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Chapter Pilot: F. CERRU *Name/Initials* *Date 28-11-2012*	
Approved for EDF by: A. MARECHAL *Name/Initials* A.J.C. Marechal *Date 28-11-2012*	Approved for AREVA by: G. CRAIG *Name/Initials* G Craig *Date 28-11-2012*

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01	Update to incorporate the detailed 2A-LOCA core study at 4500 MWth.	28-09-2009
02	Removal of RESTRICTED classification	10-06-2010
03	Consolidated Step 4 PCSR update: - Minor editorial changes - Addition of four new sections: o §3. Steam generator tube rupture (1 tube) with main steam line break o §4. Steam generator tube rupture (1 tube) with the affected steam generator main steam relief train stuck open o §5. Multiple Steam Generator Tube Rupture (10 tubes in one steam generator at power) o §6. Spurious Reactor Protection System Actuation	27-03-2011
04	Consolidated PCSR update: - References listed under each numbered section or sub-section heading numbered [Ref-1], [Ref-2], [Ref-3], etc - Minor editorial changes - Update of references (English translations) - Addition of new sections: o §7, “Safety case for heterogeneous boron dilution fault” o §8, “Spent Fuel Pool drainage faults” o §9, “Analysis of faults in essential support systems”	28-11-2012

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For information address:

AREVA NP SAS
Tour AREVA
92084 Paris La Défense Cedex
France

EDF
Division Ingénierie Nucléaire
Centre National d'Equipement Nucléaire
165-173, avenue Pierre Brossolette
BP900
92542 Montrouge
France

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SUB-CHAPTER 16.4 – SPECIFIC STUDIES

1. DOUBLE ENDED BREAK OF THE MAIN COOLANT LINE (2A-LOCA)

1.1. INTRODUCTION

1.1.1. Classification

A double-ended break on the reactor coolant system line causes a discharge of primary coolant into the containment. It results in a loss of coolant inventory and thus in a sudden primary side pressure decrease. This accident leads to uncovery of the core and might lead to the rupture of a number of fuel rod claddings. It also causes containment loads by overpressure due to mass and energy release, mechanical loads on reactor coolant system components and the associated supports and structures and on reactor pressure vessel internals.

The 2A-LOCA detailed in this section is studied as a specific study which focuses on fuel damage and cladding temperature aspects. Containment loads are studied within section 1 of Sub-chapter 6.2 and Appendix 6A.

1.1.2. Identification of causes

The accident examined is either due to:

- A non isolatable double-ended break of the main coolant line on the cold leg side between the reactor coolant pump and the reactor pressure vessel, or

- A non isolatable double-ended break of the main coolant line on the hot leg side between the reactor pressure vessel and the steam generator.

The calculations are performed considering the most pessimistic breach on the cold leg.

1.1.3. Precautions limiting the accident occurrence

Austenitic stainless steels are used for the parts of the reactor coolant system in contact with primary coolant because of their high resistance to generalised corrosion in primary water at the service temperature and in conditions of cold shutdown. Precautions are taken to avoid any other problem of localised corrosion, in terms of conditioning of the primary coolant and appropriate chemical composition of the material.

Concerning the pitting which can appear in chloride and oxygenated environment, the chloride content and oxygen content of the primary coolant avoid this form of corrosion in service conditions. Concerning the stress corrosion cracking, this conditioning of the primary coolant guarantees good behaviour of the stainless steel.

1.2. DESCRIPTION OF THE CASE STUDIED

The current paragraph relates to a typical 2A-LOCA in cold leg sequence, which is divided into the four following phases.

Phase 1 – blowdown: from the occurrence of the break to the beginning of RIS [SIS] injection,

Phase 2 – end of blowdown/refilling: from the RIS [SIS] injection to the beginning of core coolant inventory recovery,

Phase 3 – reflooding: from the beginning to the completion of the core coolant inventory recovery,

Phase 4: post-reflooding: phase after core coolant inventory recovery

BLOWDOWN (typically: 0-15 seconds)

A double-ended guillotine break in the cold leg of Loop 3 is assumed to occur at time t=0. The events following the occurrence of the break are described below:

- Initially, the mass flow rates of subcooled liquid out of the cold leg break are high and the flow becomes choked. The mass outflow from the reactor vessel side of the break is considerably greater than that from the pump side, because of the high hydraulic resistance of the pump and the SG tubes.
- Because of the high flow rates from the reactor vessel side of the cold leg break, most of the flow through the pumps in the intact loops (which are coasting down) exits through the break rather than entering the core.
- As a consequence, flow stagnation occurs in the core resulting in reversal of the inlet and outlet flows. Typically, stagnation occurs during the first 4 seconds of the accident.
- The primary system depressurises quickly. As the pressure falls, the temperature of the liquid in the upper plenum and the hot legs reaches saturation, resulting in flashing.
- Nucleate boiling and flashing start in the core, which results in a reduction in both power generation and the rate of depressurisation.
- Critical heat flux is reached in the core and heat transfer changes from nucleate to film boiling.
- Following core uncovery, the cladding temperature rises rapidly because of the initial stored energy, and the first peak in clad temperature (T1) is reached on the hot rod.
- The reactor coolant pumps are tripped at 0.0 seconds but continue to rotate in coast-down mode. The speed of the intact loop reactor coolant pumps decreases. In the broken loop, the reactor coolant pump speeds up.
- As the pressure continues to fall, saturation pressure is reached in the broken cold leg; this leads to voiding and increases the local flow resistance.

- The flow at the break changes from subcooled choked flow to saturated choked flow. As a consequence, the mass flow rate at the break falls rapidly.
- As the flow rate at the break falls, the intact loops reactor coolant pumps (in coast down mode) drive the relatively cool fluid from the lower plenum into the core. The fluid contained in the lower plenum and the downcomer is essentially incompressible (liquid) until flashing starts in these two regions. Consequently, the amount of liquid which enters the core is determined by the difference between the intact loop flow into the reactor vessel and the flow out of the break.
- The fluid entering the core cools the fuel: the cladding temperature reaches a peak and starts to decrease (first rewet typically around 5 seconds). The first rewet occurs at high pressure and low quality.
- The liquid in the intact loop begins to flash, increasing the flow resistance. The performance of the intact loop reactor coolant pumps begins to degrade due to voiding in the loops. The core inlet flow consequently falls, causing a second uncovery, typically around 10 seconds.
- When the mass flow rate from the intact hot legs into the reactor vessel exceeds the water discharge from the reactor vessel to the broken loop, this water enters the core and cools the fuel rods (typically between 10 and 15 seconds).
- Heating during the second uncovery is moderate because the greater part of the energy initially stored is removed during the first filling. Furthermore, the flashing of the liquid in the upper head, lower plenum and downcomer may cause significant steam flows and/or dispersed droplet flows which cool the core.
- 15 seconds after occurrence of the break, injection begins from the intact loop accumulators. This concludes the blowdown period.

END OF BLOWDOWN/REFILLING (typically: 15-30 seconds)

By definition, this period begins with injection of water by the Safety Injection System (RIS [SIS]).

- Initially, the steam and entrained liquid from the lower plenum and the downcomer flows out through the break in the cold leg.
- The cold RIS [SIS] water is mixed with the steam in the cold leg, generating oscillations due to direct contact condensation phenomena. Water plugs form in the cold legs as the flow of steam in the loops is condensed by the high flow of subcooled water from the RIS [SIS]. Plug formation consumes a few seconds of RIS [SIS] water injection and consequently slightly delays delivery of RIS [SIS] water into the downcomer. This delay is not detrimental as the system is at a pressure such that significant RIS [SIS] bypass would occur if RIS [SIS] water reached the downcomer. The water plugs in the cold legs oscillate, causing fluctuations in the RIS [SIS] flow into the downcomer.
- RIS [SIS] liquid which may have penetrated into the lower plenum can be swept out again. This process of “partial penetration” (or “dumping”) and “sweep out” occurs randomly. Direct contact condensation of the steam on the subcooled RIS [SIS] fluid continues.

- The rate of depressurisation decreases as the difference between the reactor vessel pressure and the containment pressure decreases. Therefore, the quantity of steam generation due to flashing falls. As a consequence, the quantity of liquid dumped in the plenum increases while that swept out decreases. The water inventory in the lower plenum falls until the lower plenum begins to fill (typically at 25 seconds), a few seconds before the end of blowdown. As a consequence, the filling and the blowdown are not consecutive, but overlap. The overlapping of blowdown and refilling reduces the core reflooding time, and hence the adiabatic fuel heat-up period, by about 4 seconds.

- The primary system pressure reaches the containment pressure approximately 25 seconds after occurrence of the breach, which is the end of blowdown.

- At the end of blowdown, the lower plenum is almost filled to the height of the bottom of the core barrel.

- The core starts to slowly heat up because of the decreasing core steam flow due to the decreasing rate of depressurisation and the depletion of liquid in the upper head and the pressuriser.

- With steam flows falling, only a small quantity of RIS [SIS] liquid is lost through the break. The majority of the injected liquid contributes to filling downcomer and the lower plenum. The end of blowdown/filling phase ends after about 30 seconds, when core reflooding begins.

REFLOODING (typically: 30-250 seconds)

The core begins to be reflooded at around 30 seconds. This phase is characterised by entrainment of water into the core. Experiments indicate that water is uniformly distributed across the core, causing thermo-hydraulic differences between the main and hot channels to disappear. For this reason the hot rod is henceforth treated as the main channel. The reflooding phase can be divided into three periods: early reflooding (injection from the accumulators), nitrogen discharge from the accumulators and late reflooding (low head safety injection). The events which take place during these reflooding periods are described below:

Early reflooding - Injection from the accumulators (typically: 30-50 seconds)

- In the reflooding phase the level of water in the downcomer increases rapidly because of the high RIS [SIS] water flow discharging from the accumulators. The water level in the downcomer stabilises at the level of the cold legs because of the discharge of water from the broken cold leg. The removal of residual heat from core barrel and reactor pressure vessel wall causes boiling, but the steam generated is completely condensed by the water from the accumulators: the downcomer is completely filled before discharge from the accumulators is complete.

- Initially, core refilling is quite fast because the downcomer is kept full by the continued RIS [SIS] injection, and there is very little pressure resistance (steam binding) due to the intact loop because of the low steam flow and the absence of liquid boiling in the steam generator.

- Because of the high temperature of the fuel rods, the core heat transfer regime during reflooding covers the whole spectrum of boiling: including single phase liquid convection, nucleate boiling, transition boiling, film boiling, churn two-phase flow, dispersed droplet flow and single phase steam flow.

- The increase in the water level in the downcomer (from 30 seconds) drives water into the core. Steam generation begins at the bottom of the core when water enters the core from the lower plenum. However, in a few seconds water entrained by the boiling process appears throughout the core and begins to cool the entire core. The steam generated in the core is vented to containment via the upper plenum and the primary loops. Some of the water in the upper region of the core is entrained in the steam flow out of the core.

- When the water level in the downcomer reaches the level of the cold leg and water exits through the break, the core flooding rate decreases rapidly. However, since steam generation in the core is essentially the same as that during early reflood, the reduction in the core flooding rate results in lower rates of water accumulation in the core and water carryover from the core.

- Water carried out of the core is either de-entrained in the upper plenum or is carried over with the steam into the primary loops. In the upper plenum, the water which de-entrains either accumulates as a two-phase mixture or falls back into the core. The water carried over to the loops either de-entrains and accumulates in the SG inlet plena or is carried over to the SG tubes where it flashes.

- In the intact cold legs, the steam flow towards the downcomer is completely condensed by subcooled water from the RIS [SIS]. Because of the high flow rate of RIS [SIS] water, condensation causes water plugs to form in the cold legs which oscillate upstream and downstream of the injection point. Therefore, the flow rate of RIS [SIS] water reaching the reactor vessel fluctuates.

- The accumulator injection rate falls sharply after 45 seconds. The accumulators are practically empty at around 50 seconds.

Nitrogen discharge from the accumulators

- When the water in the accumulators is depleted, the nitrogen which pressurises the accumulators enters the RIS [SIS] lines. Nitrogen injection typically lasts between 50 and 80 seconds. The injected nitrogen drives RIS [SIS] water from the intact cold legs into the downcomer and drives water at the top of the downcomer and in the broken cold leg towards the break. The primary system (particularly the region into which the nitrogen is injected) is pressurised for a short period of time until the nitrogen is vented from the system.

- The system pressure increases further by suppression of steam condensation. As nitrogen mixes with the steam and displaces it, the rate of condensation reduces compared with a situation involving pure steam in contact with subcooled water. The accumulation of uncondensed steam contributes to the temporary pressurisation of the downcomer and the cold legs of the primary system.

- Before the start of nitrogen discharge, the pressure above the core exceeds the pressure in the downcomer, due to the pressure drop of steam flowing from the upper plenum around the intact loops. This pressure difference maintains the water level in the core below that in the downcomer. Pressurisation by the nitrogen in the downcomer disrupts the pressure distribution and drives a small portion of the water in the downcomer into the lower plenum, displacing the water in the lower plenum into the core.

- The water in the lower plenum becomes subcooled, partly because of the pressure increase. As the water surges into the core, heat is absorbed until, after a short period of time during which the water is heated to saturation, additional steam is produced. The increase in steam production in the core increases the pressure above the core. Coupled with a fall in the nitrogen discharge rate, the pressure increase terminates the rise in core water level, and forces water out of the core back into the lower plenum.

Late reflooding (low head safety injection)

- This phase begins at about 50 seconds, when the accumulators are more or less empty and generally ends at about 250 seconds, when the entire core is quenched. During this phase the medium head safety injection (MHSI) and low head safety injection (LHSI) pumps, which have been started at around 25 seconds, inject into the three intact loops.

- As shown above, carryover of water from the core falls prior to termination of accumulator injection, when the water level in the downcomer is high. In later stages of reflooding, however, carryover of water from the core increases as the quench front reaches the upper regions of the core.

- The liquid from the two-phase pool in the upper plenum may be entrained by the steam and carried over to the hot legs. While passing through the upper plenum it may de-entrain on the structures in the upper plenum.

- As the bottom quench moves through the core, more liquid is carried over to the pool in the upper plenum. As a consequence, the level of the two-phase mixture can reach the hot leg.

- The liquid carried over by the steam through the hot legs may reach the SG where it will flash causing an increase in pressure in the SG and the upper plenum. This reduces the reflooding rate giving rise to the “steam binding” effect.

- Steam mass flow rates fall with decreasing quench velocity. This reduces the quantity of liquid entrained and carried over to the steam generator. As a result, the pressure in the steam generator and the upper plenum falls, causing an increase in the reflooding rates. This process continues with decreasing effects until the entire core is reflooded.

- In the intact cold legs, the steam is condensed by subcooled RIS [SIS] water. However, only a part of the steam flow is condensed. The remaining steam flow in the downcomer reduces the level of water in the downcomer by entraining water out of the break. The collapsed level is kept constant at the level of the upper tie plate of the core until reflooding is complete. The corresponding mixture level rises to the level of the cold leg.

- A two-phase mixture of steam and entrained water flows out the break. The pressure drop associated with this flow pressurises the downcomer relative to containment and increases the primary system pressure. Tests at Cylindrical Core Test Facility (CCTF) and Slab Core Test Facility (SCTF) show that an increase in the pressure of the system improves the cooling of the core.

- Average rods located in the main channel are quenched at 250 seconds (i.e. typically 220 seconds after the start of reflooding); the hot rod included in the same main channel is quenched at the same time.

- The peak cladding temperature reached during reflooding (T2 in early reflooding, T3 during reflooding) are below the peak temperature (T1) reached in the blowdown phase.

POST REFLOODING (typically > 250 seconds)

This phase is characterised by long term pressure oscillations in the primary system due to condensation and vaporisation.

- Initially, the production of steam in the core falls because the energy stored in the fuel rods is totally removed after reflooding. The only heat source remaining is decay heat which decreases slowly.

- The lower steam flow rate is not sufficient to carry water into the steam generator tubes. No flashing takes place in the steam generator. Hence the steam flow in the intact cold legs is much less greater than during the reflooding phase.

- All the steam flow in the cold legs is removed by condensation on the RIS [SIS] injection. Therefore, the level of the mixture in the downcomer falls for a period of time (typically 60 seconds), and no water flows out the break.

- As mentioned above, water entrainment out of the core increases after reflooding because of the increasing mixture level in the upper plenum. Firstly, water accumulates in the upper plenum until the mixture level rises up to the elevation of the hot legs. Then, water flows into the hot legs and begins to fill the hot leg (typically at 275 seconds) and SG inlet plenum. Finally, the level of the mixture reaches the SG tubes and the water is carried into the SGs (typically at 330 seconds).

- The “steam binding” effect forces water out of core into the downcomer and towards the break. The collapsed water level in core falls (typically at 350 seconds).

- Saturated water from the upper plenum falls back into the core (typically from 330 to 370 seconds) resulting in an increased production of steam compared with the previous subcooled water condition. The carry over of water into the steam generator tubes continues during this period. After around 370 seconds the upper plenum is empty, the steam flow rate falls and the cycle starts again. This cycle is repeated with decreasing amplitude but with an almost constant period of around 75 seconds.

The behaviour of a PWR during reflooding was studied in the CCTF test C2-12. The results show that the CATHARE computer code prediction of long-term oscillations is realistic. Those core level oscillations do not cause long term core heat-ups.

1.3. SAFETY CRITERIA

The acceptance criteria of this study for the fuel and cladding are as follows:

- The number of failed rods is studied. This parameter is relevant for the qualification of equipment.

- Clad temperature must not exceed 1200°C.

Although this document gives maximum cladding temperature, the focus, and methodology developed for this study, is set on the number of failed rods.

1.4. METHOD OF ANALYSIS

The method applied assesses the number of failed fuel rods. Together with this assessment, the clad temperature is considered.

1.4.1. Principle of the selected method

1.4.1.1. General principle of the method

The general principle of the method consists of establishing three-dimensional rupture surfaces which depend on:

- the hot spot factor of the fuel rods (FQ),
- the enthalpy rise factor of the fuel rods (FΔH),
- the burn-up of the fuel rods (BU).

A rupture surface is defined as a surface that separates the conditions under which the rods are intact (under the surface) from the conditions under which the fuel rods are damaged[1] (above the surface). This surface is established by:

- system calculations to simulate the thermo-hydraulic conditions that rods with the potential to rupture experience during the transient (hot residence assemblies, see section 1.4.1.2.1),
- thermo-mechanical calculations to assess whether or not a rod is damaged in this hot residence assembly as a function of its FQ, FΔH and BU, and thus establish the rupture surface (section 1.4.1.2.2).

Finally, the rupture surfaces are compared with the properties of the rods that belong to the various core maps at different times in the cycle and for different operating modes (section 1.4.1.2.3). This makes it possible to deduce the number of failed rods.

1.4.1.2. Overview of the analysis procedure

Section 16.4.1 - Figure 1 shows an overview of the analysis procedure.

1.4.1.2.1. Determination of hot residence assemblies containing potentially damaged fuel rods

For each fuel management scenario under consideration, several core maps are generated (see section 1.4.2.5.3). These scan the range of possible configurations (control rod positions, core cycle, etc.) and are dependent on the analysed axial power shapes (location of the peak power, axial offset).

For a given fuel management scenario, the hot residence assemblies are determined for each axial power shape – top-core (TC) or mid-core (MC) – under consideration.

Analysis of these core maps shows that the various fuel rods in the core are surrounded by relatively heterogeneous thermo-hydraulic environments. The thermo-hydraulic environment depends specifically on the geographical position in the core and, as a result, on the burn-up.

[1] The fuel rods are also intact on the rupture area itself.

In order to take this heterogeneity into account, several hot residence assemblies are distinguished from the various core configurations analysed, with specific reference to their burn-up levels.

A thermo-hydraulic system calculation is performed for each of these hot residence assemblies.

Different rupture surfaces are then established based on these boundary conditions in accordance with the principles described in the paragraph below.

1.4.1.2.2. Building of rupture surfaces

The purpose of the method is to determine a rupture surface which defines a surface (in terms of FQ, FΔH and BU) beyond which fuel rod clad rupture is observed.

- The different rod burn-up values in a hot residence assembly are considered. A rupture curve is established for each of these by performing thermo-mechanical calculations,
- The rupture curve is established by creating a self-adjusting mesh for the area FQ x FΔH. The flow chart in Section 16.4.1 - Figure 2 illustrates the process of establishing the rupture curve for a given rod burn-up. For each point in the mesh, a thermo-mechanical calculation is performed in order to assess whether the fuel rod is damaged,
- Rupture of a rod in the FQ x FΔH region is investigated for specific burn-up points. These are as follows:
  - The Beginning Of Life (BOL) point which corresponds to the point of maximal densification,
  - Points End Of Cycle (EOC) 1, EOC2, EOC3 and, depending on the fuel management option in question, EOC4, which correspond to times in the cycle when there is maximum deterioration in thermal conductivity,
- The connection (by linear interpolation) between these rupture curves defines the rupture surface that will then be positioned on the core map, which provides the position of every rod of the core in this space (BU, FQ, FΔH), using the assumptions used to generate this core map. The three-dimensional rupture surface thus created is shown in Section 16.4.1 - Figure 3.

<u>Note:</u> in the remainder of this document, the illustrated rupture curves correspond to the limits of variation of the analysed parameters (FQ, FΔH, BU), since the hot rods do not fail within these limits.

1.4.1.2.3. Determination of the number of failed rods

For each anticipated core configuration (axial power shape, point in the core cycle), the properties of each of the rods in terms of FQ, FΔH and BU are compared with the appropriate rupture surface. This is chosen as a function of:

- the hot residence assembly (the one which best represents the three values FQ x FΔH x BU for the assembly in which the fuel rod is actually located),
- the axial power shape under consideration,

- the type of fuel under analysis.

The principle for deducing the number of failed rods is illustrated in Section 16.4.1 - Figure 4.

1.4.2. Key assumptions

1.4.2.1. Computer code and modelling

1.4.2.1.1. Computer code

Nuclear data, i.e. the core maps providing the position of each rod of the core in the volume BU x FQ x FΔH for each selected core configuration, are calculated using the SCIENCE V2 [Ref-1] code.

Initial data for the fuel rods are calculated using the COPERNIC V2.4 [Ref-2] code. This initial data are as follows for a given rod burn-up:

- internal rod pressure, basic parameter for the analysis,
- the stored energy or mean temperature of the pellet as a function of local linear power density,
- the initial oxide thickness.

The thermo-hydraulic computer code used is CATHARE 2 V2.5_1 [Ref-3]. This code incorporates the COPERNIC V2.4 models for pellet-clad gap conductance, pellet conductivity and radial power distribution in the pellet [Ref-2] in such a way as to ensure that these two codes remain consistent with regard to the initial stored energy and pressure in the pellet/clad gap.

The CATHACOMB fuel module of the CATHARE 2 V2.5_1 [Ref-3] code is used to perform thermo-mechanical calculations. This module also uses the same models as CATHARE 2 V2.5_1 regarding thermo-mechanics (and consequently, COPERNIC V2.4 models).

Using thermo-hydraulic limit conditions calculated by the CATHARE 2 V2.5_1 system calculation, this module makes it possible to perform stand-alone thermo-mechanical calculations for different fuel rods. Section 16.4.1 - Figure 5 shows this separation principle.

1.4.2.1.2. System modelling

The EPR vessel is modelled using the CATHARE 2 V2.5_1 code and its 3D module by:

- 8 azimuths,
- 5 rings,
- 25 axial mesh sections.

The fuel core is modelled using 3 rings and 12 axial mesh sections. The central ring represents the hot assembly, whilst the 2 outer rings model the mean core (Section 16.4.1 - Figure 6).

Section 16.4.1 - Figure 7 presents the 3D vessel as modelled for the purposes of this analysis.

1.4.2.1.3. Modelling stand-alone fuel rods

Stand-alone fuel rods are meshed more finely than the rods modelled in the system calculation due to the mesh-refinement function in CATHARE 2 V2.5_1: the meshes for the stand-alone fuel rod used for thermo-mechanical calculations are four times smaller than the meshes for the system fuel rods.

1.4.2.2. Physical models

The standard CATHARE 2 V2.5_1 code is used without pessimism of the physical models.

The following models, related specifically to fuel rod rupture, are used:

- Rupture deformation models:

The Edgar model (limited by NUREG criteria) for M5 is used to calculate deformation and rupture of fuel rods modelled using CATHARE 2 V2.5_1 [Ref-1].

Clad rupture is detected if the actual stress reaches the instability threshold or if the mean deformation exceeds the limit corresponding to the total NUREG630 deformation curves adapted for M5.

- Thermo-physical laws for cladding:

The cladding for the fuel rods examined in this analysis is made from M5 [Ref-2] (see Sub-chapter 4.2).

- Ballooning model:

If a system fuel rod ruptures, the ballooning model [Ref-1] is activated.

1.4.2.3. Initial state and boundary conditions

The assumptions for the thermo-hydraulic system calculations and the thermo-mechanical calculations for hot rods are selected as best-estimate.

Assumptions regarding the initial state of the nuclear steam supply system are given in Section 16.4.1 - Table 1.

The input data and boundary conditions for thermo-hydraulic calculations are given in Section 16.4.1 - Table 2.

The CATHARE GB code is used along with the CONPATE4 code for calculating the containment backpressure. Assumptions with regard to calculating this containment backpressure are given in Section 16.4.1 - Table 3. The resulting containment backpressure laws which are used as boundary conditions for the CATHARE 2 V2.5_1 calculations are given in Section 16.4.1 - Figure 8 and Section 16.4.1 - Figure 9.

The break studied is a double-ended break located on the cold leg of loop 4. It leads to a higher break mass flow rate than a break on the hot leg.

1.4.2.4. Protection and mitigation actions

For these transients, the following F1A functions are credited:

- Reactor trip
- Accumulator injection
- Safety injection

1.4.2.4.1. F1A functions

Reactor trip

The typical sequence of events is the following (the specific assumptions concerning delays are defined in Sub-chapter 14.1 and summarised in Section 16.4.1 - Table 2):

- Reactor trip after pressuriser pressure < MIN2p
- Beginning of rod insertion
- Main feedwater isolation (main line)
- Turbine trip

Loss Of Offsite Power is not considered.

Accumulators injection

The 4 accumulators injection is actuated on a set pressure.

The accumulators head loss coefficients and the expansion coefficients are not pessimised. It is considered that one accumulator flow is lost due to the breach.

Safety injection

The following actions are taken into account:

- Medium head safety injection (MHSI)
- Low head safety injection (LHSI)

The minimal characteristic curves for safety injection used for this analysis are defined in Sub-chapter 14.1. It is considered that one safety injection train is lost due to the breach.

1.4.2.4.2. Single failure and preventive maintenance

Neither single failure nor preventive maintenance is considered in the transients studied.

1.4.2.5. Fuel management and configurations examined

1.4.2.5.1. Fuel management

The three bounding fuel management scenarios for the PCSR are considered when assessing the number of failed rods:

- IN-OUT UO_2 4500 MWth 18 months fuel management,
- IN-OUT UO_2 4500 MWth 22 months fuel management,

- IN-OUT 30% MOX 4500 MWth 18 months fuel management.

1.4.2.5.2. Axial power shapes

Two axial power shapes are analysed for the purposes of this study. These are described in paragraphs 1.4.2.5.2.1 and 1.4.2.5.2.2.

1.4.2.5.2.1. Peak power located at the top of the core (TC)

The peak power locations and axial offset values for mean cores, hot assemblies and hot rods are selected so as to be conservative within their pre-accident best estimate range of values.

The peak power is located at 3.54 m elevation and axial offset values are as follows:

- 10% for the mean core,
- 13% for the hot assembly,
- 13% for the hot rod.

1.4.2.5.2.2. Peak power located at mid-core (MC)

The peak power is located at 2.10 m elevation and the axial offset values for mean cores, hot assemblies and hot rods are equal to zero.

1.4.2.5.3. Configurations

The core maps used to cover all possible configurations for each of the given fuel management options are generated for an equilibrium cycle and for three burn-up levels (BLX, mid-cycle and end-of-cycle).

Five maps are generated at 100% power for a given fuel management option, axial power shape (see section 1.4.2.5.2) and burn-up, for realistic bounding control conditions corresponding to the following configurations:

- All rods out,
- With control banks at limit insertion for four different configurations. These four configurations are described in Section 16.4.1 - Figure 10 to Section 16.4.1 - Figure 13 and limit insertion values for 100% power are given in Section 16.4.1 - Table 4.

Fifteen core maps are thus generated for each fuel management option and for a given axial power shape.

1.5. RESULTS

This paragraph gives the results of the different fuel management options studied. Results are presented as follows:

- Determination of the fuel data and hot assemblies,
- Thermo-hydraulic results including:
  - o cladding temperature results,

  - o links to the sequences of events,

- Thermo-mechanical results including:
  - o determination of the number of failed rods.

Maximum cladding temperatures are given as an indication. The methodology used is not aimed at obtaining conservative results for this parameter.

1.5.1. Assessment of the IN-OUT UO_2 18 months fuel management – peak power located at the top of the core

1.5.1.1. Determination of the fuel data and hot assemblies

1.5.1.1.1. Fuel data

The fuel data used to estimate the number of failed rods for the IN-OUT UO_2 18 months fuel management scenario were generated using the COPERNIC V2.4 [Ref-1] code.

Section 16.4.1 - Figure A 1 shows the power history for UO_2 rods in the IN-OUT UO_2 18 months fuel management.

The five calculation points to establish the rupture surfaces (see section 1.4.1.2.2) are described in Section 16.4.1 - Table A 1.

1.5.1.1.2. Determination of hot assemblies

Section 16.4.1 - Figure A 2 shows the enthalpy rise factor for the various fuel assemblies in the core for the fifteen possible configurations as a function of their mean burn-up.

Section 16.4.1 - Figure A 3 shows the hot spot factor of the various fuel assemblies in the core for all the same configurations.

Two assemblies were selected as the residence assemblies for the stand-alone fuel rods:

- *Zone 1:* mean burn-up of the assembly less than 30,000 MWd/tU,

- *Zone 2:* mean burn-up of the assembly greater than 30,000 MWd/tU.

The power of each zone is modelled by a rod with an enthalpy rise factor that bounds the enthalpy rise factor of all the assemblies in this zone.

Similarly, the hot spot factor for each zone is modelled by a rod with a hot spot factor that bounds all the assemblies in this zone.

Finally, the burn-up in each zone is selected as the calculation point that is the closest to the mean burn-up for the zone in question. The mean burn-up is determined as the mean burn-up value for the rods in the zone under consideration.

The properties of the two hot residence assemblies (HA1 for zone 1 and HA2 for zone 2) are given in Section 16.4.1 - Table A 2. The mean core is then imposed such as to obtain an enthalpy rise factor that is equal to the value of a mean core rod ($F\Delta H=1$) and a hot spot factor equal to the ratio between FQ and $F\Delta H$ for the hot assembly.

1.5.1.2. Thermo-hydraulic results

1.5.1.2.1. Hot residence assembly 1

The main illustrations corresponding to the system calculation are shown on Section 16.4.1 - Figure A 4 to Section 16.4.1 - Figure A 16.

The main events of the transient are summarised in Section 16.4.1 - Table A 3.

The peak clad temperatures reached in the central hot assembly (the properties of which are reiterated in Section 16.4.1 - Table A 2) are given in Section 16.4.1 - Table A 4.

The maximum hot assembly peak clad temperature is obtained for the T1 peak, around 670°C.

1.5.1.2.2. Hot residence assembly 2

The main illustrations corresponding to the system calculation are shown on Section 16.4.1 - Figure A 17 to Section 16.4.1 - Figure A 29.

The main events of the transient are summarised in Section 16.4.1 - Table A 5.

The peak clad temperatures reached in the central hot assembly (the properties of which are reiterated in Section 16.4.1 - Table A 2) are given in Section 16.4.1 - Table A 6.

The maximum hot assembly peak clad temperature is obtained for the T1 peak at 609°C.

1.5.1.3. Thermo-mechanical results

1.5.1.3.1. Indicator rods

Before establishing rupture surfaces, thermo-mechanical calculations are performed for each fuel rod burn-up point (BOL, EOC1, EOC2, EOC3 and EOC4), for each hot residence assembly and in each azimuth.

These indicator rods are those that have the maximum linear power density and thus the highest hot spot factor of all the rods in all core configurations.

The bounding properties of these hot rods are described in Section 16.4.1 - Table A 7.

They are used to determine the most onerous azimuth in terms of thermo-mechanical calculations. The results obtained for these bounding rods in the sections below were obtained using the most onerous azimuth.

1.5.1.3.1.1. Indicator rods inserted in hot residence assembly 1

Section 16.4.1 - Table A 8 shows the results obtained for each indicator rod inserted in the thermo-hydraulic environment of hot assembly 1 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure A 30 to Section 16.4.1 - Figure A 34 present the maximum clad temperature transient, for each of the indicator rods.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 793°C.

1.5.1.3.1.2. Indicator rods inserted in hot residence assembly 2

Section 16.4.1 - Table A 9 shows the results obtained for each indicator rod inserted in the thermo-hydraulic environment of hot assembly 2 in terms of:

- peak clad temperature ,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure A 35 to Section 16.4.1 - Figure A 39 present the maximum clad temperature transient, for each of the indicator rods.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 753°C.

1.5.1.3.1.3. Conclusions

Peak cladding temperature:

The maximum peak cladding temperature for all the cases analysed remains well below 1200°C.

This acceptance criterion is met with margins.

Fuel rod cladding rupture:

Regardless of the hot residence assembly and the burn-up of the indicator rod in question, no rupture occurs during the transient.

Using this overall separate approach it can be concluded that rupture will not occur during the transient. This result is confirmed by the study of rupture surfaces, as described in the following section.

1.5.1.3.2. Rupture surfaces

1.5.1.3.2.1. Introduction

In this study, the range of variation for parameters FQ and FΔH is selected as being arbitrarily constant for all cycles for reasons of simplification. This range is taken from the pre-accident configuration analysis for the core.

The range of variation for the FΔH parameter selected for the purpose of establishing rupture surfaces is [1,1.8], the maximum enthalpy rise factor from the core maps being 1.71.

The range of variation for the FQ parameter selected for the purpose of establishing rupture surfaces is [1,2.7], the maximum hot spot factor from the core maps being 2.57.

1.5.1.3.2.2. Rupture curves

The curves shown in this section represent limits beneath which fuel rod rupture has not occurred. The rupture curves are thus above the curves shown here.

Since failures were not detected in the selected FQ x FΔH range of variation, the curves shown here coincide with the upper limit of the range in question and are thus identical for the two hot residence assemblies.

In addition, these curves were only studied and established for rods with an enthalpy rise factor which is at least equal to that of a mean rod in the core (FΔH>1), i.e. for all rods with an FΔH less than 1, the hot spot factor is less than 1.905 and thus the rods do not rupture.

These limit curves at each burn-up point are compared with the properties of the rods in the core maps. The following curves represent limit curves, but also show the properties of the fuel rods in terms of FQ and FΔH for all core configurations. These have been grouped according to BU zone to make them easier to read:

- Section 16.4.1 - Figure A 40 shows the limit curve for the rod at the beginning of life compared with rods for which the BU is less than the maximum BOL burn-up,
- Section 16.4.1 - Figure A 41 shows the limit curve for the rod at the end of cycle 1 with maximum BU compared with the properties of rods for which the BU is between the maximum BOL burn-up and the maximum BU for EOC1,
- Section 16.4.1 - Figure A 42 shows the limit curve for the rod at the end of cycle 2 with maximum BU compared with the properties of rods for which the BU is between the maximum EOC1 burn-up and the maximum BU for EOC2,
- Section 16.4.1 - Figure A 43 shows the limit curve for the rod at the end of cycle 3 with maximum BU compared with the properties of rods for which the BU is between the maximum EOC2 burn-up and the maximum BU for EOC3,
- The rods at the end of cycle 4 all have an enthalpy rise factor of less than 1.

1.5.1.3.3. Conclusion

The analysis conducted for IN-OUT UO_2 18 months fuel management shows that there is no rupture of UO_2 rods with peak power located at the top of the core during the 2A-LOCA transient.

The maximum cladding temperature remains below 1200°C with margins.

1.5.2. Assessment for IN-OUT UO_2 22 months fuel management – peak power located at the top of the core

1.5.2.1. Determination of the fuel data and hot assemblies

1.5.2.1.1. Fuel data

The fuel data used to estimate the number of failed rods for the IN-OUT UO_2 22 months fuel management scenario were generated using the COPERNIC V2.4 [Ref-1] code.

Section 16.4.1 - Figure B 1 shows the history of power data for UO_2 rods in the IN-OUT UO_2 22 months fuel management.

The four calculation points to establish the rupture surfaces (see section 1.4.1.2.2) are described in Section 16.4.1 - Table B 1.

1.5.2.1.2. Determination of hot assemblies

Section 16.4.1 - Figure B 2 shows the enthalpy rise factor for the various fuel assemblies in the core for the fifteen possible configurations as a function of their mean burn-up.

Section 16.4.1 - Figure B 3 shows the hot spot factor of the various fuel assemblies in the core for all the core configurations, for all fifteen possible configurations.

Two assemblies were selected as the residence assemblies for the calculated fuel rods:

- *Zone 1:* mean burn-up of the assembly less than 30,000 MWd/tU,
- *Zone 2:* mean burn-up of the assembly greater than 30,000 MWd/tU.

The power of each zone is modelled by a rod with an enthalpy rise factor that bounds the enthalpy rise factor of all the assemblies in this zone.

Similarly, the hot spot factor for each zone is modelled by a rod with a hot spot factor that bounds all the assemblies in this zone.

Finally, the burn-up in each zone is selected as the calculation point that is the closest to the mean burn-up for the zone in question. The mean burn-up is determined as the mean burn-up value for the rods in the zone under consideration.

The properties of the two hot residence assemblies (HA1 for zone 1 and HA2 for zone 2) are given in Section 16.4.1 - Table B 2. The mean core is then imposed such as to obtain an enthalpy rise factor that is equal to the value for a mean core rod ($F\Delta H$ =1) and a hot spot factor equal to the ratio between FQ and $F\Delta H$ for the hot assembly.

1.5.2.2. Thermo-hydraulic results

1.5.2.2.1. Hot residence assembly 1

The main illustrations corresponding to the system calculation are shown on Section 16.4.1 - Figure B 4 to Section 16.4.1 - Figure B 16.

The main events of the transient are summarised in Section 16.4.1 - Table B 3.

The peak clad temperatures reached in the central hot assembly (the properties of which are reiterated in Section 16.4.1 - Table B 2) are given in Section 16.4.1 - Table B 4.

The maximum hot assembly peak clad temperature is obtained for the T1 peak at around 665°C.

1.5.2.2.2. Hot residence assembly 2

The main illustrations corresponding to the system calculation are shown on Section 16.4.1 - Figure B 17 to Section 16.4.1 - Figure B 29.

The main events of the transient are summarised in Section 16.4.1 - Table B 5.

The peak clad temperatures reached in the central hot assembly (the properties of which are reiterated in Section 16.4.1 - Table B 2) are given in Section 16.4.1 - Table B 6.

The maximum hot assembly peak clad temperature is obtained for the T1 peak at around 617°C.

1.5.2.3. Thermo-mechanical results

1.5.2.3.1. Indicator rods

Before establishing rupture surfaces, thermo-mechanical calculations are performed for each fuel rod burn-up point (BOL, EOC1, EOC2 and EOC3), for each hot residence assembly and in each azimuth.

These indicator rods are those that have the maximum linear power density and thus the highest hot spot factor of all the rods in all core configurations.

The bounding properties of these hot rods are described in Section 16.4.1 - Table B 7.

They are used to determine the most onerous azimuth in terms of thermo-mechanical calculations. The results obtained for these bounding rods in the sections below were obtained using the most onerous azimuth.

1.5.2.3.1.1. Indicator rods inserted in hot residence assembly 1

Section 16.4.1 - Table B 8 shows the results obtained for each indicator rod inserted in the thermo-hydraulic environment of hot assembly 1 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure B 30 to Section 16.4.1 - Figure B 33 present the maximum clad temperature transient, for each of the indicator rods.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 750°C.

1.5.2.3.1.2. Indicator rods inserted in hot residence assembly 2

Section 16.4.1 - Table B 9 shows the results obtained for each indicator rod inserted in the thermo-hydraulic environment of hot assembly 2 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure B 34 to Section 16.4.1 - Figure B 37 present the maximum clad temperature transient, for each of the indicator rods.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 758°C.

1.5.2.3.1.3. Conclusions

<u>Peak cladding temperature:</u>

The maximum peak cladding temperature for all the cases analysed remains well below 1200°C.

This acceptance criterion is met with margins.

<u>Fuel rod cladding rupture:</u>

Regardless of the hot residence assembly and the burn-up of the indicator rod in question, no rupture occurs during the transient.

Using this overall separate approach it can be concluded that rupture will not occur during the transient. This result is confirmed by the study of rupture surfaces, as described in the following section.

1.5.2.3.2. Rupture surfaces

1.5.2.3.2.1. Introduction

In this study, the range of variation for parameters FQ and F∆H is selected as being arbitrarily constant for all cycles for reasons of simplification. This range is taken from the pre-accident configuration analysis for the core.

The range of variation for the F∆H parameter selected for the purpose of establishing rupture surfaces is [1,1.8], the maximum enthalpy rise factor from the core maps being 1.555.

The range of variation for the FQ parameter selected for the purpose of establishing rupture surfaces is [1,2.7], the maximum hot spot factor from the core maps being 2.458.

1.5.2.3.2.2. Rupture curves

The procedure used is similar to the procedure outlined in section 1.5.1.3.2.2.

The curves shown in this section represent limits beneath which fuel rod rupture has not occurred. The rupture curves are thus above the limit curves shown here.

These limit curves at each burn-up point are compared with the properties of the rods in the core maps. The following curves represent limit curves, but also show the properties of the fuel rods in terms of FQ and FΔH for all core configurations. These have been grouped according to BU zone to make them easier to read:

- Section 16.4.1 - Figure B 38 shows the limit curve for the rod at the beginning of life compared with rods for which the BU is less than the maximum BOL burn-up,

- Section 16.4.1 - Figure B 39 shows the limit curve for the rod at the end of cycle 1 with maximum BU compared with the properties of rods for which the BU is between the maximum BOL burn-up and the maximum BU for EOC1,

- Section 16.4.1 - Figure B 40 shows the limit curve for the rod at the end of cycle 2 with maximum BU compared with the properties of rods for which the BU is between the maximum EOC1 burn-up and the maximum BU for EOC2,

The rods at the end of cycle 3 all have an enthalpy rise factor of less than 1.

1.5.2.3.3. Conclusion

The analysis conducted for IN-OUT UO_2 22 months fuel management shows that there is no rupture of UO_2 rods with peak power located at the top of the core during the 2A-LOCA transient.

The maximum cladding temperature remains below 1200°C with margins.

1.5.3. Assessment for IN-OUT UO_2 18 months fuel management – peak power located at mid-core

1.5.3.1. Determination of the fuel data and hot assemblies

1.5.3.1.1. Fuel data

The fuel data used to estimate the number of failed rods for the IN-OUT UO_2 18 months fuel management scenario were generated using the COPERNIC V2.4 [Ref-1] code.

Section 16.4.1 - Figure C 1 shows the power history for UO_2 rods in the IN-OUT UO_2 18 months fuel management.

The five calculation points to establish the rupture surfaces (see section 1.4.1.2.2) are described in Section 16.4.1 - Table C 1.

1.5.3.1.2. Determination of hot assemblies

Section 16.4.1 - Figure C 2 shows the enthalpy rise factor for the various fuel assemblies in the core for the fifteen possible configurations as a function of their mean burn-up.

Section 16.4.1 - Figure C 3 shows the hot spot factor of the various fuel assemblies in the core for all the same configurations.

Two assemblies were selected as the residence assemblies for the stand-alone fuel rods:

- *Zone 1:* mean burn-up of the assembly less than 30,000 MWd/tU,
- *Zone 2:* mean burn-up of the assembly greater than 30,000 MWd/tU.

The power of each zone is modelled by a rod with an enthalpy rise factor that bounds the enthalpy rise factor of all the assemblies in this zone.

Similarly, the hot spot factor for each zone is modelled by a rod with a hot spot factor that bounds all the assemblies in this zone.

Finally, the burn-up in each zone is selected as the calculation point that is the closest to the mean burn-up for the zone in question. The mean burn-up is determined as the mean burn-up value for the rods in the zone under consideration.

The properties of the two hot residence assemblies (HA1 for zone 1 and HA2 for zone 2) are given in Section 16.4.1 - Table C 2. The mean core is then imposed such as to obtain an enthalpy rise factor that is equal to the value for a mean core rod ($F\Delta H$=1) and a hot spot factor equal to the ratio between FQ and $F\Delta H$ for the hot assembly.

1.5.3.2. Thermo-hydraulic results

1.5.3.2.1. Hot residence assembly 1

The main illustrations corresponding to the system calculation are shown on Section 16.4.1 - Figure C 4 to Section 16.4.1 - Figure C 16.

The main events of the transient are summarised in Section 16.4.1 - Table C 3.

The peak clad temperatures reached in the central hot assembly (the properties of which are reiterated in Section 16.4.1 - Table C 2) are given in Section 16.4.1 - Table C 4.

The maximum hot assembly peak clad temperature is obtained for the T1 peak at around 634°C.

1.5.3.2.2. Hot residence assembly 2

The main illustrations corresponding to the system calculation are shown on Section 16.4.1 - Figure C 17 to Section 16.4.1 - Figure C 29.

The main events of the transient are summarised in Section 16.4.1 - Table C 5.

The peak clad temperatures reached in the central hot assembly (the properties of which are reiterated in Section 16.4.1 - Table C 2) are given in Section 16.4.1 - Table C 6.

The maximum hot assembly peak clad temperature is obtained for the T1 peak at around 611°C.

1.5.3.3. Thermo-mechanical results

1.5.3.3.1. Indicator rods

Before establishing rupture surfaces, thermo-mechanical calculations are performed for each fuel rod burn-up point (BOL, EOC1, EOC2, EOC3 and EOC4), for each hot residence assembly and in each azimuth.

These indicator rods are those which have the maximum linear power density and thus the highest hot spot factor of all the rods in all core configurations.

The bounding properties of these hot rods are described in Section 16.4.1 - Table C 7.

They are used to determine the most onerous azimuth in terms of thermo-mechanical calculations. The results obtained for these bounding rods in the sections below were obtained using the most onerous azimuth.

1.5.3.3.1.1. Indicator rods inserted in hot residence assembly 1

Section 16.4.1 - Table C 8 shows the results obtained for each indicator rod inserted in the thermo-hydraulic environment of hot assembly 1 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure C 30 to Section 16.4.1 - Figure C 34 present the maximum clad temperature transient, for each of the indicator rods.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 765°C.

1.5.3.3.1.2. Indicator rods inserted in hot residence assembly 2

Section 16.4.1 - Table C 9 shows the results obtained for each indicator rod inserted in the thermo-hydraulic environment of hot assembly 2 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure C 35 to Section 16.4.1 - Figure C 39 present the maximum clad temperature transient, for each of the indicator rods.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 754°C.

1.5.3.3.1.3. Conclusions

<u>Peak cladding temperature:</u>

The maximum peak cladding temperature for all the cases analysed remains well below 1200°C.

This acceptance criterion is met with margins.

<u>Fuel rod cladding rupture:</u>

Regardless of the hot residence assembly and the burn-up of the indicator rod in question, no rupture occurs during the transient.

Using this overall separate approach it can be concluded that rupture will not occur during the transient. This result is confirmed by the study of rupture surfaces, as described in the following section.

1.5.3.3.2. Rupture surfaces

1.5.3.3.2.1. Introduction

In this study, the range of variation for parameters FQ and FΔH is selected as being arbitrarily constant for all cycles for reasons of simplification. This range is taken from the pre-accident configuration analysis for the core.

The range of variation for the FΔH parameter selected for the purpose of establishing rupture surfaces is [1,1.8], the maximum enthalpy rise factor from the core maps being 1.71.

The range of variation for the FQ parameter selected for the purpose of establishing rupture surfaces is [1,2.7], the maximum hot spot factor from the core maps being 2.42.

1.5.3.3.2.2. Rupture curves

The procedure used is similar to the procedure outlined in section 1.5.1.3.2.2.

The curves shown in this section represent limits beneath which fuel rod rupture has not occurred. The rupture curves are thus above the limit curves shown here.

These limit curves at each burn-up point are compared with the properties of the rods in the core maps. The following curves represent limit curves, but also show the properties of the fuel rods in terms of FQ and FΔH for all core configurations. These have been grouped according to BU zone to make them easier to read:

- Section 16.4.1 - Figure C 40 shows the limit curve for the rod at the beginning of life compared with rods for which the BU is less than the maximum BOL burn-up,

- Section 16.4.1 - Figure C 41 shows the limit curve for the rod at the end of cycle 1 with maximum BU compared with the properties of rods for which the BU is between the maximum BOL burn-up and the maximum BU for EOC1,

- Section 16.4.1 - Figure C 42 shows the limit curve for the rod at the end of cycle 2 with maximum BU compared with the properties of rods for which the BU is between the maximum EOC1 burn-up and the maximum BU for EOC2,

- Section 16.4.1 - Figure C 43 shows the limit curve for the rod at the end of cycle 3 with maximum BU compared with the properties of rods for which the BU is between the maximum EOC2 burn-up and the maximum BU for EOC3,

The rods at the end of cycle 4 all have an enthalpy rise factor of less than 1.

1.5.3.3.3. Conclusion

The analysis conducted for the IN-OUT UO_2 18 months fuel management scenario shows that there is no failed UO_2 rod with peak power located at mid-core during the 2A-LOCA transient.

The maximum cladding temperature remains below 1200°C with margins.

1.5.4. Assessment for IN-OUT UO_2 22 months fuel management – peak power located at mid-core

1.5.4.1. Determination of the fuel data and hot assemblies

1.5.4.1.1. Fuel data

The fuel data used to estimate the number of failed rods for the IN-OUT UO_2 22 months fuel management scenario were generated using the COPERNIC V2.4 code.

Section 16.4.1 - Figure D 1 shows the power history for UO_2 rods in the IN-OUT UO_2 22 months fuel management.

The four calculation points to establish the rupture surfaces (see section 1.4.1.2.2) are described in Section 16.4.1 - Table D 1.

1.5.4.1.2. Determination of hot assemblies

Section 16.4.1 - Figure D 2 shows the enthalpy rise factor for the various fuel assemblies in the core for the fifteen possible configurations as a function of their mean burn-up.

Section 16.4.1 - Figure D 3 shows the hot spot factor of the various fuel assemblies in the core for all the core configurations, for all fifteen possible configurations.

Two assemblies were selected as the residence assemblies for the calculated fuel rods:

- *Zone 1:* mean burn-up of the assembly less than 30,000 MWd/tU,
- *Zone 2:* mean burn-up of the assembly greater than 30,000 MWd/tU.

The power of each zone is modelled by a rod with an enthalpy rise factor that bounds the enthalpy rise factor of all the assemblies in this zone.

Similarly, the hot spot factor for each zone is modelled by a rod with a hot spot factor that bounds all the assemblies in this zone.

Finally, the burn-up in each zone is selected as the calculation point that is the closest to the mean burn-up for the zone in question. The mean burn-up is determined as the mean burn-up value for the rods in the zone under consideration.

The properties of the two hot residence assemblies (HA1 for zone 1 and HA2 for zone 2) are given in Section 16.4.1 - Table D 2. The mean core is then imposed such as to obtain an enthalpy rise factor that is equal to the value for a mean core rod ($F\Delta H=1$) and a hot spot factor equal to the ratio between FQ and $F\Delta H$ for the hot assembly.

1.5.4.2. Thermo-hydraulic results

1.5.4.2.1. Hot residence assembly 1

The main illustrations corresponding to the system calculation are shown on Section 16.4.1 - Figure D 4 to Section 16.4.1 - Figure D 16.

The main events of the transient are summarised in Section 16.4.1 - Table D 3.

The peak clad temperatures reached in the central hot assembly (the properties of which are reiterated in Section 16.4.1 - Table D 2) are given in Section 16.4.1 - Table D 4.

The maximum hot assembly peak clad temperature is obtained for the T1 peak at around 666°C.

1.5.4.2.2. Hot residence assembly 2

The main illustrations corresponding to the system calculation are shown on Section 16.4.1 - Figure D 17 to Section 16.4.1 - Figure D 29.

The main events of the transient are summarised in Section 16.4.1 - Table D 5.

