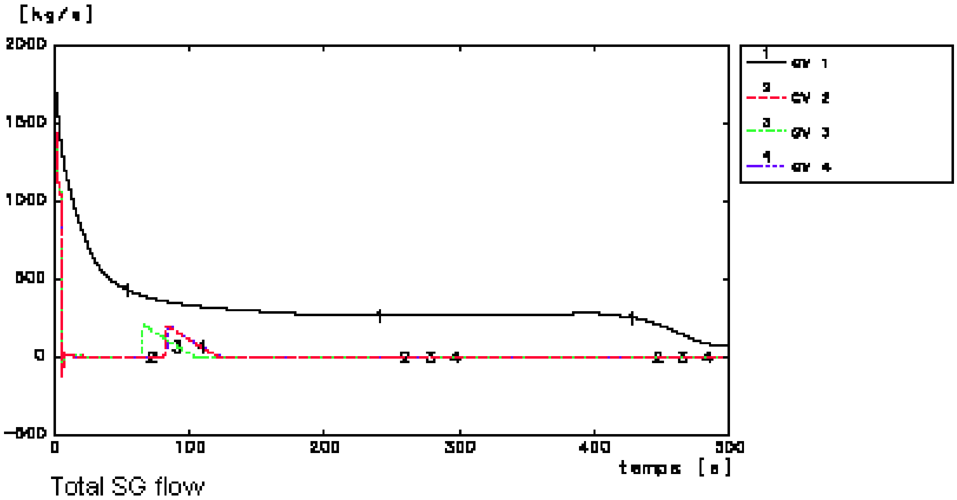
APPENDIX 14C - FIGURE 45 [REF-1]

SG Pressure and Flow Rate

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) – EPR 4500 MWth



SG pressure

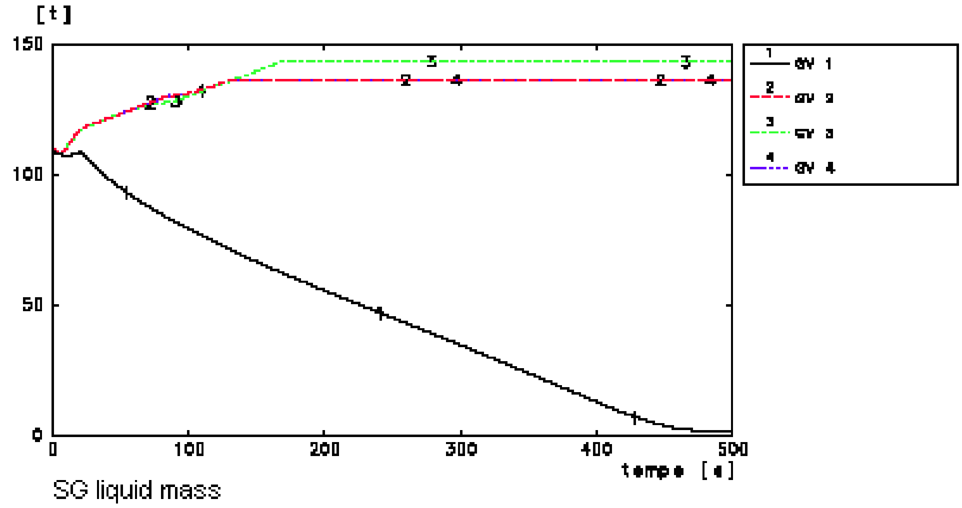


Total SG flow

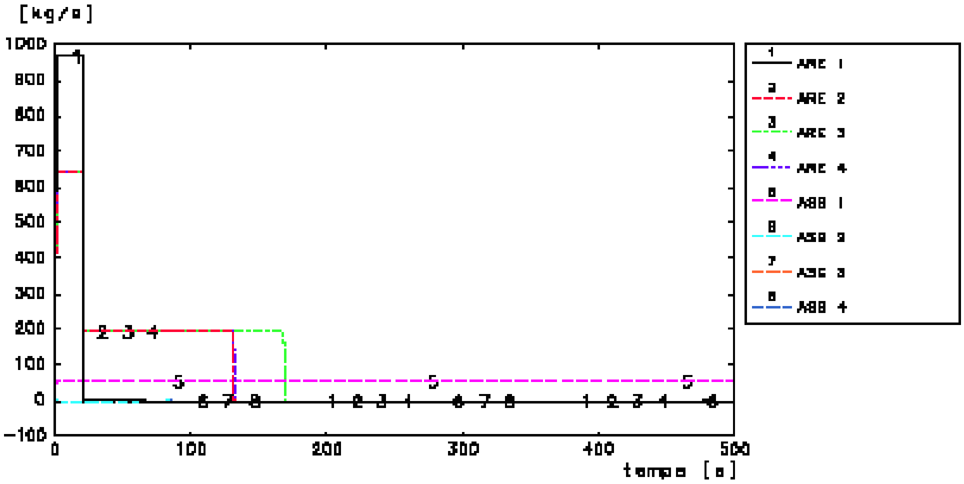
APPENDIX 14C - FIGURE 46 [REF-1]