The peak clad temperatures reached in the central hot assembly (the properties of which are reiterated in Section 16.4.1 - Table D 2) are given in Section 16.4.1 - Table D 6.

The maximum hot assembly peak clad temperature is obtained for the T1 peak at around 613°C.

1.5.4.3. Thermo-mechanical results

1.5.4.3.1. Indicator rods

Before establishing rupture surfaces, thermo-mechanical calculations are performed for each fuel rod burn-up point (BOL, EOC1, EOC2 and EOC3), for each hot residence assembly and in each azimuth.

These indicator rods are those that have the maximum linear power density and thus the highest hot spot factor of all the rods in all core configurations.

The bounding properties of these hot rods are described in Section 16.4.1 - Table D 7.

They are used to determine the most onerous azimuth in terms of thermo-mechanical calculations. The results obtained for these bounding rods in the sections below were obtained using the most onerous azimuth.

1.5.4.3.1.1. Indicator rods inserted in hot residence assembly 1

Section 16.4.1 - Table D 8 shows the results obtained for each indicator rod inserted in the thermo-hydraulic environment of hot assembly 1 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure D 30 to Section 16.4.1 - Figure D 33 present the maximum clad temperature transient, for each of the indicator rods.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 755°C.

1.5.4.3.1.2. Indicator rods inserted in hot residence assembly 2

Section 16.4.1 - Table D 9 shows the results obtained for each indicator rod inserted in the thermo-hydraulic environment of hot assembly 2 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure D 34 to Section 16.4.1 - Figure D 37 present the maximum clad temperature transient, for each of the indicator rods.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 749°C.

1.5.4.3.1.3. Conclusions

Peak cladding temperature:

The maximum peak cladding temperature for all the cases analysed remains well below 1200°C.

This acceptance criterion is met with margins.

Fuel rod cladding rupture:

Regardless of the hot residence assembly and the burn-up of the indicator rod in question, no rupture occurs during the transient.

Using this overall separate approach it can be concluded that rupture will not occur during the transient. This result is confirmed by the study of rupture surfaces, as described in the following section.

1.5.4.3.2. Rupture surfaces

1.5.4.3.2.1. Introduction

In this study, the range of variation for parameters FQ and FΔH is selected as being arbitrarily constant for all cycles for reasons of simplification. This range is taken from the pre-accident configuration analysis for the core.

The range of variation for the FΔH parameter selected for the purpose of establishing rupture surfaces is [1:1.8], the maximum enthalpy rise factor from the core maps being 1.562.

The range of variation for the FQ parameter selected for the purpose of establishing rupture surfaces is [1:2.7], the maximum hot spot factor from the core maps being 2.417.

1.5.4.3.2.2. Rupture curves

The procedure used is similar to the procedure outlined in section 1.5.1.3.2.2.

The curves shown in this section represent limits beneath which fuel rod rupture has not occurred. The rupture curves are thus above the limit curves shown here.

These limit curves at each burn-up point are compared with the properties of the rods in the core maps. The following curves represent limit curves, but also show the properties of the fuel rods in terms of FQ and FΔH for all core configurations. These have been grouped according to BU zone to make them easier to read:

- Section 16.4.1 - Figure D 38 shows the limit curve for the rod at the beginning of life compared with rods for which the BU is less than the maximum BOL burn-up,
- Section 16.4.1 - Figure D 39 shows the limit curve for the rod at the end of cycle 1 with maximum BU compared with the properties of rods for which the BU is between the maximum BOL burn-up and the maximum BU for EOC1,
- Section 16.4.1 - Figure D 40 shows the limit curve for the rod at the end of cycle 2 with maximum BU compared with the properties of rods for which the BU is between the maximum EOC1 burn-up and the maximum BU for EOC2,

The rods at the end of cycle 3 all have an enthalpy rise factor of less than 1.

1.5.4.3.3. Conclusion

The analysis conducted for IN-OUT UO_2 22 months fuel management scenario shows that there is no rupture of UO_2 rods with peak power located at mid-core during the 2A-LOCA transient.

The maximum cladding temperature remains below 1200°C with margins.

1.5.5. Assessment for IN-OUT 30% MOX 18 months fuel management – peak power located at the top of the core

1.5.5.1. Determination of the fuel data and hot assemblies

1.5.5.1.1. Fuel data

A distinction is made between two types of MOX rods: rods with a high Pu content and rods with an average Pu content, which are representative of core assemblies (Section 16.4.1 - Table E 1).

Section 16.4.1 - Figure E 1 shows the power history for MOX rods with a high Pu content in the IN-OUT 30% MOX 18 months fuel management scenario. Section 16.4.1 - Figure E 2 shows the power history for MOX rods with an average Pu content.

The fuel data used to estimate the number of failed rods for the IN-OUT 30% MOX 18 months fuel management scenario were generated using the COPERNIC V2.4 [Ref-1] code.

Calculation points are shown in Section 16.4.1 - Table E 2 for rods with a high Pu content and in Section 16.4.1 - Table E 3 for rods with an average Pu content. These two contents are examined systematically as part of the thermo-mechanical behaviour analysis for the fuel rods.

The fuel data supplied by the COPERNIC V2.4 code are valid for MOX rods up to a maximum burn-up of 65,000 MWd/tU [Ref-1], referred to as C3.

Bounding data are used beyond this burn-up level. This is used for thermo-mechanical calculations for rods at a high burn-up. This makes it possible to define the initial fuel state in terms of pressure, temperature and oxide thickness. This data are selected as being arbitrarily high to simulate the potential behaviour of fuel with a high burn-up.

This data are shown in Section 16.4.1 - Table E 4 (rods with a high Pu content) and in Section 16.4.1 - Table E 5 (rods with an average Pu content).

1.5.5.1.2. Determination of hot assemblies

Section 16.4.1 - Figure E 3 shows the enthalpy rise factor for the various fuel assemblies in the core for the fifteen possible configurations as a function of their mean burn-up.

Section 16.4.1 - Figure E 4 shows the hot spot factor of the various fuel assemblies in the core for all the same configurations.

Two assemblies were selected as the residence assemblies for the stand-alone fuel rods:

- *Zone 1:* mean burn-up of the assembly less than 25,000 MWd/tU,
- *Zone 2:* mean burn-up of the assembly greater than 25,000 MWd/tU.

The power of each zone is modelled by a rod with an enthalpy rise factor that bounds the enthalpy rise factor of all the assemblies in this zone.

Similarly, the hot spot factor for each zone is modelled by a rod with a hot spot factor that bounds all the assemblies in this zone.

In addition, the fuel rod selected to model the residence assembly is a MOX rod with an average Pu content, since by definition this represents the average content of a core assembly.

Finally, the burn-up in each zone is selected as the calculation point that is the closest to the mean burn-up for the zone in question. The mean burn-up itself is determined as the mean burn-up value for the rods in the zone under consideration.

The properties of the two hot residence assemblies (HA1 for zone 1 and HA2 for zone 2) are given in Section 16.4.1 - Table E 6. The mean core is then imposed so as to obtain an enthalpy rise factor that is equal to the value for a mean core rod ($F\Delta H=1$) and a hot spot factor equal to the ratio between FQ and $F\Delta H$ for the hot assembly. The mean core burn-up corresponds to the mean burn-up of the core: the mean core is selected at end of cycle 2. Additionally, since the majority of assemblies in the core comprise $U0_2$-type rods, the mean core is modelled using UO_2-type fuel (Section 16.4.1 - Figure E 3 and Section 16.4.1 - Figure E 4).

1.5.5.2. Thermo-hydraulic results

1.5.5.2.1. Hot residence assembly 1

The main illustrations corresponding to the system calculation are shown on Section 16.4.1 - Figure E 5 to Section 16.4.1 - Figure E 17.

The main events of the transient are summarised in Section 16.4.1 - Table E 7.

The peak clad temperatures reached in the central hot assembly (the properties of which are reiterated in Section 16.4.1 - Table E 6) are given in Section 16.4.1 - Table E 8.

The maximum hot assembly peak clad temperature is obtained for the T1 peak at around 665°C.

1.5.5.2.2. Hot residence assembly 2

The main illustrations corresponding to the system calculation are shown on Section 16.4.1 - Figure E 18 to Section 16.4.1 - Figure E 30.

The main events of the transient are summarised in Section 16.4.1 - Table E 9.

The peak clad temperatures reached in the central hot assembly (the properties of which are reiterated in Section 16.4.1 - Table E 6) are given in Section 16.4.1 - Table E 10.

The maximum hot assembly peak clad temperature is obtained for the T1 peak at around 686°C.

1.5.5.3. Thermo-mechanical results

1.5.5.3.1. Indicator rods

Before establishing rupture surfaces, thermo-mechanical calculations are performed on the rods for each fuel rod burn-up point (BOL, EOC1, EOC2, C3 and EOC4), for each hot residence assembly and in each azimuth.

These indicator rods are those that have the maximum linear power density and thus the highest hot spot factor of all the rods in all core configurations.

The bounding properties for these hot rods are shown in Section 16.4.1 - Table E 11 for rods with a high Pu content and in Section 16.4.1 - Table E 12 for rods with an average Pu content.

They are used to determine the most onerous azimuth in terms of thermo-mechanical calculations. The results obtained for these bounding rods in the sections below were obtained using the most onerous azimuth.

1.5.5.3.1.1. Indicator rods inserted in hot residence assembly 1

1.5.5.3.1.1.1. Indicator rods with a high Pu content

Section 16.4.1 - Table E 13 shows the results obtained for each indicator rod with a high Pu content inserted in the thermo-hydraulic environment of hot assembly 1 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure E 31 to Section 16.4.1 - Figure E 35 present the maximum clad temperature transient, for each of the indicator rods.

In particular, the following figures apply to fuel rods at EOC4 which is limiting with respect to the clad temperature and deformation:

- Section 16.4.1 - Figure E 36 shows the true stress compared with the rupture stress at the peak power height,
- Section 16.4.1 - Figure E 37 shows the clad deformation transient at the peak power height.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 771°C.

1.5.5.3.1.1.2. Indicator rods with an average Pu content

Section 16.4.1 - Table E 14 shows the results obtained for each indicator rod with an average Pu content inserted in the thermo-hydraulic environment of hot assembly 1 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure E 38 to Section 16.4.1 - Figure E 42 present the maximum clad temperature transient, for each of the indicator rods.

In particular, the following figures apply to fuel rods at EOC4, which is limiting with respect to the clad temperature and deformation:

- Section 16.4.1 - Figure E 43 shows the true stress compared with the rupture stress at the peak power height,
- Section 16.4.1 - Figure E 44 shows the clad deformation transient at the peak power height.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 782°C.

1.5.5.3.1.2. Indicator rods inserted in hot residence assembly 2

1.5.5.3.1.2.1. Indicator rods with a high Pu content

Section 16.4.1 - Table E 15 shows the results obtained for each indicator rod with a high Pu content inserted in the thermo-hydraulic environment of hot assembly 2 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure E 45 to Section 16.4.1 - Figure E 49 present the maximum clad temperature transient, for each of the indicator rods.

In particular, the following figures apply to fuel rods at EOC4, which is limiting with respect to the clad temperature and deformation:

- Section 16.4.1 - Figure E 50 shows the true stress compared with the rupture stress at the peak power height,
- Section 16.4.1 - Figure E 51 shows the clad deformation transient at the peak power height.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 775°C

1.5.5.3.1.2.2. Indicator rods with an average Pu content

Section 16.4.1 - Table E 16 shows the results obtained for each indicator rod with an average Pu content inserted in the thermo-hydraulic environment of hot assembly 2 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure E 52 to Section 16.4.1 - Figure E 56 present the maximum clad temperature transient, for each of the indicator rods.

In particular, the following figures apply to fuel rods at EOC4, which is limiting with respect to the clad temperature and deformation:

- Section 16.4.1 - Figure E 57 shows the true stress compared with the rupture stress at the peak power height,
- Section 16.4.1 - Figure E 58 shows the clad deformation transient at the peak power height.

The hot rod maximum peak cladding temperature is obtained for Ti peak at around 786°C.

1.5.5.3.1.3. Conclusions

<u>Peak cladding temperature:</u>

The maximum peak cladding temperature for all the cases analysed remains well below 1200°C.

This acceptance criterion is met with margins.

<u>Fuel rod cladding rupture:</u>

Regardless of the hot residence assembly, the burn-up of the indicator rod in question and its Pu content, no rupture occurs during the transient.

Using this overall separate approach it can be concluded that rupture will not occur during the transient. This result is confirmed by the study of rupture surfaces, as described in the following section.

1.5.5.3.2. Rupture surfaces

1.5.5.3.2.1. Introduction

In this study, the range of variation for parameters FQ and FΔH is selected as being arbitrarily constant for all cycles for reasons of simplification. This range is taken from the pre-accident configuration analysis for the core.

The range of variation for the FΔH parameter selected for the purpose of establishing rupture surfaces is [1,1.626], the maximum enthalpy rise factor from the core maps being 1.626.

The range of variation for the FQ parameter selected for the purpose of establishing rupture surfaces is [1,2.353], the maximum hot spot factor from the core maps being 2.353.

1.5.5.3.2.2. Rupture curves[2]

The curves shown in this section represent limits beneath which fuel rod rupture has not occurred. The rupture curves are thus above the curves shown here.

Since ruptures were not detected in the selected FQ x FΔH range of variation, the curves shown here coincide with the upper limit of the range in question and are thus identical for the two hot residence assemblies.

In addition, these curves were only studied and established for rods with an enthalpy rise factor that is at least equal to that of a mean rod in the core (FΔH>1), i.e. for all rods with an FΔH less than 1, the hot spot factor is also less than 1.459 and thus the rods do not rupture.

These limit curves at each burn-up point are compared with the properties of the rods in the core maps. The following curves represent limit curves, but also show the properties of the fuel rods in terms of FQ and FΔH for all core configurations. These have been grouped according to BU zone to make them easier to read:

- Section 16.4.1 - Figure E 59 shows the limit curve for the rod at the beginning of life compared with rods for which the BU is less than the maximum BOL burn-up,
- Section 16.4.1 - Figure E 60 shows the limit curve for the rod at the end of cycle 1 with maximum BU compared with the properties of rods for which the BU is between the maximum BOL burn-up and the maximum BU for EOC1,
- Section 16.4.1 - Figure E 61 shows the limit curve for the rod at the end of cycle 2 with maximum BU compared with the properties of rods for which the BU is between the maximum EOC1 burn-up and the maximum BU for EOC2,
- Section 16.4.1 - Figure E 62 shows the limit curve for the rod at the end of cycle 3 with maximum BU compared with the properties of rods for which the BU is between the maximum EOC2 burn-up and the maximum BU for C3,
- The rods at the end of cycle 4 all have an enthalpy rise factor of less than 1.

1.5.5.3.3. Behaviour of the bounding UO_2 rod

In the thermo-hydraulic environments represented by the HA1 and HA2 residence assemblies, a thermo-mechanical verification calculation was carried out for a bounding UO_2-type fuel. The fuel data used for this calculation are from the IN-OUT UO_2 18 months fuel management (section 1.5), which bounds data for the IN-OUT 30% MOX 18 months fuel management in terms of FQ and FΔH.

This confirms that this rod does not rupture under these conditions.

The results obtained for this bounding rod are given in Section 16.4.1 - Table E 17.

1.5.5.3.4. Conclusion

The analysis conducted for IN-OUT 30% MOX 18 months fuel management scenario shows that there is no rupture of MOX rods with peak power located at the top of the core during the 2A-LOCA transient. Similarly, the bounding UO_2 fuel rod remains intact during the transient.

[2] The rupture curves obtained are identical for MOX rods with a high Pu content and MOX rods with an average Pu content.

The maximum cladding temperature remains below 1200°C with margins.

1.5.6. Assessment for IN-OUT 30% MOX 18 months fuel management – peak power located at mid-core

1.5.6.1. Determination of the fuel data and hot assemblies

1.5.6.1.1. Fuel data

A distinction is made between two types of MOX rods: rods with a high Pu content and rods with an average Pu content, which are representative of core assemblies (Section 16.4.1 - Table F 1).

Section 16.4.1 - Figure F 1 shows the power history for MOX rods with a high Pu content in the IN-OUT 30% MOX 18 months fuel management scenario. Section 16.4.1 - Figure F 2 shows the power history for MOX rods with an average Pu content.

The fuel data used to estimate the number of failed rods for the IN-OUT 30% MOX 18 months fuel management were generated using the COPERNIC V2.4 [Ref-1] code.

Calculation points are shown in Section 16.4.1 - Table F 2 for rods with a high Pu content and in Section 16.4.1 - Table F 3 for rods with an average Pu content. These two contents are examined systematically as part of the thermo-mechanical behaviour analysis for the fuel rods.

The fuel data supplied by the COPERNIC V2.4 code are valid for MOX rods up to a maximum burn-up of 65,000 MWd/tU [Ref-1] (referred to as C3).

Bounding data are used beyond this burn-up level. This is used for thermo-mechanical calculations for rods with a high burn-up. This makes it possible to define the initial fuel state in terms of pressure, temperature and oxide thickness. This data are selected as being arbitrarily high to simulate the potential behaviour of fuel with a high burn-up.

This data are shown in Section 16.4.1 - Table F 4 (rods with a high Pu content) and in Section 16.4.1 - Table F 5 (rods with an average Pu content).

1.5.6.1.2. Determination of hot assemblies

Section 16.4.1 - Figure F 3 shows the enthalpy rise factor for the various fuel assemblies in the core for the fifteen possible configurations as a function of their mean burn-up.

Section 16.4.1 - Figure F 4 shows the hot spot factor of the various fuel assemblies in the core for all the same configurations.

Two assemblies were selected as the residence assemblies for the stand-alone fuel rods:

- *Zone 1:* mean burn-up of the assembly less than 25,000 MWd/tU,
- *Zone 2:* mean burn-up of the assembly greater than 25,000 MWd/tU.

The power of each zone is modelled by a rod with an enthalpy rise factor that bounds the enthalpy rise factor of all the assemblies in this zone.

Similarly, the hot spot factor for each zone is modelled by a rod with a hot spot factor that bounds all the assemblies in this zone.

In addition, the fuel rod selected to model the residence assembly is a MOX rod with an average Pu content, since by definition this represents the average content of a core assembly.

Finally, the burn-up in each zone is selected as the calculation point that is the closest to the mean burn-up for the zone in question. The mean burn-up itself is determined as the mean burn-up value for the rods in the zone under consideration.

The properties of the two hot residence assemblies (HA1 for zone 1 and HA2 for zone 2) are given in Section 16.4.1 - Table F 6. The mean core is then imposed such as to obtain an enthalpy rise factor that is equal to the value for a mean core rod (F∆H=1) and a hot spot factor equal to the ratio between FQ and F∆H for the hot assembly. The mean core burn-up corresponds to the mean burn-up of the core: the mean core is selected at the end of cycle 2. Additionally, since the majority of assemblies in the core comprise UO_2-type rods, the mean core is modelled using UO_2-type fuel (Section 16.4.1 - Figure F 3 and Section 16.4.1 - Figure F 4).

1.5.6.2. Thermo-hydraulic results

1.5.6.2.1. Hot residence assembly 1

The main illustrations corresponding to the system calculation are shown on Section 16.4.1 - Figure F 5 to Section 16.4.1 - Figure F 17.

The main events of the transient are summarised in Section 16.4.1 - Table F 7.

The peak clad temperatures reached in the central hot assembly (the properties of which are reiterated in Section 16.4.1 - Table F 6) are given in Section 16.4.1 - Table F 8.

The maximum hot assembly peak clad temperature is obtained for the T1 peak at around 642°C.

1.5.6.2.2. Hot residence assembly 2

The main illustrations corresponding to the system calculation are shown on Section 16.4.1 - Figure F 18 to Section 16.4.1 - Figure F 30.

The main events of the transient are summarised in Section 16.4.1 - Table F 9.

The peak clad temperatures reached in the central hot assembly (the properties of which are reiterated in Section 16.4.1 - Table F 6) are given in Section 16.4.1 - Table F 10.

The maximum hot assembly peak clad temperature is obtained for the T1 peak at around 660°C.

1.5.6.3. Thermo-mechanical results

1.5.6.3.1. Indicator rods

Before establishing rupture surfaces, thermo-mechanical calculations are performed on the rods for each fuel rod burn-up point (BOL, EOC1, EOC2, C3 and EOC4), for each hot residence assembly and in each azimuth.

These indicator rods are those that have the maximum linear power density and thus the highest hot spot factor of all the rods in all core configurations.

The bounding properties for these hot rods are shown in Section 16.4.1 - Table F 11 for rods with a high Pu content and in Section 16.4.1 - Table F 12 for rods with an average Pu content.

They are used to determine the most onerous azimuth in terms of thermo-mechanical calculations. The results obtained for these bounding rods in the sections below were obtained using the most onerous azimuth.

1.5.6.3.1.1. Indicator rods inserted in hot residence assembly 1

1.5.6.3.1.1.1. Indicator rods with an average Pu content

Section 16.4.1 - Table F 13 shows the results obtained for each indicator rod with a high Pu content inserted in the thermo-hydraulic environment of hot assembly 1 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure F 31 to Section 16.4.1 - Figure F 35 present the maximum clad temperature transient, for each of the indicator rods with a high Pu content.

In particular, the following figures apply to fuel rods at EOC4, which is limiting with respect to the clad temperature and deformation:

- Section 16.4.1 - Figure F 36 shows the true stress compared with the rupture stress at the peak power height,
- Section 16.4.1 - Figure F 37 shows the clad deformation transient at the peak power height.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 766°C.

1.5.6.3.1.1.2. Indicator rods with an average Pu content

Section 16.4.1 - Table F 14 shows the results obtained for each indicator rod with an average Pu content inserted in the thermo-hydraulic environment of hot assembly 1 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure F 38 to Section 16.4.1 - Figure F 42 present the maximum clad temperature transient, for each of the indicator rods with an average Pu content.

In particular, the following figures apply to fuel rods at EOC4, which is limiting with respect to the clad temperature and deformation:

- Section 16.4.1 - Figure F 43 shows the true stress compared with the rupture stress at the peak power height,
- Section 16.4.1 - Figure F 44 shows the clad deformation transient at the peak power height.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 774°C.

1.5.6.3.1.2. Indicator rods inserted in hot residence assembly 2

1.5.6.3.1.2.1. Indicator rods with a high Pu content

Section 16.4.1 - Table F 15 shows the results obtained for each indicator rod with a high Pu content inserted in the thermo-hydraulic environment of hot assembly 2 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure F 45 to Section 16.4.1 - Figure F 49 present the maximum clad temperature transient, for each of the indicator rods with a high Pu content.

In particular, the following figures apply to fuel rods at EOC4, which is limiting with respect to the clad temperature and deformation:

- Section 16.4.1 - Figure F 50 shows the true stress compared with the rupture stress at the peak power height,
- Section 16.4.1 - Figure F 51 shows the clad deformation transient at the peak power height.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 764°C.

1.5.6.3.1.2.2. Indicator rods with an average Pu content

Section 16.4.1 - Table F 16 shows the results obtained for each indicator rod with an average Pu content inserted in the thermo-hydraulic environment of hot assembly 2 in terms of:

- peak clad temperature,
- maximum deformation achieved,
- rod clad rupture.

Section 16.4.1 - Figure F 52 to Section 16.4.1 - Figure F 56 present the maximum clad temperature transient, for each of the indicator rods with an average Pu content.

In particular, the following figures apply to fuel rods at EOC4, which is limiting with respect to the clad temperature and deformation:

- Section 16.4.1 - Figure F 57 shows the true stress compared with the rupture stress at the peak power height,
- Section 16.4.1 - Figure F 58 shows the clad deformation transient at the peak power height.

The hot rod maximum peak cladding temperature is obtained for the T1 peak at around 772°C.

1.5.6.3.1.3. Conclusions

Peak cladding temperature:

The maximum peak cladding temperature for all the cases analysed remains well below 1200°C.

This acceptance criterion is met with margins.

Fuel rod cladding rupture:

Regardless of the hot residence assembly, the burn-up of the indicator rod in question and its Pu content, no rupture occurs during the transient.

Using this overall separate approach it can be concluded that rupture will not occur during the transient. This result is confirmed by the study of rupture surfaces, as described in the following section.

1.5.6.3.2. Rupture surfaces

1.5.6.3.2.1. Introduction

In this study, the range of variation for parameters FQ and FΔH is selected as being arbitrarily constant for all cycles for reasons of simplification. This range is taken from the pre-accident configuration analysis for the core.

The range of variation for the FΔH parameter selected for the purpose of establishing rupture surfaces is [1,1.635], the maximum enthalpy rise factor from the core maps being 1.635.

The range of variation for the FQ parameter selected for the purpose of establishing rupture surfaces is [1,2.257], the maximum hot spot factor from the core maps being 2.257.

1.5.6.3.2.2. Rupture curves[3]

The curves shown in this section represent limits beneath which fuel rod rupture has not occurred. The rupture curves are thus above the curves shown here.

Since ruptures were not detected in the selected FQ x FΔH range of variation, the curves shown here coincide with the upper limit of the range in question and are thus identical for the two hot residence assemblies.

In addition, these curves were only studied and established for rods with an enthalpy rise factor which is at least equal to that of an average rod in the core (FΔH>1), i.e. for all rods with an FΔH less than 1, the hot spot factor is also less than 1.414 and thus the rods do not rupture.

These limit curves at each burn-up point are compared with the properties of the rods in the core maps. The following curves represent limit curves, but also show the properties of the fuel rods in terms of FQ and FΔH for all core configurations. These have been grouped according to BU zone to make them easier to read:

- Section 16.4.1 - Figure F 59 shows the limit curve for the rod at the beginning of life compared with rods for which the BU is less than the maximum BOL burn-up,

[3] The rupture curves obtained are identical for MOX rods with a high Pu content and MOX rods with an average Pu content.

- Section 16.4.1 - Figure F 60 shows the limit curve for the rod at the end of cycle 1 with maximum BU compared with the properties of rods for which the BU is between the maximum BOL burn-up and the maximum BU for EOC1,

- Section 16.4.1 - Figure F 61 shows the limit curve for the rod at the end of cycle 2 with maximum BU compared with the properties of rods for which the BU is between the maximum EOC1 burn-up and the maximum BU for EOC2,

- Section 16.4.1 - Figure F 62 shows the limit curve for the rod at the end of cycle 3 with maximum BU compared with the properties of rods for which the BU is between the maximum EOC2 burn-up and the maximum BU for C3,

- The rods at the end of cycle 4 all have an enthalpy rise factor of less than 1.

1.5.6.3.3. Behaviour of the bounding UO_2 rod

In the thermo-hydraulic environments represented by the HA1 and HA2 residence assemblies, a thermo-mechanical verification calculation was carried out for the bounding UO_2-type fuel. The fuel data used for this calculation are from the IN-OUT UO_2 18 months fuel management (section 1.5), which bounds data for the IN-OUT 30% MOX 18 months fuel management in terms of FQ and FΔH.

This confirms that this rod does not rupture under these conditions.

The results obtained for this bounding rod are given in Section 16.4.1 - Table F 17.

1.5.6.3.4. Conclusion

The analysis conducted for IN-OUT 30% MOX 18 months fuel management scenario shows that there is no failed rod with peak power located at mid-core during the 2A-LOCA transient. Similarly, the bounding UO_2 fuel rod remains intact during the transient.

The maximum cladding temperature remains below 1200°C with margins.

1.6. CONCLUSIONS

The three fuel management options, IN-OUT UO_2, 4500 MWth 18 months, IN-OUT UO_2, 4500 MWth 22 months and IN-OUT 30% MOX, 4500 MWth 18 months were examined as part of this study to estimate the number of failed rods in a 2A-LOCA transient (a double-ended guillotine break in the cold leg).

In view of the results obtained (very minimal deformation, very low stresses on the clad, not very high temperature levels and no rewetting during blowdown), the analysis conducted concludes that no rods (either UO_2 or MOX) rupture during the 2A-LOCA transient for the three bounding fuel management configurations proposed.

SECTION 16.4.1 - TABLE 1
ASSUMPTIONS CONCERNING THE INITIAL STATE OF THE NSSS

Initial state	Value
Total primary power (%) Total primary power (MWth)	100 4500 [Ref-1]
Thermal power of the four primary pumps (MWth)	24 [Ref-1]
PZR pressure (bar abs)	155 [Ref-1]
PZR level – liquid volume (m^3)	40 [Ref-1]
Core bypass flow rate (%)	3.46 [Ref-1]
Loop flow rate (m^3/h per loop)	28315 [Ref-1]
Core outlet temperature (°C)	330.1 [Ref-1]
Core inlet temperature (°C)	296.1 [Ref-1]
Average temperature (°C)	312.6 [Ref-1]
Upper head temperature (°C)	329 [Ref-1]
Operation of primary pumps	Nominal
Nominal height (m)	102.1
Rated torque (N.m.)	45575
Inertia (kg.m^2)	5238
SG liquid level (m)	15.7 [Ref-1]
MFWS temperature at the SG inlet (°C)	230 [Ref-1]
Steam flow rate per SG (kg/s)	639 [Ref-1]
SG saturation pressure (bar abs)	78 [Ref-1]
SG plugging	0
SG fouling	0
Mean linear power density (W/cm)	163.4 [Ref-1]
Initial stored energy	No uncertainty
Pellet-clad gap pressure	No uncertainty
Spalling	Not taken into account

SECTION 16.4.1 - TABLE 2
ASSUMPTIONS CONCERNING BOUNDARY CONDITIONS

Parameters	Values
Residual power	ORIGEN-S without uncertainty [Ref-1]
Fission power	Void effect, curve from the Preliminary Safety Analysis Report [Ref-1] Doppler coefficient temperature -4.03 pcm/°C
RT signal due to “Pressuriser low pressure” (bar abs)	135 – 1.5 (degraded atmosphere)=133.5 [Ref-1]
RT delay after signal (s)	1.2 (0.9 for I&C + 0.3 for the RT breakers to open and the gripper assemblies to release [Ref-1])
Turbine trip on receipt of the RT signal (s)	1.2 (0.9 for I&C + 0.3 for the turbine valves to close) [Ref-1]
MFWS isolation delay on receipt of the RT signal (s)	15.9 (0.9 for I&C + 15 for the valves to close) [Ref-1]
Reactor Coolant Pump trip signal (% of nom. ΔP for the Reactor Coolant Pump)	80 – 2 (degraded atmosphere)=78 [Ref-1]
Reactor Coolant Pump trip delay after signal (s)	1.05 (0.9 for I&C + 0.15 for the breaker to open) [Ref-1]
SI signal due to “Pressuriser low pressure” (bar abs)	115 – 1.5 (degraded atmosphere)=113.5 [Ref-1]
SI delay after signal (s)	15.9 (0.9 for I&C + 15 for the pump to start up to full flow) [Ref-1]
Partial cooldown	Not modelled
Number of SI trains	Three trains inject into the primary system One train lost at the break
SI flows	Minimum
SI temperature (°C)	15
Accumulators: Initial pressure (bar abs) Initial temperature (°C)	 47.5 [Ref-1] 30 [Ref-1]

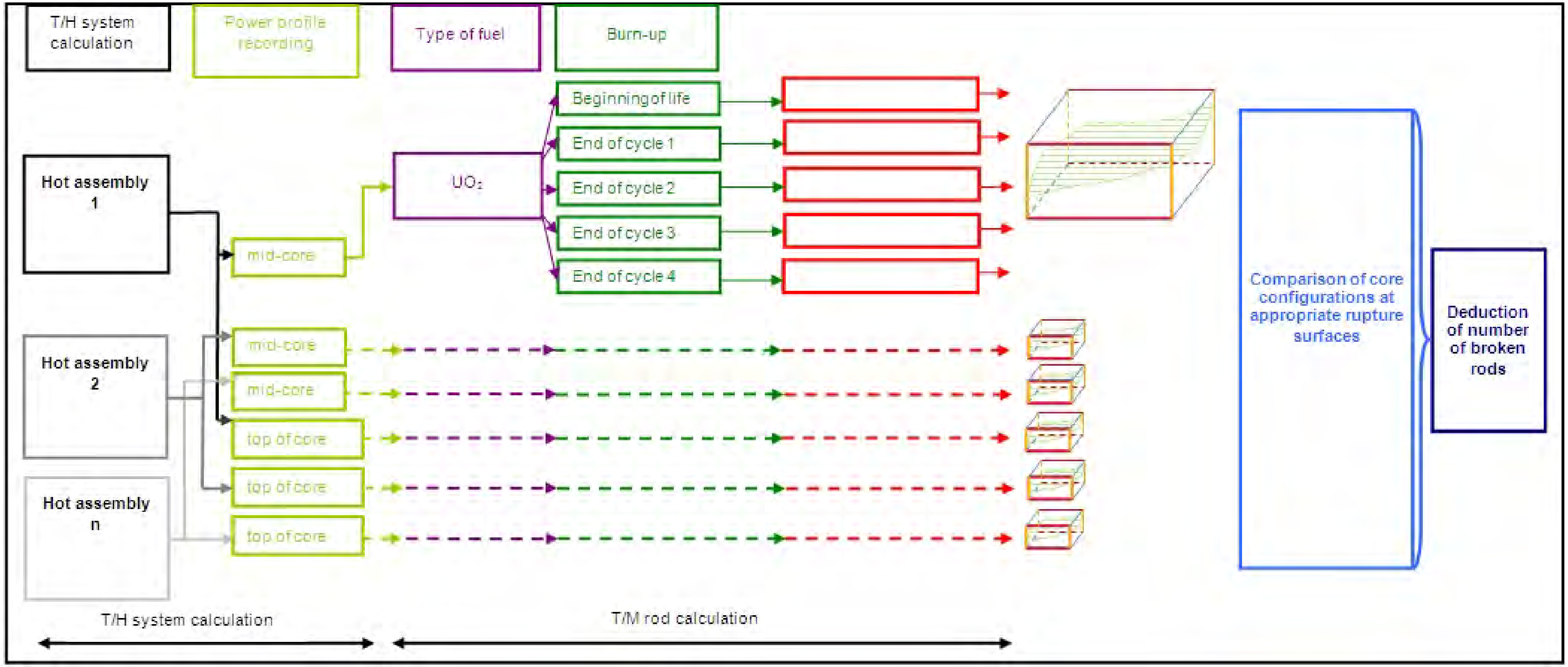
Parameters	Values
Total volume (m^3)	47 [Ref-1]
Liquid volume (m^3)	32.5 [Ref-1]
Expansion coefficient	1.4
Discharge line resistance (m^{-4})	2100 [Ref-1]

SECTION 16.4.1 - TABLE 3
ASSUMPTIONS WITH REGARD TO CALCULATING THE BACKPRESSURE IN THE CONTAINMENT

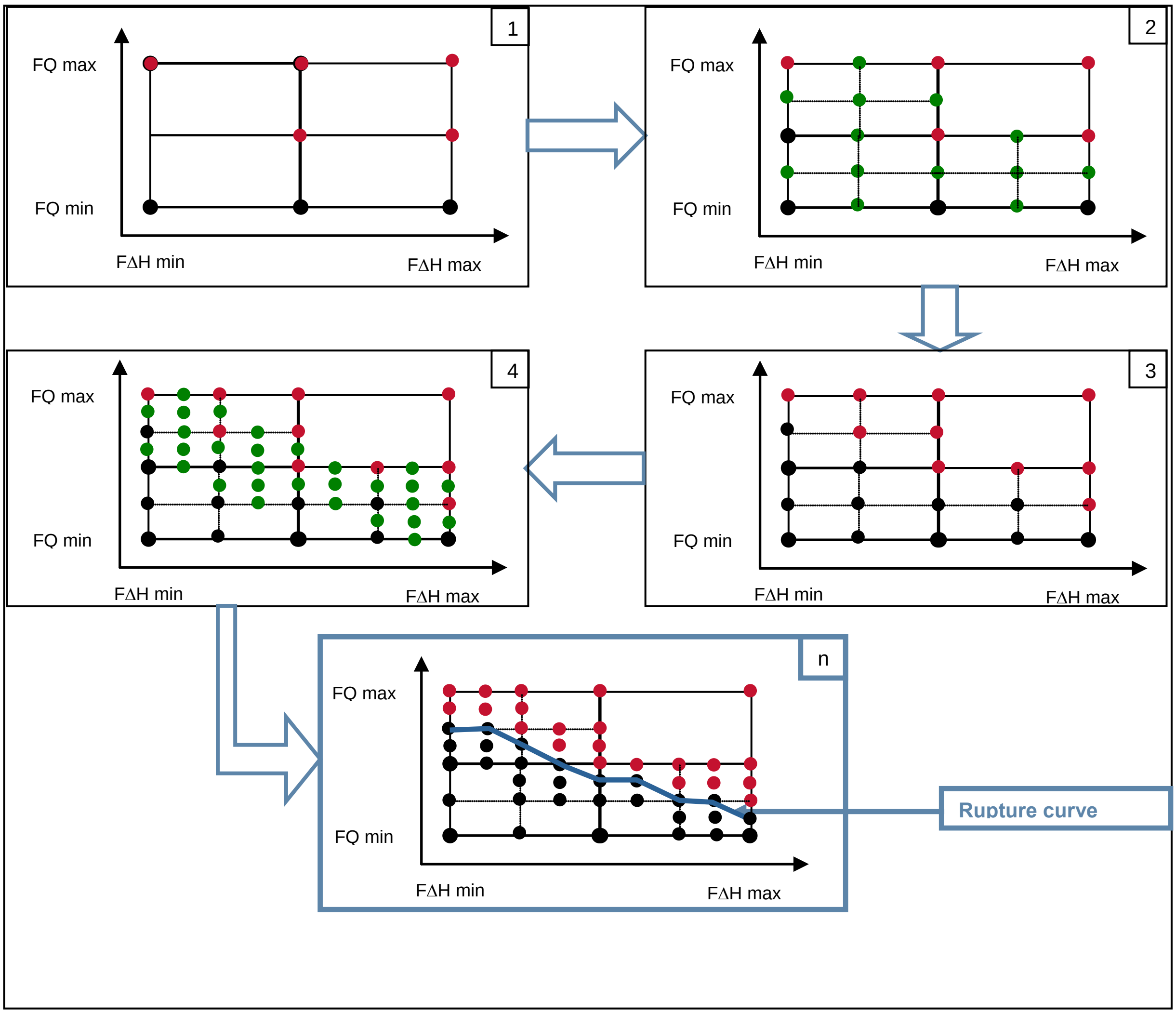
Parameters	Values
Free volume in the containment vessel (m^3)	80000 [Ref-1]
In-containment refuelling water storage tank (IRWST) volume (m^3)	1530 [Ref-1]
Multiplication factor for the Tagami correlation	4
Multiplication factor for the Uchida correlation	1.2
Initial temperature (°C)	25
Initial pressure (bar abs)	1

SECTION 16.4.1 - TABLE 4
LIMIT INSERTIONS AT 100% POWER

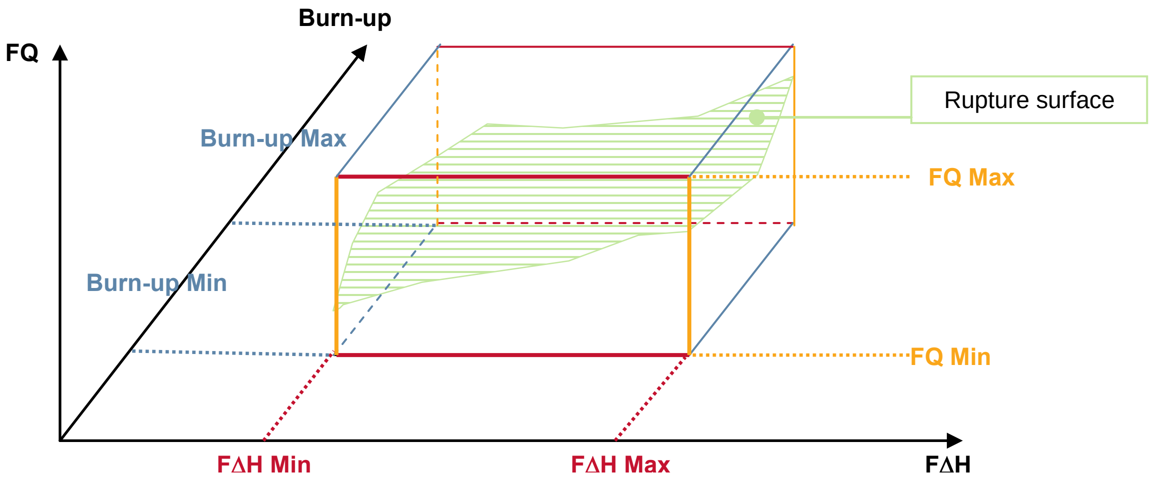
	Beginning of life	Mid-life	End of life
P1	200 steps	200 steps	100 steps
P2	100 steps	100 steps	50 steps
P3	100 steps	100 steps	50 steps
P4	100 steps	100 steps	50 steps
P5	100 steps	100 steps	50 steps



SECTION 16.4.1 - FIGURE 1
OVERVIEW OF THE STUDY TO ESTIMATE THE NUMBER OF FAILED RODS

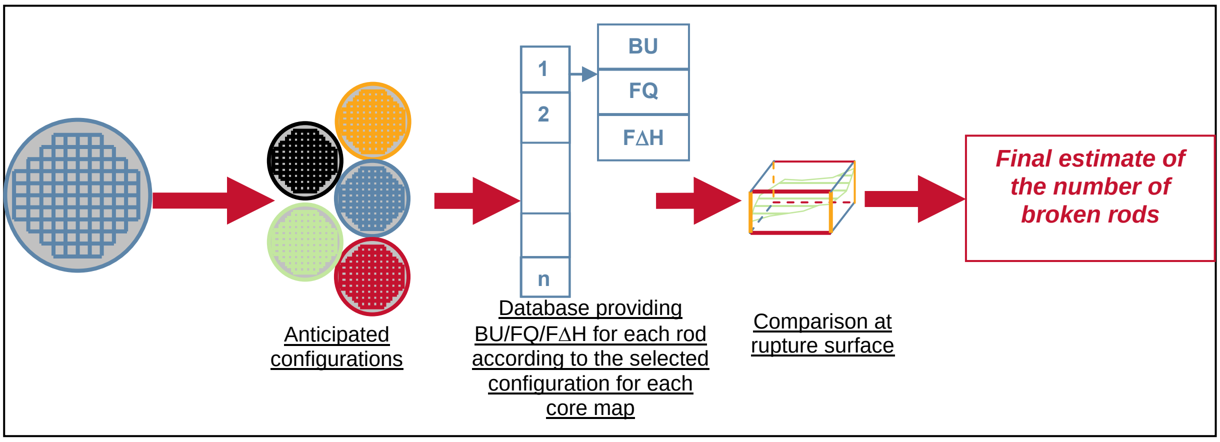


SECTION 16.4.1 - FIGURE 2
PRINCIPLE OF THE ITERATIVE DETERMINATION OF THE RUPTURE CURVE WITH A FIXED BURN-UP VALUE

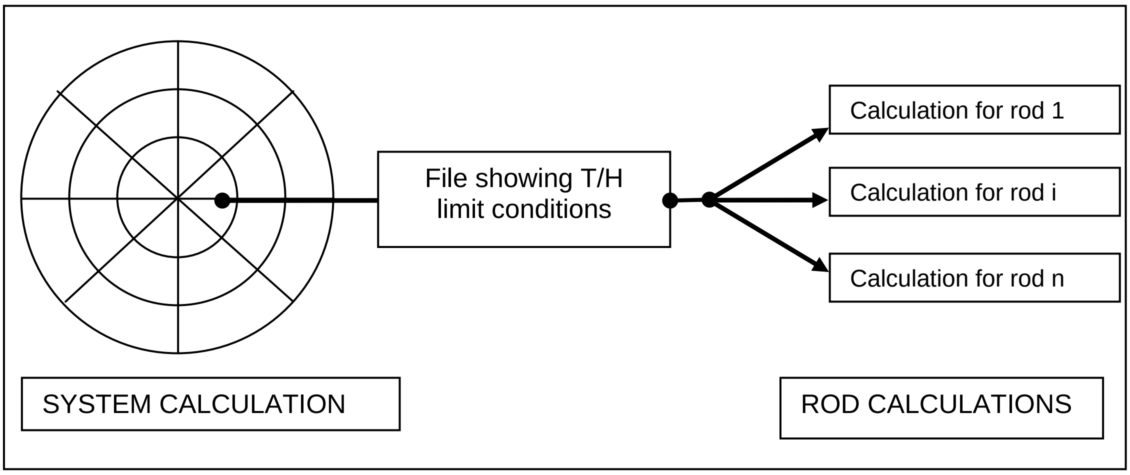


Burn-up Min	: minimum BU value observed for rods at BOL
Burn-up Max	: maximum BU value observed for rods at EOL
FΔH Min	: minimum FΔH value for all rods on the processed core maps
FΔH max	: maximum FΔH value for all rods on the processed core maps
FQ min	: minimum FQ value for all rods on the processed core maps
FQ max	: maximum FQ value for all rods on the processed core maps

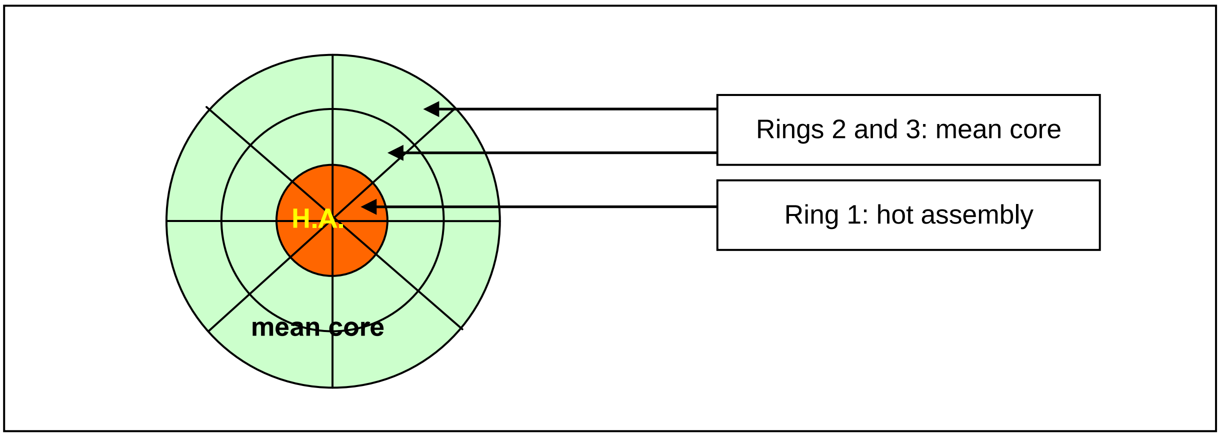
SECTION 16.4.1 - FIGURE 3
SCHEMATIC DIAGRAM SHOWING THE RUPTURE SURFACE



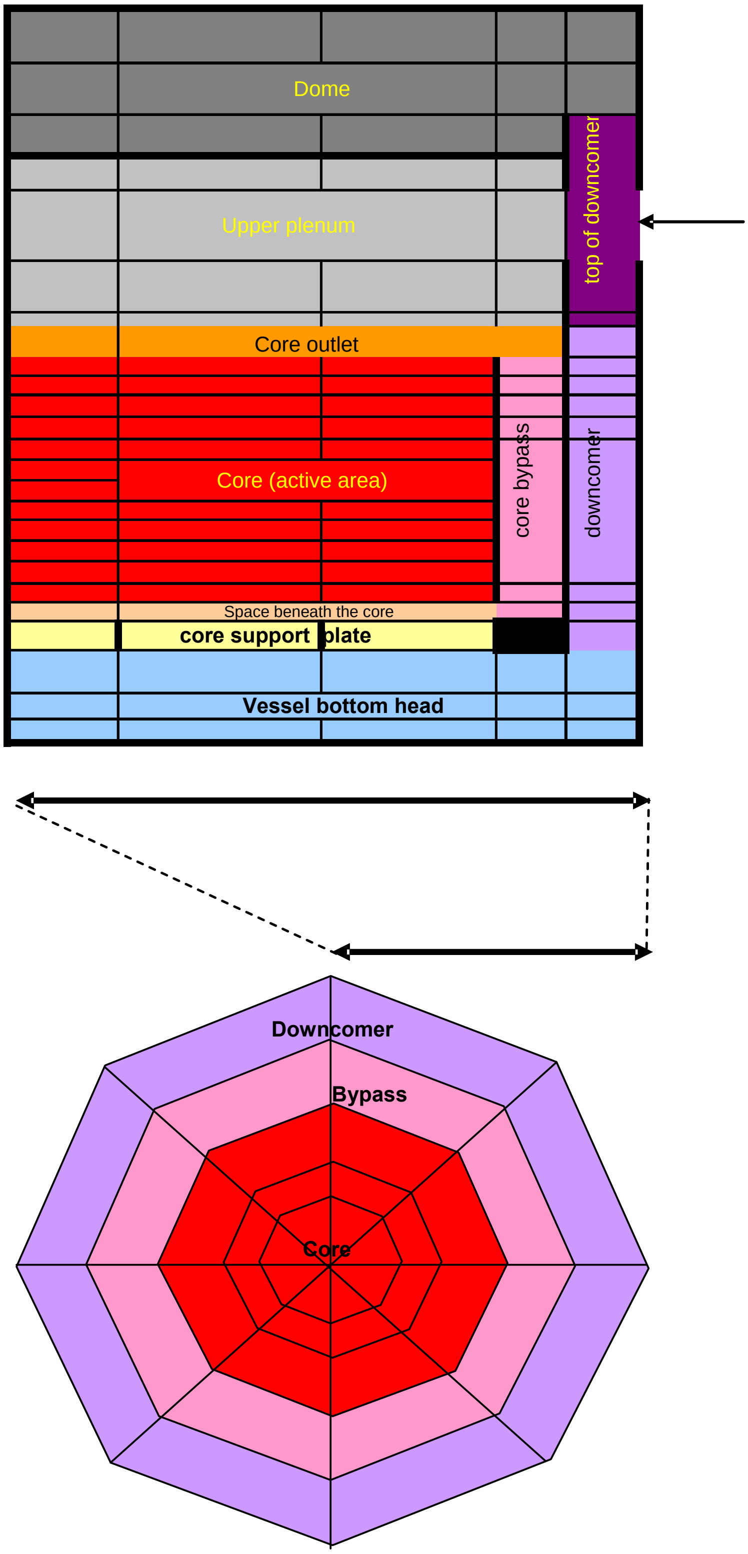
SECTION 16.4.1 - FIGURE 4
DETERMINATION OF THE NUMBER OF FAILED RODS BY COMPARISON WITH THE RUPTURE SURFACE



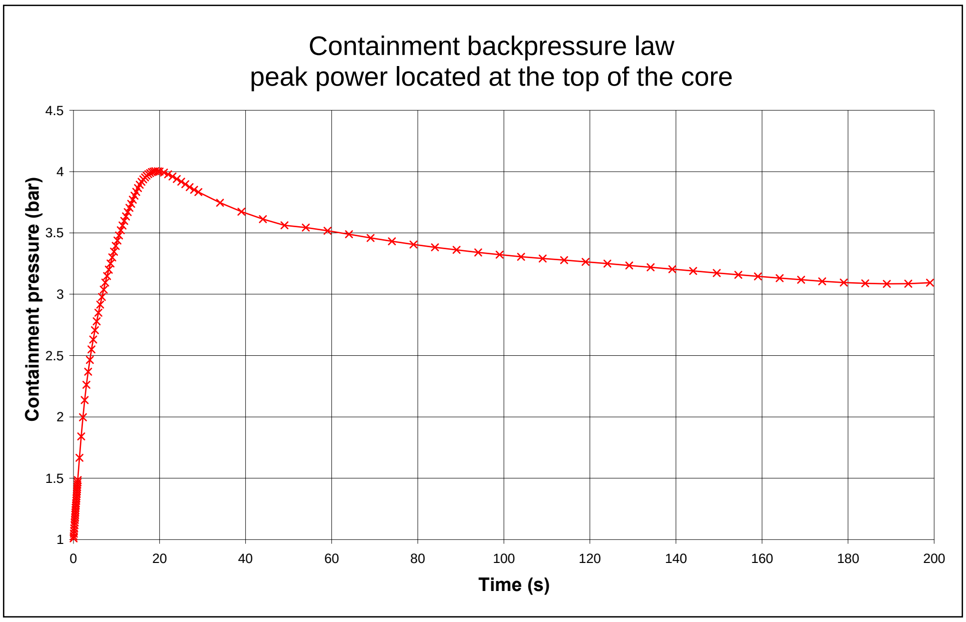
SECTION 16.4.1 - FIGURE 5
SEPARATION PRINCIPLE BETWEEN CATHARE 2 V2.5_1 AND ITS STAND-ALONE FUEL MODULE



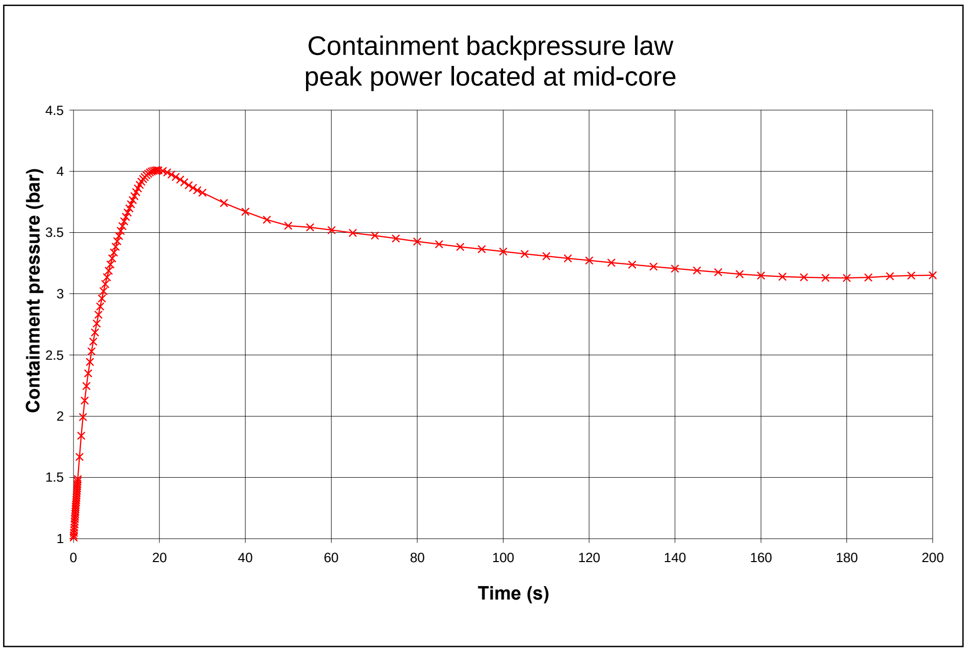
SECTION 16.4.1 - FIGURE 6
3D CORE MODELLING



SECTION 16.4.1 - FIGURE 7
DIAGRAM SHOWING MODELLED 3D VESSEL



SECTION 16.4.1 - FIGURE 8
CONTAINMENT BACKPRESSURE LAW – PEAK POWER LOCATED AT THE TOP OF THE CORE



SECTION 16.4.1 - FIGURE 9
CONTAINMENT BACKPRESSURE LAW – PEAK POWER LOCATED AT MID-CORE

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T
17									N								
16						N		N		N		N					
15			N		P1		P3		P4		P3		P5		N		
14				N		N						N		N			
13			P5		P5		P4		P5		P2		P5		P1		
12		N		N				N		N				N		N	
11			P3		P2		N				N		P4		P3		
10		N				N		N		N		N				N	
09	N		P4		P5				N				P5		P4		N
08		N				N		N		N		N				N	
07			P3		P4		N				N		P2		P3		
06		N		N				N		N				N		N	
05			P1		P5		P2		P5		P4		P5		P5		
04				N		N						N		N			
03			N		P5		P3		P4		P3		P1		N		
02						N		N		N		N					
01									N								

Rod cluster control assembly: P
Shutdown RCC assembly: N

SECTION 16.4.1 - FIGURE 10
DEFINITION OF CONTROL ROD SEQUENCE NO.1

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T
17									N								
16						N		N		N		N					
15			N		P4		P3		P4		P1		P5		N		
14				N		N						N		N			
13			P5		P2		P5		P5		P3		P2		P4		
12		N		N				N		N				N		N	
11			P1		P3		N				N		P5		P3		
10		N				N		N		N		N				N	
09	N		P4		P5				N				P5		P4		N
08		N				N		N		N		N				N	
07			P3		P5		N				N		P3		P1		
06		N		N				N		N				N		N	
05			P4		P2		P3		P5		P5		P2		P5		
04				N		N						N		N			
03			N		P5		P1		P4		P3		P4		N		
02						N		N		N		N					
01									N								

Rod cluster control assembly: P
Shutdown RCC assembly: N

SECTION 16.4.1 - FIGURE 11
DEFINITION OF CONTROL ROD SEQUENCE NO.2

17									N								
16						N		N		N		N					
15			N		P3		P4		P1		P4		P5		N		
14				N		N						N		N			
13			P5		P5		P3		P5		P2		P5		P3		
12		N		N				N		N				N		N	
11			P4		P2		N				N		P3		P4		
10		N				N		N		N		N				N	
09	N		P1		P5				N				P5		P1		N
08		N				N		N		N		N				N	
07			P4		P3		N				N		P2		P4		
06		N		N				N		N				N		N	
05			P3		P5		P2		P5		P3		P5		P5		
04				N		N						N		N			
03			N		P5		P4		P1		P4		P3		N		
02						N		N		N		N					
01									N								
	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T

Rod cluster control assembly: P
Shutdown RCC assembly: N

SECTION 16.4.1 - FIGURE 12
DEFINITION OF CONTROL ROD SEQUENCE NO.3

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T
17									N								
16						N		N		N		N					
15			N		P4		P1		P5		P3		P5		N		
14				N		N						N		N			
13			P5		P2		P4		P5		P3		P2		P4		
12		N		N				N		N				N		N	
11			P3		P3		N				N		P4		P1		
10		N				N		N		N		N				N	
09	N		P5		P5				N				P5		P5		N
08		N				N		N		N		N				N	
07			P1		P4		N				N		P3		P3		
06		N		N				N		N				N		N	
05			P4		P2		P3		P5		P4		P2		P5		
04				N		N						N		N			
03			N		P5		P3		P5		P1		P4		N		
02						N		N		N		N					
01									N								

Rod cluster control assembly: P
Shutdown RCC assembly: N

SECTION 16.4.1 - FIGURE 13
DEFINITION OF CONTROL ROD SEQUENCE NO.4

SECTION 16.4.1 - TABLE A 1: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – FUEL CALCULATION POINTS

CORE CYCLE TIME	ROD BURN-UP
BEGINNING OF LIFE (BOL)	150 MWD/TU
END OF CYCLE 1 (EOC1)	27,000 MWD/TU
END OF CYCLE 2 (EOC2)	47,750 MWD/TU
END OF CYCLE 3 (EOC3)	64,650 MWD/TU
END OF CYCLE 4 (EOC4)	72,250 MWD/TU

SECTION 16.4.1 - TABLE A 2: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – PROPERTIES OF HOT ASSEMBLIES

HOT ASSEMBLY	FΔH	FQ	BU
HA1	1.47	2.24	EOC1
HA2	1.20	1.91	EOC2

SECTION 16.4.1 - TABLE A 3: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – MAIN EVENTS OF THE TRANSIENT

EVENT	TIME
RT SIGNAL	**3.6 S**
SI SIGNAL	**7.1 S**
START OF ACCUMULATOR INJECTION	**14.1 S**
START O F SA FETY I NJECTION SY STEM INJECTION	**23.0 S**
START OF REFLOOD	**27.7 S**
END OF REFLOOD	**103.5 S**
DURATION OF REFLOOD	**75.8 S**
END OF ACCUMULATOR INJECTION	**44.0 S**

SECTION 16.4.1 - TABLE A 4: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – PEAK CLAD TEMPERATURES

	T1	T2	T3
TIME (S)	**3.7**	**29.6**	**48.9**
MESH	**11**	**11**	**11**
VALUE (°C)	**670**	**615**	**511**

SECTION 16.4.1 - TABLE A 5: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – MAIN EVENTS OF THE TRANSIENT

EVENT	TIME
RT SIGNAL	**3.6 S**
SI SIGNAL	**7.1 S**
START OF ACCUMULATOR INJECTION	**14.0 S**
START O F SAFETY I NJECTION SY STEM INJECTION	**23.0 S**
START OF REFLOOD	**27.3 S**
END OF REFLOOD	**81.7 S**
DURATION OF REFLOOD	**54.4 S**
END OF ACCUMULATOR INJECTION	**43.8 S**

SECTION 16.4.1 - TABLE A 6: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – PEAK CLAD TEMPERATURES

	T1	T2	T3
TIME (S)	**3.97**	**27.3**	**54**
MESH	**11**	**11**	**11**
VALUE (°C)	**609**	**537**	**488**

SECTION 16.4.1 - TABLE A 7: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – PROPERTIES OF THE INDICATOR RODS

BURN-UP	FQ	FΔH	PRESSURE IN THE GAP (BAR)	MAXIMUM LINEAR POWER DENSITY (W/CM)	MAXIMUM INITIAL PELLET TEMPERATURE (°C)
BOL	**2.57**	**1.71**	**56.8**	**431**	**957**
EOC1	**2.57**	**1.71**	**68.9**	**431**	**958**
EOC2	**2.57**	**1.71**	**79.6**	**431**	**1031**
EOC3	**2.57**	**1.71**	**101.1**	**431**	**1128**
EOC4	**2.57**	**1.71**	**103.5**	**431**	**1175**

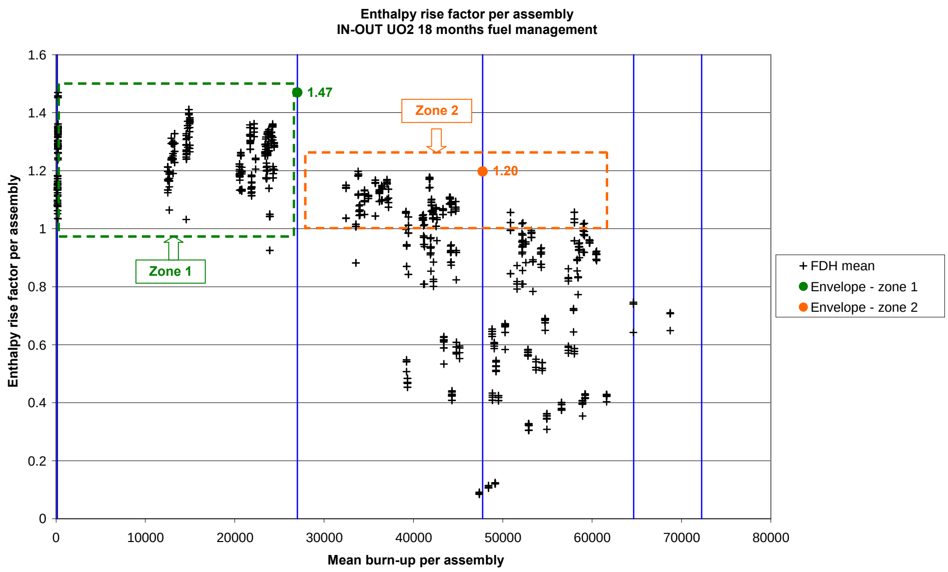
SECTION 16.4.1 - TABLE A 8: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – RESULTS OBTAINED FOR INDICATOR RODS

BURN-UP	T1 (°C)	T2 (°C)	T3 (°C)	MAXIMUM DEFORMATION (%)	RUPTURE
BOL	**712**	**656**	**533**	**0.40**	**NO**
EOC1	**716**	**663**	**538**	**0.30**	**NO**
EOC2	**743**	**674**	**542**	**0.30**	**NO**
EOC3	**770**	**700**	**554**	**0.40**	**NO**
EOC4	**793**	**714**	**579**	**0.50**	**NO**

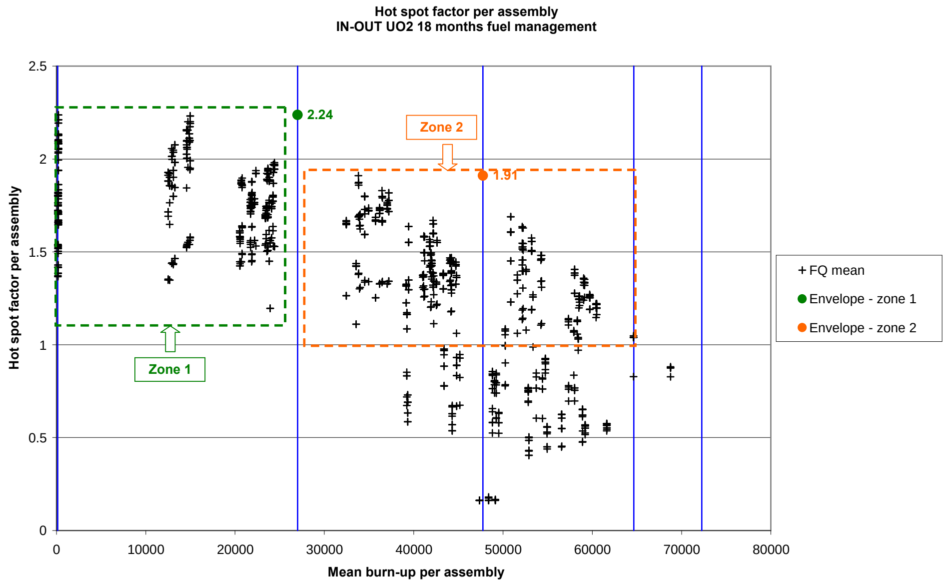
SECTION 16.4.1 - TABLE A 9: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – RESULTS OBTAINED FOR INDICATOR RODS

BURN-UP	T1 (°C)	T2 (°C)	T3 (°C)	MAXIMUM DEFORMATION (%)	RUPTURE
BOL	**686**	**613**	**561**	**0.34**	**NO**
EOC1	**685**	**616**	**563**	**0.33**	**NO**
EOC2	**713**	**626**	**567**	**0.34**	**NO**
EOC3	**740**	**652**	**579**	**0.39**	**NO**
EOC4	**753**	**666**	**586**	**0.43**	**NO**

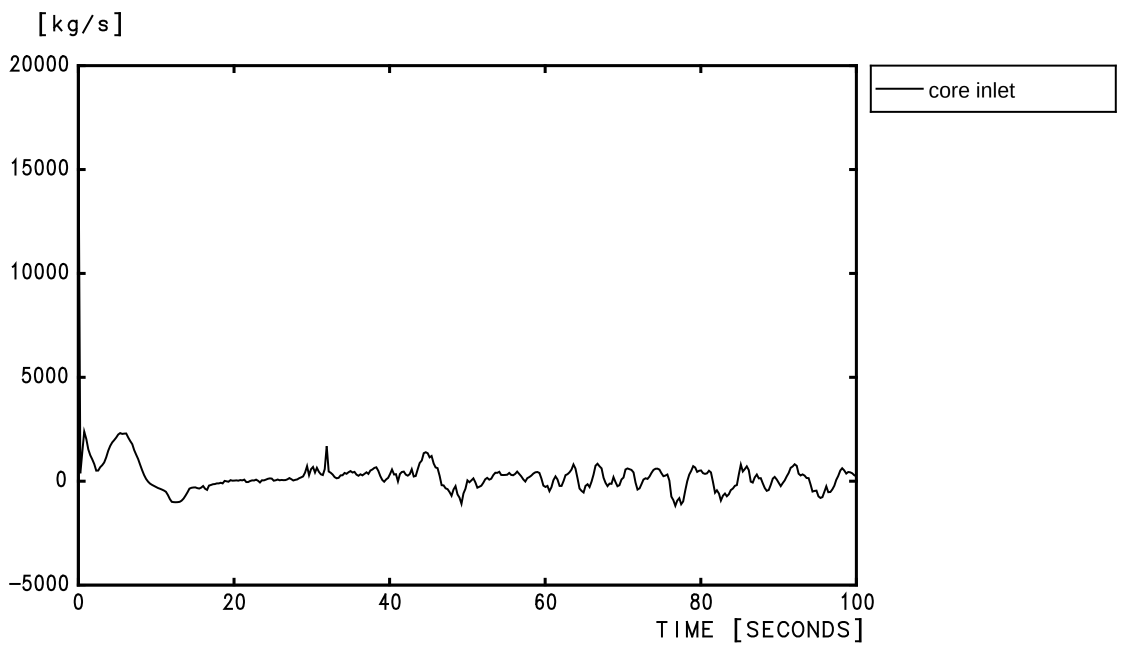
SECTION 16.4.1 - FIGURE A 1: IN-OUT UO2 18 MONTHS FUEL MANAGEMENT – TC – POWER HISTORY OF UO2 RODS



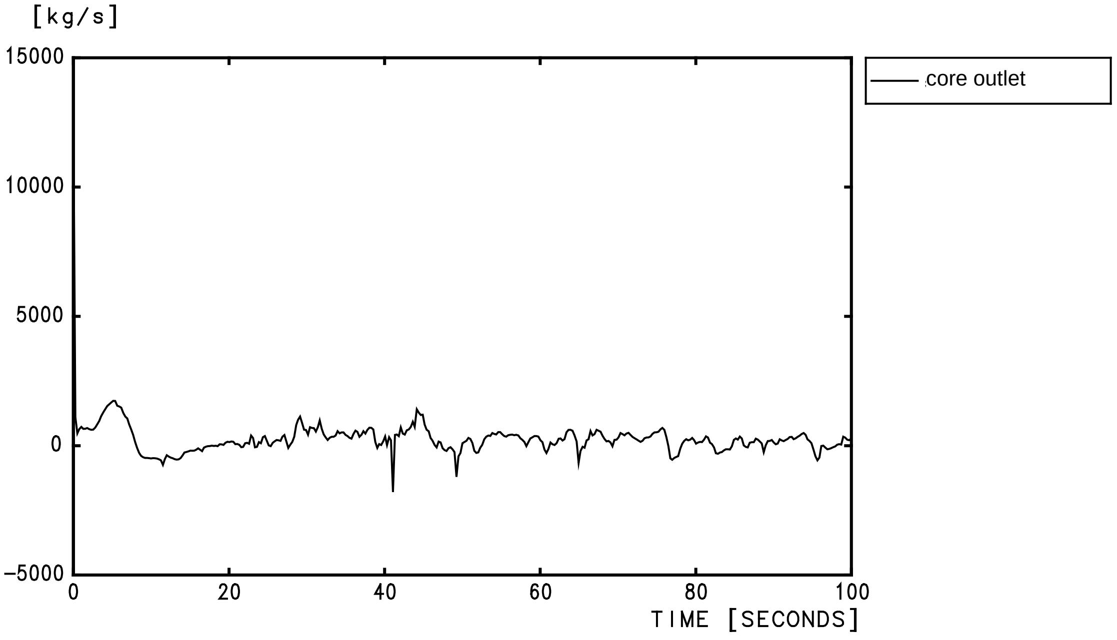
SECTION 16.4.1 - FIGURE A 2: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – FΔH OF ASSEMBLIES AS A FUNCTION OF BURN-UP



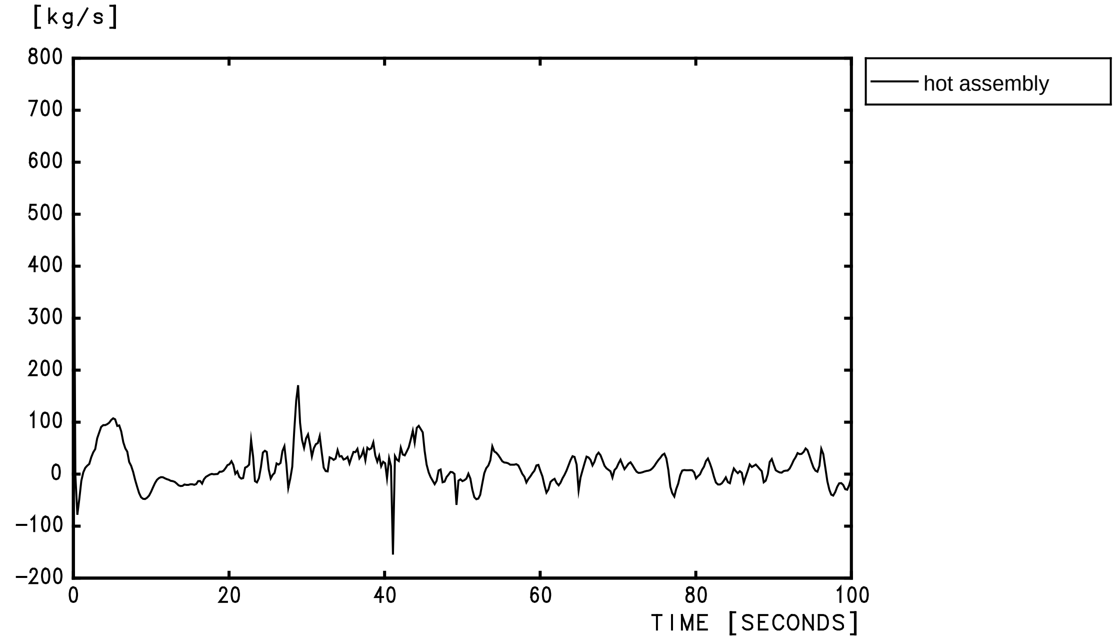
SECTION 16.4.1 - FIGURE A 3: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – FQ OF ASSEMBLIES AS A FUNCTION OF BURN-UP



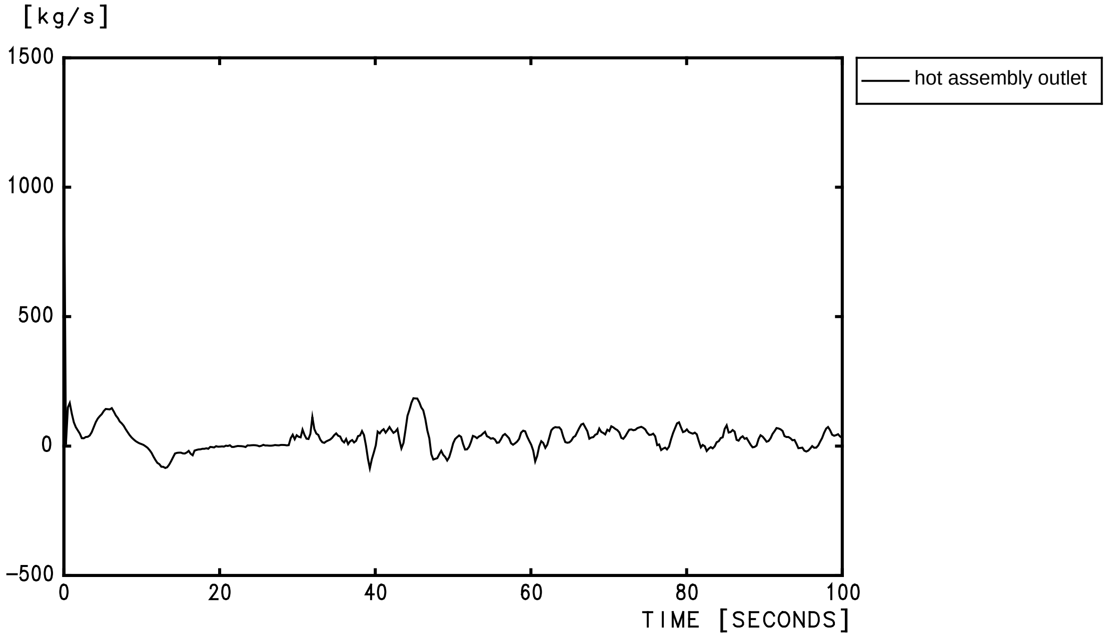
SECTION 16.4.1 - FIGURE A 4: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE CORE INLET



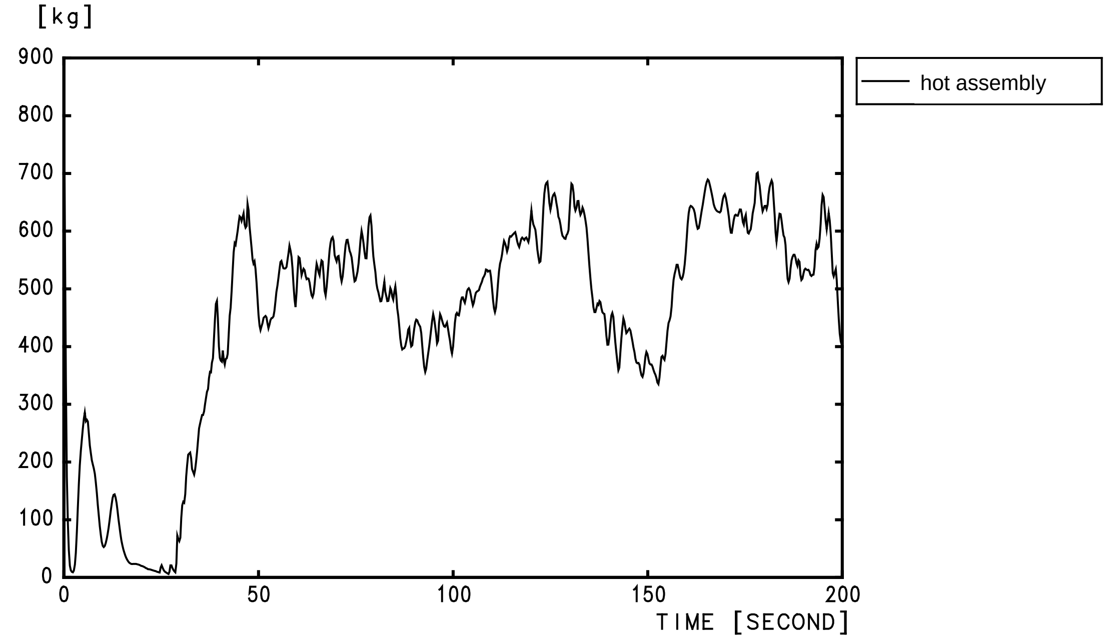
SECTION 16.4.1 - FIGURE A 5: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE CORE OUTLET



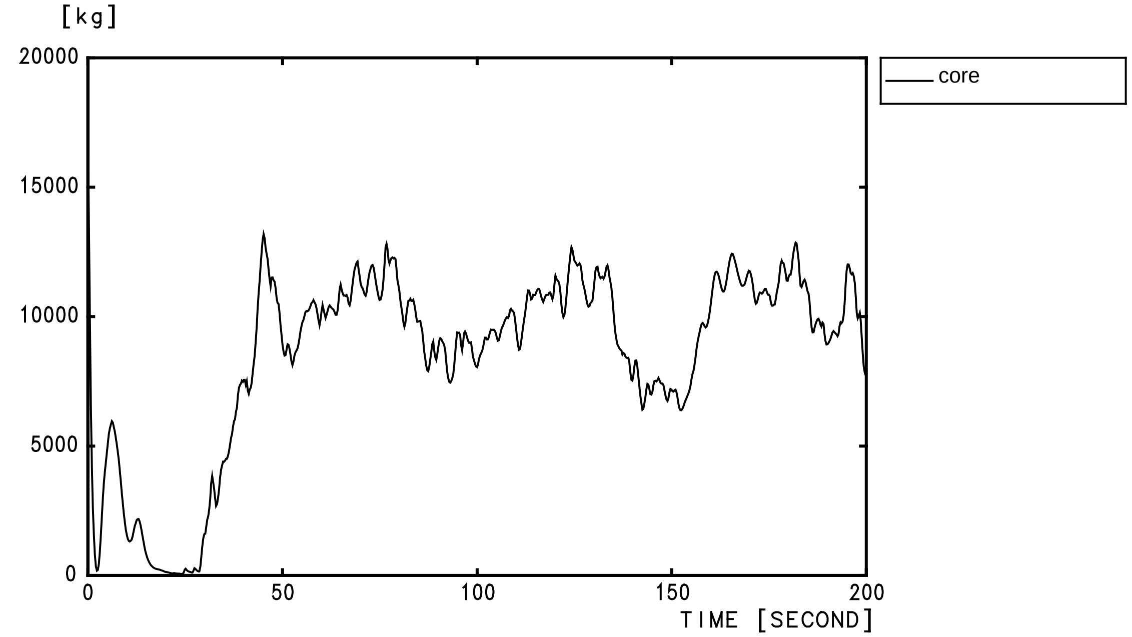
SECTION 16.4.1 - FIGURE A 6: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE HOT ASSEMBLY INLET



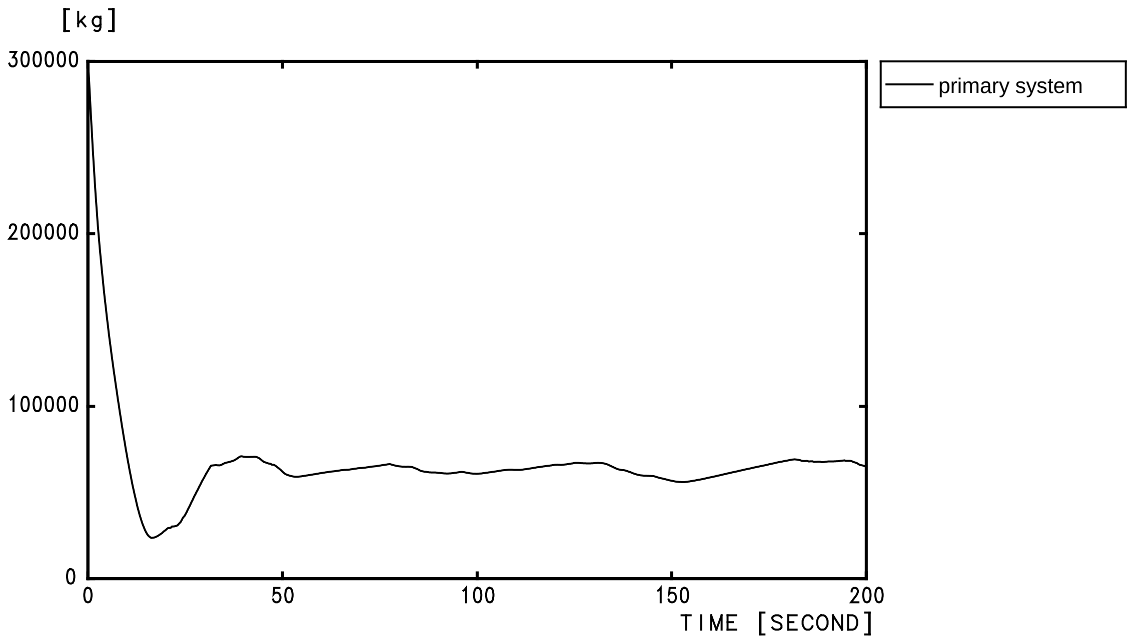
SECTION 16.4.1 - FIGURE A 7: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE HOT ASSEMBLY OUTLET



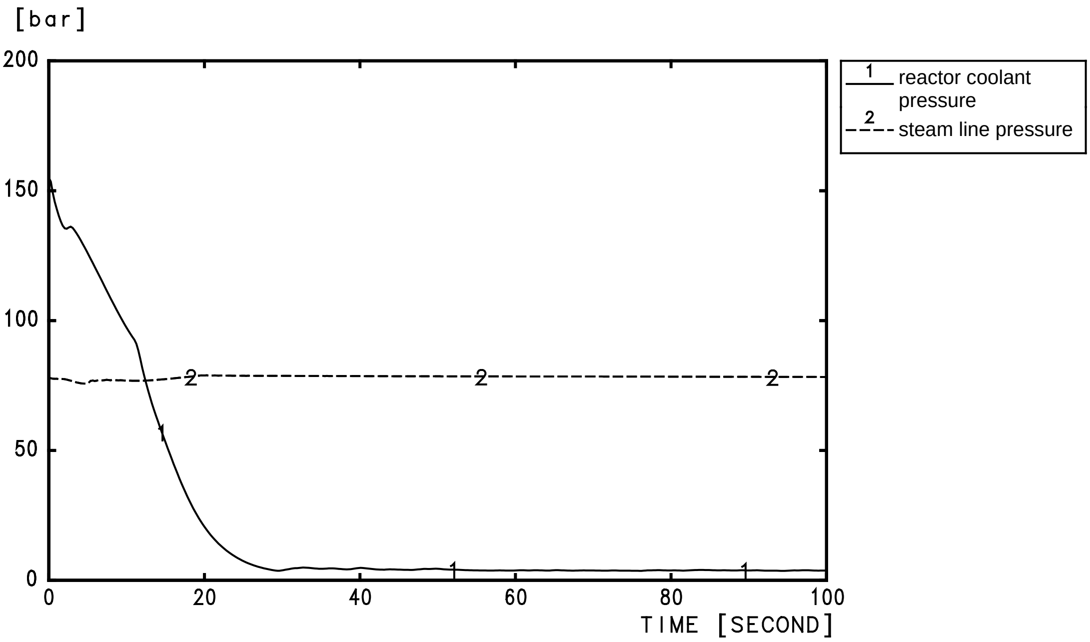
SECTION 16.4.1 - FIGURE A 8: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – LIQUID MASS IN THE HOT ASSEMBLY



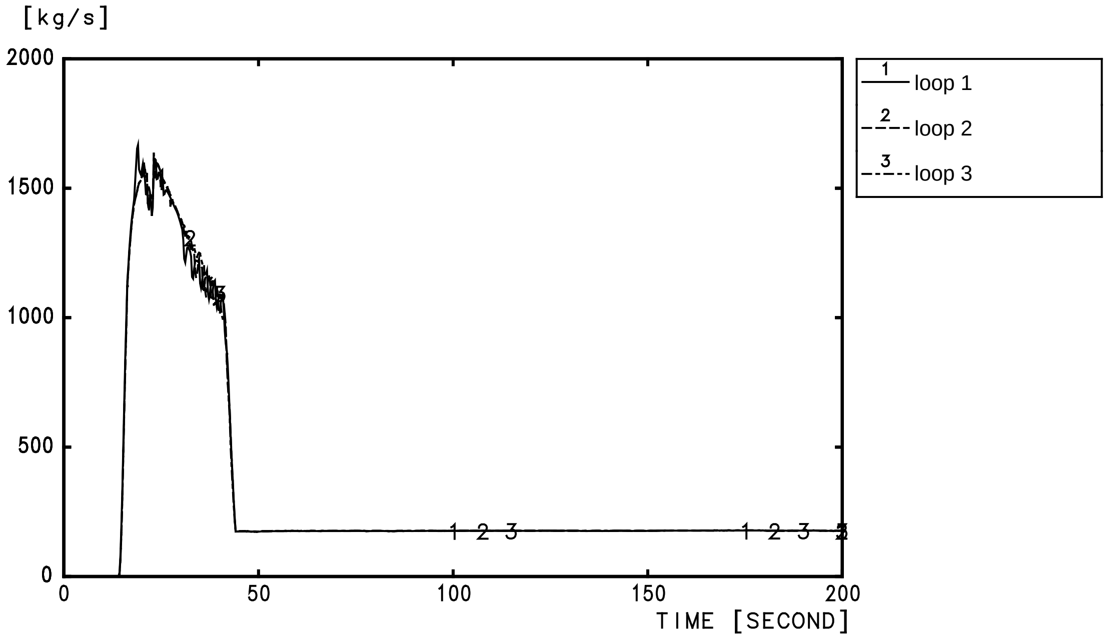
SECTION 16.4.1 - FIGURE A 9: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – LIQUID MASS IN THE CORE



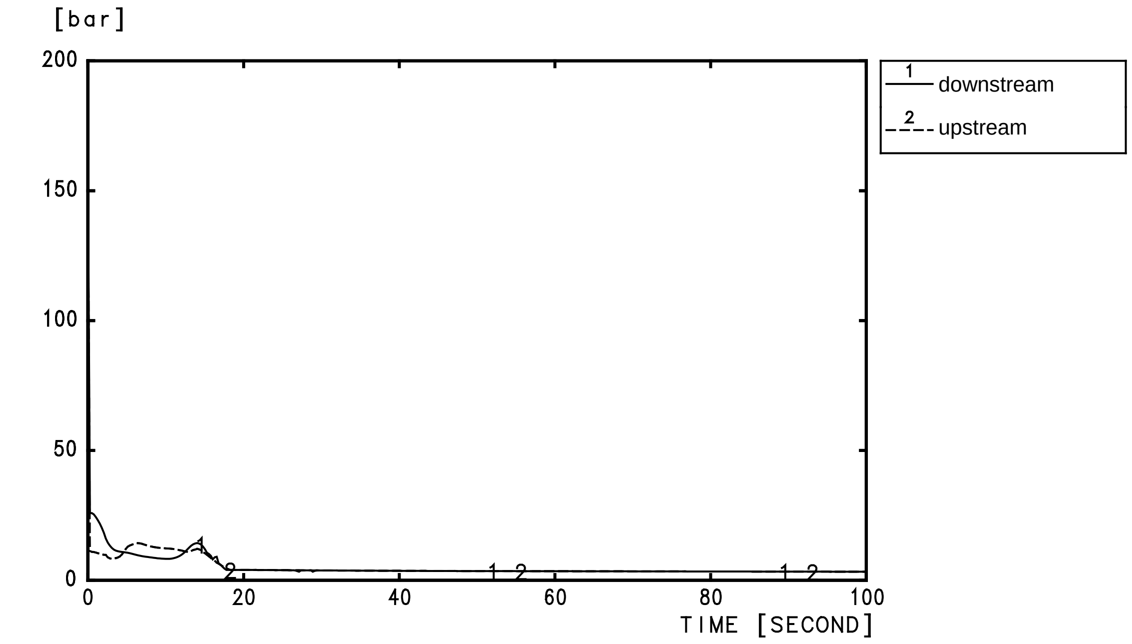
SECTION 16.4.1 - FIGURE A 10: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL MASS IN THE PRIMARY SYSTEM



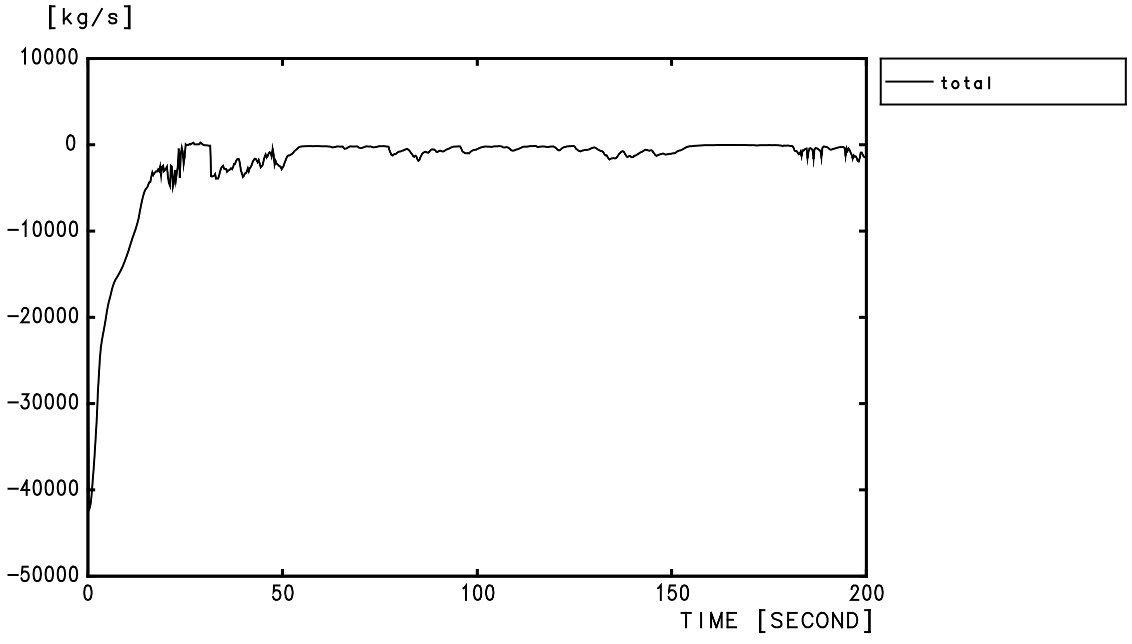
SECTION 16.4.1 - FIGURE A 11: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – REACTOR COOLANT PRESSURE AND STEAM LINE PRESSURE



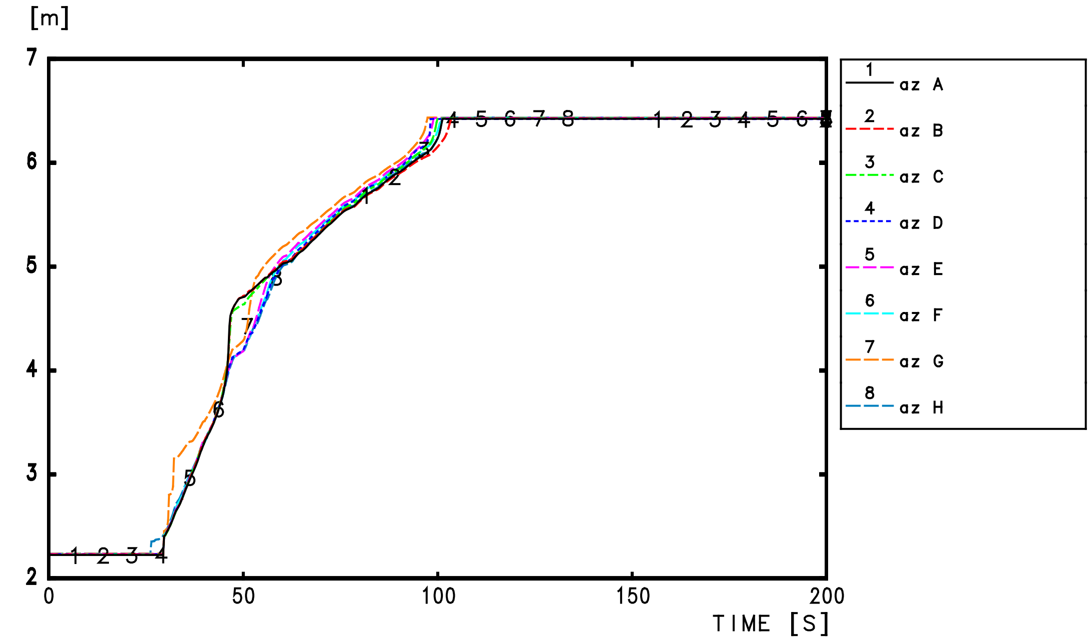
SECTION 16.4.1 - FIGURE A 12: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL RIS [SIS] FLOW RATE (ACCUMULATORS AND SI)



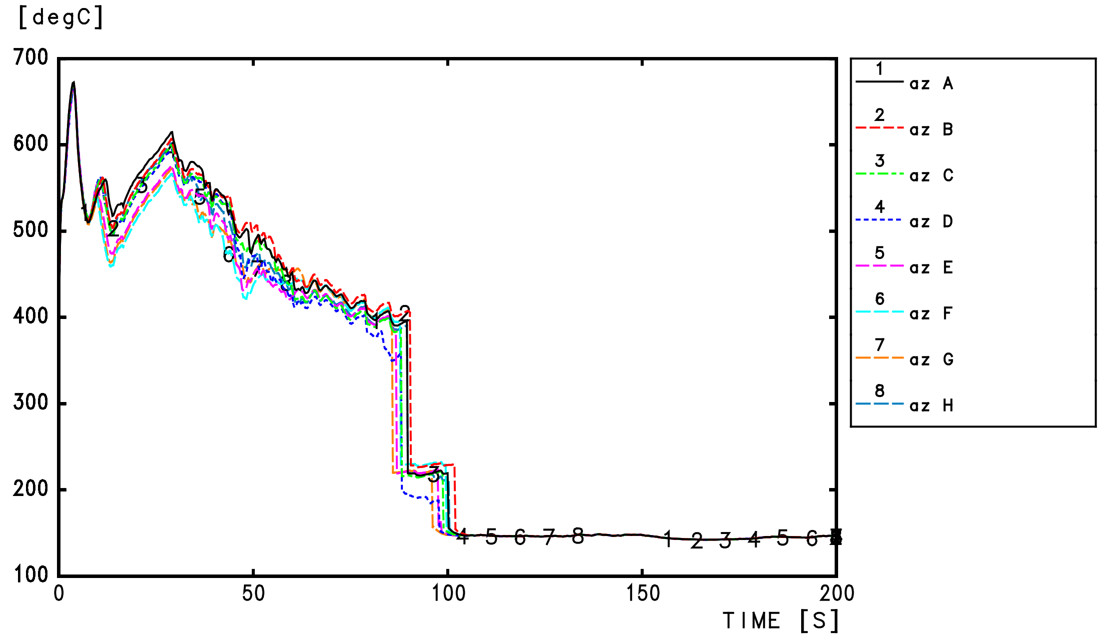
SECTION 16.4.1 - FIGURE A 13: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – PRESSURE AT THE BREAK



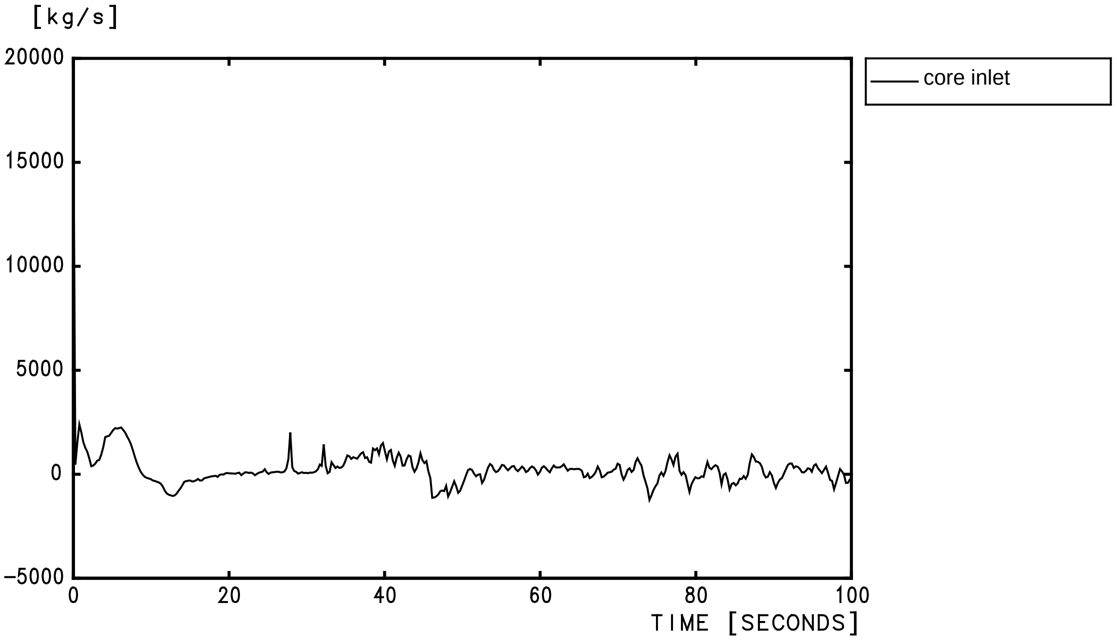
SECTION 16.4.1 - FIGURE A 14: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE BREAK



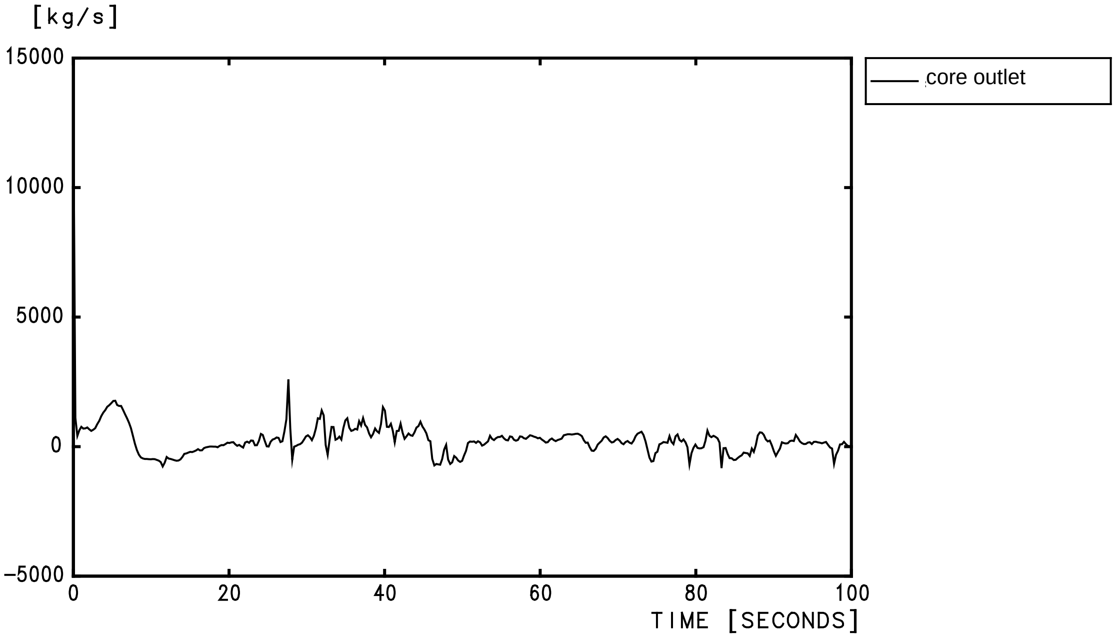
SECTION 16.4.1 - FIGURE A 15: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – QUENCH FRONT LEVEL IN HOT ASSEMBLIES



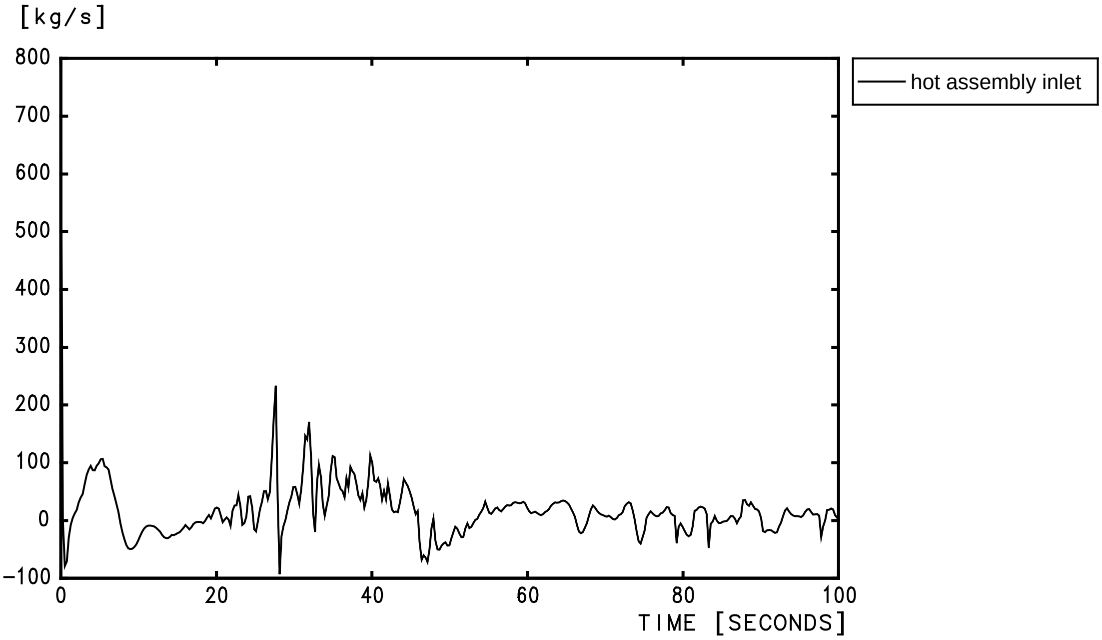
SECTION 16.4.1 - FIGURE A 16: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – MAXIMUM CLAD TEMPERATURES FOR HOT ASSEMBLIES