Feedwater Flow

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) – EPR 4500 MWth



SG liquid mass

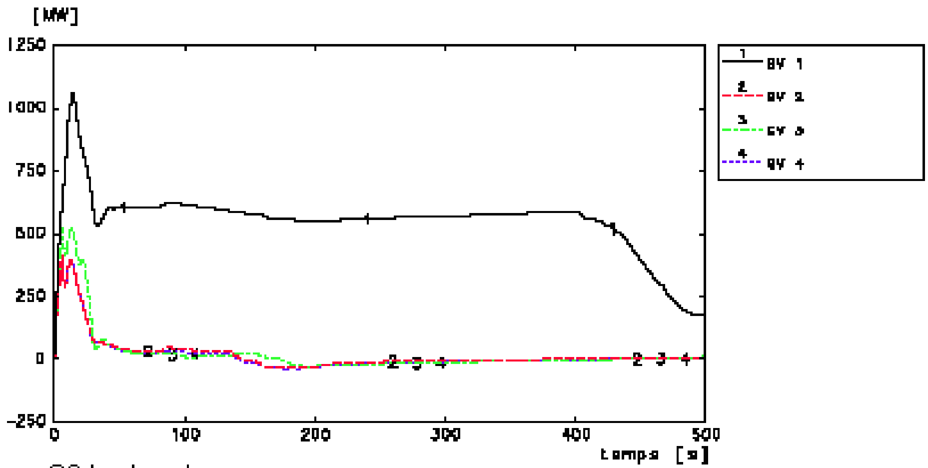


Feedwater flow

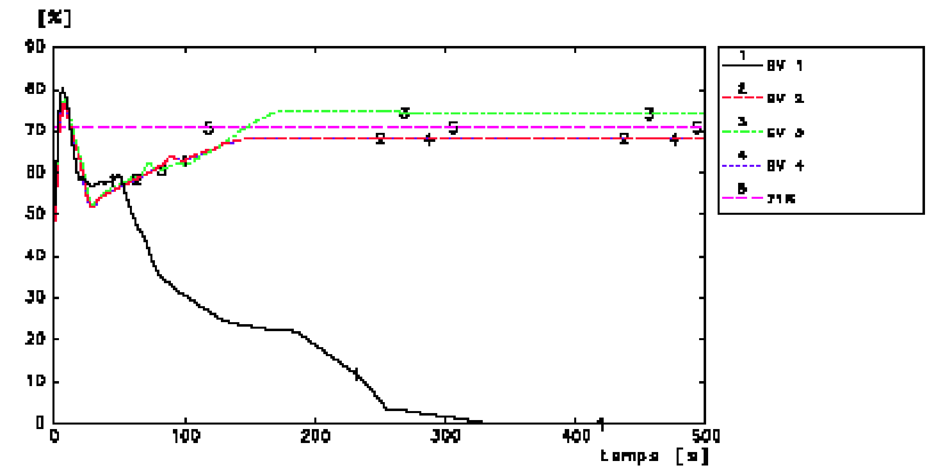
APPENDIX 14C - FIGURE 47 [REF-1]

SG Level

Double-Ended Guillotine Break Upstream of the VIV [MSIV] (SF on stuck rod) – EPR 4500 MWth



SG heat exchange



SG level (NR)

APPENDIX 14C – 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. METHODS AND ASSUMPTIONS

3.1. METHODS OF ANALYSIS

[Ref-1] S Balleydier. MANTA Code synthetic qualification assessment. NFPSD DC 85 Revision D. AREVA. September 2008. (E).

[Ref-2] SCIENCE V2 Nuclear Code Package - Qualification Report. NFPSD DC 89 Revision A. AREVA. March 2004. (E)

[Ref-3] M N Juillon. Qualification Report FLICA IIIF – Version 3. NFPSD DC 188 Revision A. AREVA. January 2006. (E).

4. DESCRIPTION OF CASES STUDIED (FROM THE INITIATING EVENT TO THE CONTROLLED STATE)

4.4. INITIAL STATE

[Ref-1] M Petiard – H. Lafitte. EPR Preliminary Safety Analysis Report, Section 15.2.4b.1- Steam system piping break (State A) Edition 2003. PSRR DC 13 Revision B. AREVA. November 2003. (E).

4.5. SPECIFIC ASSUMPTIONS

4.5.1. Neutronic data and decay heat

[Ref-1] P Dormeau. Neutronic data for transient analyses. NFEPC DC 128 Revision B. AREVA. December 2003. (E).

4.5.2. Assumptions related to non-F1 systems

4.5.2.1. Main Feedwater

[Ref-1] P Dormeau. EPR RF002 – NSSS operating parameters. EPC DC 1690 Revision C. AREVA. September 2003. (E).

[Ref-2] M Petiard – H Lafitte. EPR Preliminary Safety Analysis Report, Section 15.2.4b.1- Steam system piping break (State A) Edition 2003. PSRR DC 13 Revision B. AREVA. November 2003 (E).