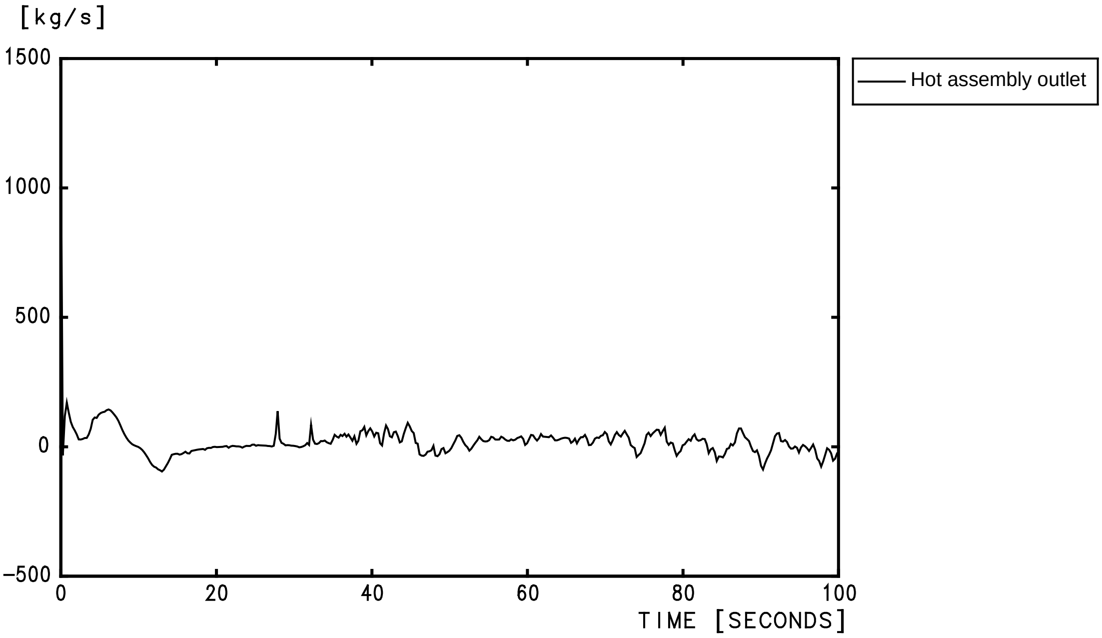
SECTION 16.4.1 - FIGURE A 17: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE CORE INLET



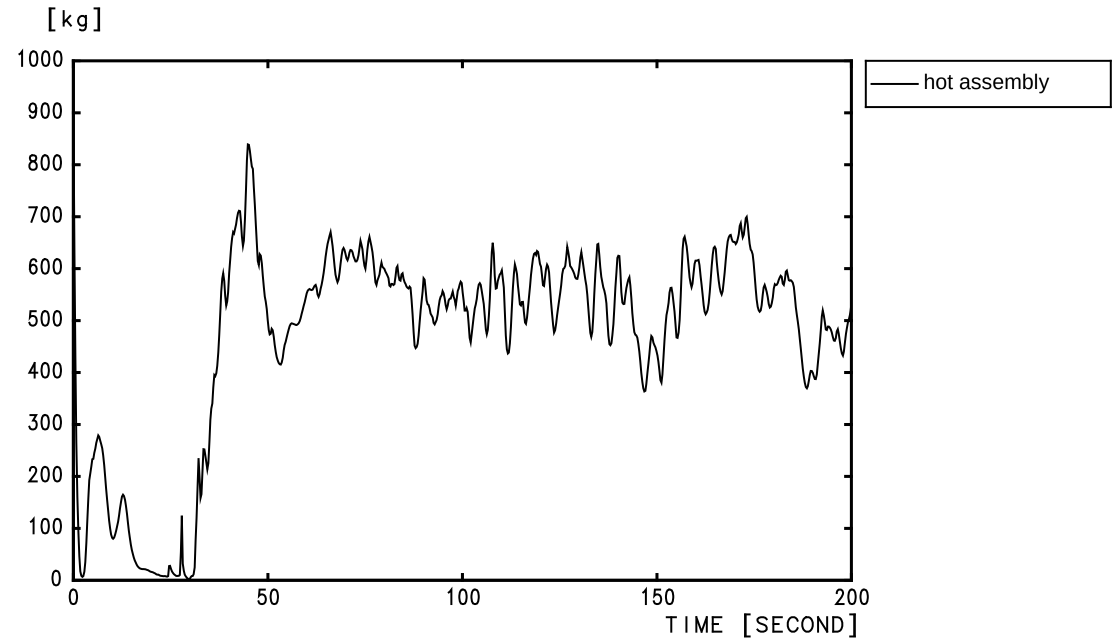
SECTION 16.4.1 - FIGURE A 18: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE CORE OUTLET



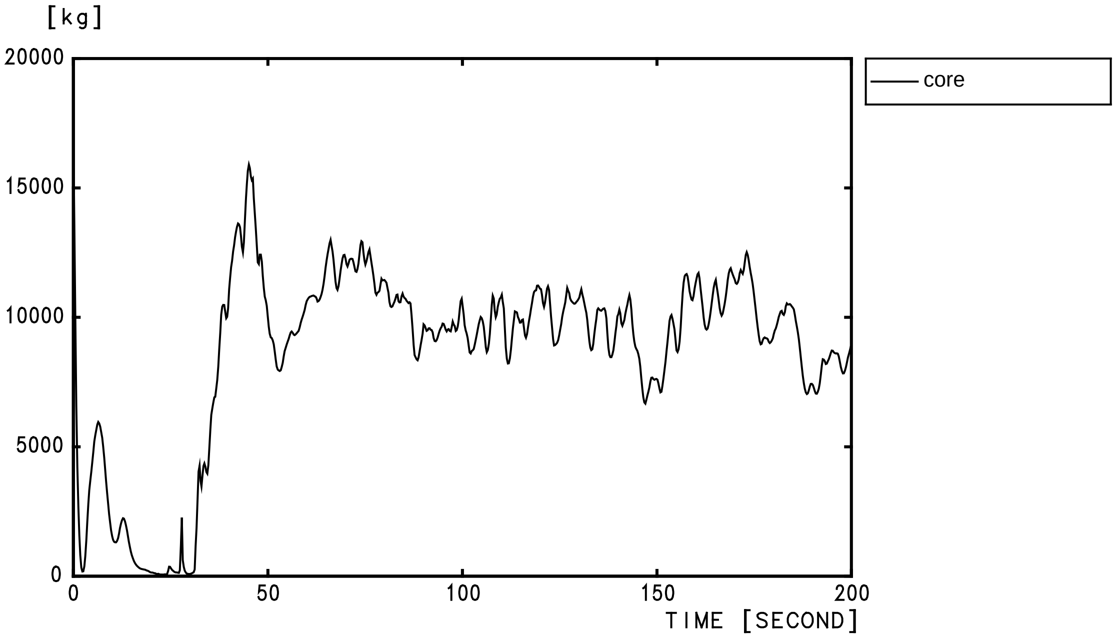
SECTION 16.4.1 - FIGURE A 19: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE HOT ASSEMBLY INLET



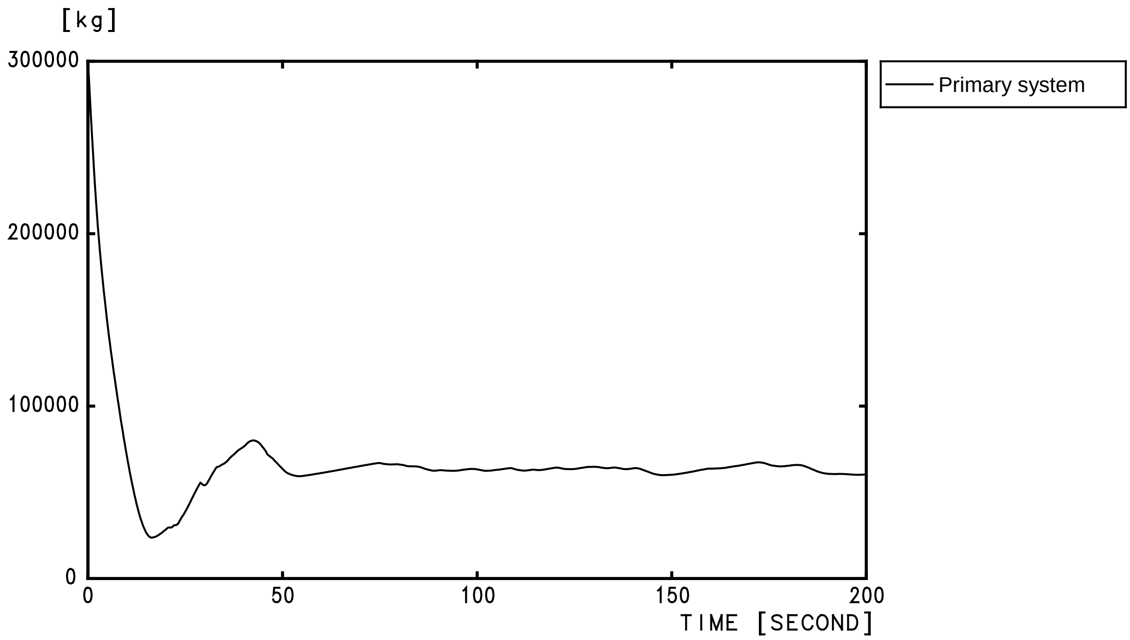
SECTION 16.4.1 - FIGURE A 20: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE HOT ASSEMBLY OUTLET



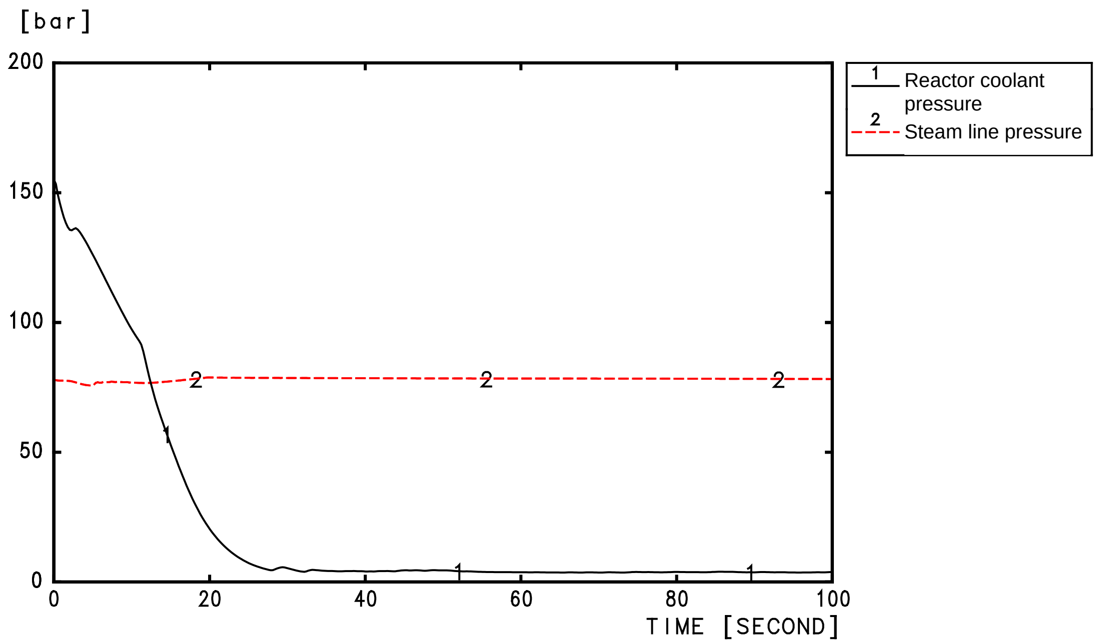
SECTION 16.4.1 - FIGURE A 21: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – LIQUID MASS IN THE HOT ASSEMBLY



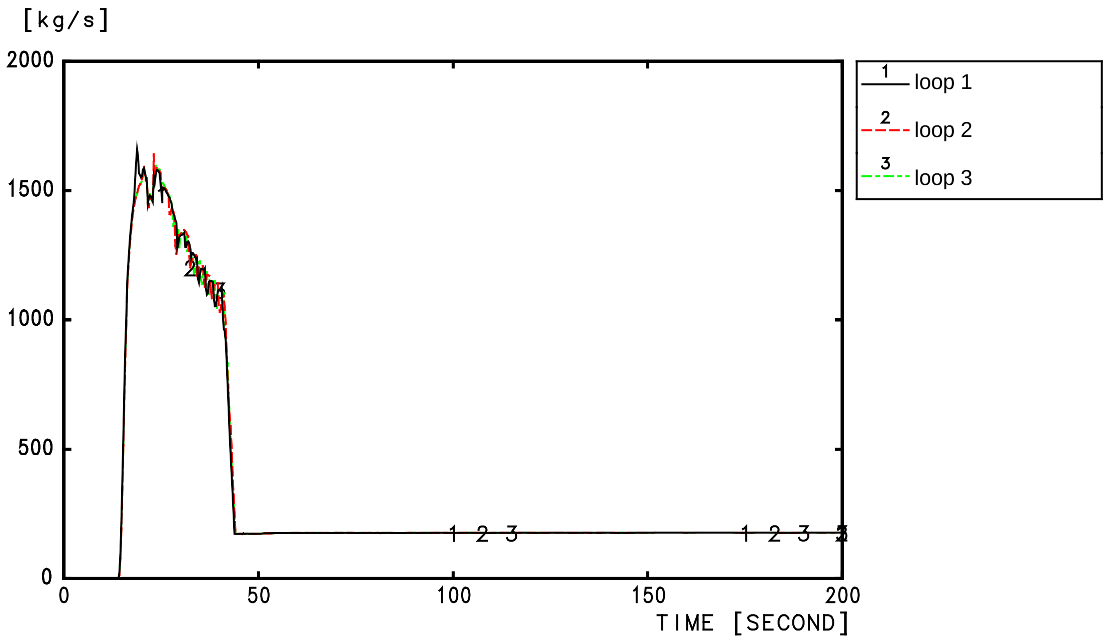
SECTION 16.4.1 - FIGURE A 22: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – LIQUID MASS IN THE CORE



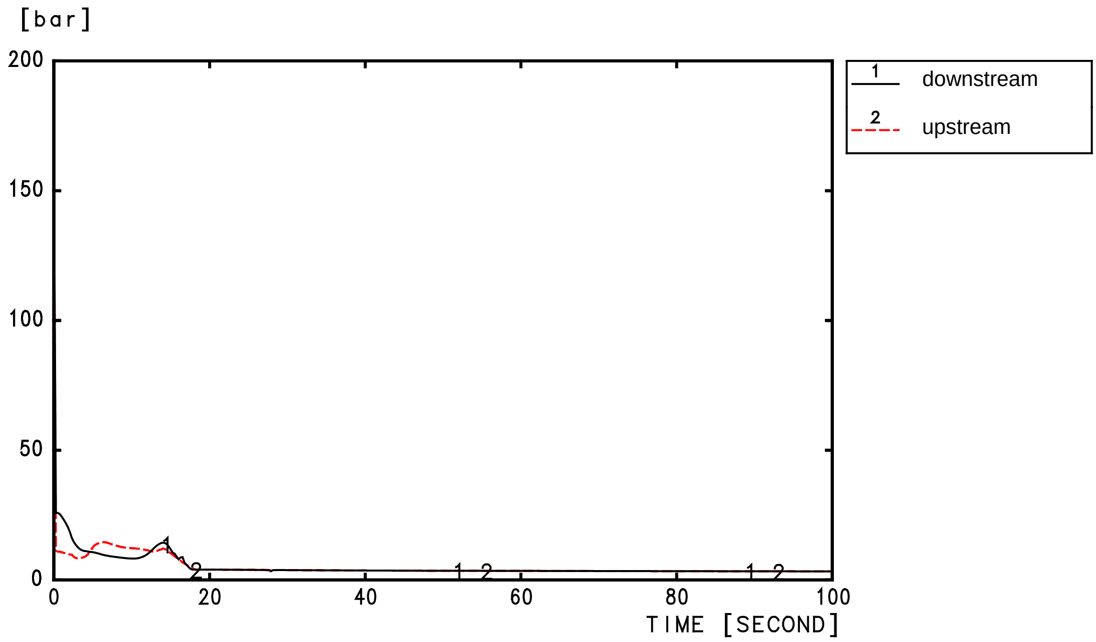
SECTION 16.4.1 - FIGURE A 23: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL MASS IN THE PRIMARY SYSTEM



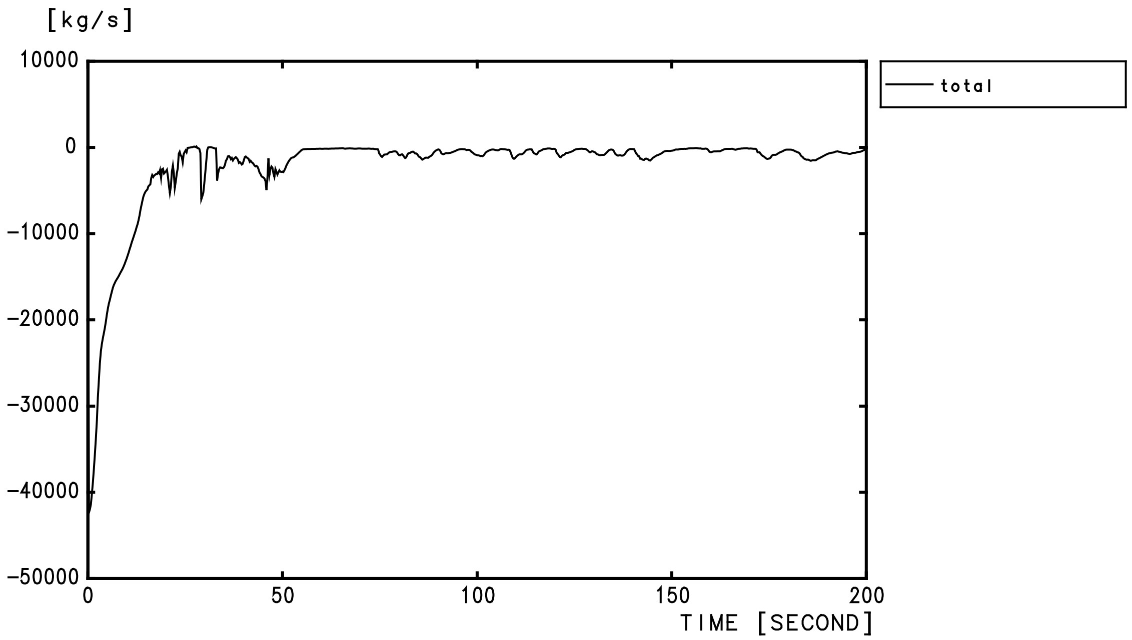
SECTION 16.4.1 - FIGURE A 24: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – REACTOR COOLANT PRESSURE AND STEAM LINE PRESSURE



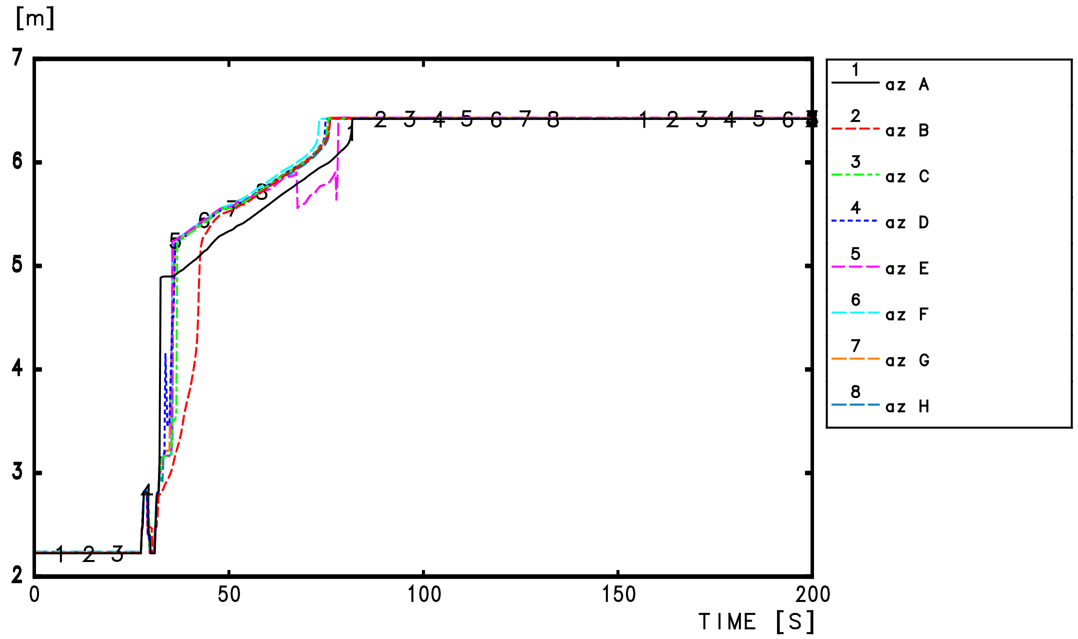
SECTION 16.4.1 - FIGURE A 25: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL RIS [SIS] FLOW RATE (ACCUMULATORS AND SI)



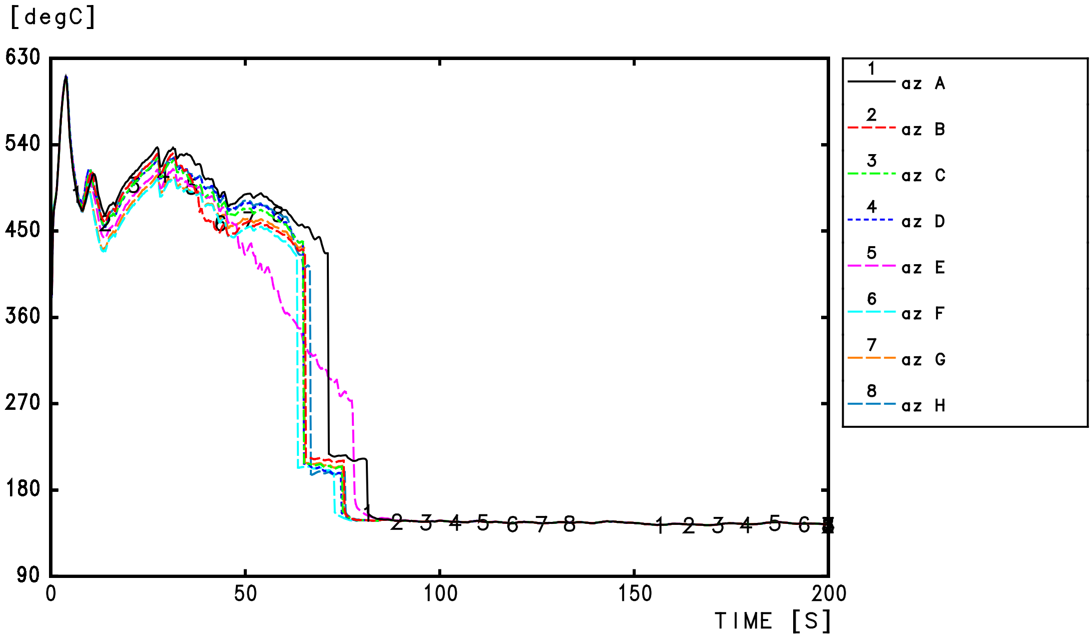
SECTION 16.4.1 - FIGURE A 26: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – PRESSURE AT THE BREAK



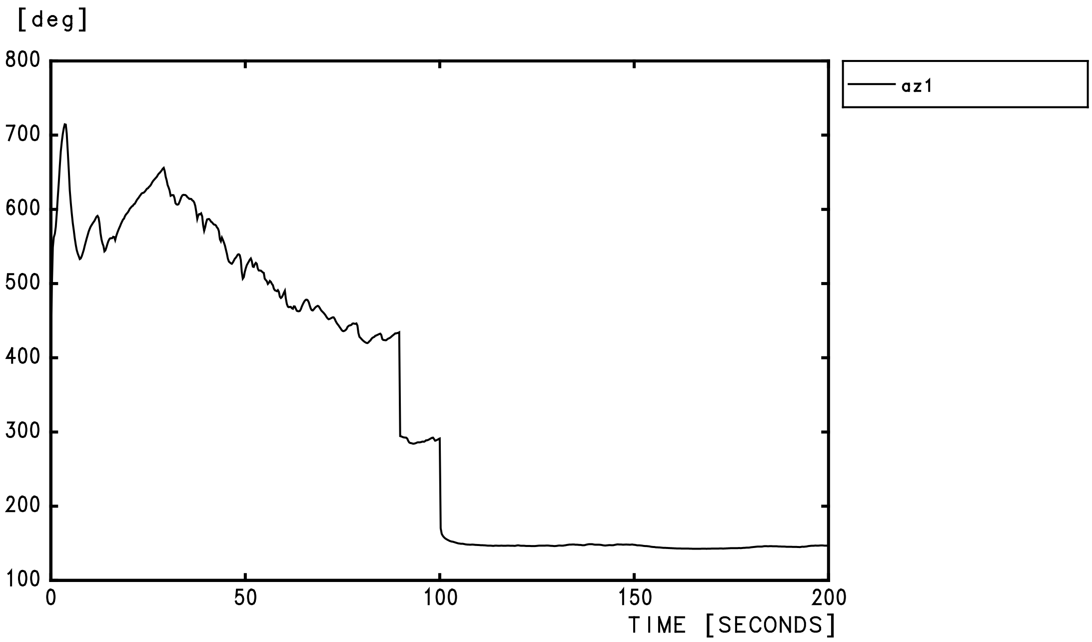
SECTION 16.4.1 - FIGURE A 27: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE BREAK



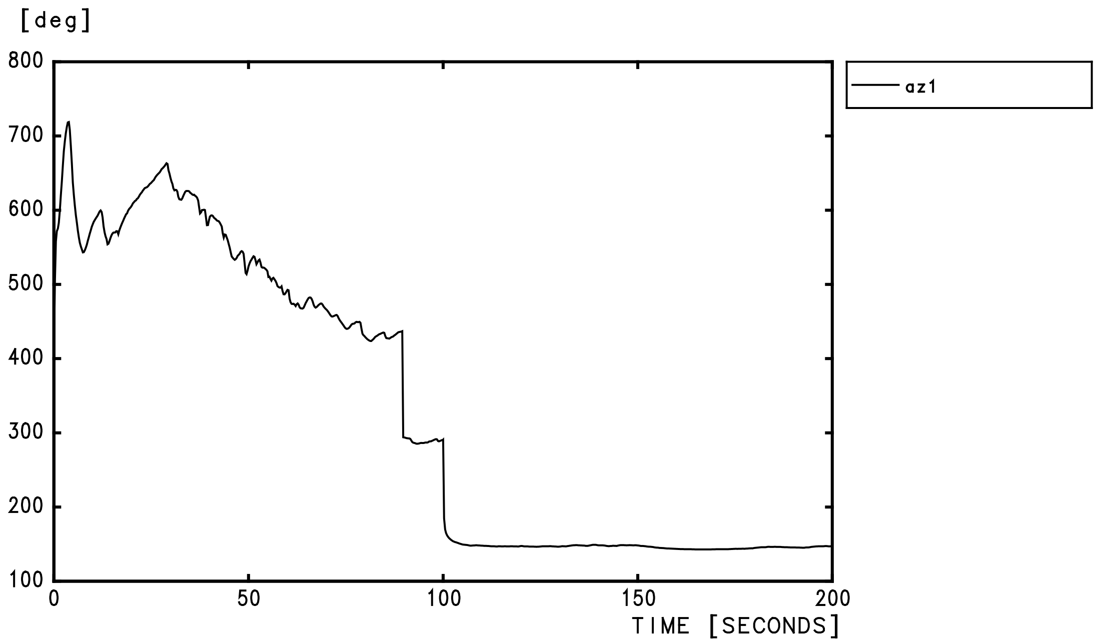
SECTION 16.4.1 - FIGURE A 28: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – QUENCH FRONT LEVEL IN HOT ASSEMBLIES



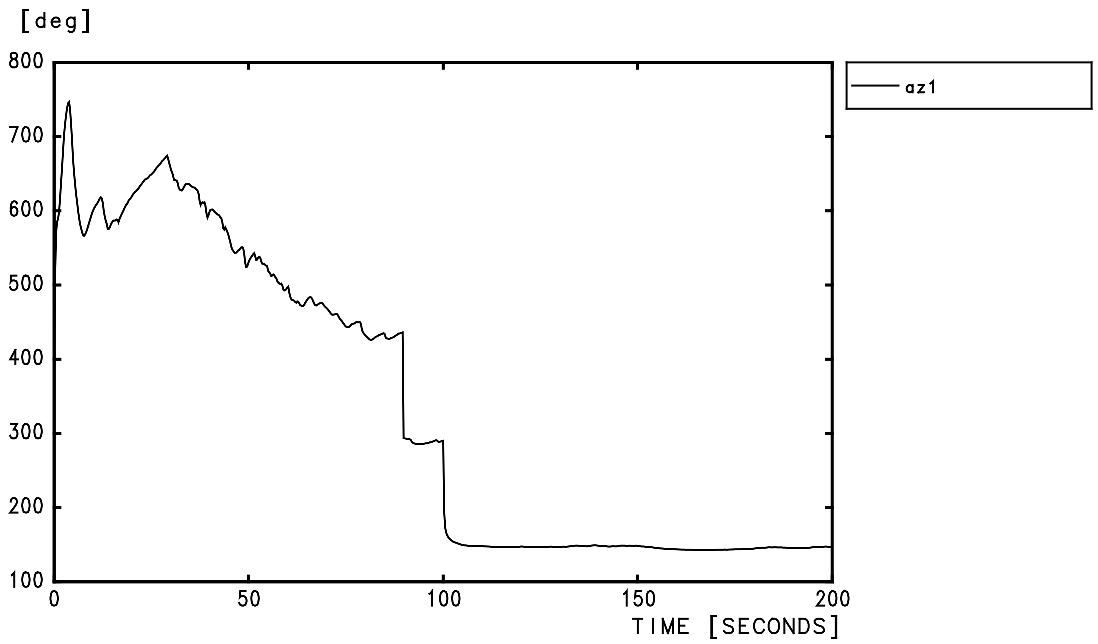
SECTION 16.4.1 - FIGURE A 29: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – MAXIMUM CLAD TEMPERATURES FOR HOT ASSEMBLIES



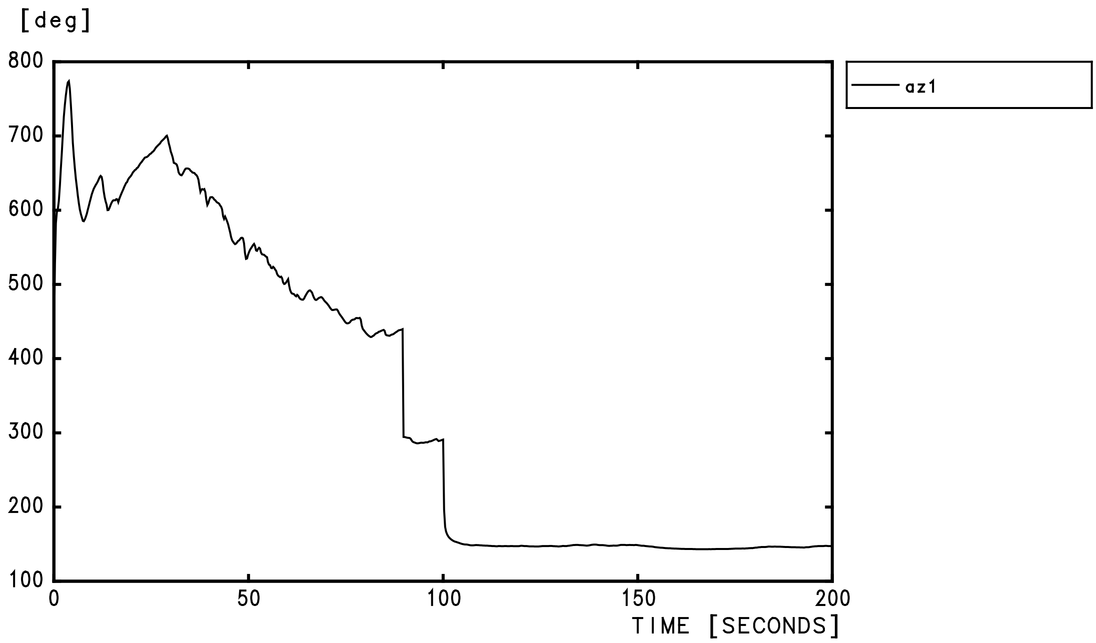
SECTION 16.4.1 - FIGURE A 30: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – BOL INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



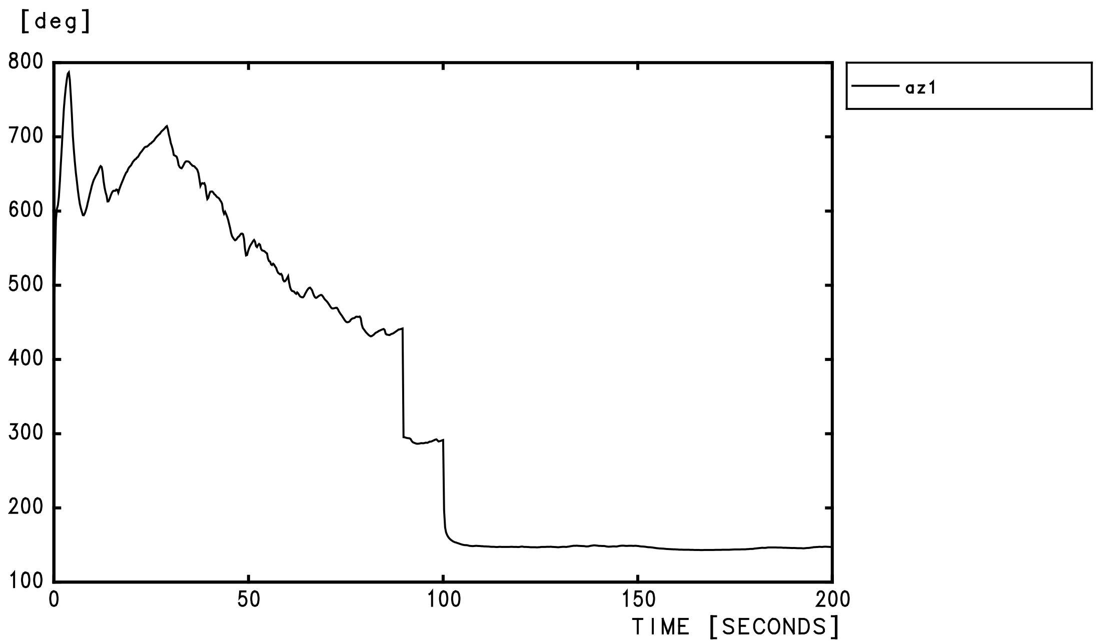
SECTION 16.4.1 - FIGURE A 31: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC1 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



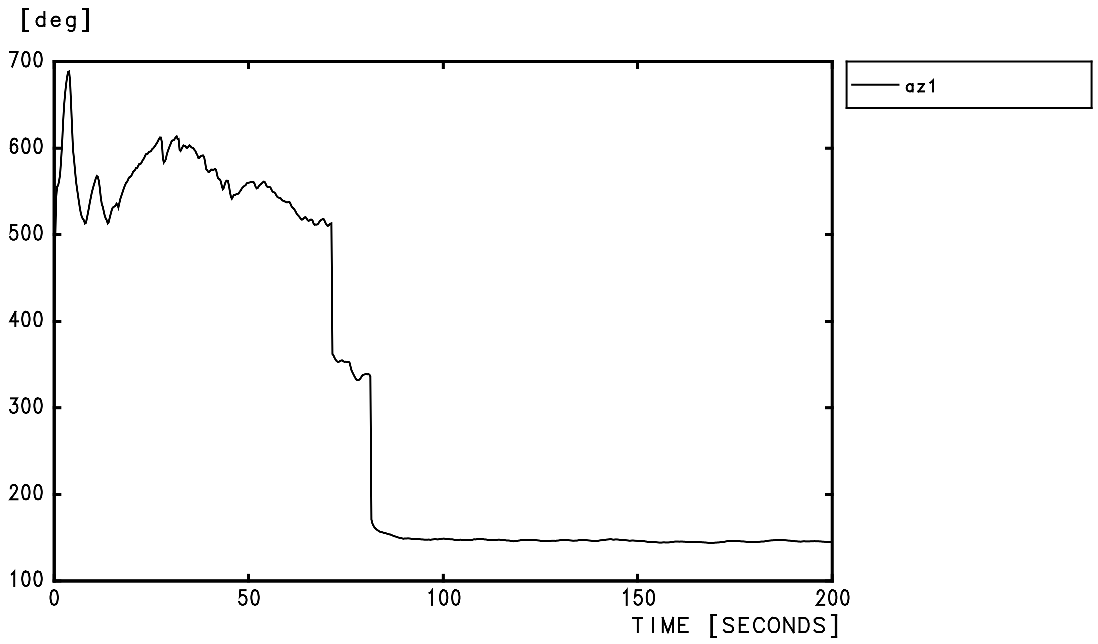
SECTION 16.4.1 - FIGURE A 32: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC2 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



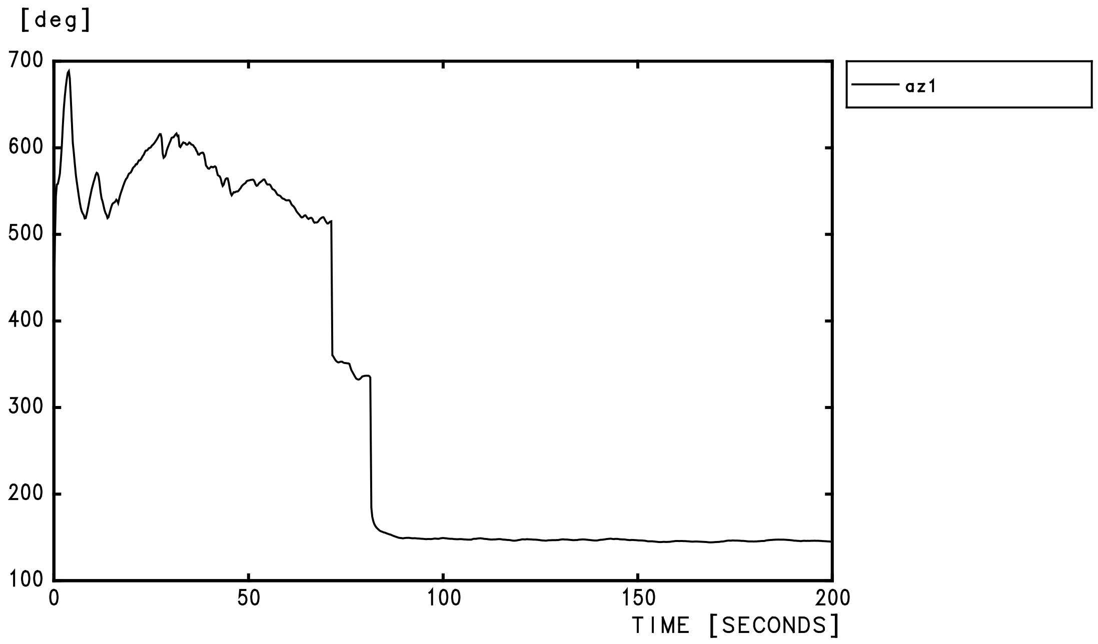
SECTION 16.4.1 - FIGURE A 33: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC3 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



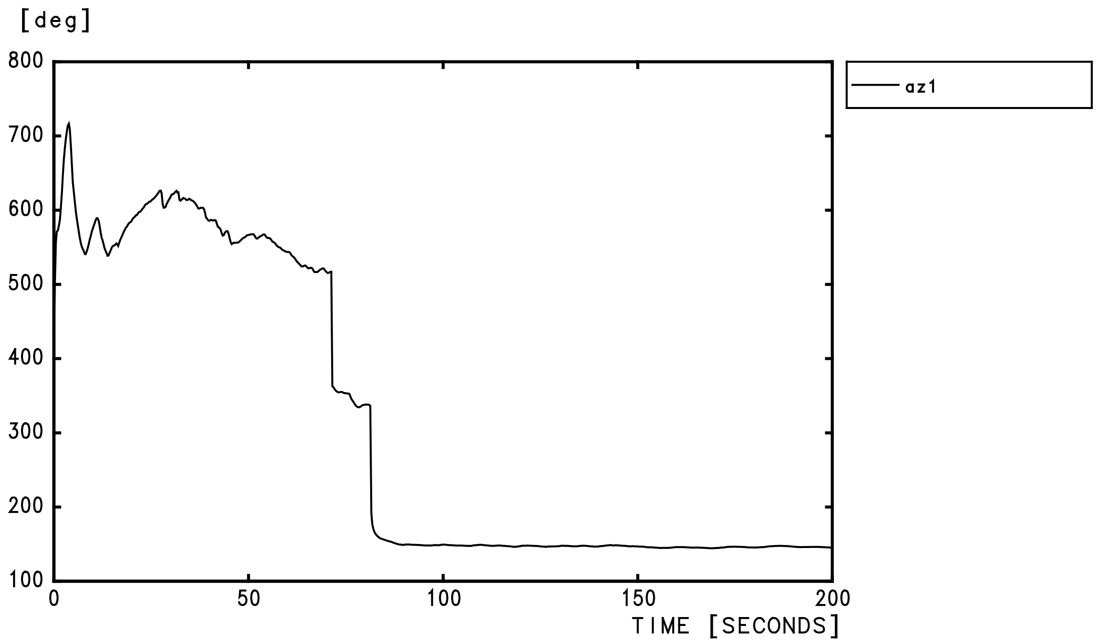
SECTION 16.4.1 - FIGURE A 34: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC4 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



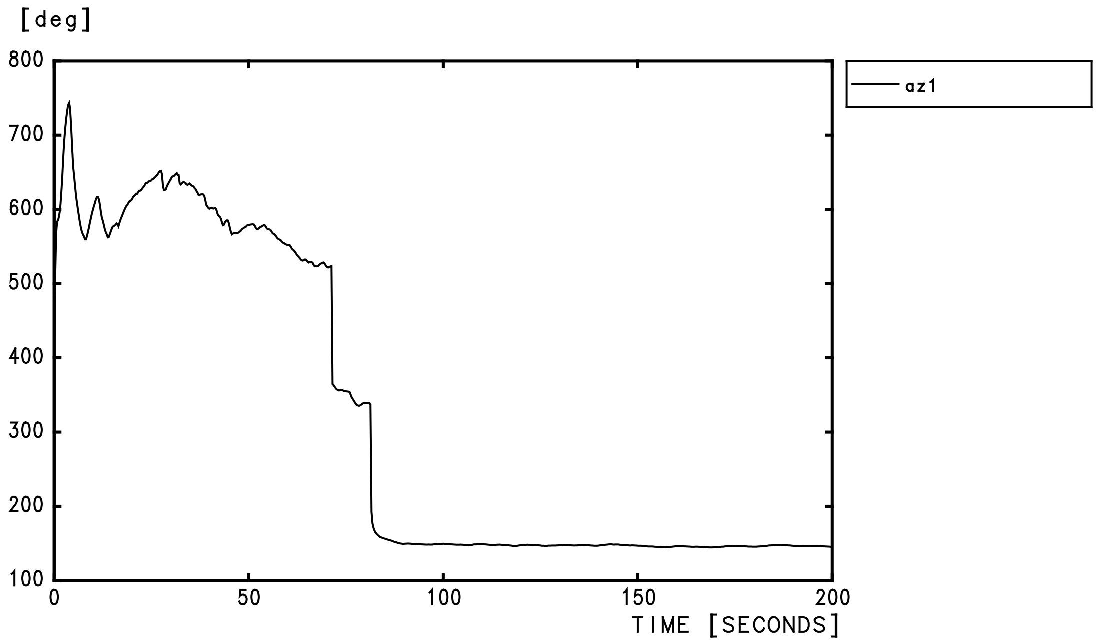
SECTION 16.4.1 - FIGURE A 35: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – BOL INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



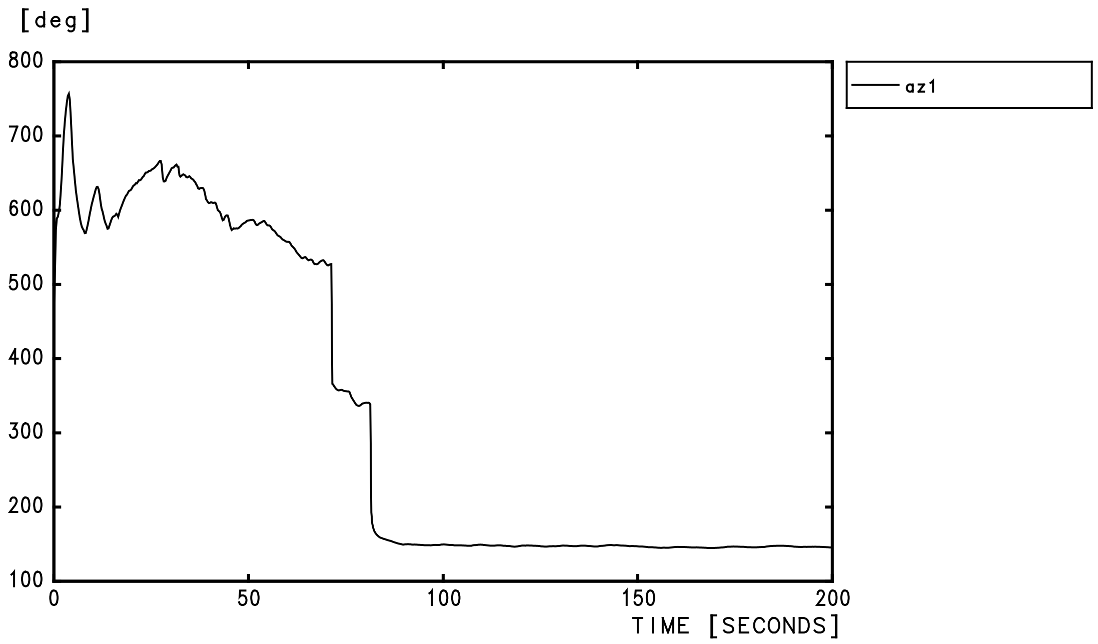
SECTION 16.4.1 - FIGURE A 36: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC1 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



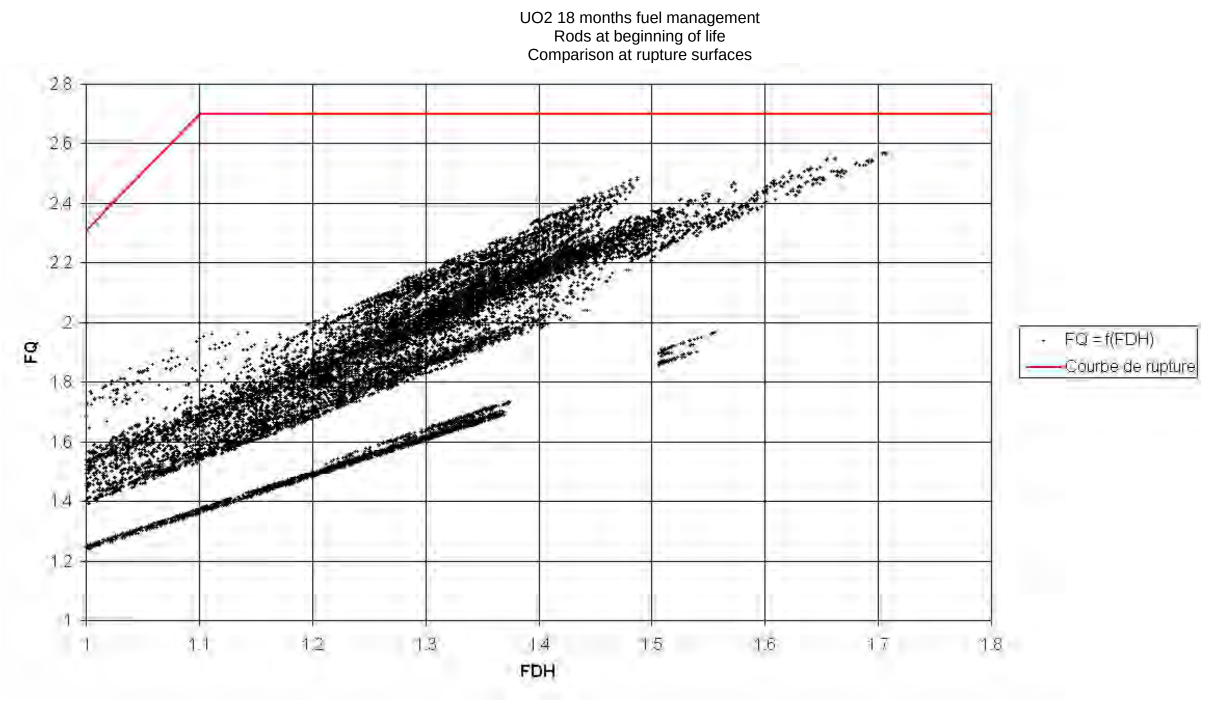
SECTION 16.4.1 - FIGURE A 37: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC2 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



SECTION 16.4.1 - FIGURE A 38: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC3 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE

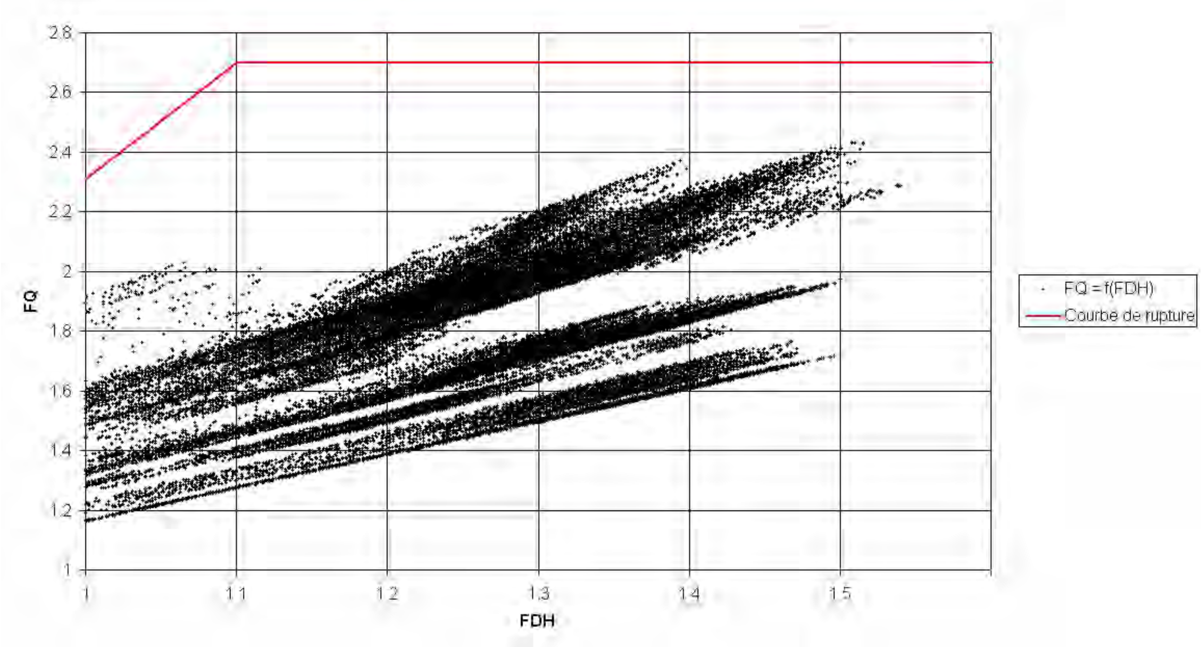


SECTION 16.4.1 - FIGURE A 39: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC4 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



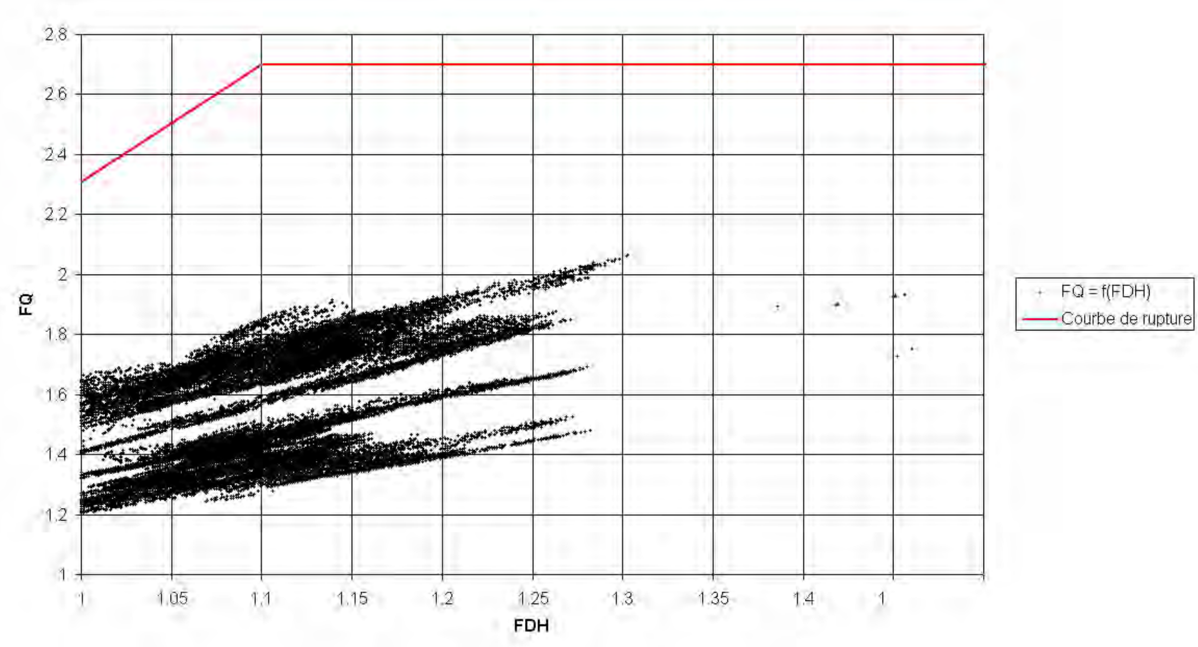
SECTION 16.4.1 - FIGURE A 40: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – RUPTURE CURVE – BOL

UO2 18 months fuel management
Rods at end of cycle 1
Comparison at rupture surfaces

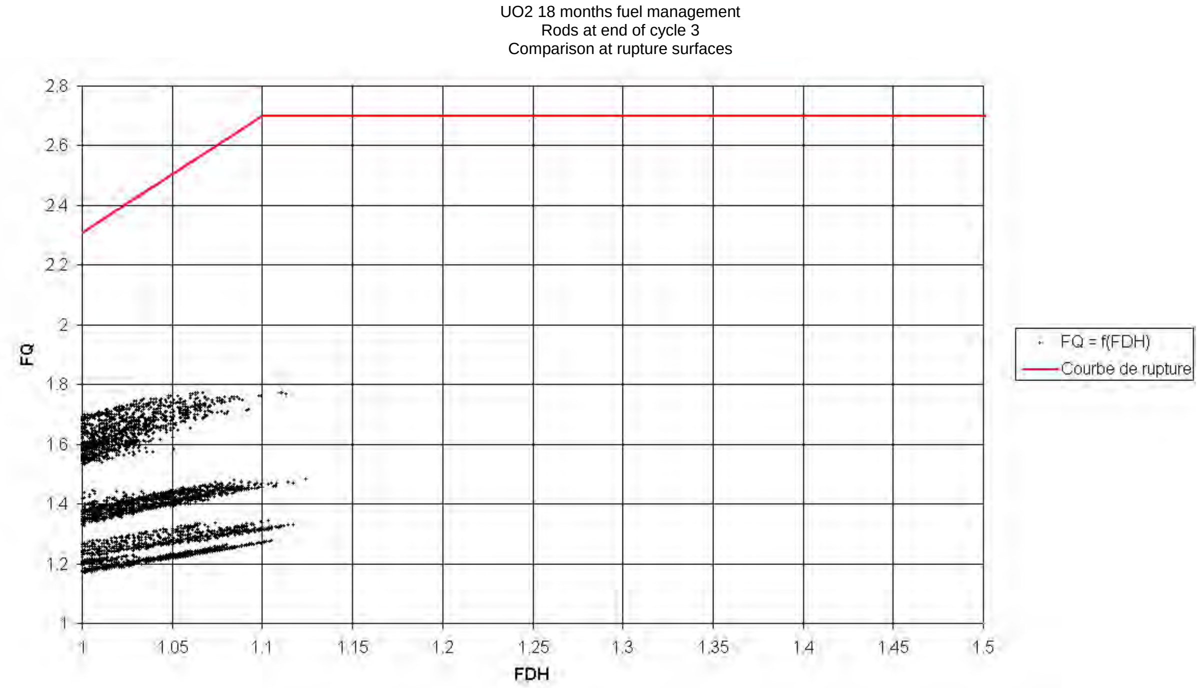


SECTION 16.4.1 - FIGURE A 41: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – RUPTURE CURVE – EOC1

UO2 18 months fuel management
Rods at end of cycle 2
Comparison at rupture surfaces



SECTION 16.4.1 - FIGURE A 42: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – TC – RUPTURE CURVE – EOC2



SECTION 16.4.1 - FIGURE A 43: IN-OUT UO2 18 MONTHS FUEL MANAGEMENT – TC – RUPTURE CURVE – EOC3

SECTION 16.4.1 - TABLE B 1: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – FUEL CALCULATION POINTS

Core cycle time	Rod burn-up
Beginning of life (BOL)	150 MWd/tU
End of Cycle 1 (EOC1)	32,300 MWd/tU
End of Cycle 2 (EOC2)	57,900 MWd/tU
End of Cycle 3 (EOC3)	71,800 MWd/tU

SECTION 16.4.1 - TABLE B 2: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – PROPERTIES OF HOT ASSEMBLIES

Hot assembly	FΔH	FQ	BU
HA1	1.39	2.22	EOC1
HA2	1.19	1.85	EOC2

SECTION 16.4.1 - TABLE B 3: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – MAIN EVENTS OF THE TRANSIENT

Event	Time
RT signal	3.6 s
SI signal	7.1 s
Start of accumulator injection	14.1 s
Start of safety injection system injection	23.0 s
Start of reflood	24.9 s
End of reflood	113.9 s
Duration of reflood	88.9 s
End of accumulator injection	44.0 s

SECTION 16.4.1 - TABLE B 4: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – PEAK CLAD TEMPERATURES

	T1	T2	T3
Time (s)	3.9	24.5	51.4
Mesh	11	11	11
Value (°C)	665	607	580

SECTION 16.4.1 - TABLE B 5: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – MAIN EVENTS OF THE TRANSIENT

Event	Time
RT signal	3.6 s
SI signal	7.12 s
Start of accumulator injection	14.1 s
Start of safety injection system injection	23.0 s
Start of reflood	23.9 s
End of reflood	97.7 s
Duration of reflood	73.8 s
End of accumulator injection	43.9 s

SECTION 16.4.1 - TABLE B 6: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – PEAK CLAD TEMPERATURES

	T1	T2	T3
Time (s)	4.0	30.1	53.5
Mesh	11	11	11
Value (°C)	617	548	508

SECTION 16.4.1 - TABLE B 7: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – PROPERTIES OF THE INDICATOR RODS

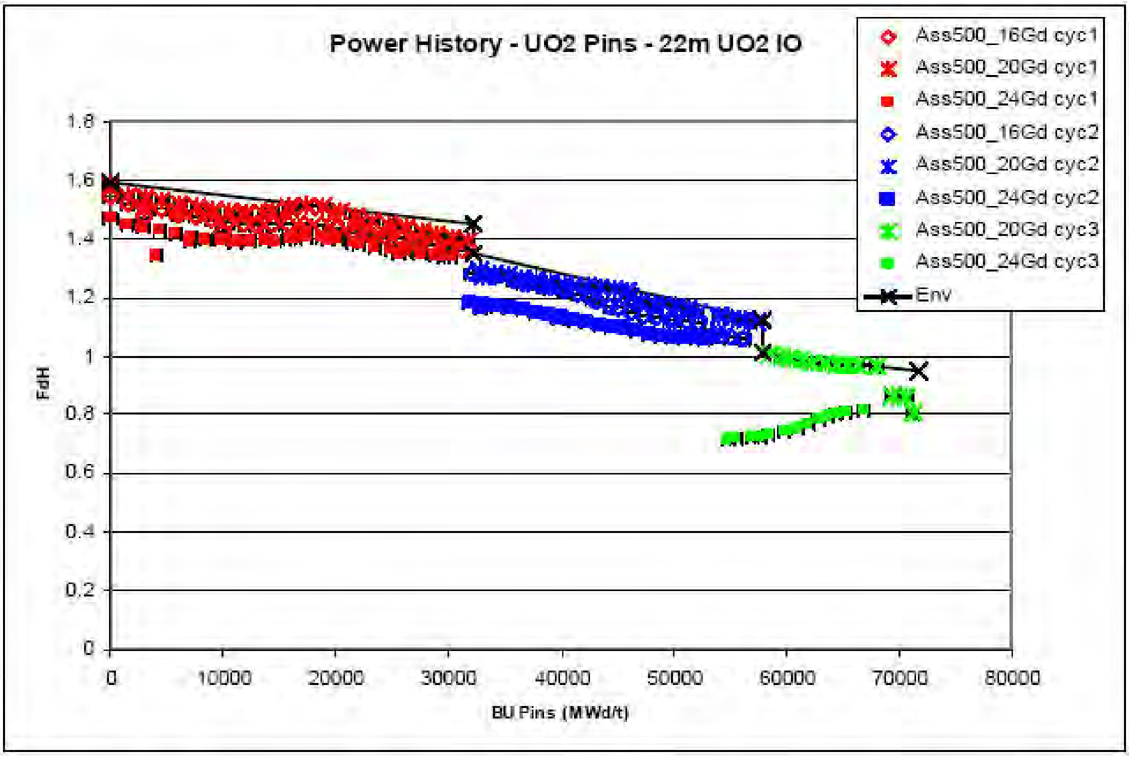
Burn-up	FQ	FΔH	Pressure in the gap (bar)	Maximum linear power density (W/cm)	Maximum initial pellet temperature (°C)
BOL	2.458	1.555	56.9	412	940
EOC1	2.458	1.555	71.8	412	944
EOC2	2.458	1.555	93.4	412	1045
EOC3	2.458	1.555	107.8	412	1133

SECTION 16.4.1 - TABLE B 8: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – RESULTS OBTAINED FOR INDICATOR RODS

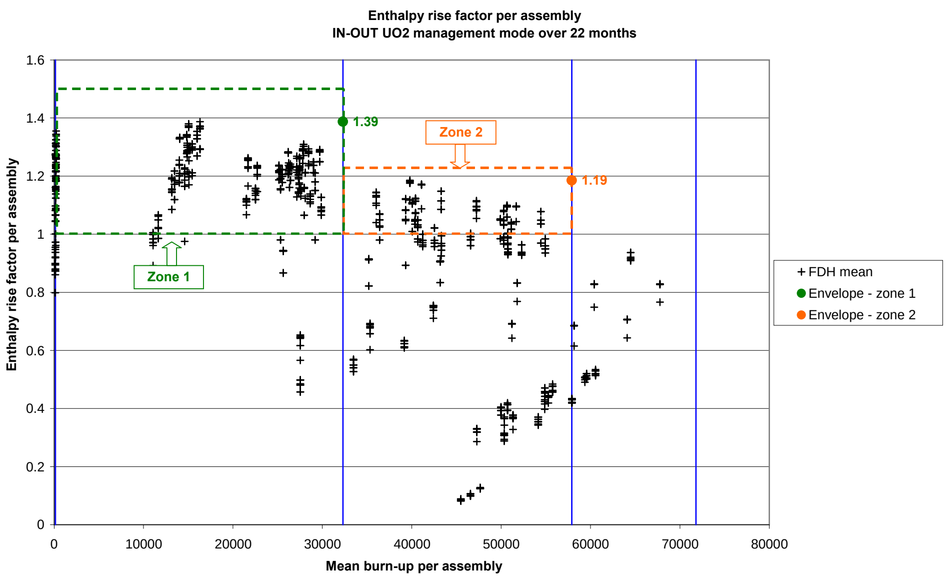
Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Rupture
BOL	692	633	544	0.36	NO
EOC1	691	638	547	0.35	NO
EOC2	725	654	556	0.39	NO
EOC3	750	679	569	0.49	NO

SECTION 16.4.1 - TABLE B 9: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – RESULTS OBTAINED FOR INDICATOR RODS

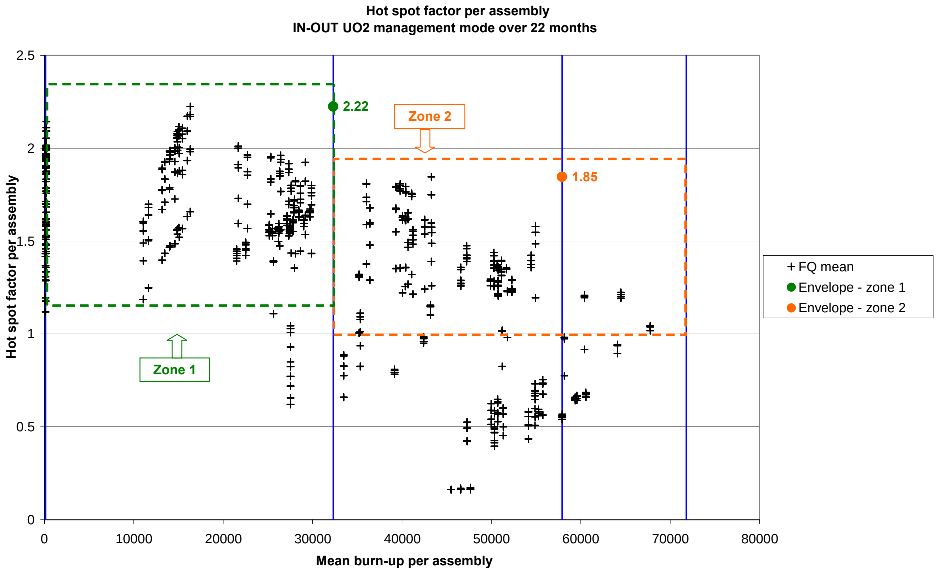
Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Rupture
BOL	691	612	575	0.34	NO
EOC1	691	618	578	0.33	NO
EOC2	729	633	586	0.36	NO
EOC3	758	657	600	0.40	NO



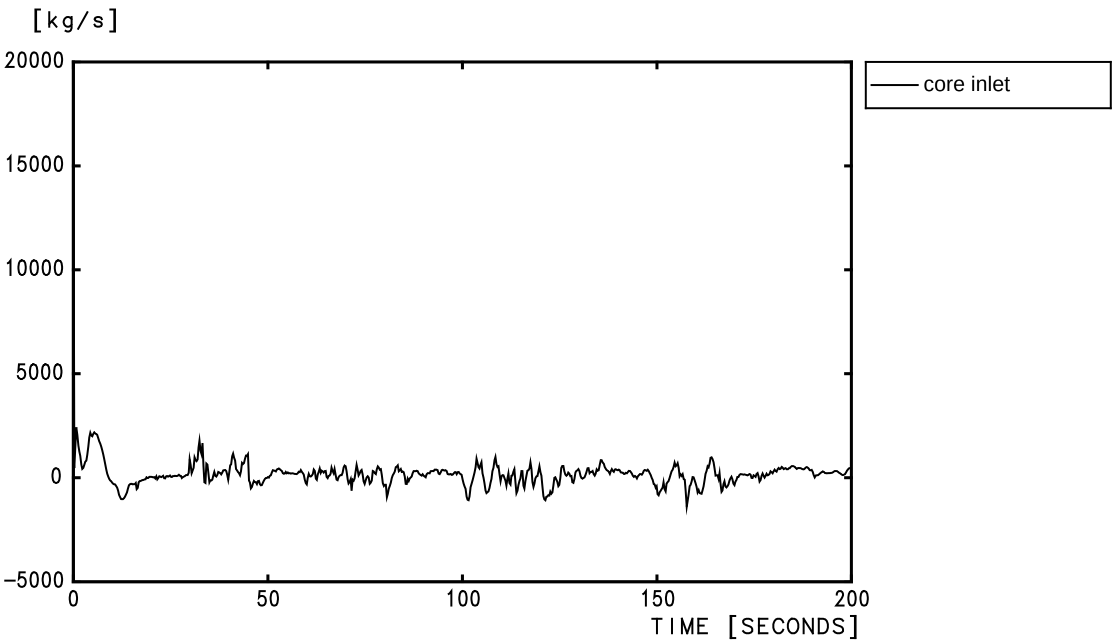
SECTION 16.4.1 - FIGURE B 1: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – POWER HISTORY OF UO_2 RODS



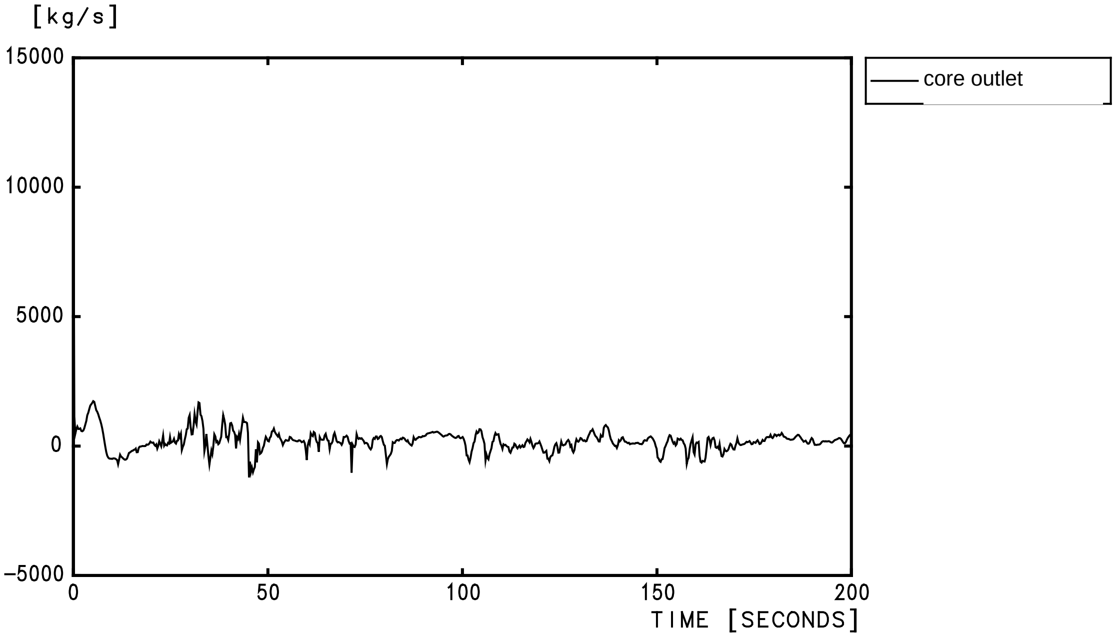
SECTION 16.4.1 - FIGURE B 2: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – FΔH FOR ASSEMBLIES AS A FUNCTION OF BURN-UP



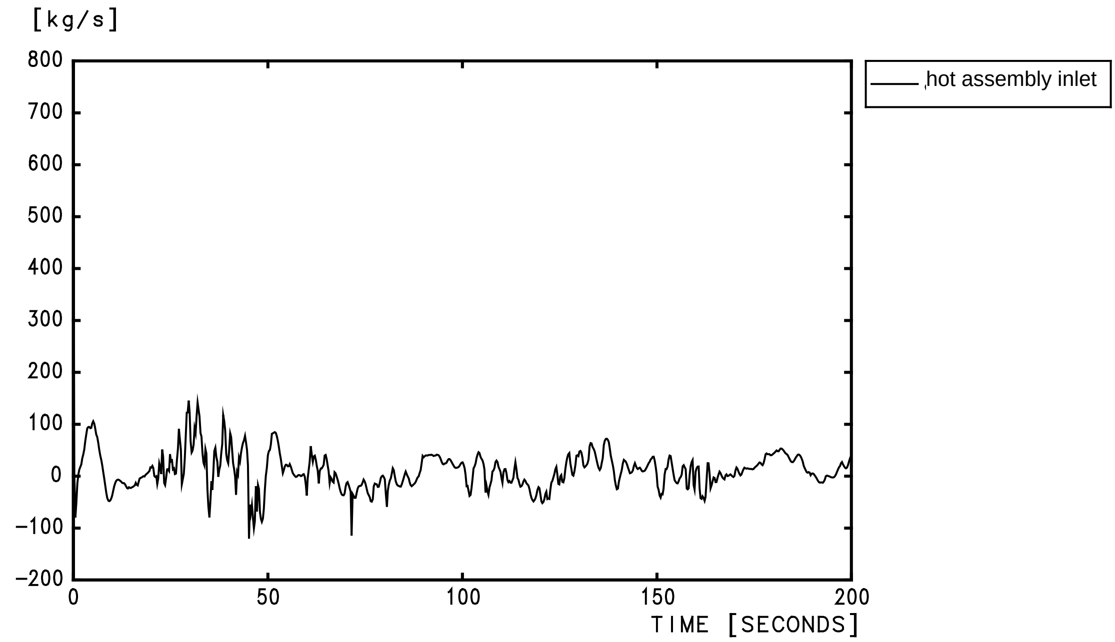
SECTION 16.4.1 - FIGURE B 3: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – FQ FOR ASSEMBLIES AS A FUNCTION OF BURN-UP



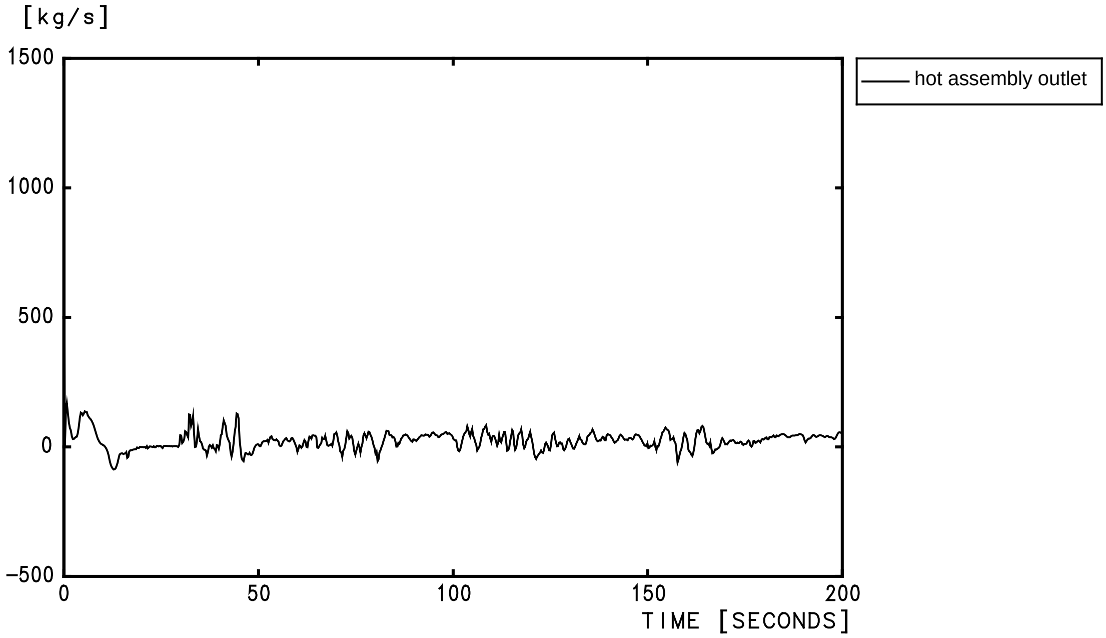
SECTION 16.4.1 - FIGURE B 4: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE CORE INLET



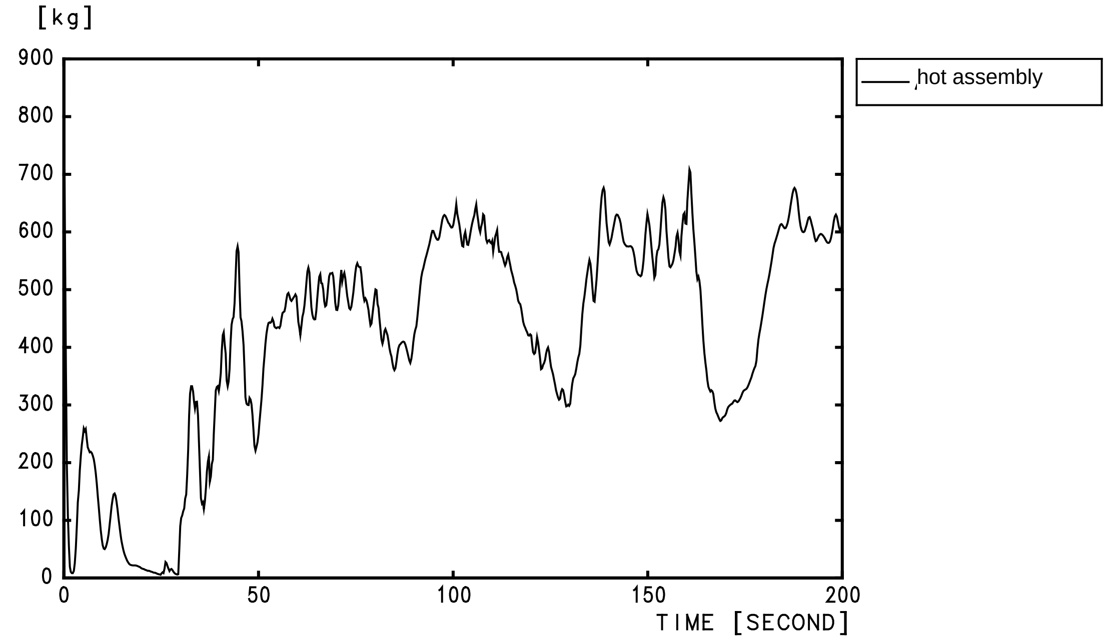
SECTION 16.4.1 - FIGURE B 5: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE CORE OUTLET



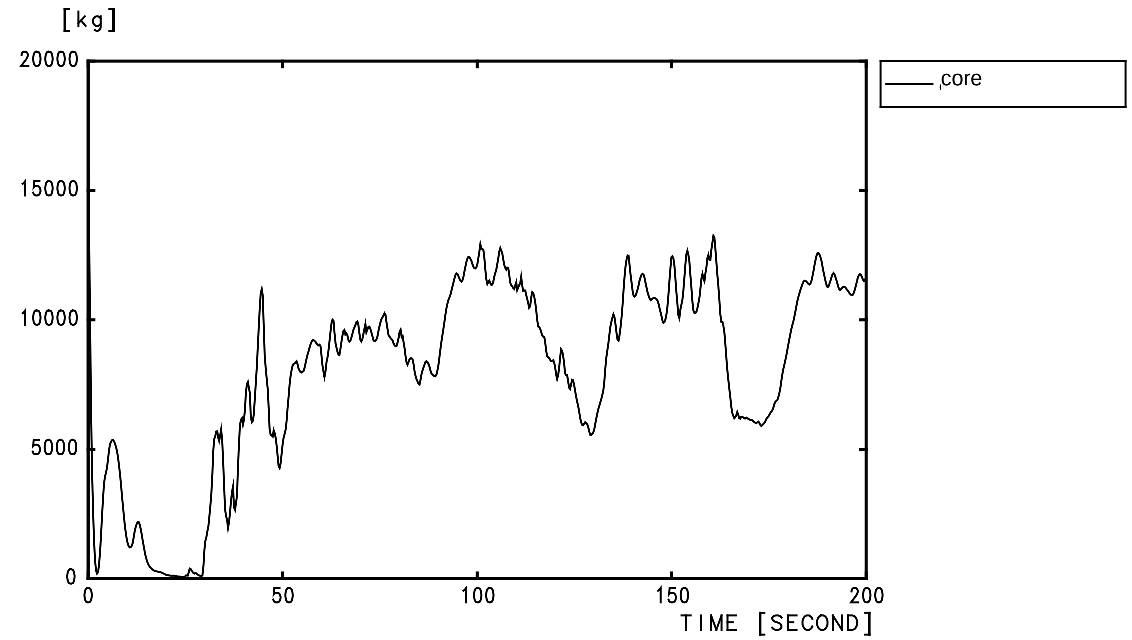
SECTION 16.4.1 - FIGURE B 6: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE HOT ASSEMBLY INLET



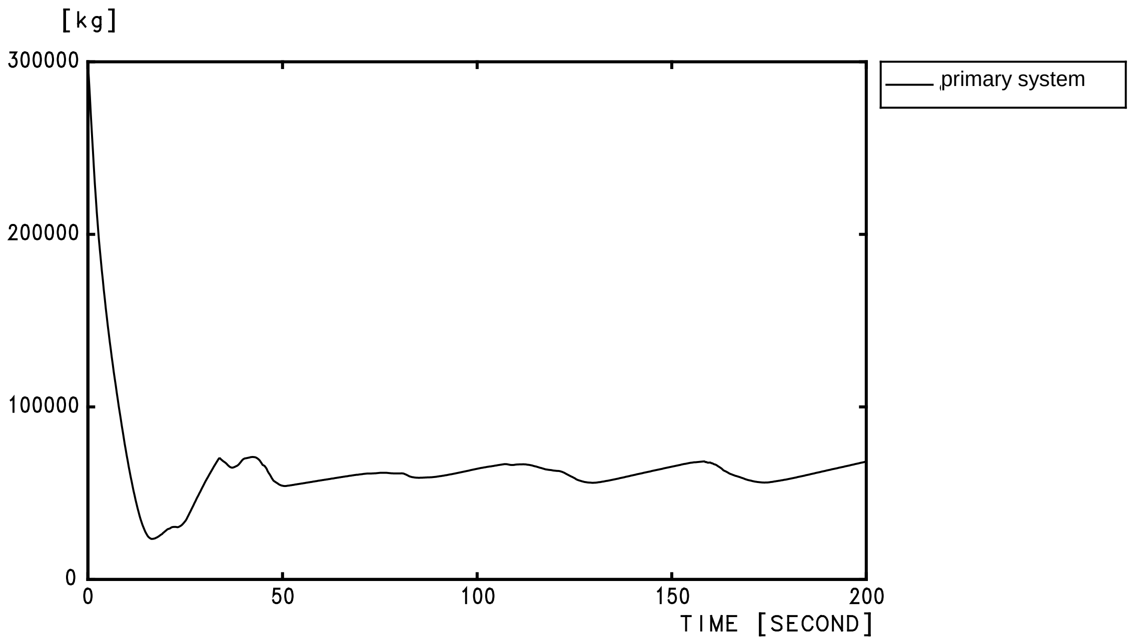
SECTION 16.4.1 - FIGURE B 7: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE HOT ASSEMBLY OUTLET



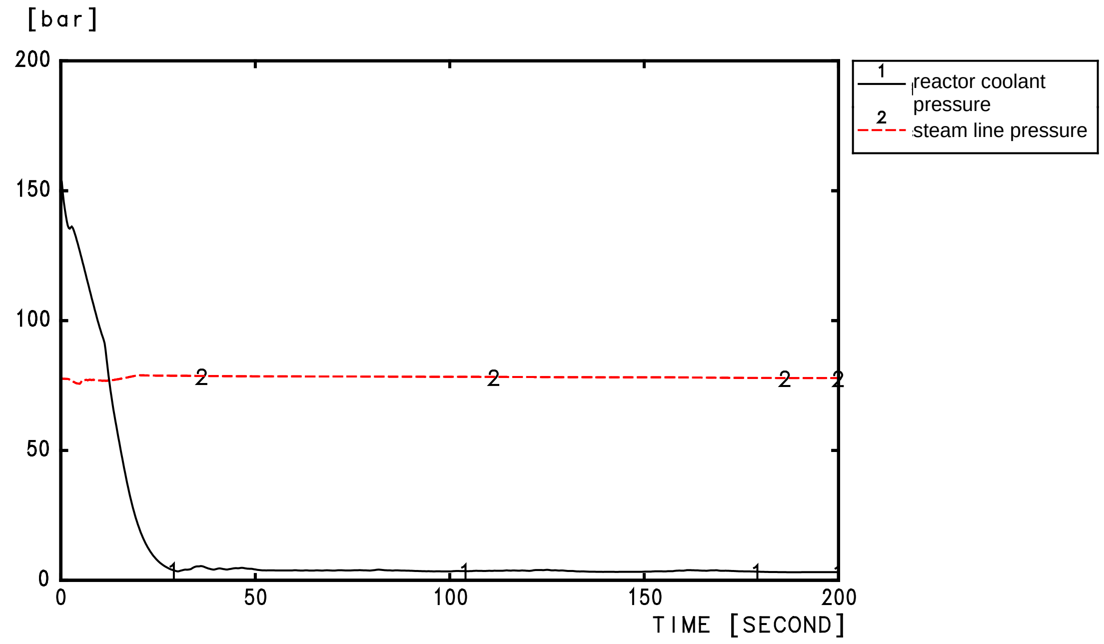
SECTION 16.4.1 - FIGURE B 8: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – LIQUID MASS IN THE HOT ASSEMBLY



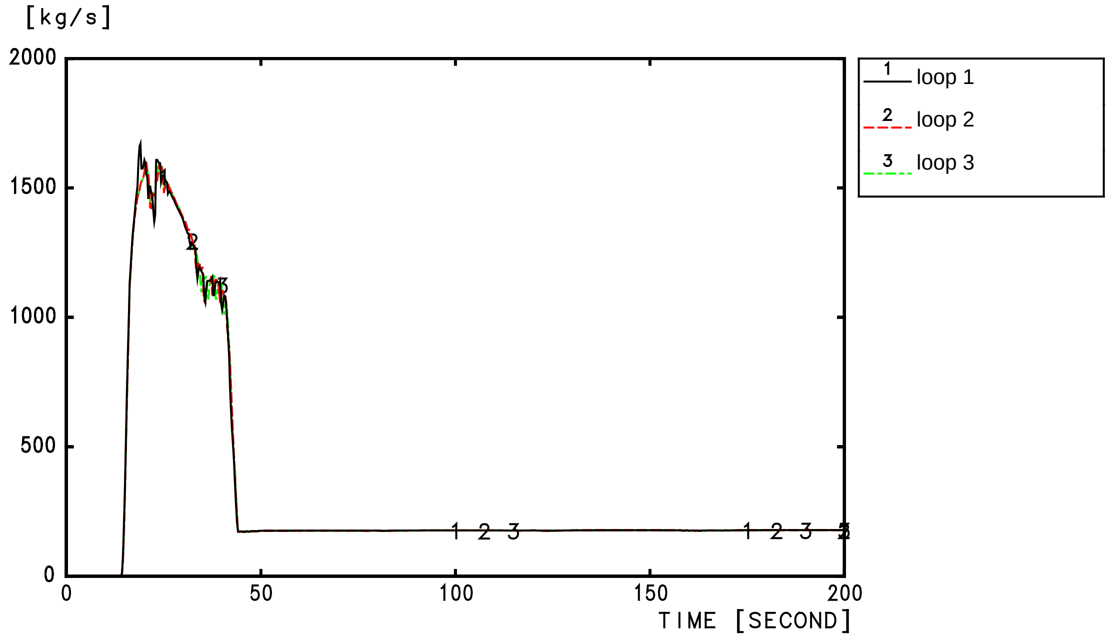
SECTION 16.4.1 - FIGURE B 9: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – LIQUID MASS IN THE CORE



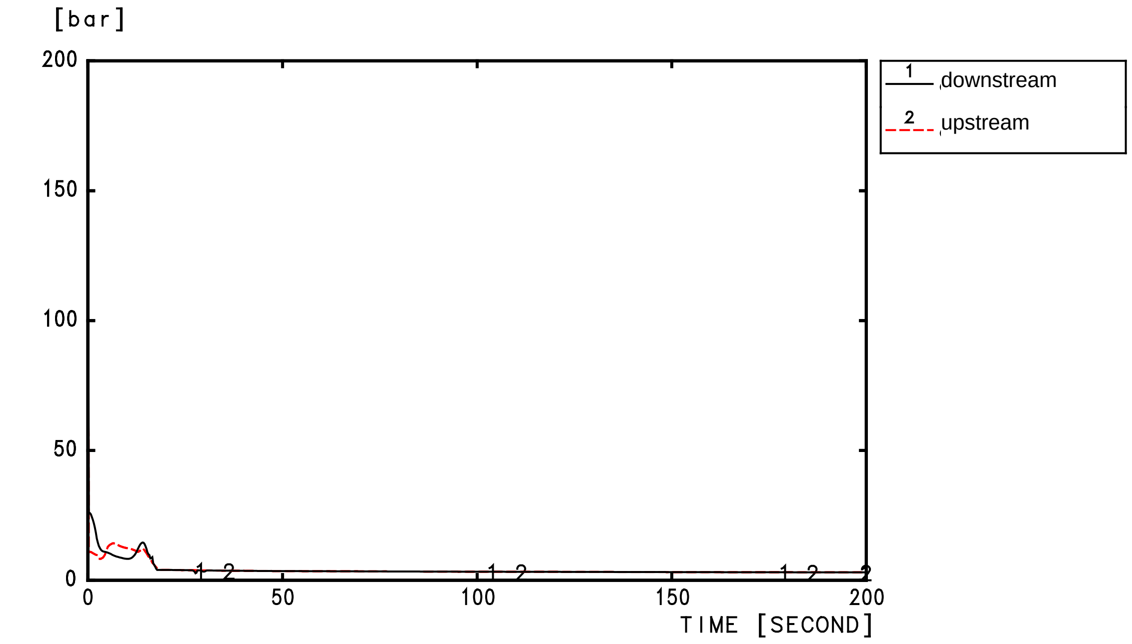
SECTION 16.4.1 - FIGURE B 10: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL MASS IN THE PRIMARY SYSTEM



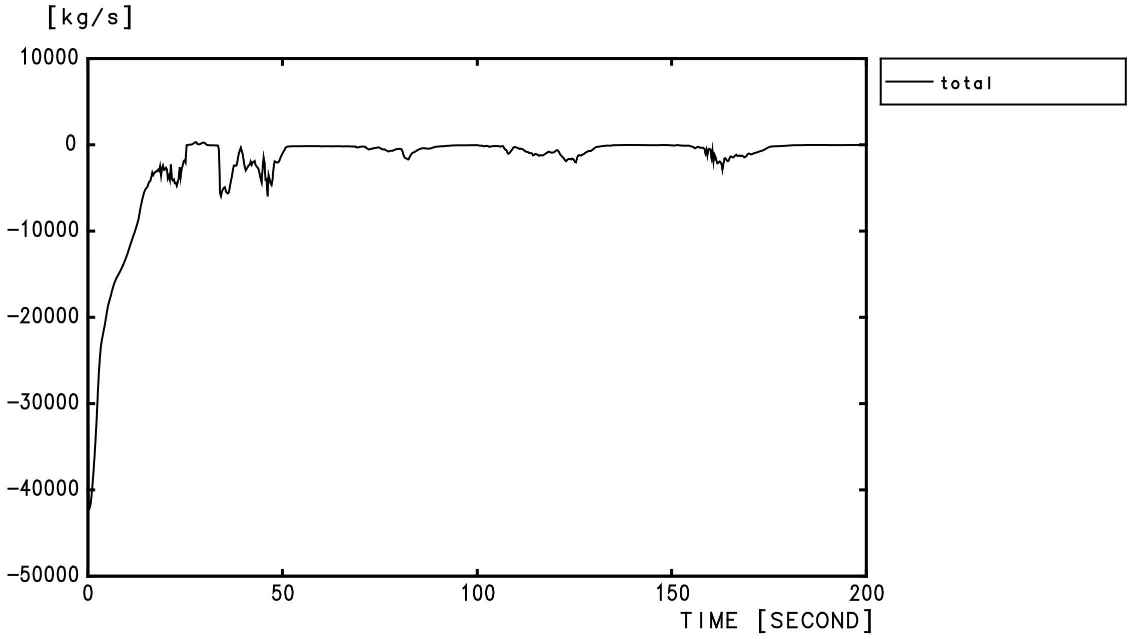
SECTION 16.4.1 - FIGURE B 11: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – REACTOR COOLANT PRESSURE AND STEAM LINE PRESSURE



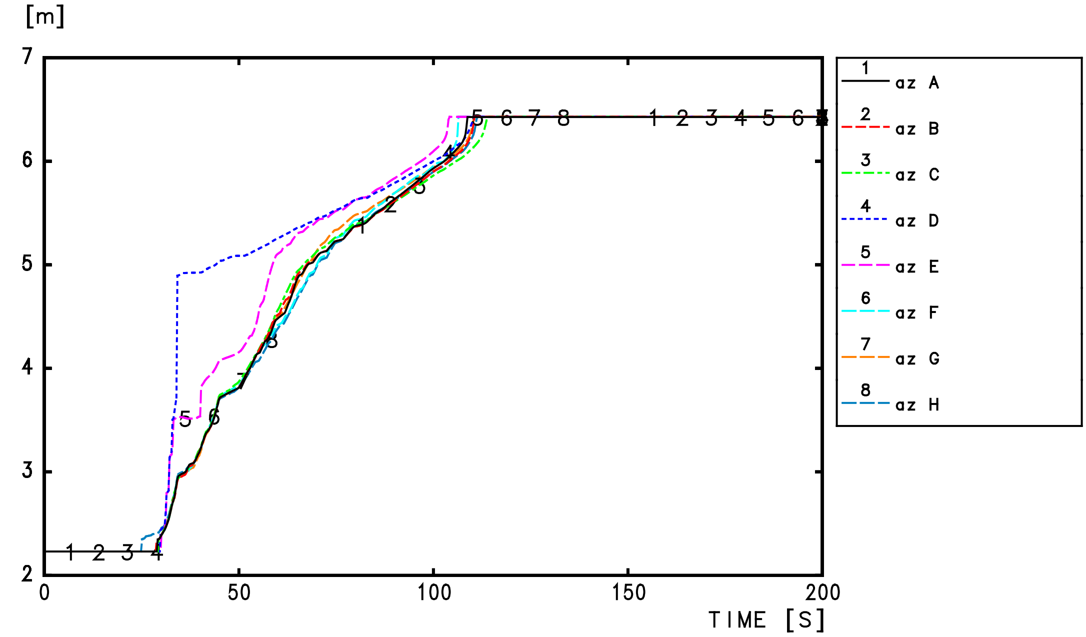
SECTION 16.4.1 - FIGURE B 12: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL RIS [SIS] FLOW RATE (ACCUMULATORS AND SI)



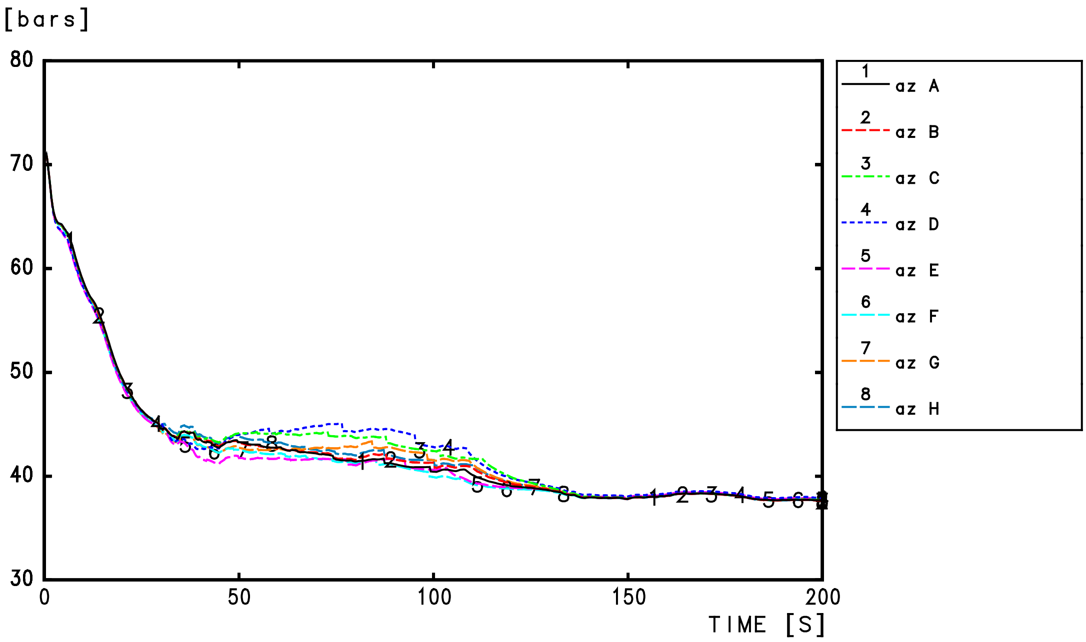
SECTION 16.4.1 - FIGURE B 13: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – PRESSURE AT THE BREAK



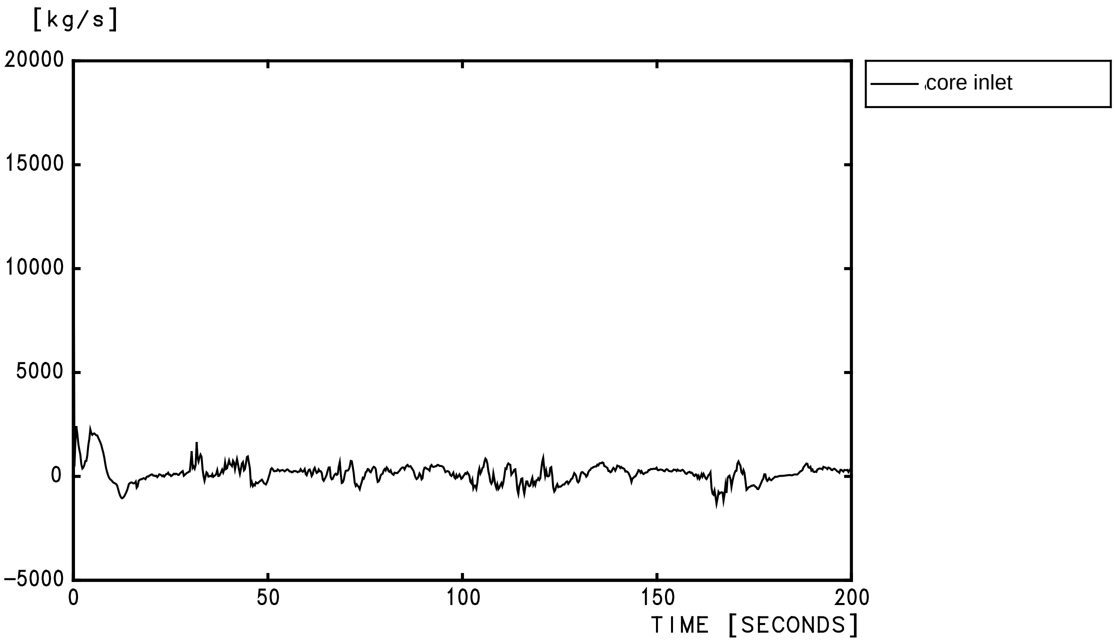
SECTION 16.4.1 - FIGURE B 14: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE BREAK



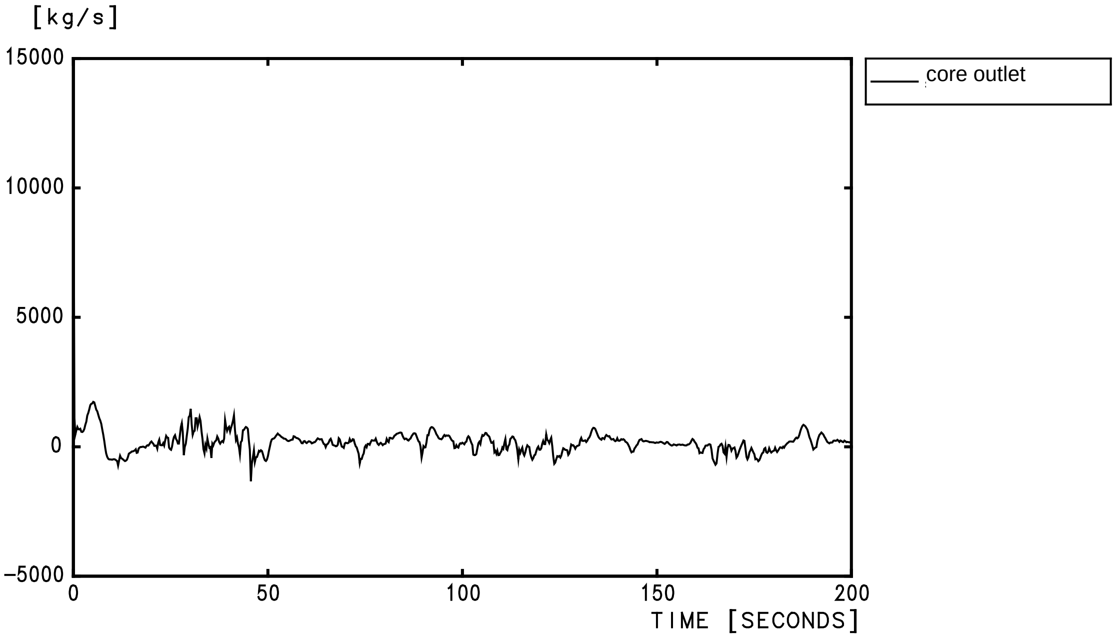
SECTION 16.4.1 - FIGURE B 15: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – QUENCH FRONT LEVEL IN HOT ASSEMBLIES



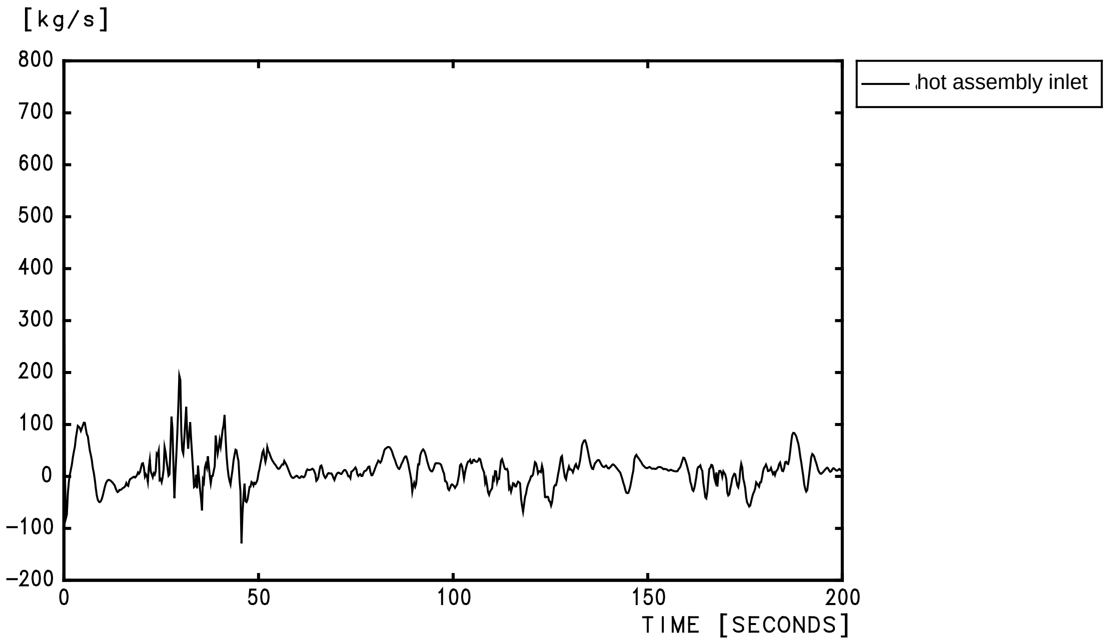
SECTION 16.4.1 - FIGURE B 16: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – MAXIMUM CLAD TEMPERATURES FOR HOT ASSEMBLIES



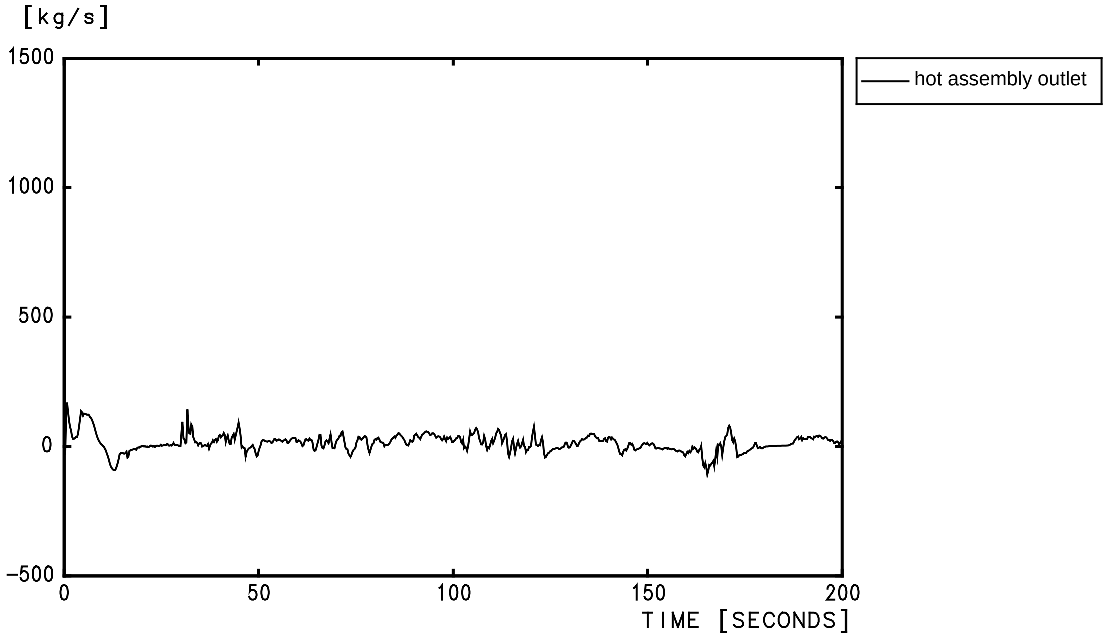
SECTION 16.4.1 - FIGURE B 17: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE CORE INLET