4.5.3. Assumptions related to F1 systems

4.5.3.1. VIV [MSIV] (F1A)

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E).

4.5.3.2. ARE [MFW] high-load line isolation (F1A)

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E).

4.5.3.3. ARE [MFW] low-load line isolation (F1A)

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E).

4.5.3.4. MHSI (F1A)

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E).

4.5.3.5. ASG [EFWS] (F1A)

[Ref-1] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E).

4.5.4. Other assumptions

4.5.4.2. Break flow (or stuck-open valve flow) correlation:

[Ref-1] D Venturi. 79/19TE Steam generator secondary side pressure vessel details right and left SGs. NEEG-F DC 1210 Revision F. AREVA. April 2008. (E).

[Ref-2] M Petiard – H Lafitte. EPR Preliminary Safety Analysis Report, Section 15.2.4b.1- Steam system piping break (State A) Edition 2003. PSRR DC 13 Revision B. AREVA. November 2003. (E).

4.5.4.3. Other assumptions maximising core cooling:

[Ref-1] N. Goreaud. EPR – Lower Plenum Hydraulics Design of the Flow Distribution Device. EPRR DC 1651. Revision D. AREVA. December 2001. (E)

5. RESULTS

5.1. DOUBLE-ENDED GUILLOTINE BREAK UPSTREAM OF THE VIV [MSIV] (WITH SF ON STUCK ROD)

5.1.1. Short term phase

[Ref-1] M Petiard – H Lafitte. EPR Preliminary Safety Analysis Report, Section 15.2.4b.1-Steam system piping break (State A) Edition 2003. PSRR DC 13 Revision B. AREVA. November 2003. (E).

5.1.2. Long term phase

[Ref-1] M Petiard – H Lafitte. EPR Preliminary Safety Analysis Report, Section 15.2.4b.1-Steam system piping break (State A) Edition 2003. PSRR DC 13 Revision B. AREVA. November 2003. (E).

[Ref-2] R Gagner. EPR sizing at 4500MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E).

5.2. DOUBLE-ENDED GUILLOTINE BREAK DOWNSTREAM OF THE VIV [MSIV] (WITH SF OF A VIV [MSIV])

[Ref-1] M Petiard – H Lafitte. EPR Preliminary Safety Analysis Report, Section 15.2.4b.1-Steam system piping break (State A) Edition 2003. PSRR DC 13 Revision B. AREVA. November 2003. (E).

5.3. SPURIOUS MSSV OPENING (WITH SF ON VDA [MSRT])

[Ref-1] M Petiard – H Lafitte. EPR Preliminary Safety Analysis Report, Section 15.2.4b.1-Steam system piping break (State A) Edition 2003. PSRR DC 13 Revision B. AREVA. November 2003. (E).

5.4. BORON DILUTION WITH SF ON VDA [MSRT]

[Ref-1] M Petiard – H Lafitte. EPR Preliminary Safety Analysis Report, Section 15.2.4b.1-Steam system piping break (State A) Edition 2003. PSRR DC 13 Revision B. AREVA. November 2003. (E).

8. EXTENSION TO 4500 MWTH

8.1. ASSUMPTIONS FOR 4500 MWTH POWER OPERATION

8.1.2. Neutronic data and decay heat

[Ref-1] S Laurent. Neutronic data for transient analyses (update 4500 MWth). NFPSC DC 286 Revision B. AREVA. February 2006. (E).

8.2. RESULTS

8.2.3. Spurious MSSV opening (with SF on VDA [MSRT])

[Ref-1] S Laurent. Neutronic data for transient analyses (update 4500 MWth). NFPSC DC 286 Revision B. AREVA. February 2006. (E).

APPENDIX 14C - TABLES 1 TO 7

[Ref-1] M Petiard – H Lafitte. EPR Preliminary Safety Analysis Report, Section 15.2.4b.1- Steam system piping break (State A) Edition 2003. PSRR DC 13 Revision B. AREVA. November 2003. (E).

APPENDIX 14C - TABLES 8 TO 10

[Ref-1] H Lafitte. Main Steam Line Break (States A, B).
NFPSR DC 1057 Revision B. AREVA. June 2005. (E)

APPENDIX 14C - FIGURE 1

[Ref-1] Functional Validation of MANTA-SMART-FLICA Coupling.
NEPD-F DC 10157 Revision A. AREVA. October 2008. (E)

APPENDIX 14C - FIGURES 3 TO 40

[Ref-1] M Petiard – H Lafitte. EPR Preliminary Safety Analysis Report, Section 15.2.4b.1- Steam system piping break (State A) Edition 2003. PSRR DC 13 Revision B. AREVA. November 2003. (E).

APPENDIX 14C - FIGURES 41 TO 47

[Ref-1] H Lafitte. Main Steam Line Break (States A, B).
NFPSR DC 1057 Revision B. AREVA. June 2005. (E)