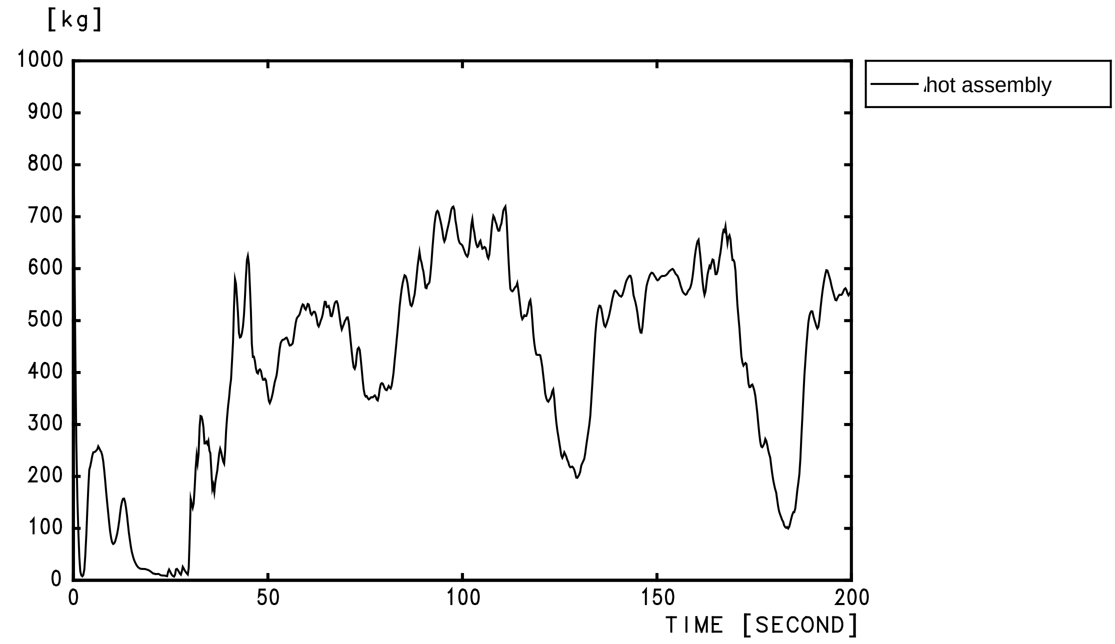
SECTION 16.4.1 - FIGURE B 18: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE CORE OUTLET



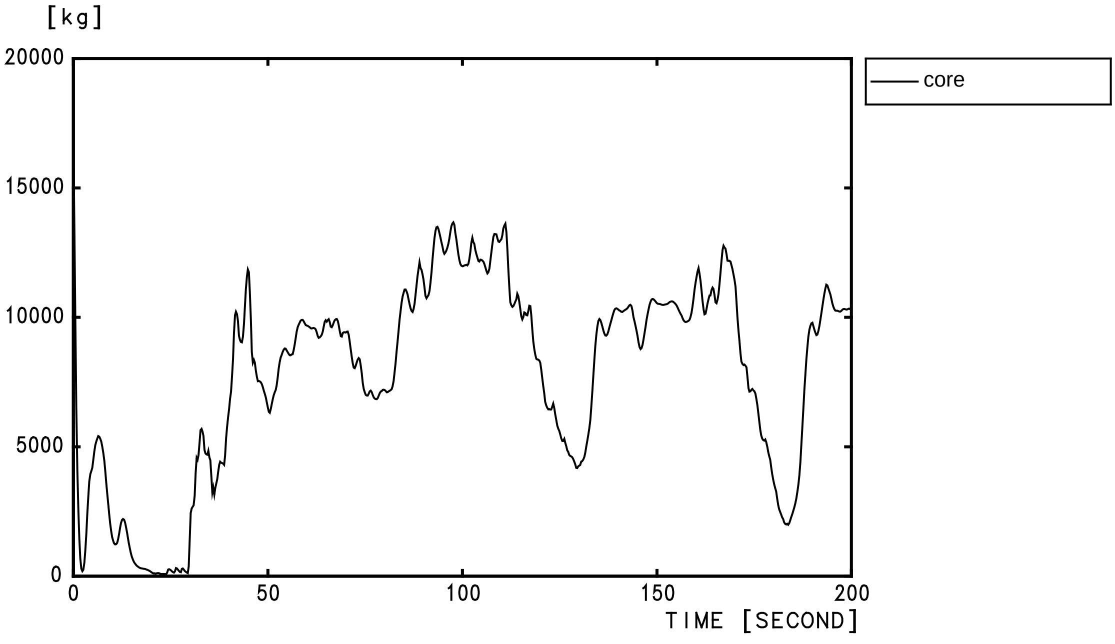
SECTION 16.4.1 - FIGURE B 19: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE HOT ASSEMBLY INLET



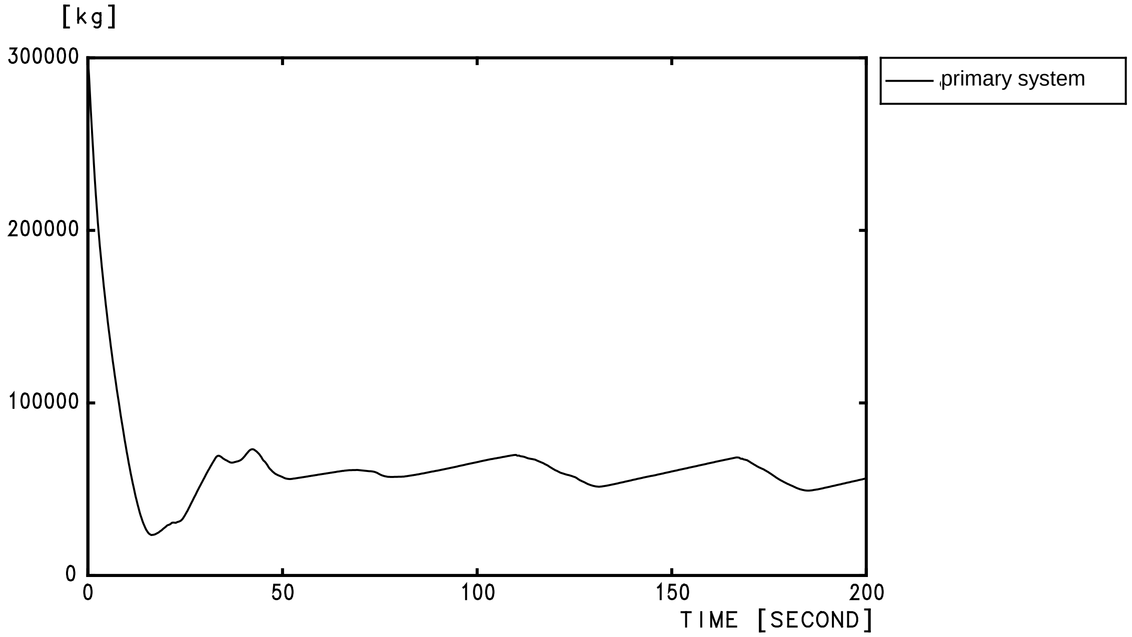
SECTION 16.4.1 - FIGURE B 20: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE HOT ASSEMBLY OUTLET



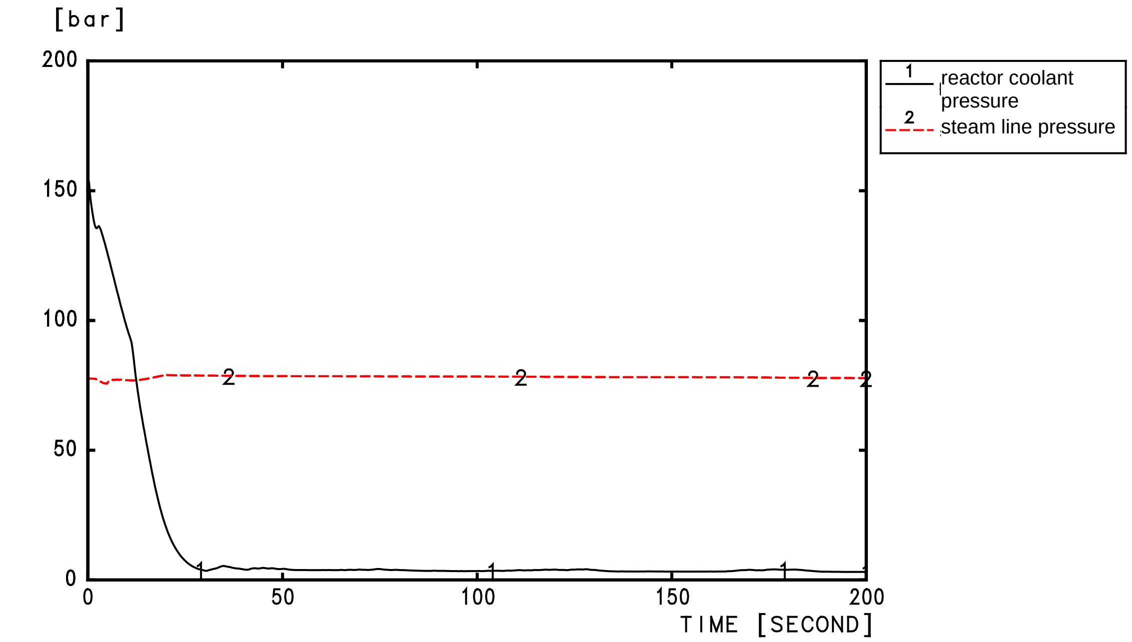
SECTION 16.4.1 - FIGURE B 21: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – LIQUID MASS IN THE HOT ASSEMBLY



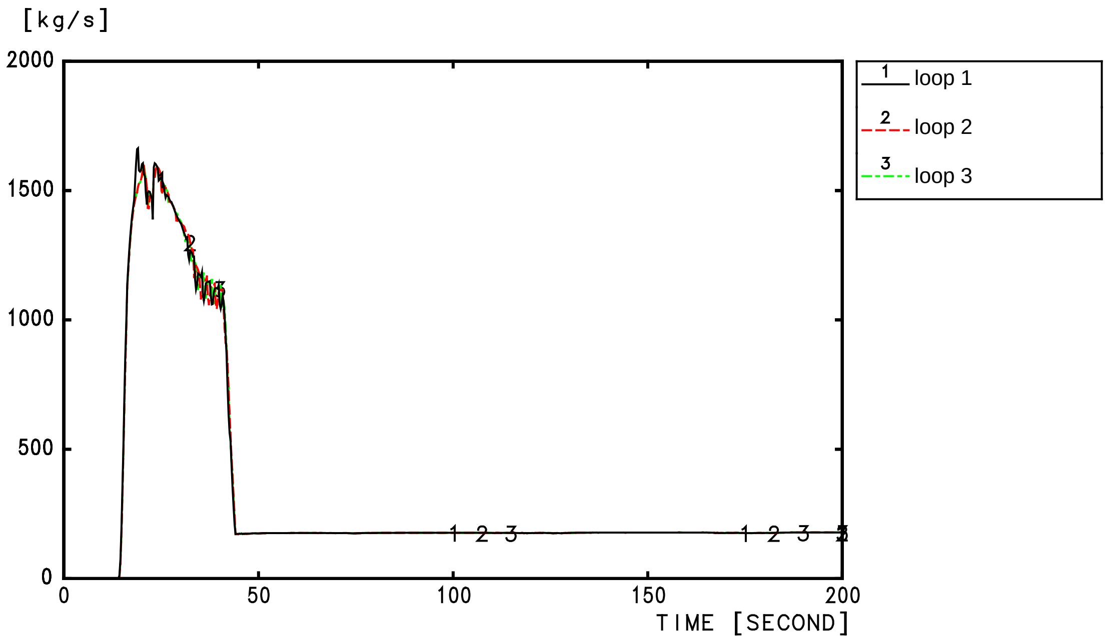
SECTION 16.4.1 - FIGURE B 22: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – LIQUID MASS IN THE CORE



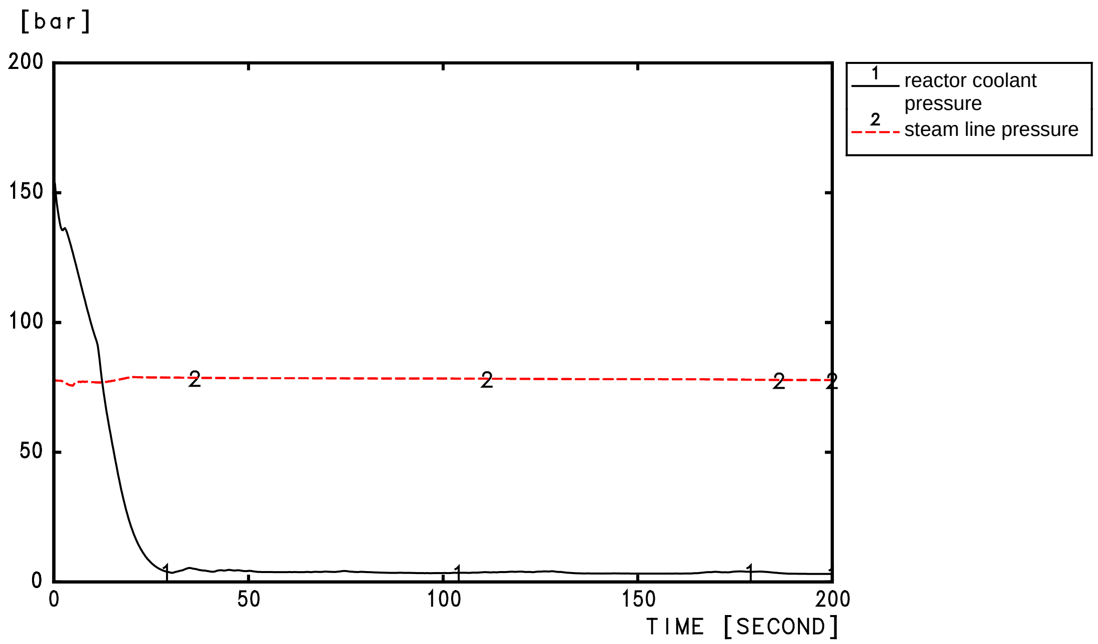
SECTION 16.4.1 - FIGURE B 23: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL MASS IN THE PRIMARY SYSTEM



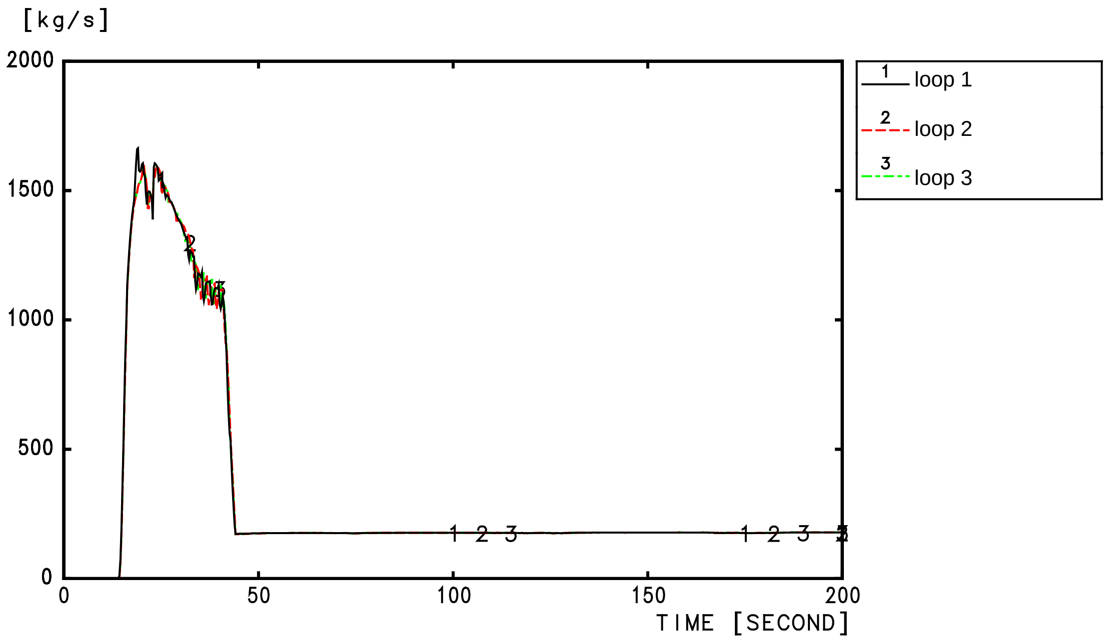
SECTION 16.4.1 - FIGURE B 24: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – REACTOR COOLANT PRESSURE AND STEAM LINE PRESSURE



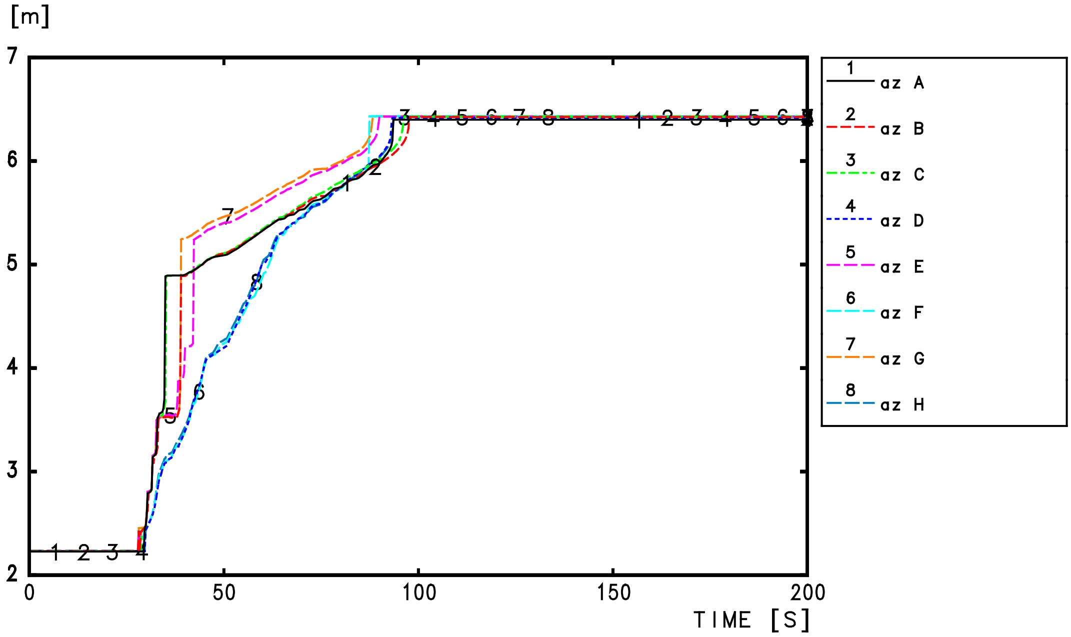
SECTION 16.4.1 - FIGURE B 25: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL RIS [SIS] FLOW RATE (ACCUMULATORS AND SI)



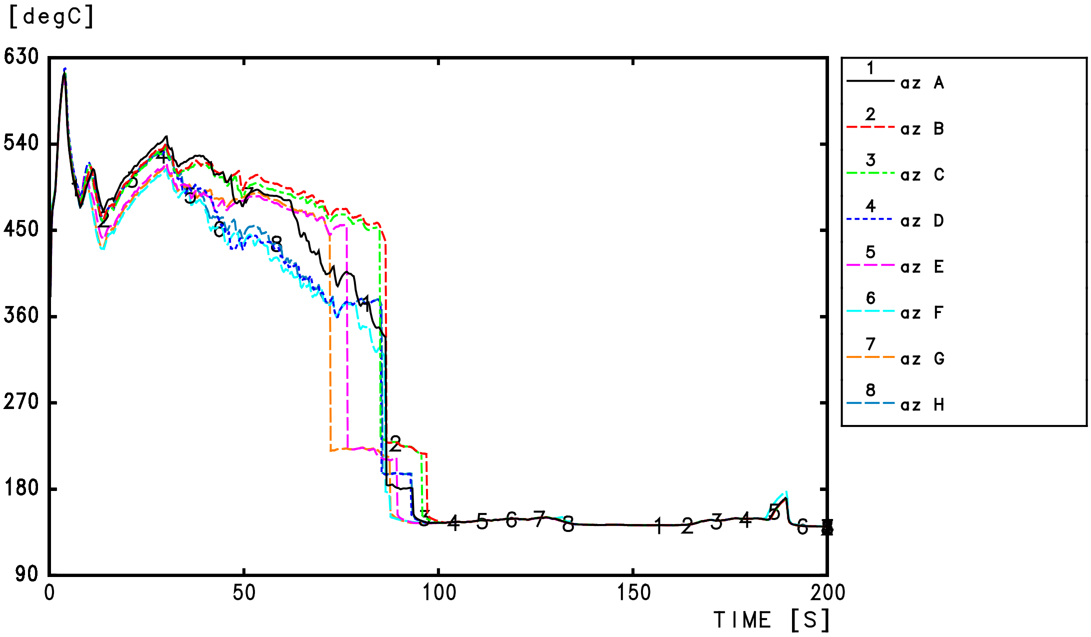
SECTION 16.4.1 - FIGURE B 26: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – PRESSURE AT THE BREAK



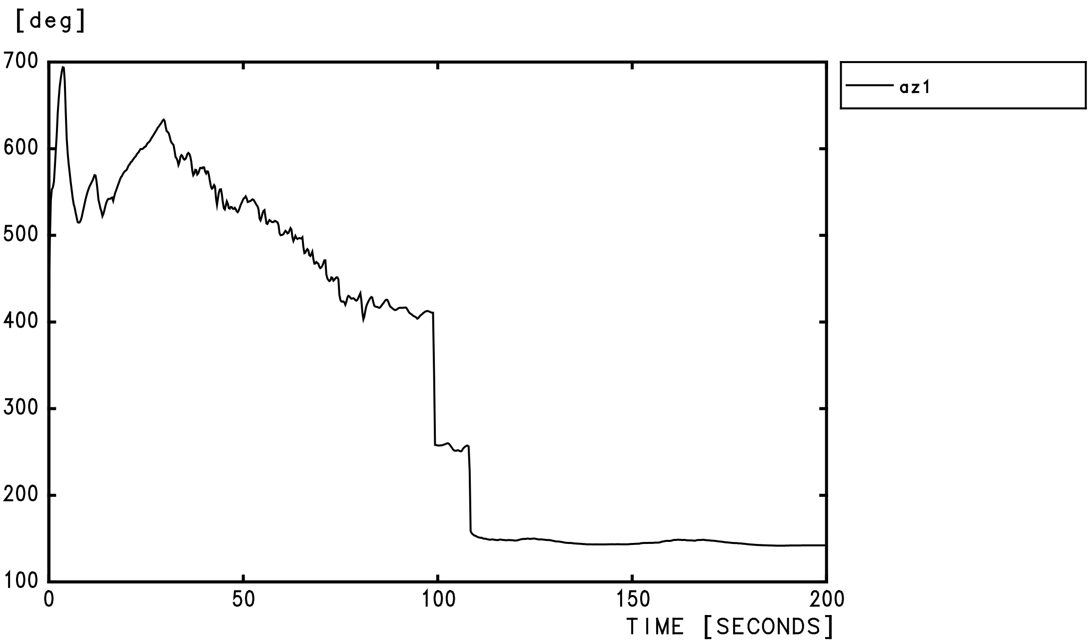
SECTION 16.4.1 - FIGURE B 27: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE BREAK



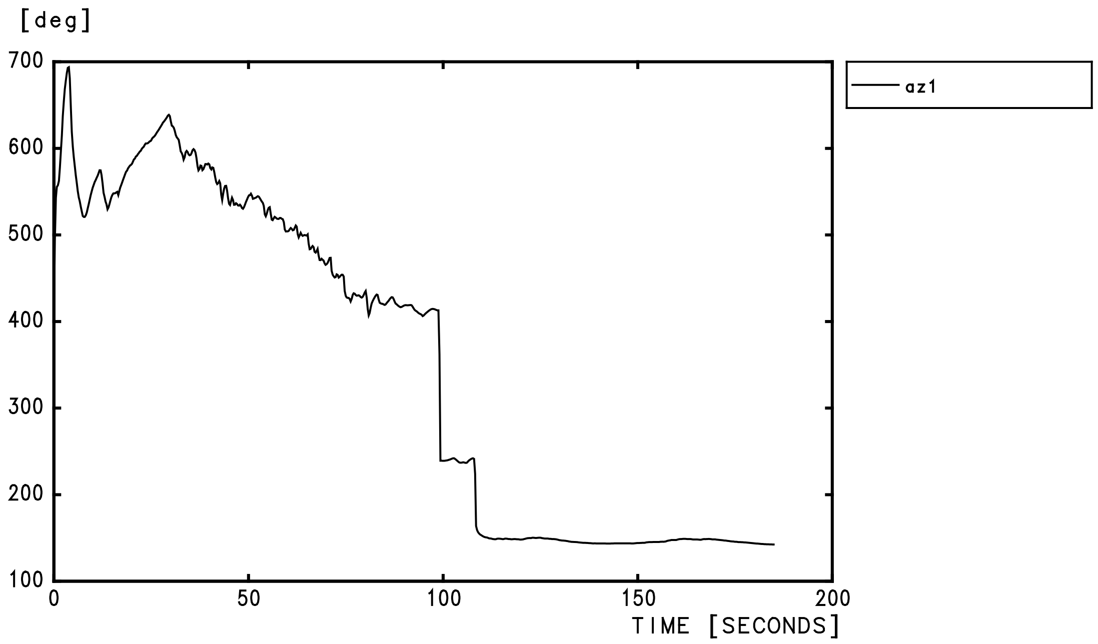
SECTION 16.4.1 - FIGURE B 28: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – QUENCH FRONT LEVEL IN HOT ASSEMBLIES



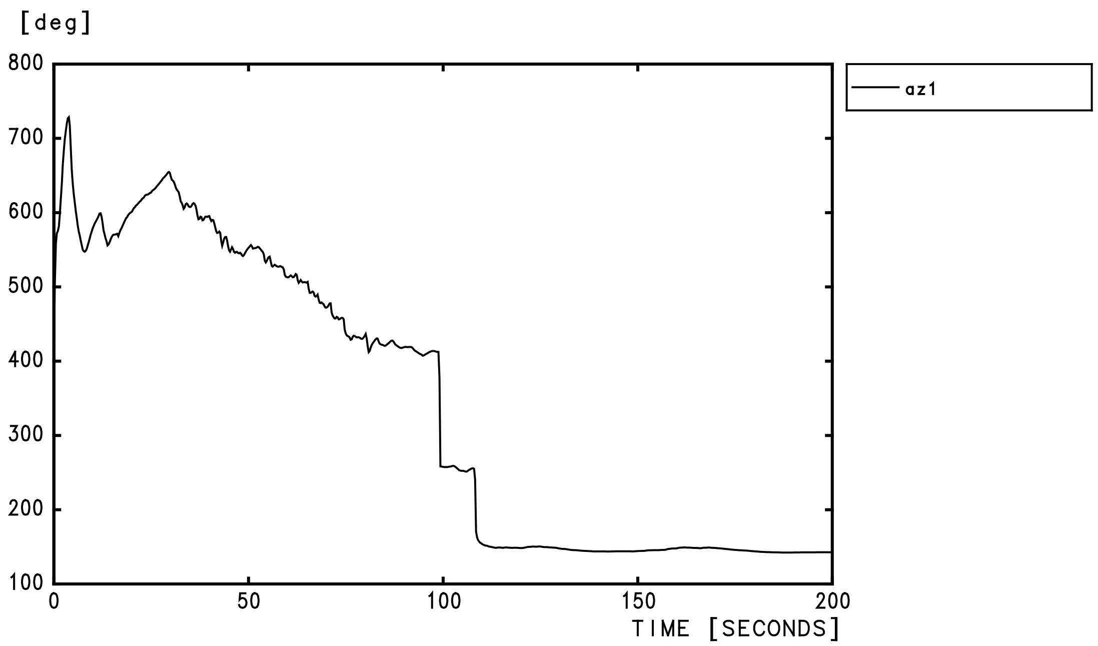
SECTION 16.4.1 - FIGURE B 29: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – MAXIMUM CLAD TEMPERATURES FOR HOT ASSEMBLIES



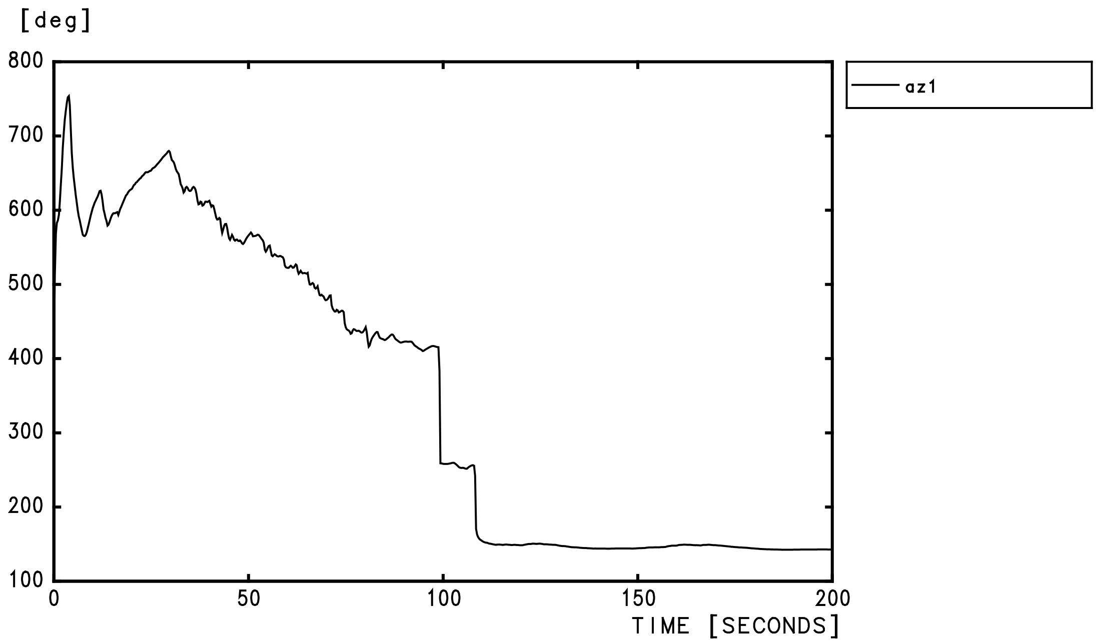
SECTION 16.4.1 - FIGURE B 30: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – BOL INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



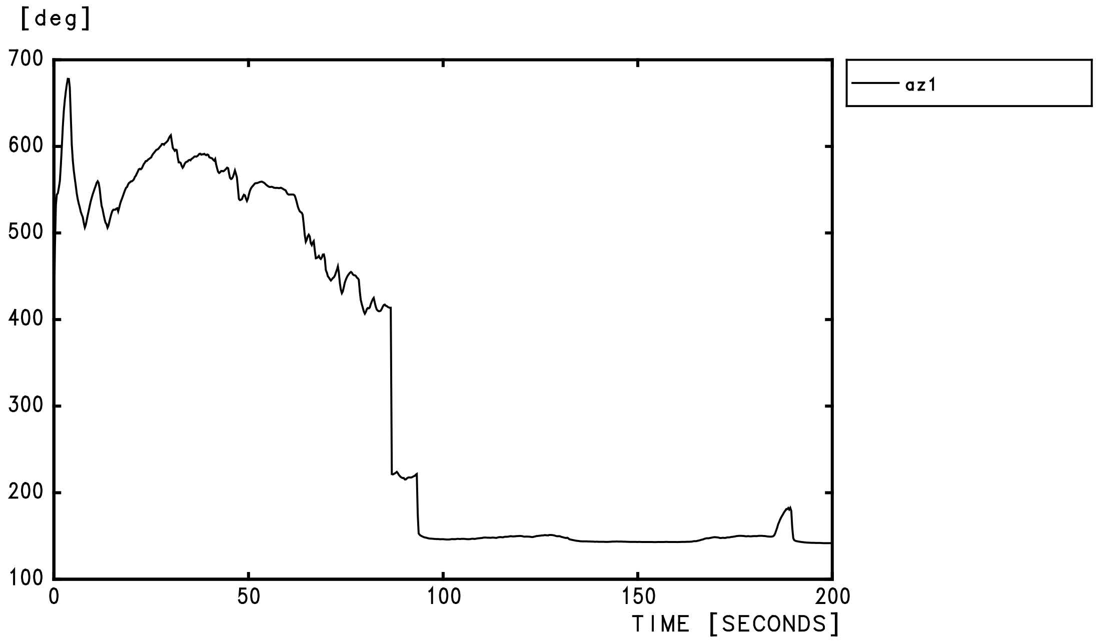
SECTION 16.4.1 - FIGURE B 31: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC1 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



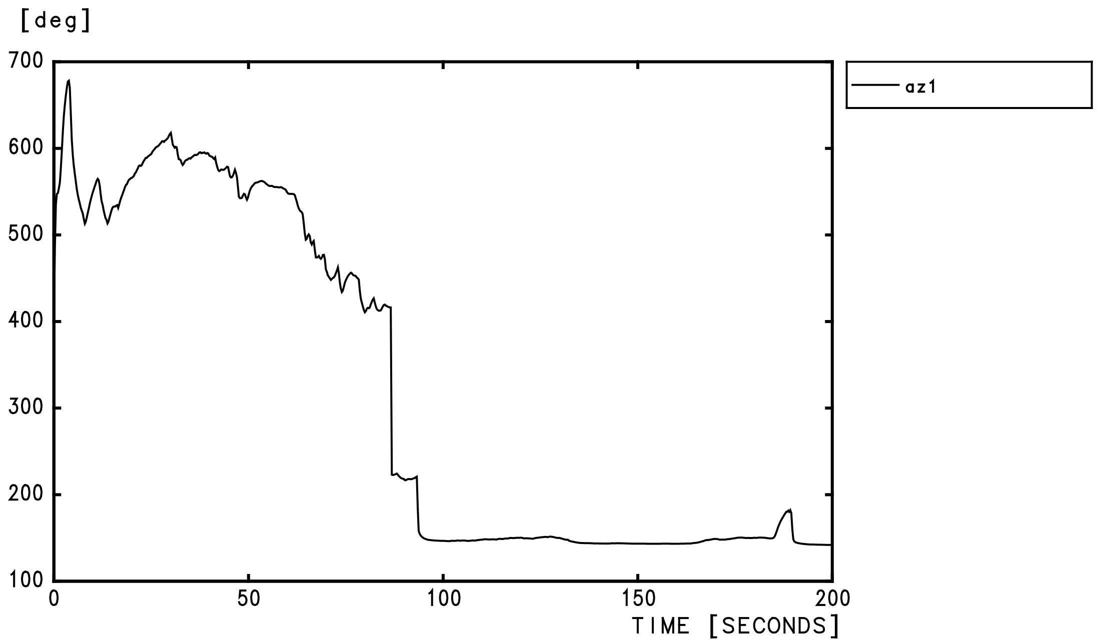
SECTION 16.4.1 - FIGURE B 32: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC2 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



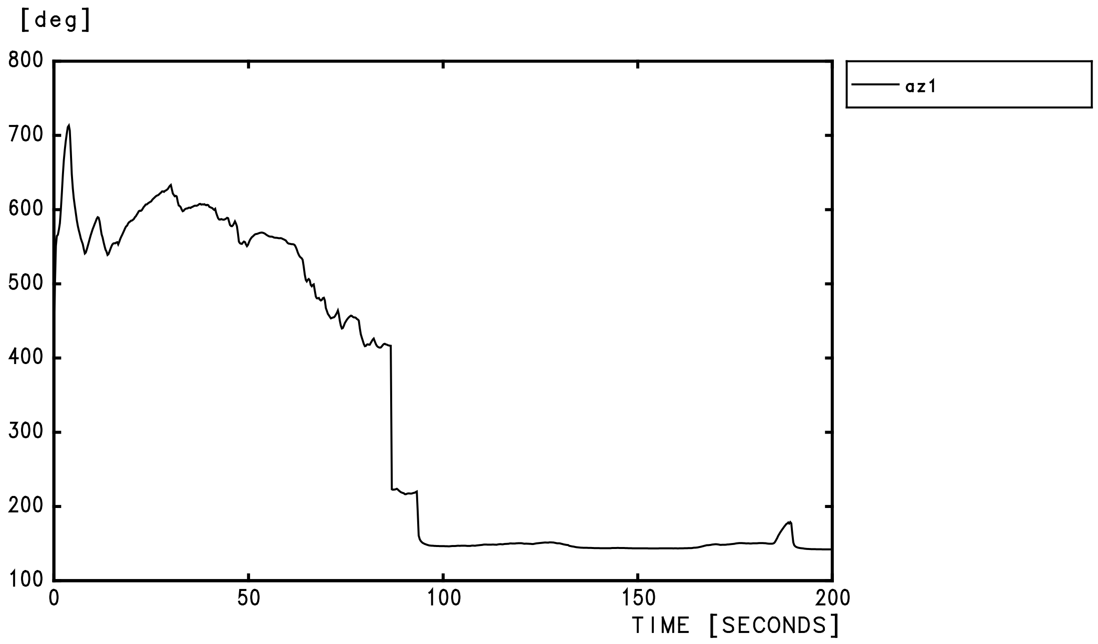
SECTION 16.4.1 - FIGURE B 33: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC3 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



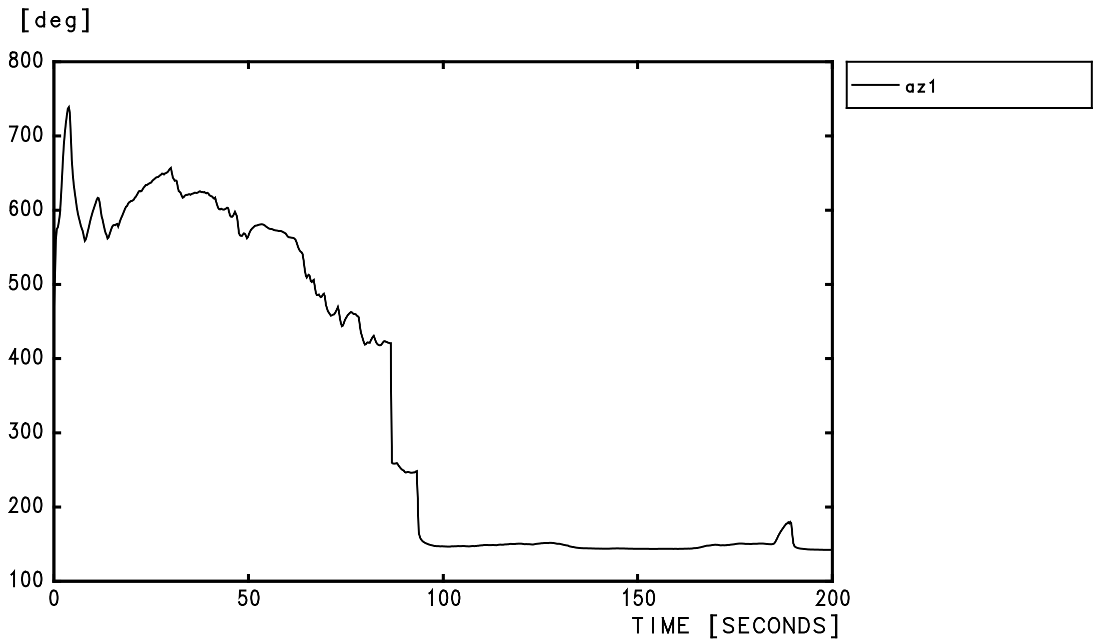
SECTION 16.4.1 - FIGURE B 34: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – BOL INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



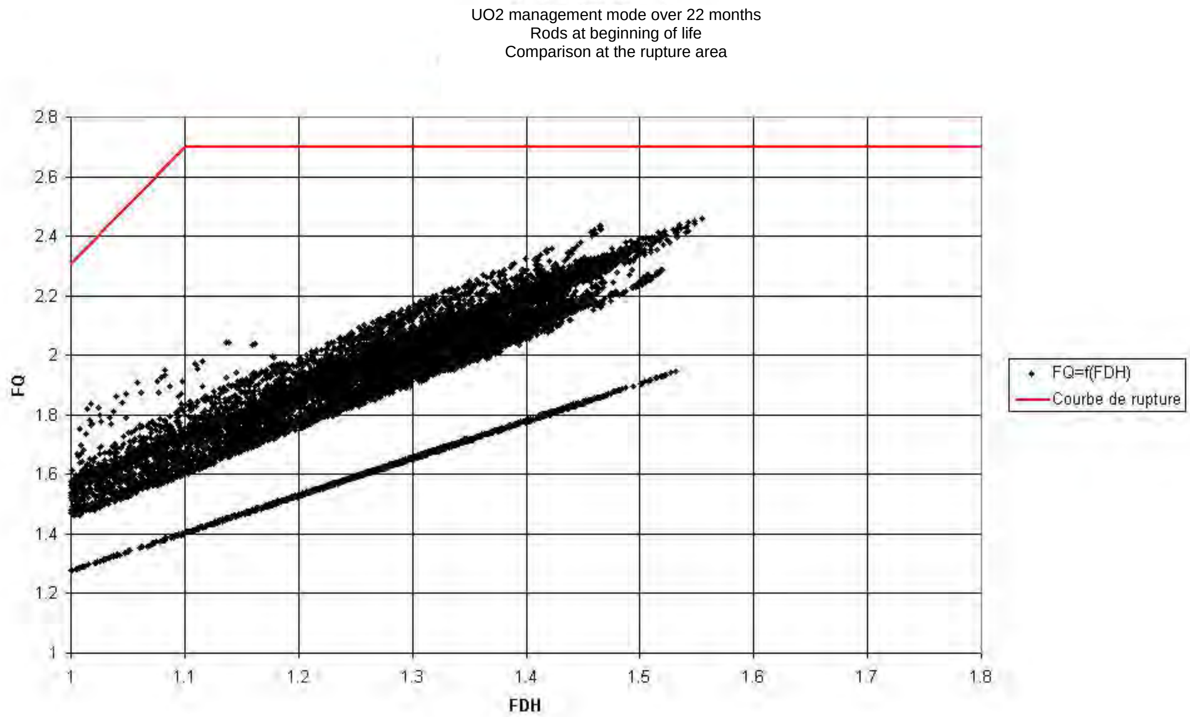
SECTION 16.4.1 - FIGURE B 35: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC1 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



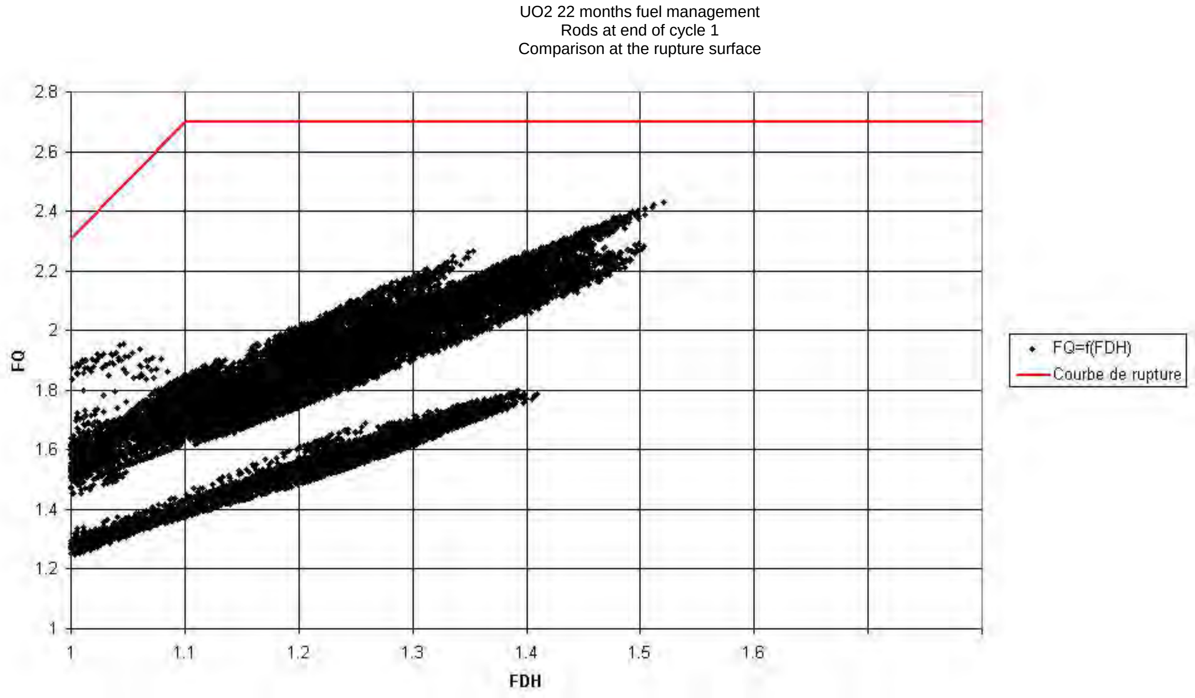
SECTION 16.4.1 - FIGURE B 36: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC2 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



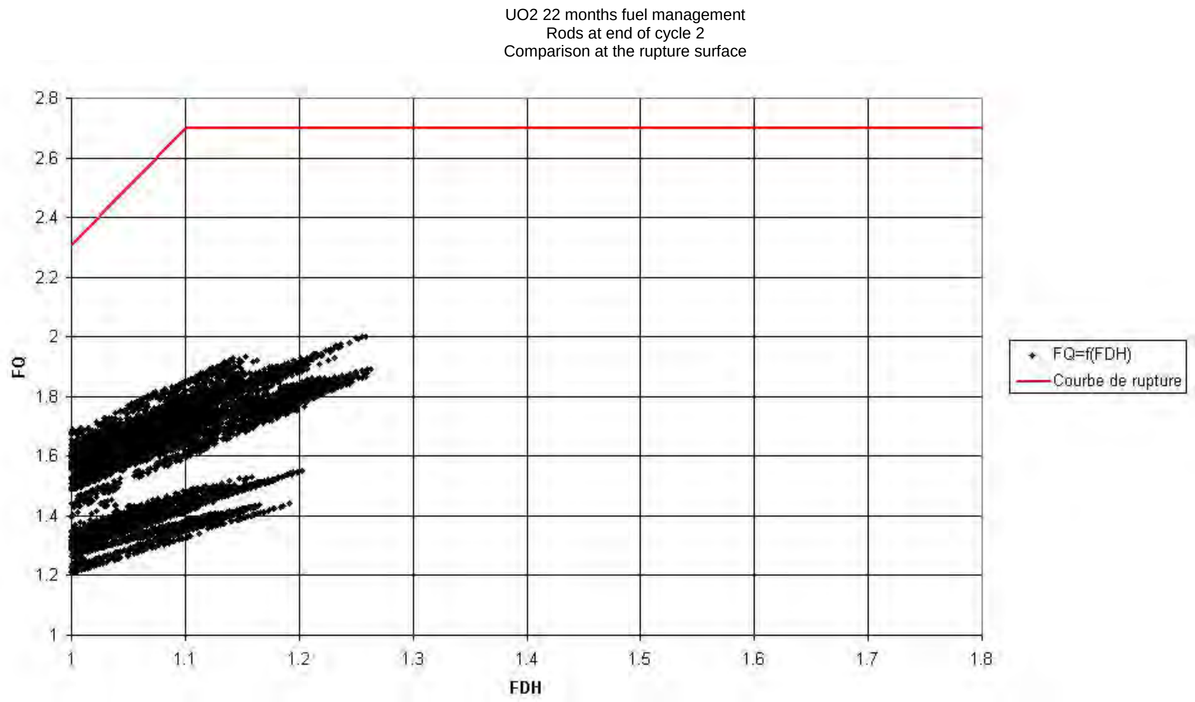
SECTION 16.4.1 - FIGURE B 37: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC3 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



SECTION 16.4.1 - FIGURE B 38: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – RUPTURE CURVE – BOL



SECTION 16.4.1 - FIGURE B 39: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – TC – RUPTURE CURVE – EOC1



SECTION 16.4.1 - FIGURE B 40: IN-OUT UO2 22 MONTHS FUEL MANAGEMENT – TC – RUPTURE CURVE – EOC2

SECTION 16.4.1 - TABLE C 1: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – FUEL CALCULATION POINTS

Core cycle time	Rod burn-up
Beginning of life (BOL)	150 MWd/tU
End of Cycle 1 (EOC1)	27,000 MWd/tU
End of Cycle 2 (EOC2)	47,750 MWd/tU
End of Cycle 3 (EOC3)	64,650 MWd/tU
End of Cycle 4 (EOC4)	72,250 MWd/tU

SECTION 16.4.1 - TABLE C 2: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – PROPERTIES OF HOT ASSEMBLIES

Hot assembly	F∆H	FQ	BU
HA1	1.47	2.10	EOC1
HA2	1.20	1.80	EOC2

SECTION 16.4.1 - TABLE C 3: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – MAIN EVENTS OF THE TRANSIENT

Event	Time at which the event occurs
RT signal	3.6 s
SI signal	7.1 s
Start of accumulator injection	14.2 s
Start of safety injection system injection	23.0 s
Start of reflood	30.7 s
End of reflood	105.0 s
Duration of reflood	74.3 s
End of accumulator injection	44.1 s

SECTION 16.4.1 - TABLE C 4: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – PEAK CLAD TEMPERATURES

	T1	T2	T3
Time (s)	3.4	30.3	50.6
Mesh	7	7	7
Value (°C)	634	571	500

SECTION 16.4.1 - TABLE C 5: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – MAIN EVENTS OF THE TRANSIENT

Event	Time at which the event occurs
RT signal	3.6 s
SI signal	7.1 s
Start of accumulator injection	14.0 s
Start of safety injection system injection	23.0 s
Start of reflood	27.5 s
End of reflood	119.0 s
Duration of reflood	91.5 s
End of accumulator injection	44.1 s

SECTION 16.4.1 - TABLE C 6: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – PEAK CLAD TEMPERATURES

	T1	T2	T3
Time (s)	4.0	27.3	54.0
Mesh	6	7	7
Value (°C)	611	533	497

SECTION 16.4.1 - TABLE C 7: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – PROPERTIES OF THE INDICATOR RODS

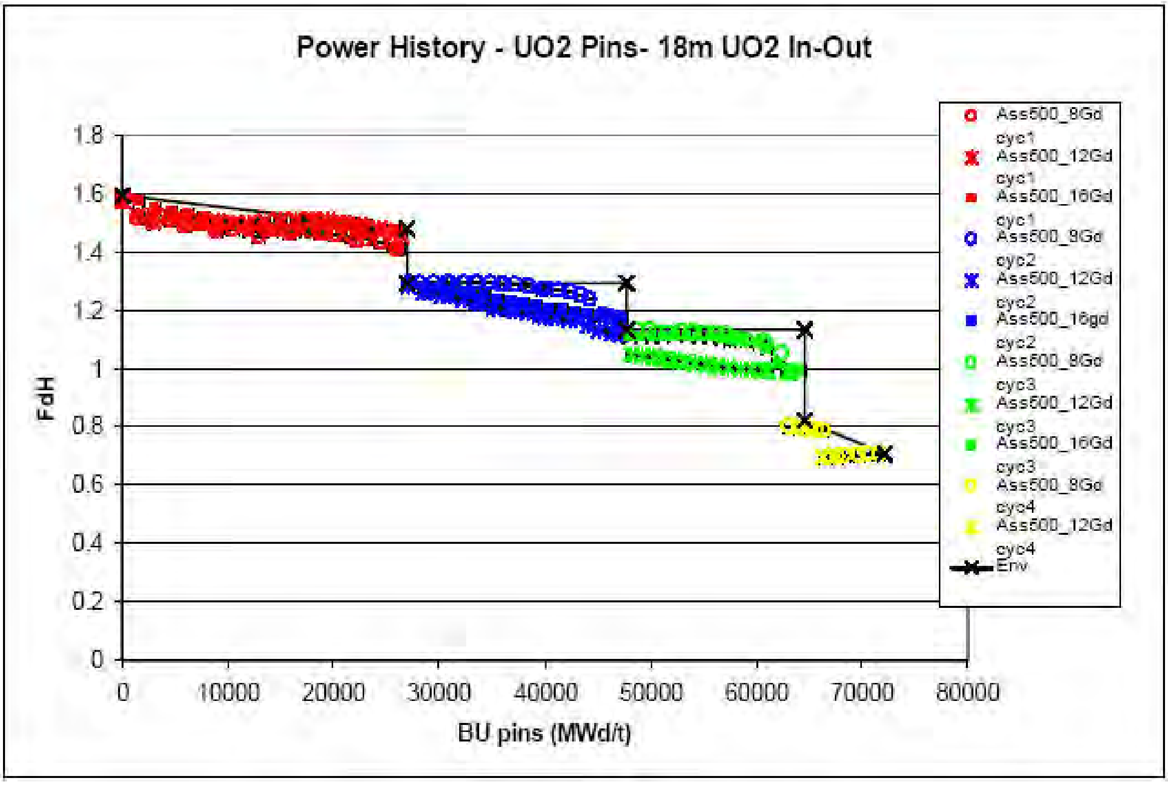
Burn-up	FQ	FΔH	Pressure in the gap (bar)	Maximum linear power density (W/cm)	Maximum initial pellet temperature (°C)
BOL	2.42	1.71	56.1	395.80	934
EOC1	2.42	1.71	68.6	395.80	925
EOC2	2.42	1.71	79.7	395.80	994
EOC3	2.42	1.71	101.3	395.80	1094
EOC4	2.42	1.71	103.8	395.80	1139

SECTION 16.4.1 - TABLE C 8: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – RESULTS OBTAINED FOR INDICATOR RODS

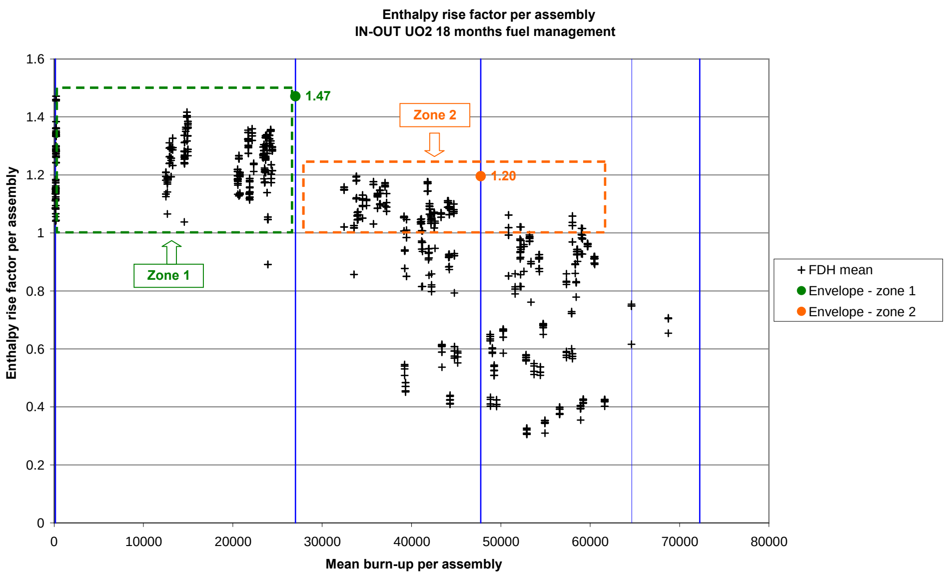
Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Rupture
BOL	693	621	545	0.40	NO
EOC1	686	623	546	0.40	NO
EOC2	714	633	552	0.40	NO
EOC3	743	658	564	0.50	NO
EOC4	765	671	572	0.50	NO

SECTION 16.4.1 - TABLE C 9: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – RESULTS OBTAINED FOR INDICATOR RODS

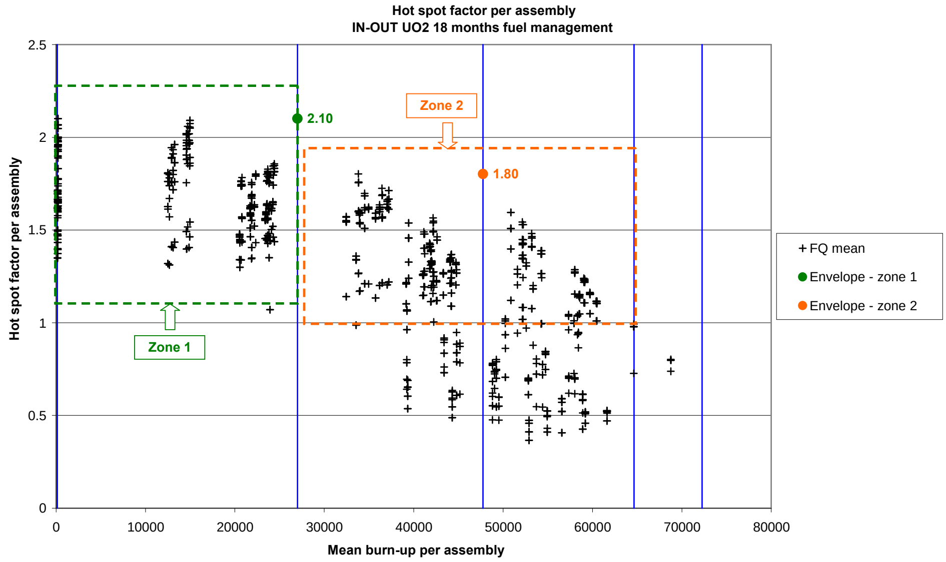
Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Rupture
BOL	687	608	557	0.34	NO
EOC1	692	602	553	0.32	NO
EOC2	712	619	561	0.34	NO
EOC3	741	645	575	0.38	NO
EOC4	754	658	581	0.40	NO



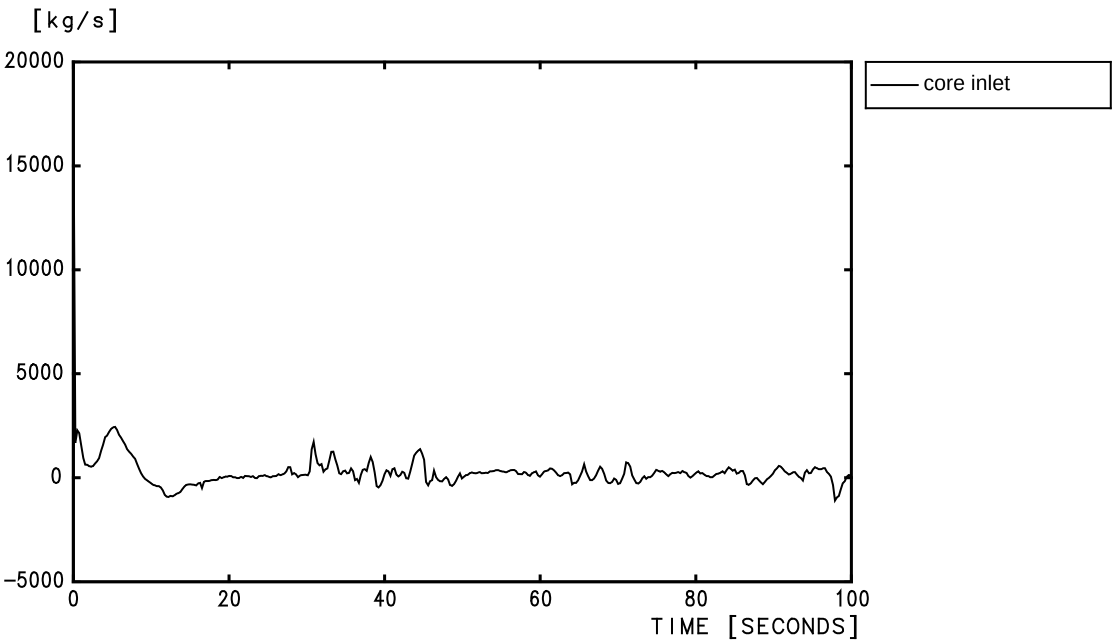
SECTION 16.4.1 - FIGURE C 1: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – POWER HISTORY OF UO_2 RODS



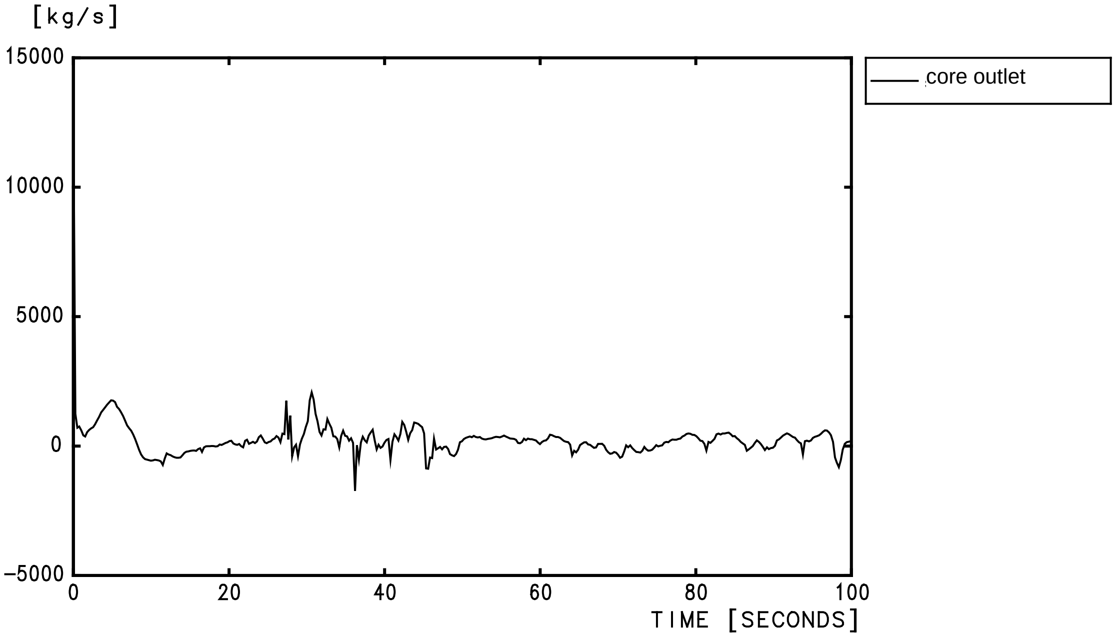
SECTION 16.4.1 - FIGURE C 2: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – F∆H FOR ASSEMBLIES AS A FUNCTION OF BURN-UP



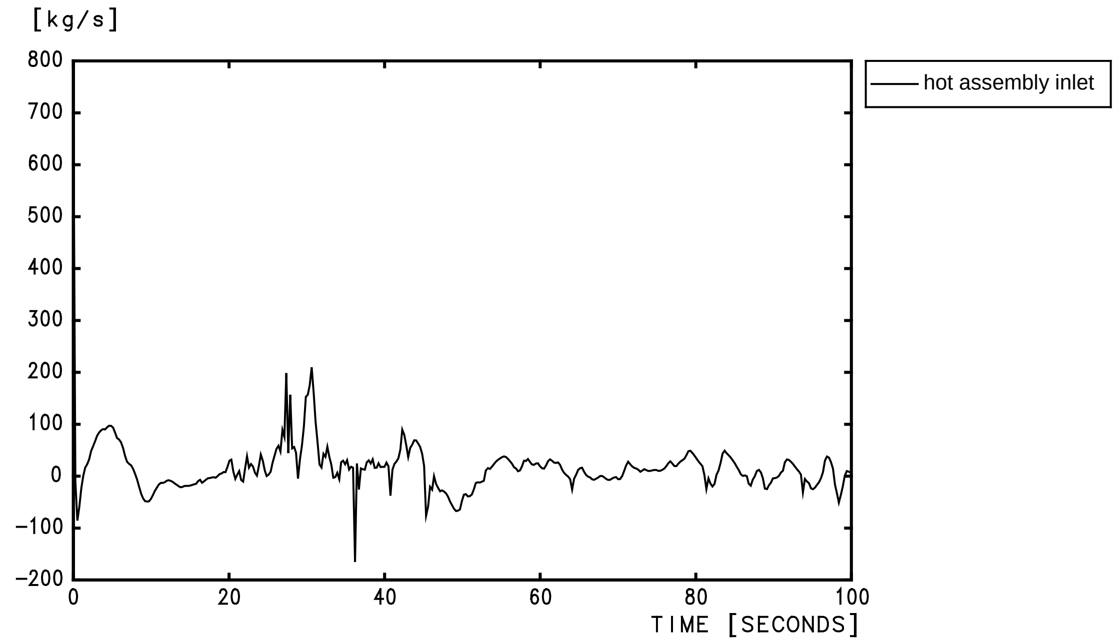
SECTION 16.4.1 - FIGURE C 3: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – FQ FOR ASSEMBLIES AS A FUNCTION OF BURN-UP



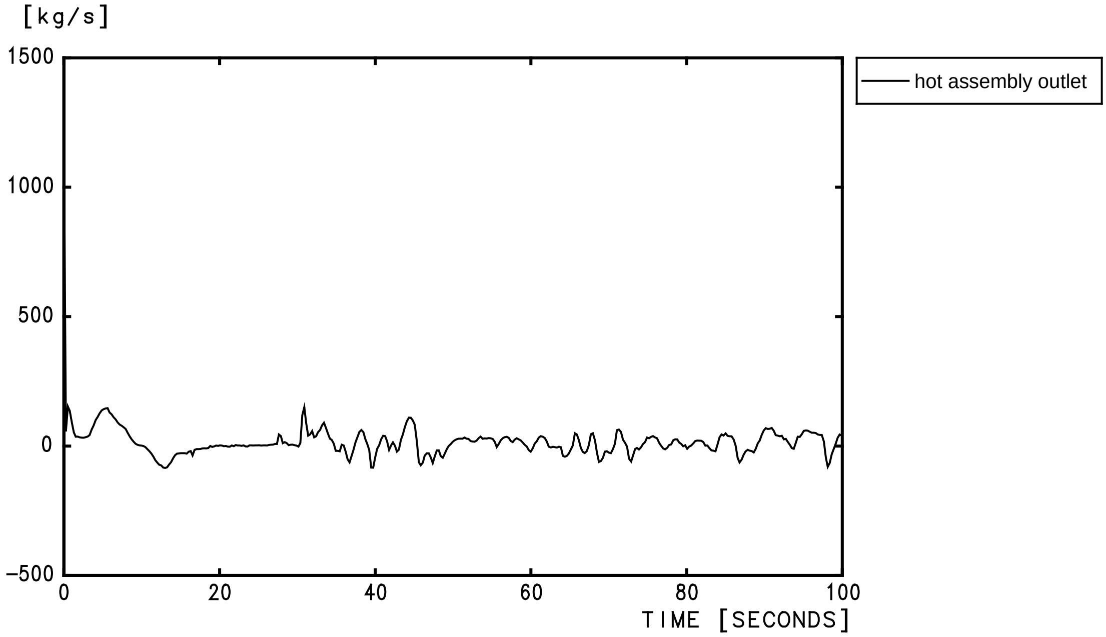
SECTION 16.4.1 - FIGURE C 4: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE CORE INLET



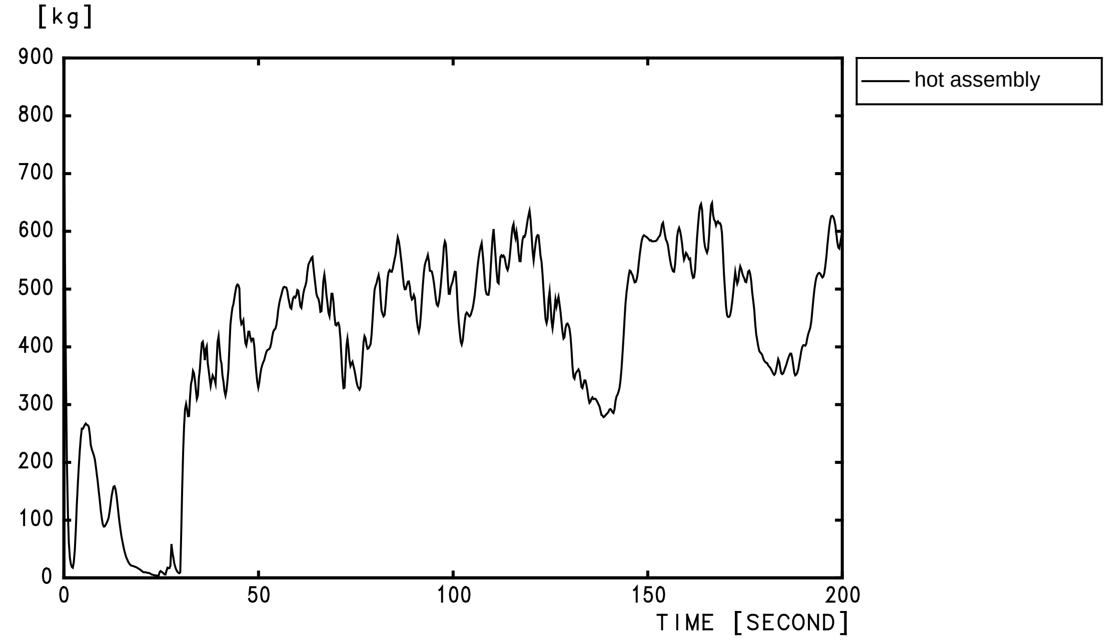
SECTION 16.4.1 - FIGURE C 5: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE CORE OUTLET



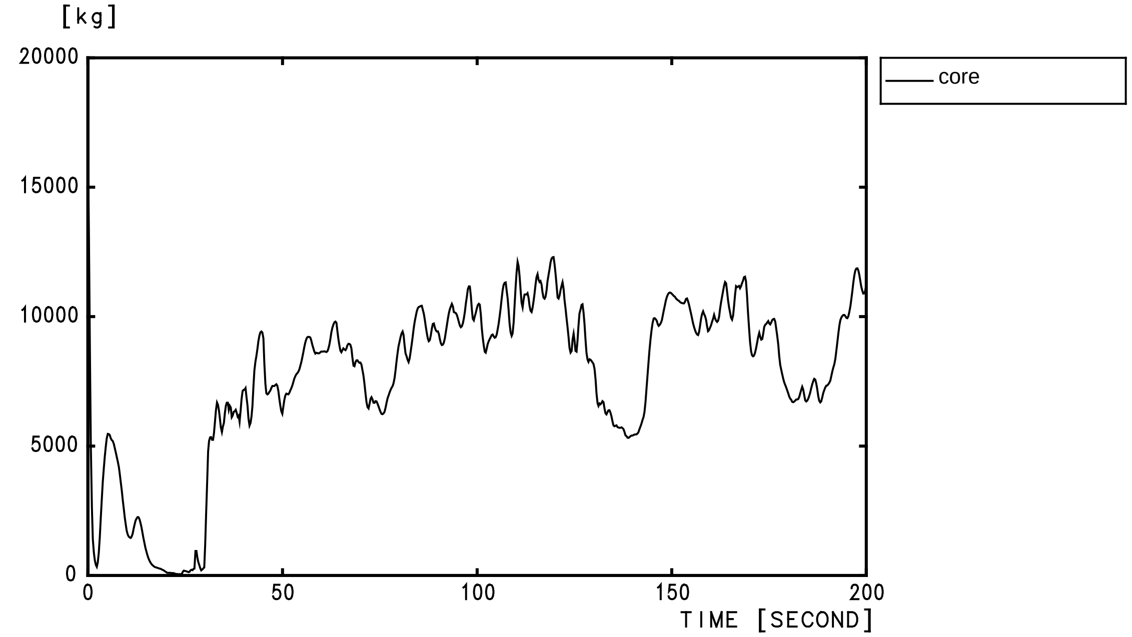
SECTION 16.4.1 - FIGURE C 6: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE HOT ASSEMBLY INLET



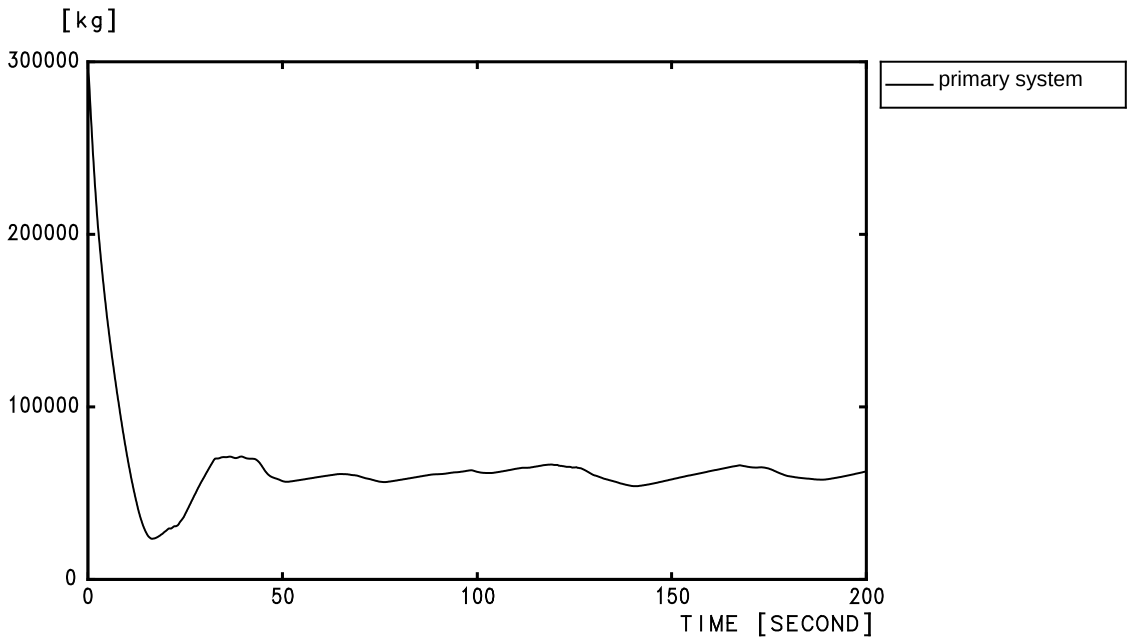
SECTION 16.4.1 - FIGURE C 7: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE HOT ASSEMBLY OUTLET



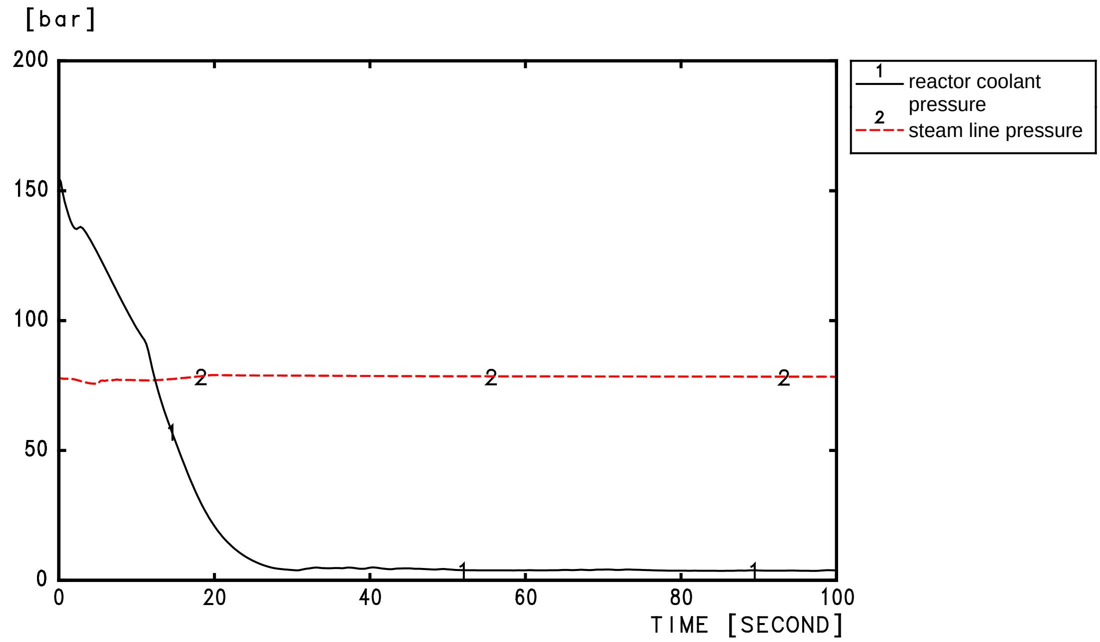
SECTION 16.4.1 - FIGURE C 8: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – LIQUID MASS IN THE HOT ASSEMBLY



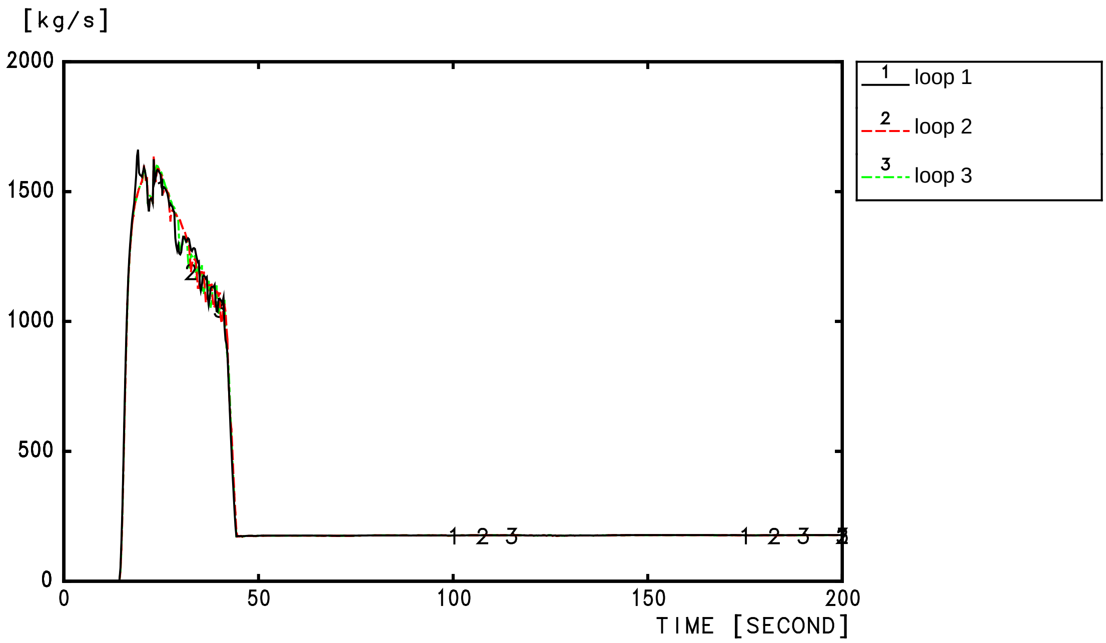
SECTION 16.4.1 - FIGURE C 9: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – LIQUID MASS IN THE CORE



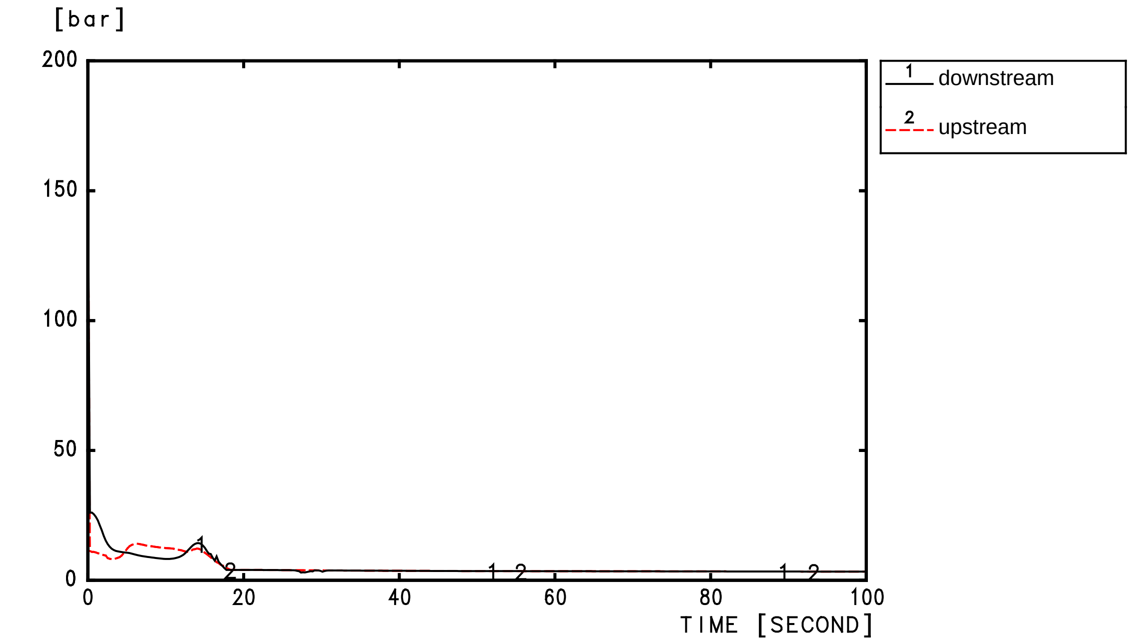
SECTION 16.4.1 - FIGURE C 10: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL MASS IN THE PRIMARY SYSTEM



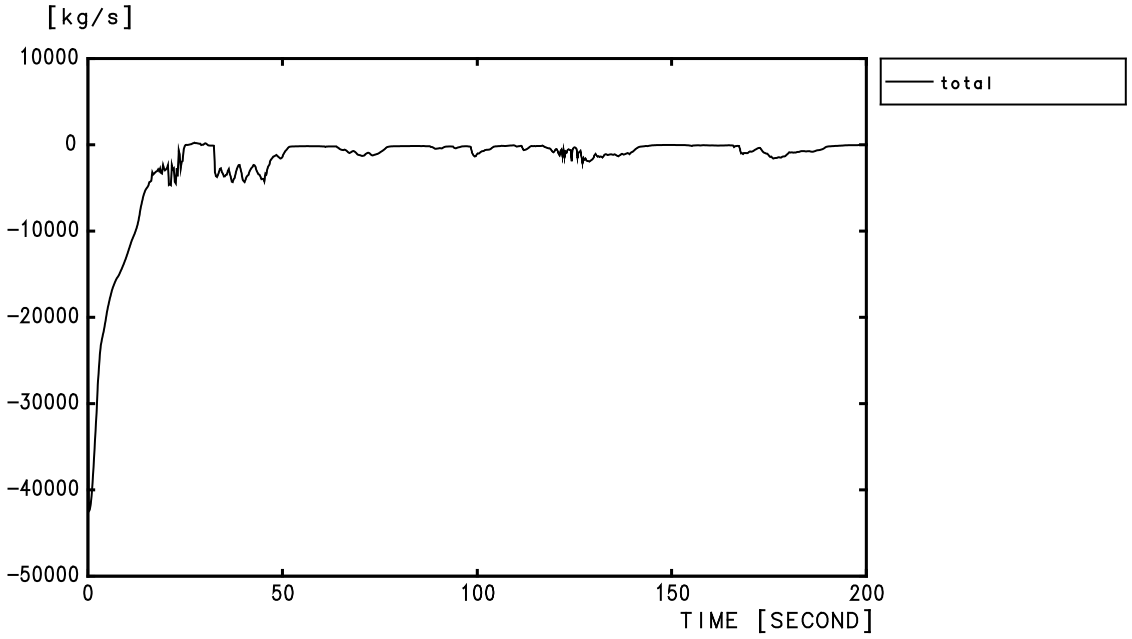
SECTION 16.4.1 - FIGURE C 11: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – REACTOR COOLANT PRESSURE AND STEAM LINE PRESSURE



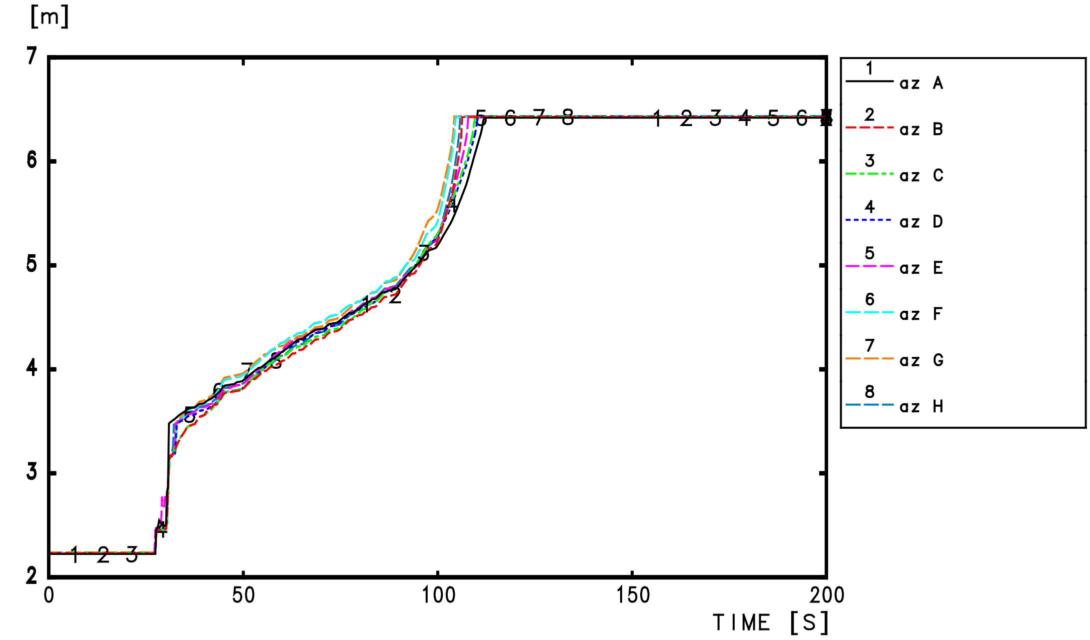
SECTION 16.4.1 - FIGURE C 12: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL RIS [SIS] FLOW RATE (ACCUMULATORS AND SI)



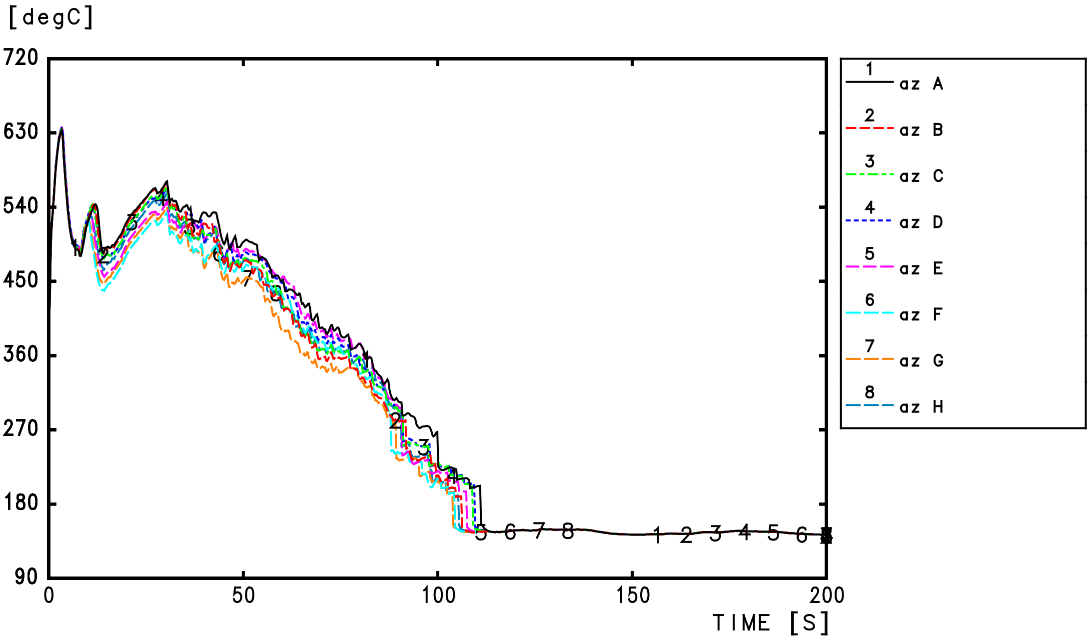
SECTION 16.4.1 - FIGURE C 13: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – PRESSURE AT THE BREAK



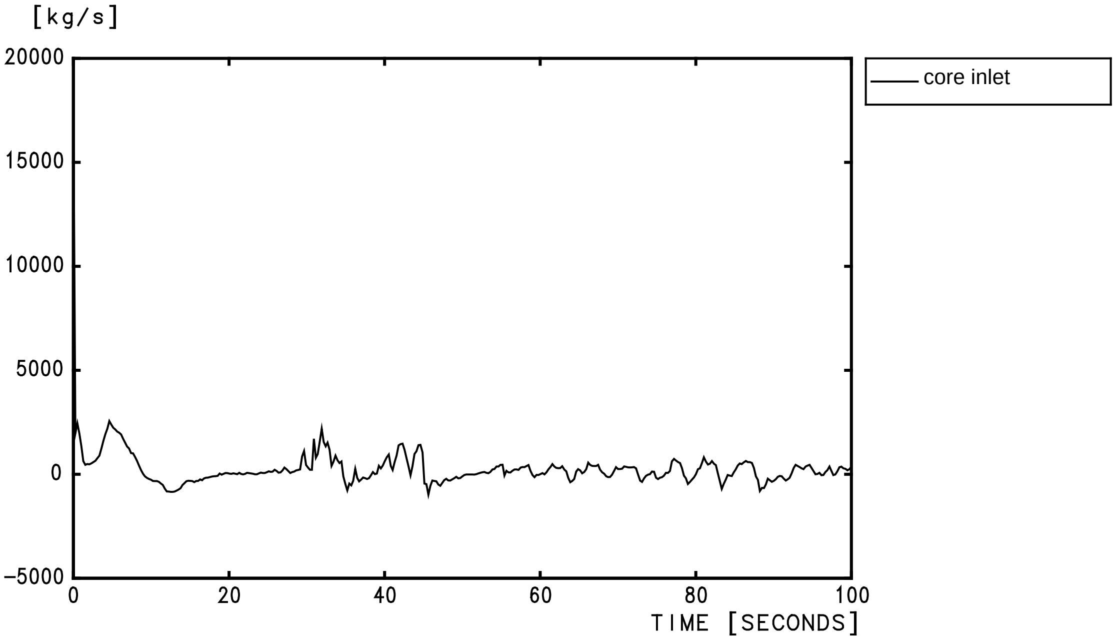
SECTION 16.4.1 - FIGURE C 14: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE BREAK



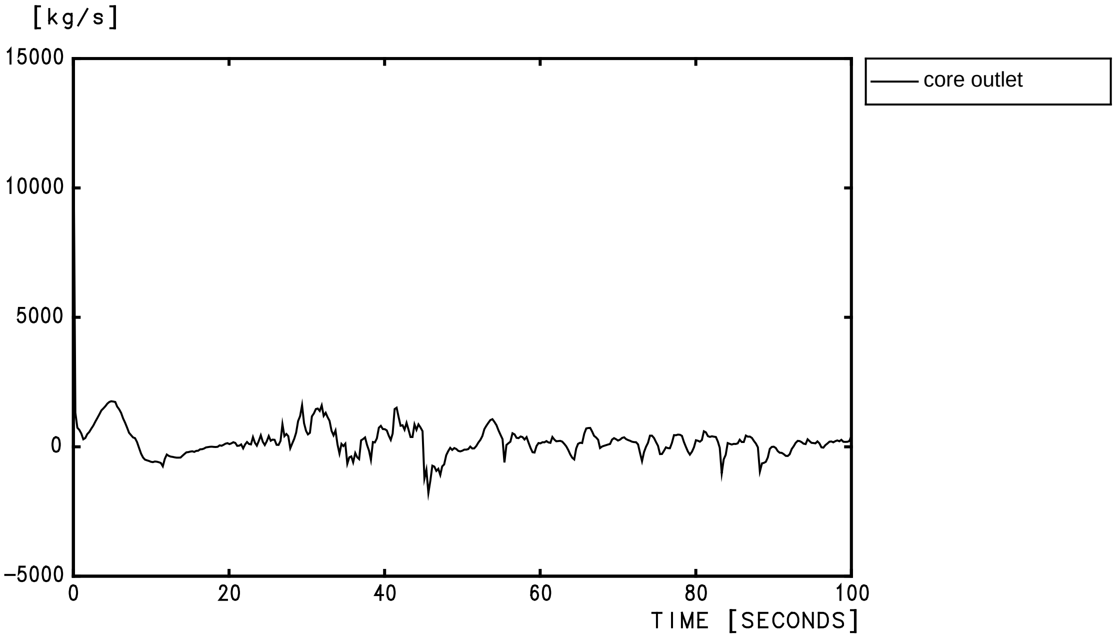
SECTION 16.4.1 - FIGURE C 15: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – QUENCH FRONT LEVEL IN HOT ASSEMBLIES



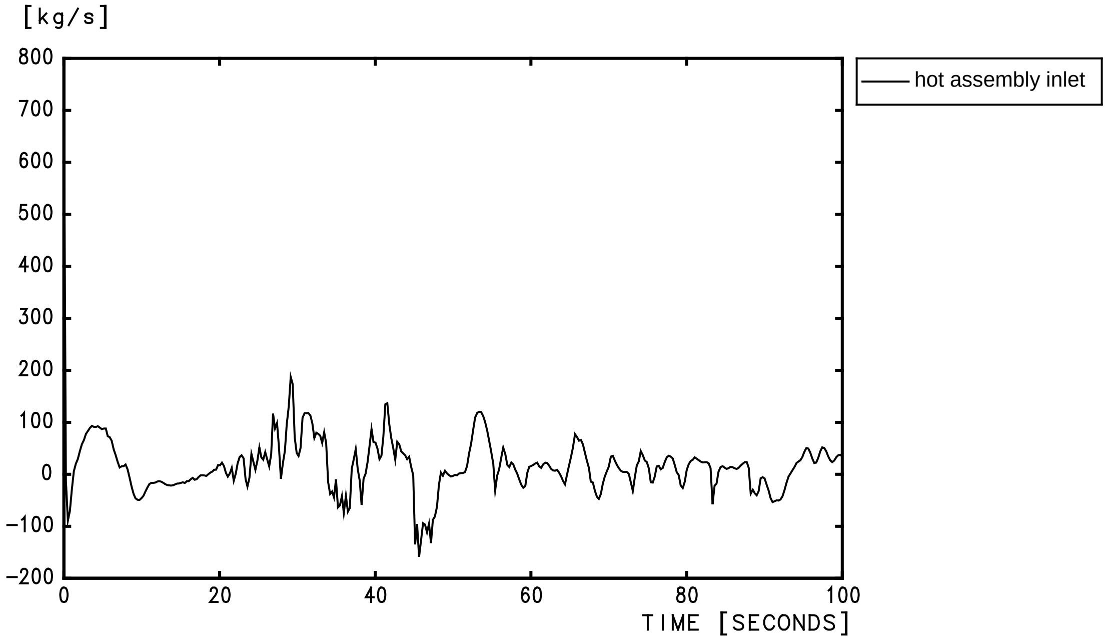
SECTION 16.4.1 - FIGURE C 16: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – MAXIMUM CLAD TEMPERATURES FOR HOT ASSEMBLIES



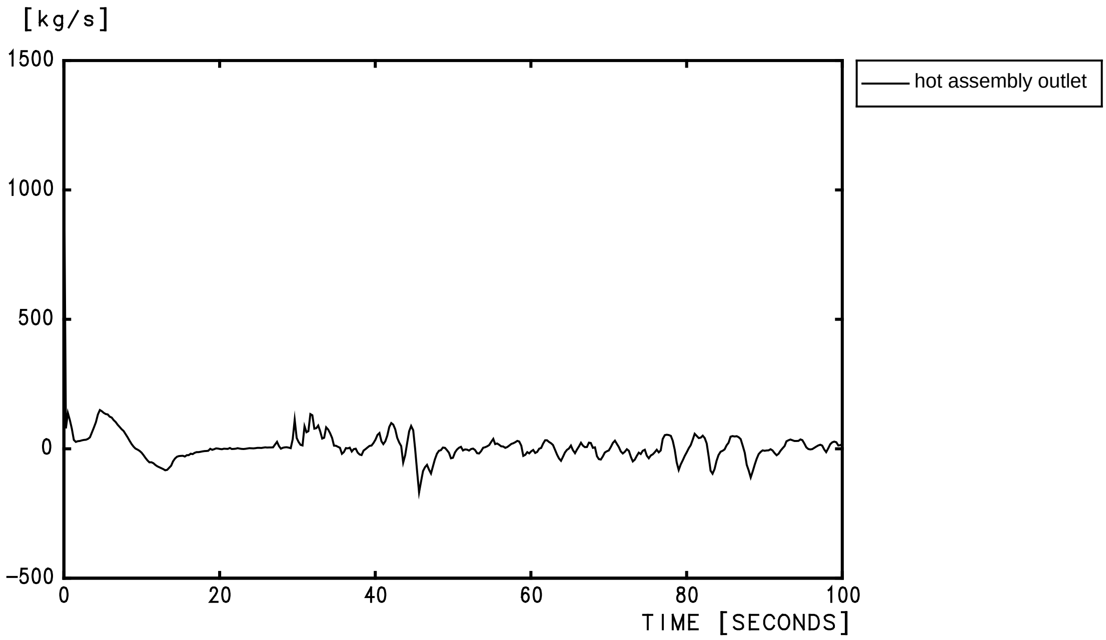
SECTION 16.4.1 - FIGURE C 17: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE CORE INLET



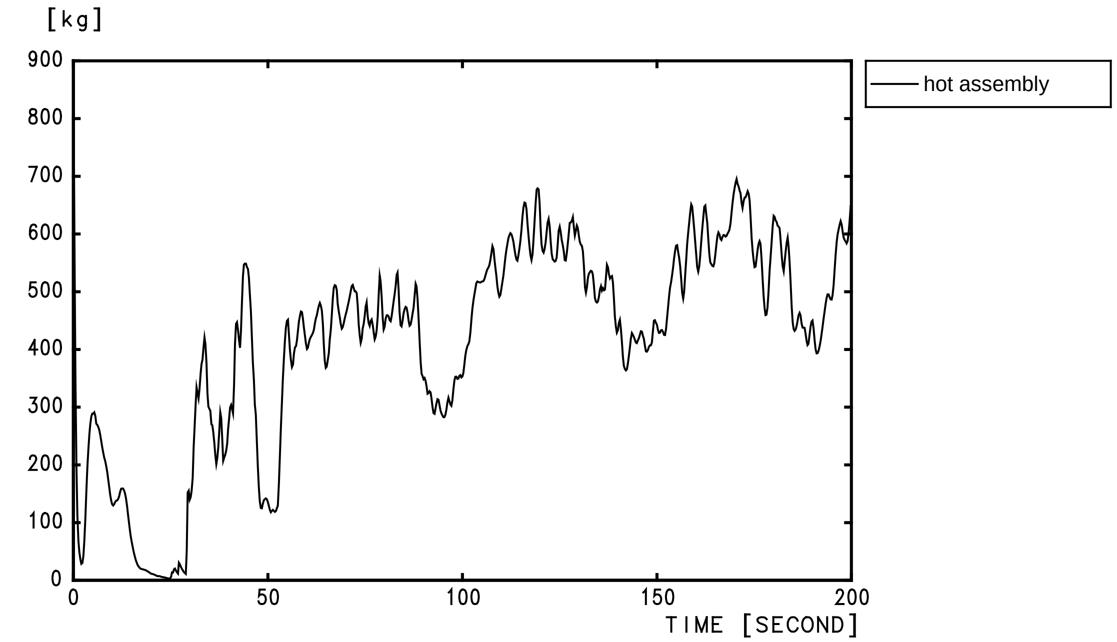
SECTION 16.4.1 - FIGURE C 18: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE CORE OUTLET



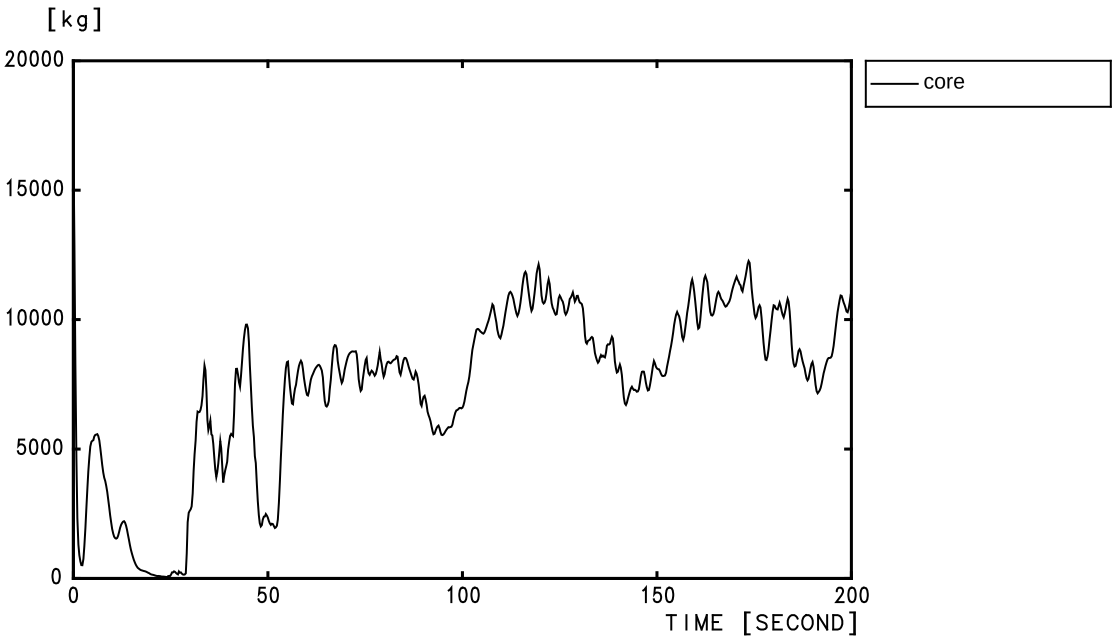
SECTION 16.4.1 - FIGURE C 19: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE HOT ASSEMBLY INLET



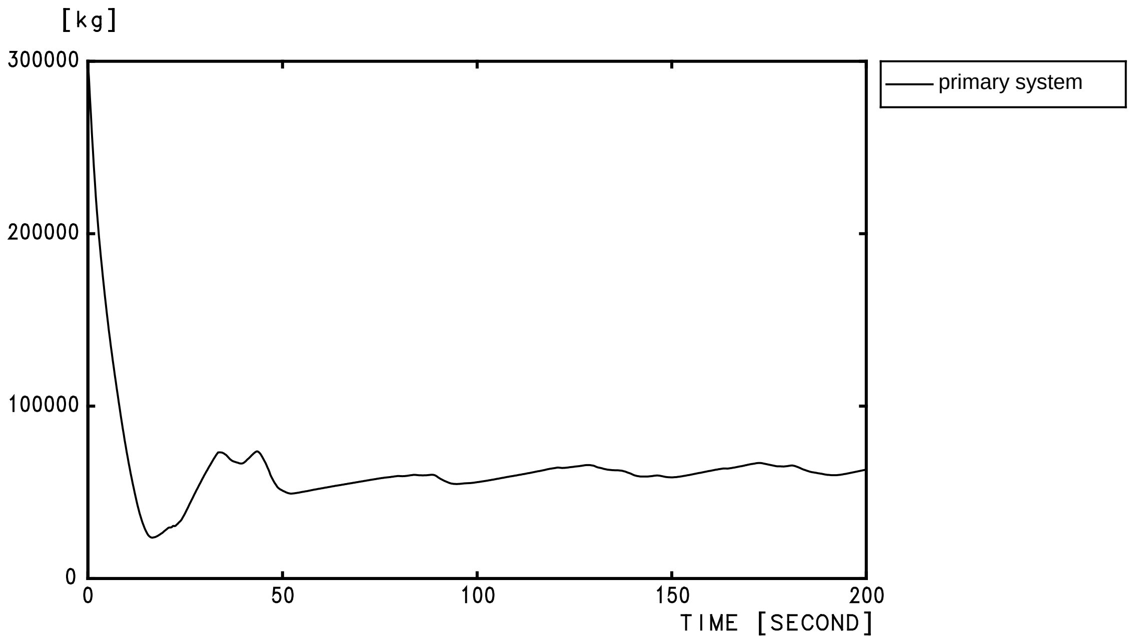
SECTION 16.4.1 - FIGURE C 20: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE HOT ASSEMBLY OUTLET



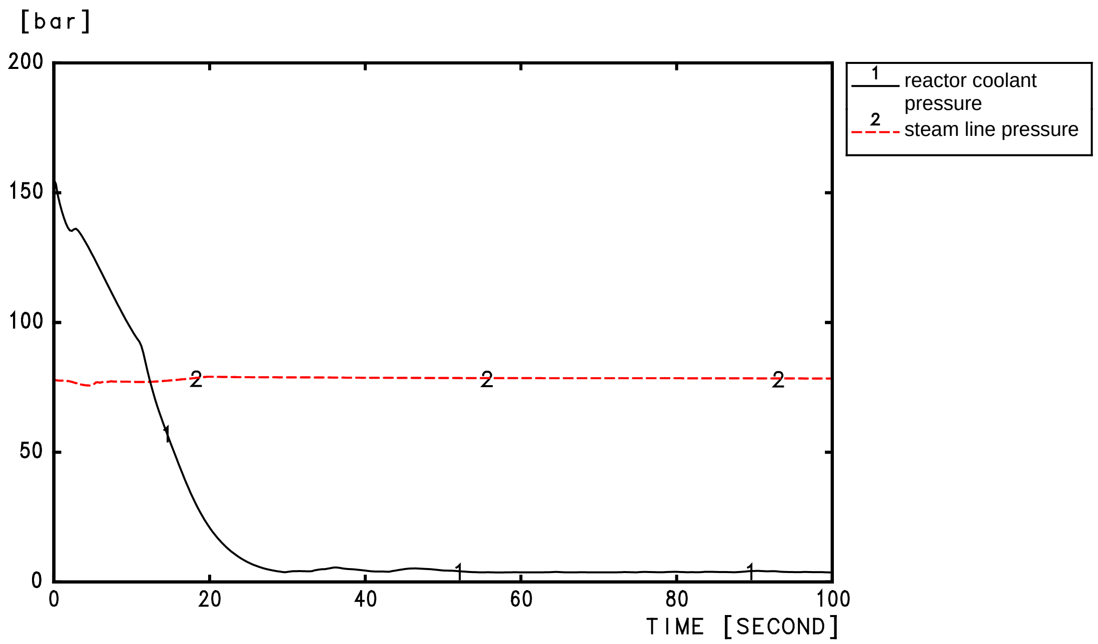
SECTION 16.4.1 - FIGURE C 21: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – LIQUID MASS IN THE HOT ASSEMBLY



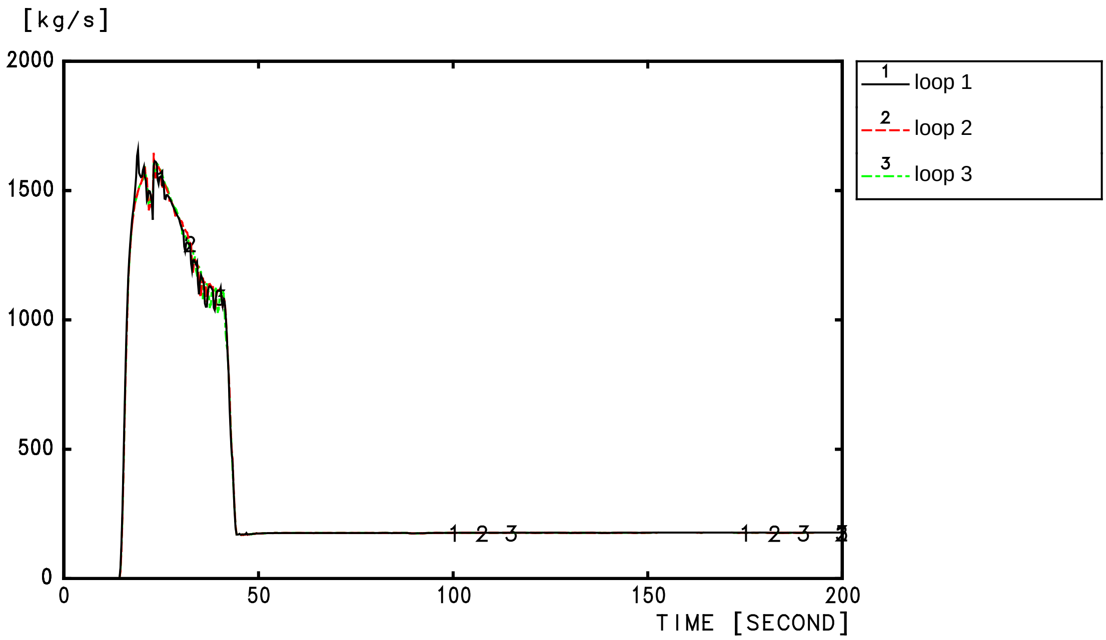
SECTION 16.4.1 - FIGURE C 22: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – LIQUID MASS IN THE CORE



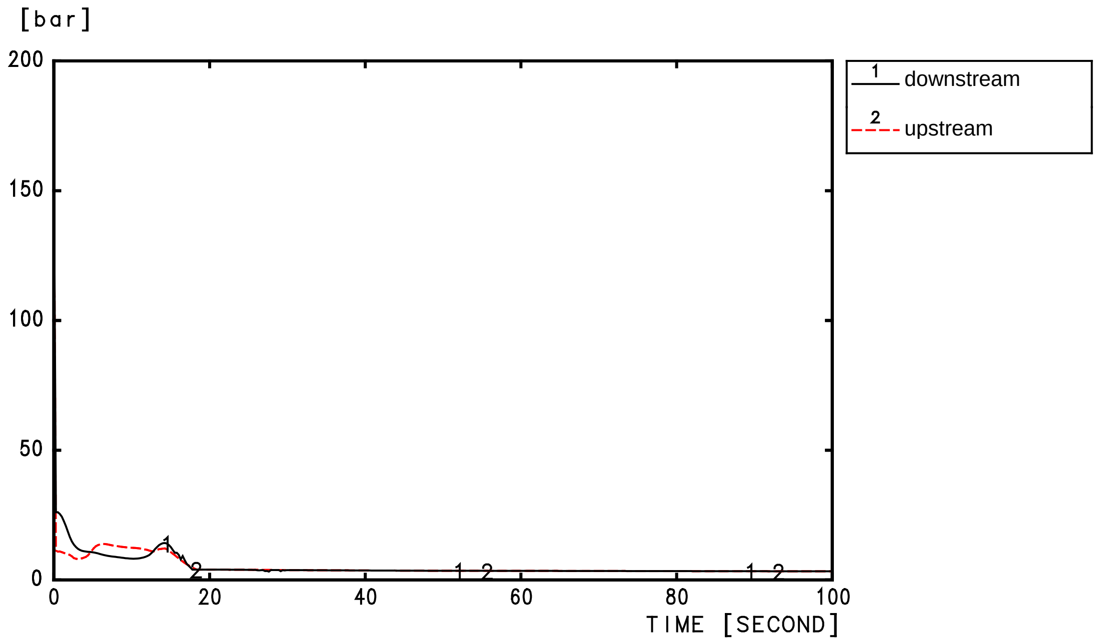
SECTION 16.4.1 - FIGURE C 23: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL MASS IN THE PRIMARY SYSTEM



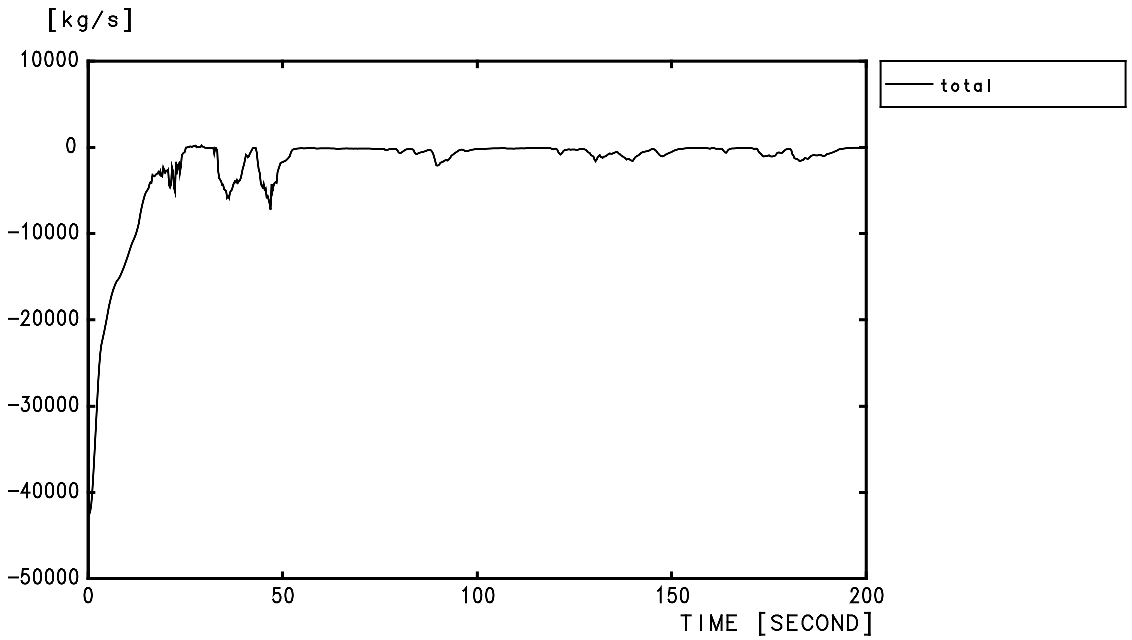
SECTION 16.4.1 - FIGURE C 24: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – REACTOR COOLANT PRESSURE AND STEAM LINE PRESSURE



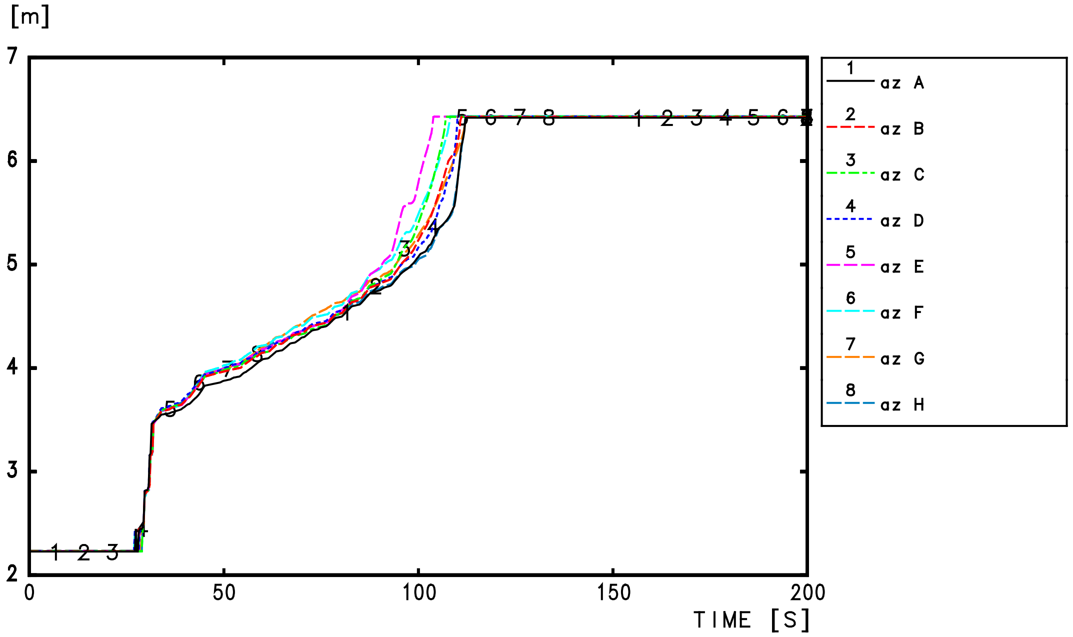
SECTION 16.4.1 - FIGURE C 25: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL RIS [SIS] FLOW RATE (ACCUMULATORS AND SI)



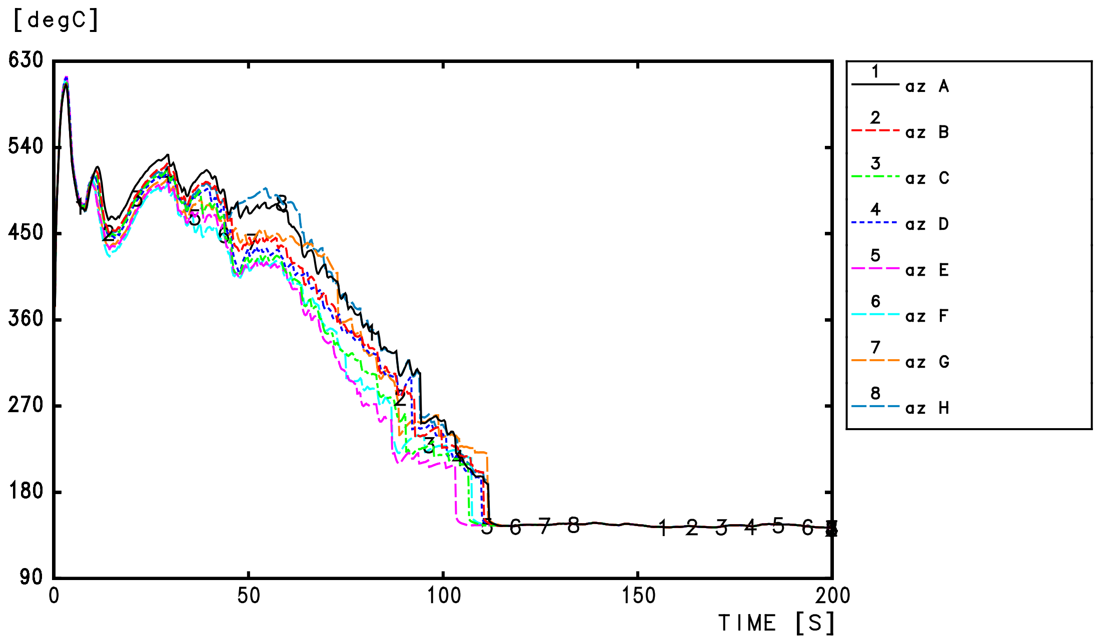
SECTION 16.4.1 - FIGURE C 26: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – PRESSURE AT THE BREAK



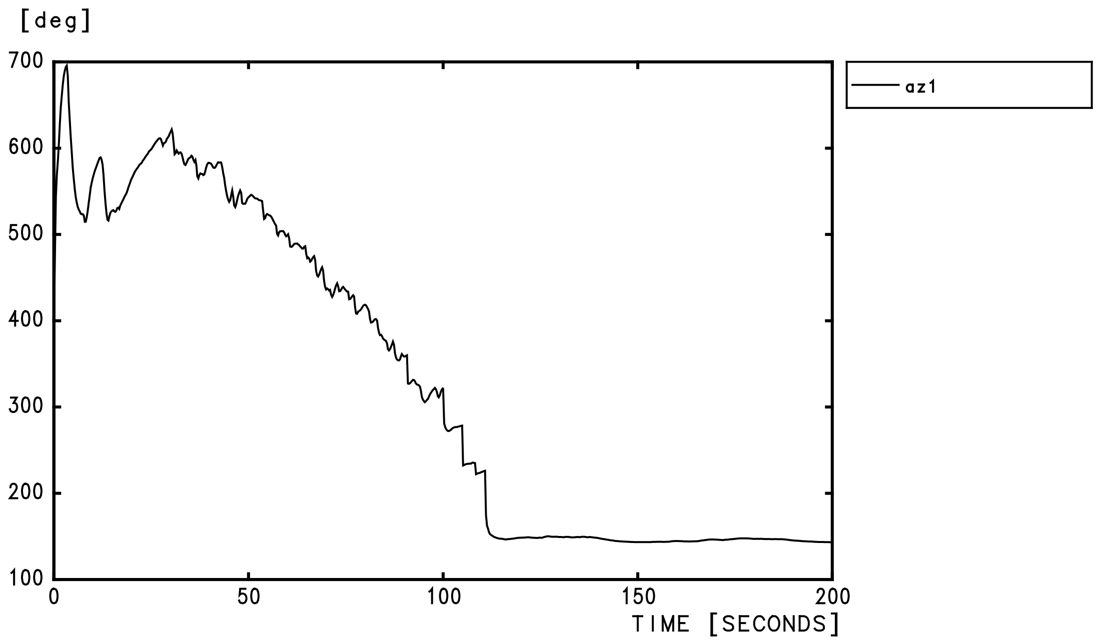
SECTION 16.4.1 - FIGURE C 27: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE BREAK



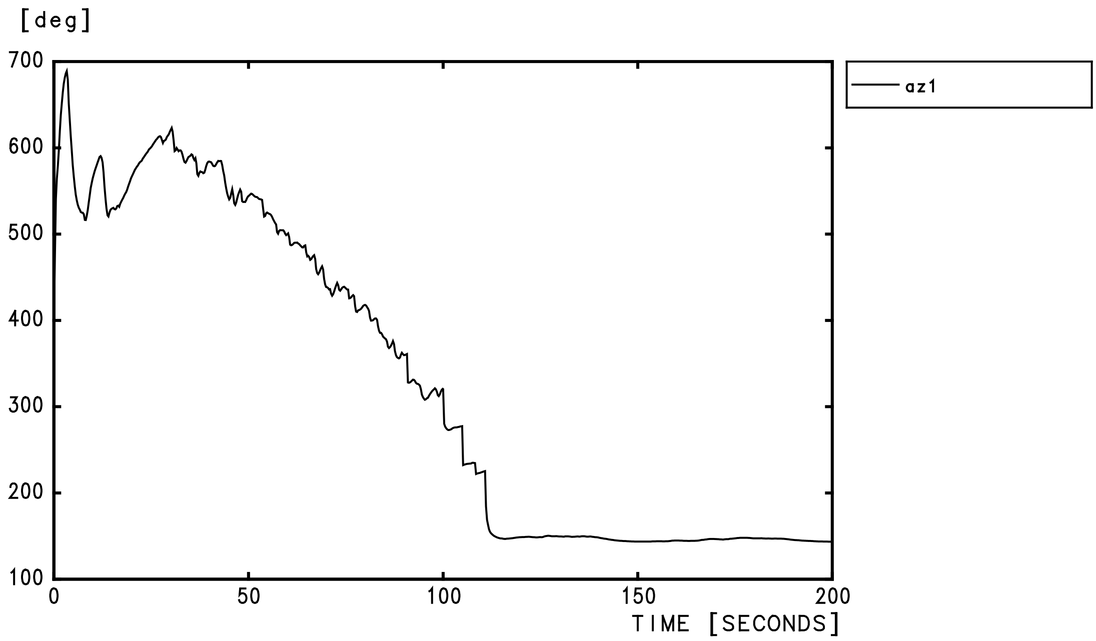
SECTION 16.4.1 - FIGURE C 28: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – QUENCH FRONT LEVEL IN HOT ASSEMBLIES



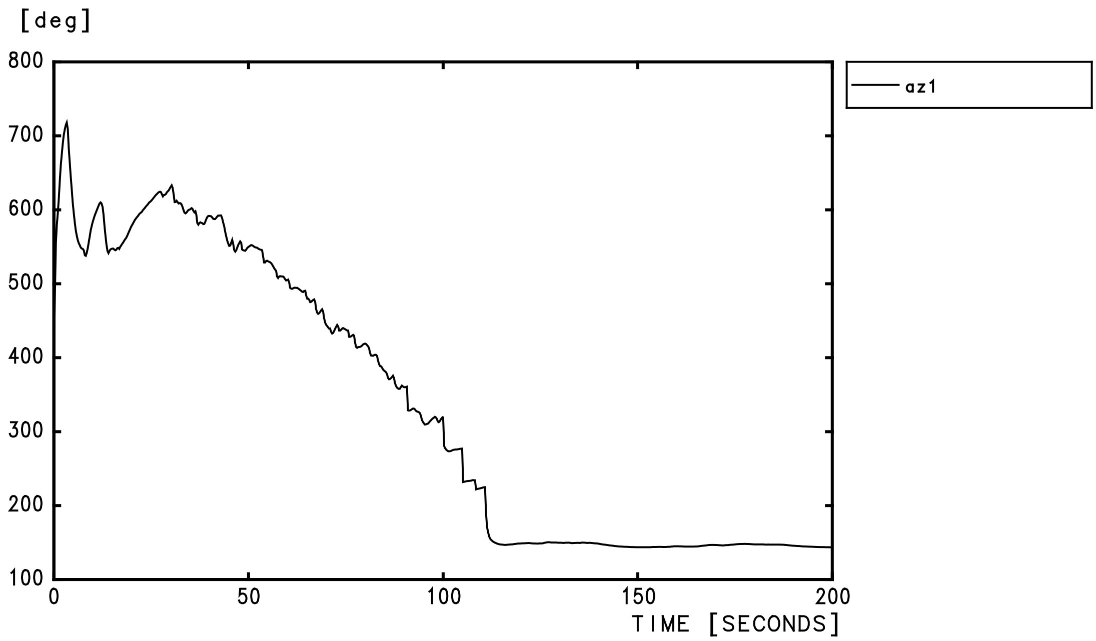
SECTION 16.4.1 - FIGURE C 29: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – MAXIMUM CLAD TEMPERATURES FOR HOT ASSEMBLIES



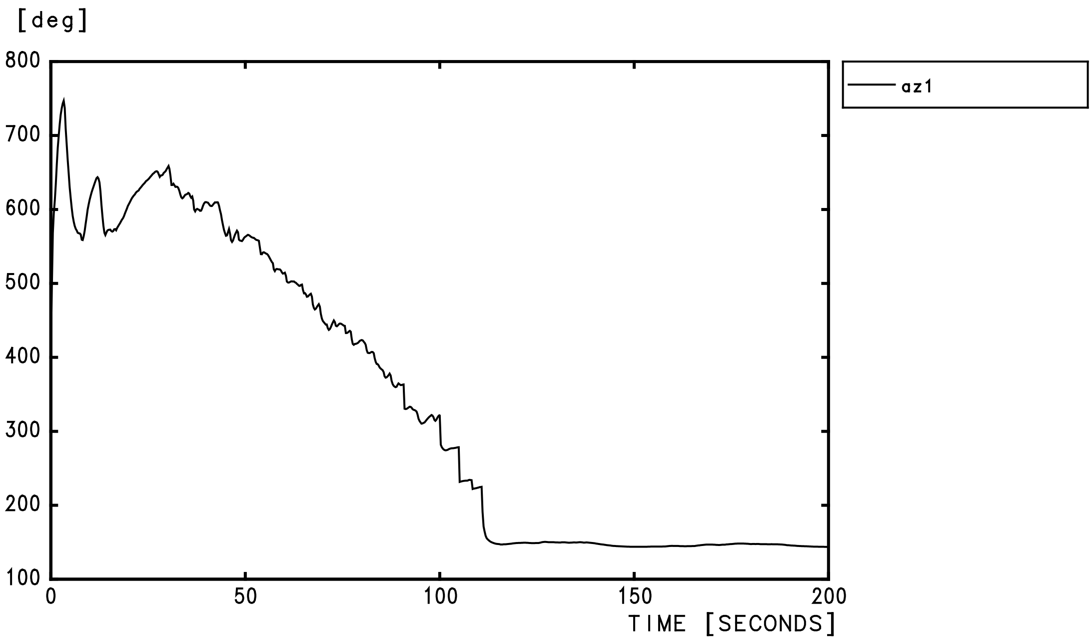
SECTION 16.4.1 - FIGURE C 30: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – BOL INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



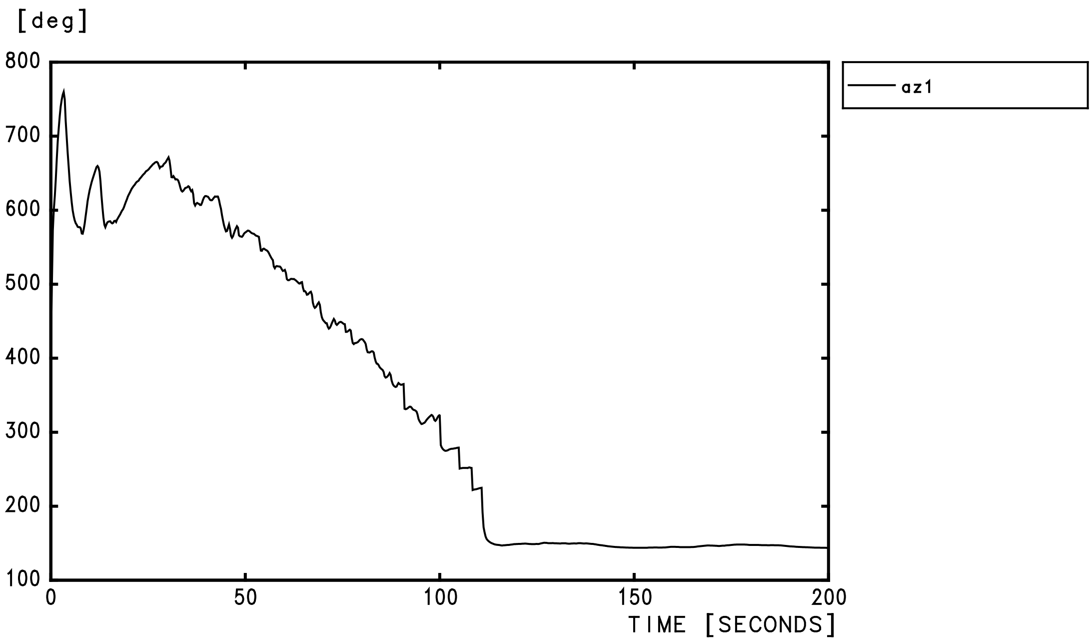
SECTION 16.4.1 - FIGURE C 31: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC1 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



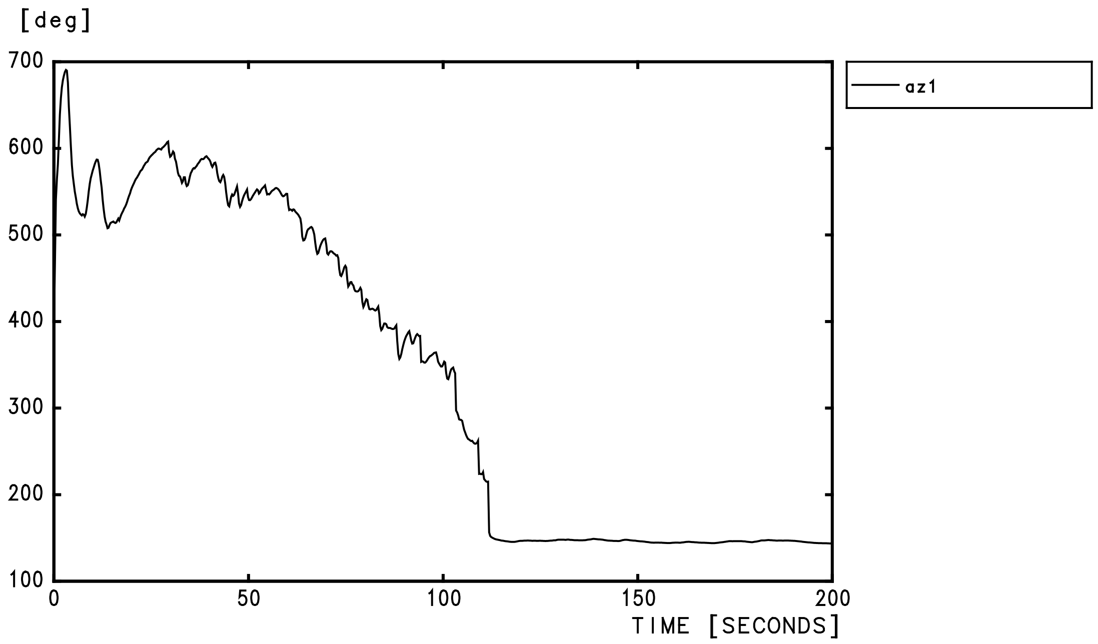
SECTION 16.4.1 - FIGURE C 32: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC2 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



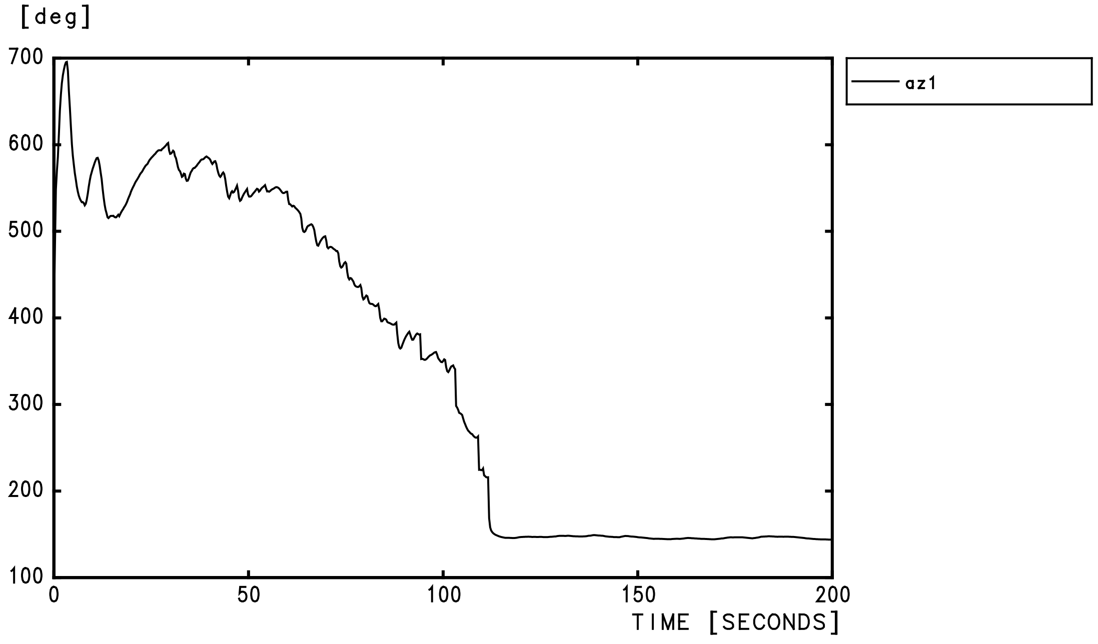
SECTION 16.4.1 - FIGURE C 33: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC3 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



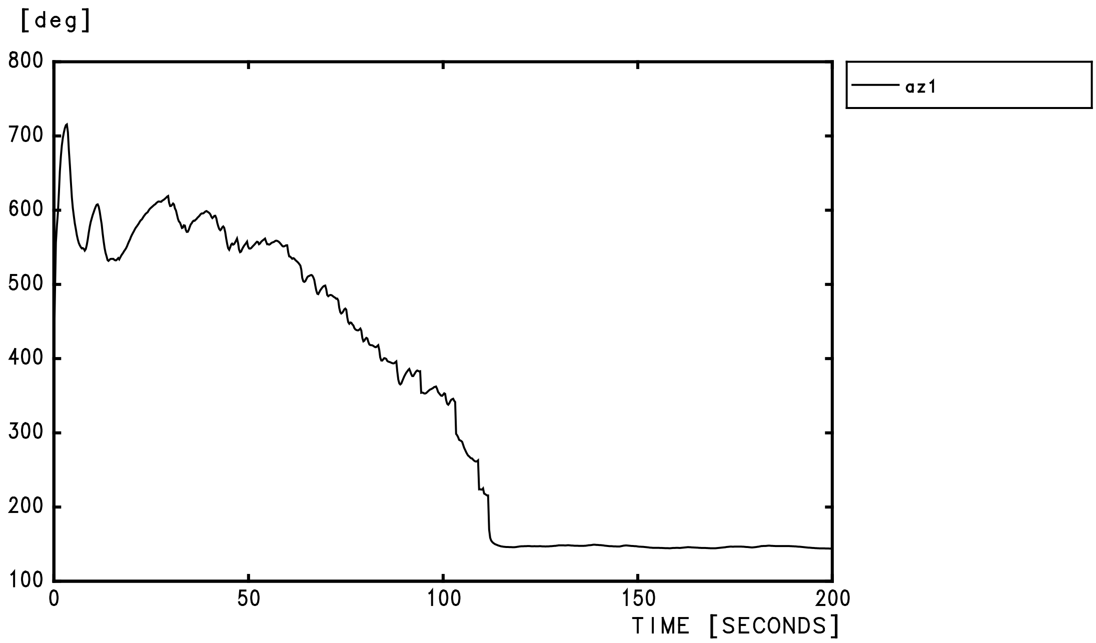
SECTION 16.4.1 - FIGURE C 34: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC4 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



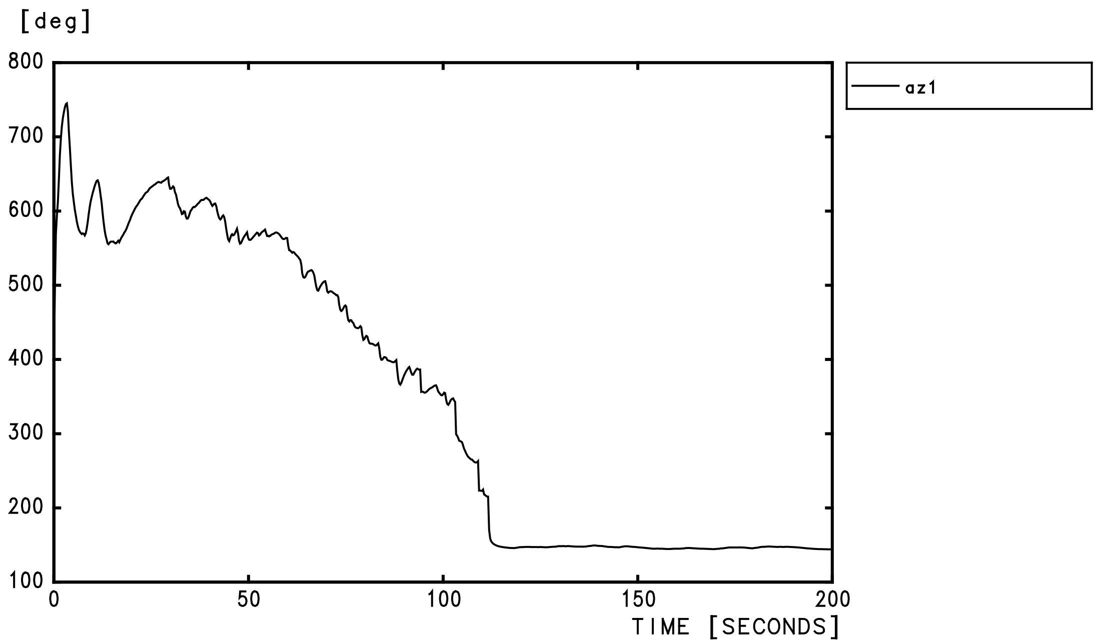
SECTION 16.4.1 - FIGURE C 35: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – BOL INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



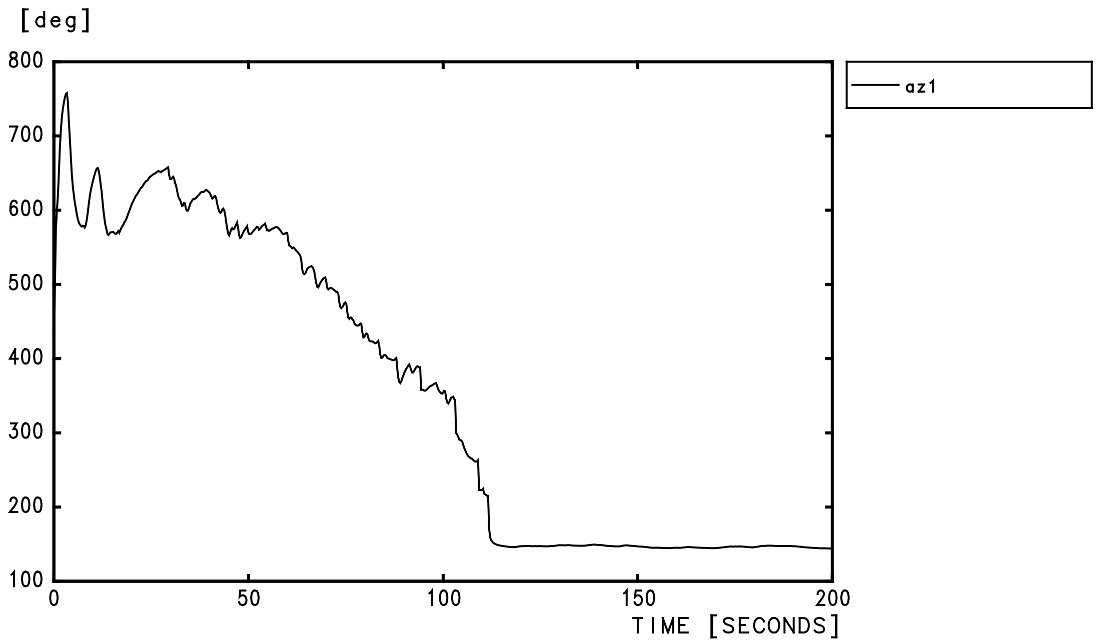
SECTION 16.4.1 - FIGURE C 36: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC1 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



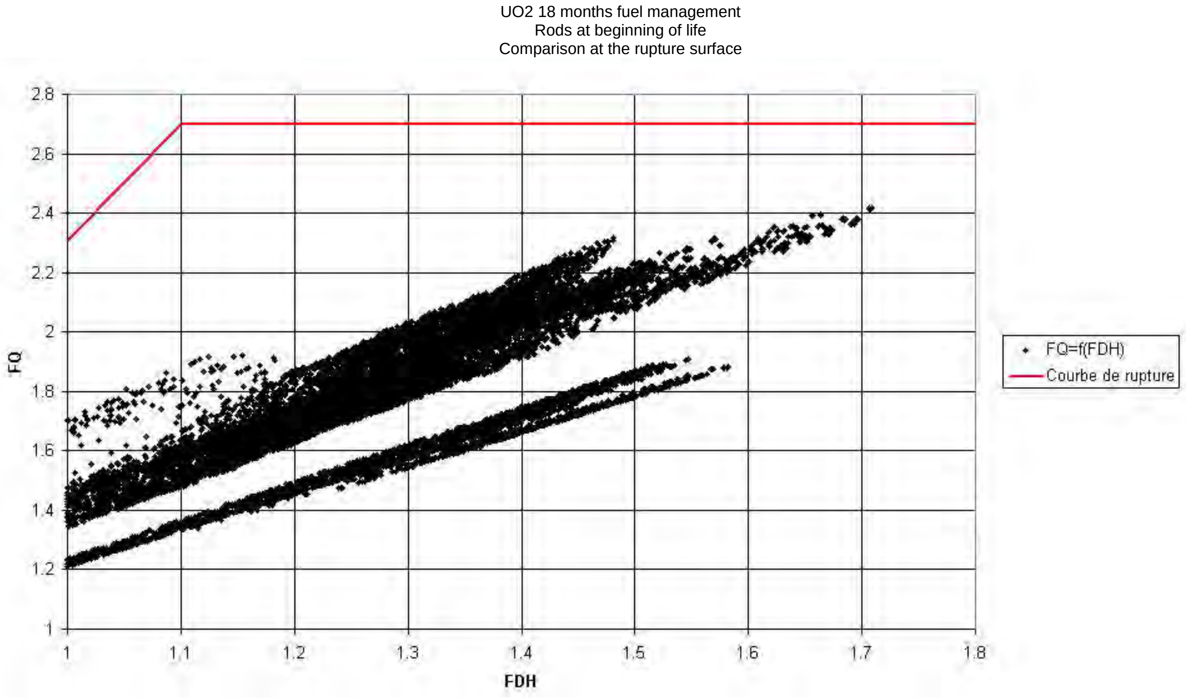
SECTION 16.4.1 - FIGURE C 37: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC2 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



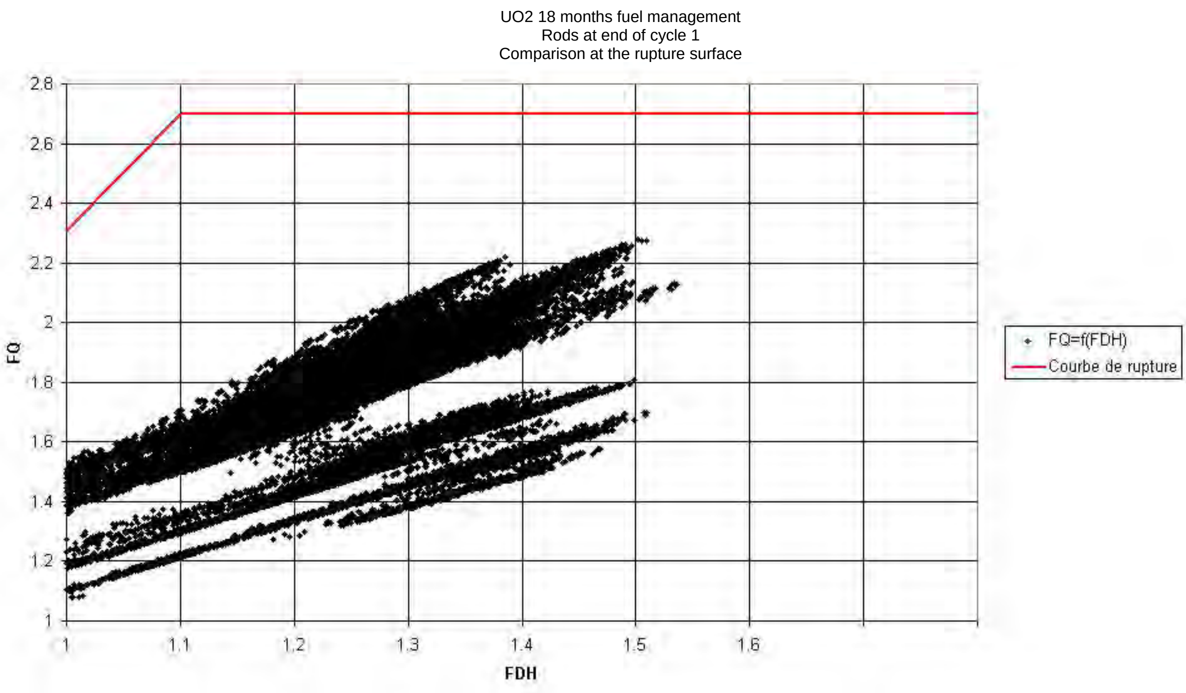
SECTION 16.4.1 - FIGURE C 38: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC3 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



SECTION 16.4.1 - FIGURE C 39: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC4 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE

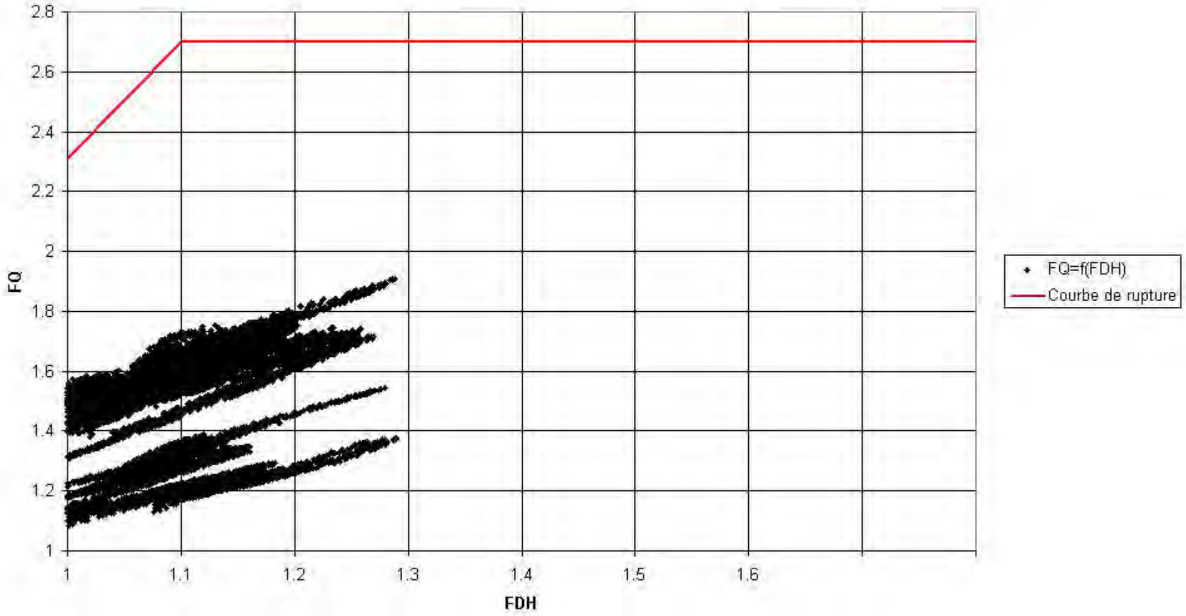


SECTION 16.4.1 - FIGURE C 40: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – RUPTURE CURVE – BOL



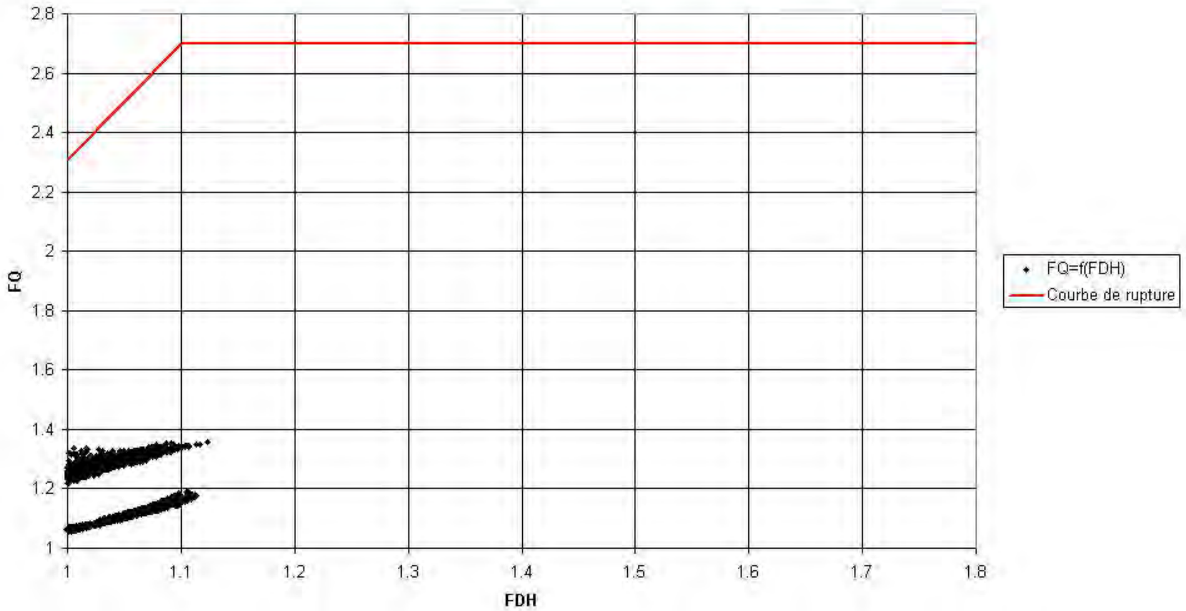
SECTION 16.4.1 - FIGURE C 41: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – RUPTURE CURVE – EOC1

UO2 18 months fuel management
Rods at end of cycle 2
Comparison at the rupture surface



SECTION 16.4.1 - FIGURE C 42: IN-OUT UO_2 18 MONTHS FUEL MANAGEMENT – MC – RUPTURE CURVE – EOC2

UO2 18 months fuel management
Rods at end of cycle 3
Comparison at the rupture surface



SECTION 16.4.1 - FIGURE C 43: IN-OUT UO2 18 MONTHS FUEL MANAGEMENT – MC – RUPTURE CURVE – EOC3

SECTION 16.4.1 - TABLE D 1: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – FUEL CALCULATION POINTS

Core cycle time	Rod burn-up
Beginning of life (BOL)	150 MWd/tU
End of Cycle 1 (EOC1)	32,300 MWd/tU
End of Cycle 2 (EOC2)	57,900 MWd/tU
End of Cycle 3 (EOC3)	71,800 MWd/tU

SECTION 16.4.1 - TABLE D 2: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – PROPERTIES OF HOT ASSEMBLIES

Hot assembly	F∆H	FQ	BU
HA1	1.39	2.14	EOC1
HA2	1.19	1.77	EOC2

SECTION 16.4.1 - TABLE D 3: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – MAIN EVENTS OF THE TRANSIENT

Event	Time at which the event occurs
RT signal	3.6 s
SI signal	7.1 s
Start of accumulator injection	14.2 s
Start of safety injection system injection	23.0 s
Start of reflood	24.3 s
End of reflood	125.9 s
Duration of reflood	101.6 s
End of accumulator injection	44.0 s

SECTION 16.4.1 - TABLE D 4: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – PEAK CLAD TEMPERATURES

	T1	T2	T3
Time (s)	3.2	27.8	51.3
Mesh	7	7	7
Value (°C)	666	589	518

SECTION 16.4.1 - TABLE D 5: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – MAIN EVENTS OF THE TRANSIENT

Event	Time at which the event occurs
RT signal	3.6 s
SI signal	7.1 s
Start of accumulator injection	14.2 s
Start of safety injection system injection	23.0 s
Start of reflood	23.9 s
End of reflood	113.3 s
Duration of reflood	89.4 s
End of accumulator injection	44.1 s

SECTION 16.4.1 - TABLE D 6: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – PEAK CLAD TEMPERATURES

	T1	T2	T3
Time (s)	3.2	29.6	49.5
Mesh	6	7	7
Value (°C)	613	541	513

SECTION 16.4.1 - TABLE D 7: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – PROPERTIES OF THE INDICATOR RODS

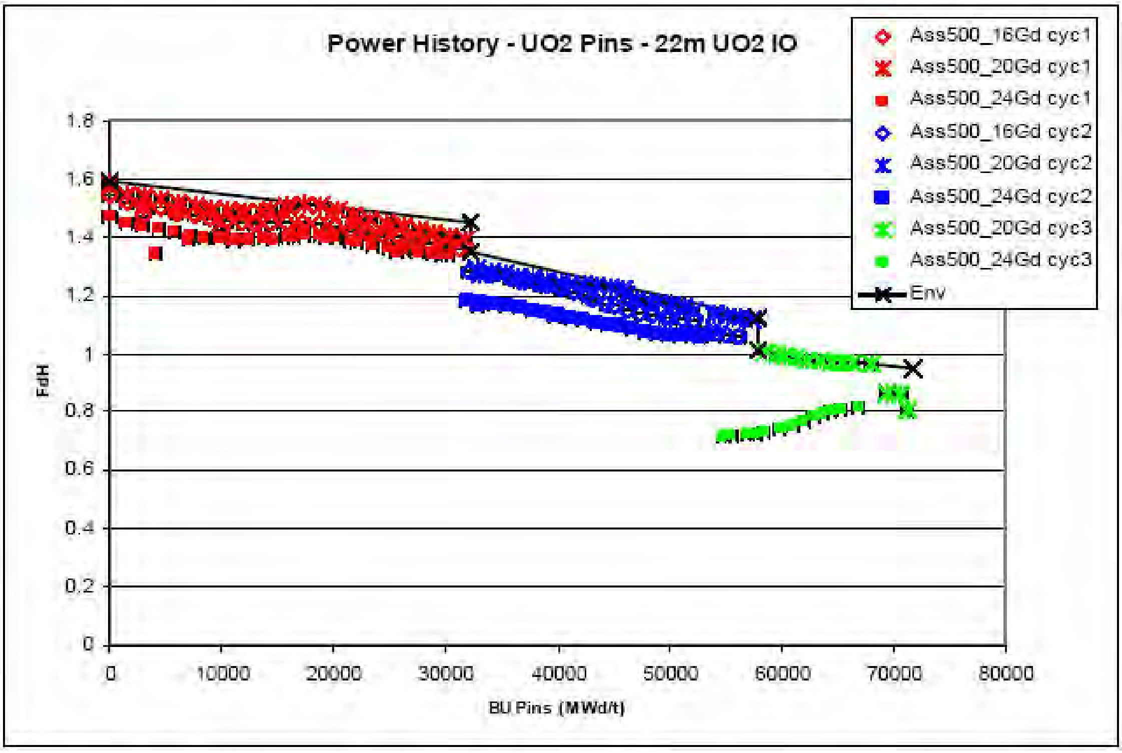
Burn-up	FQ	FH	Pressure in the gap (bar)	Maximum linear power density (W/cm)	Maximum initial pellet temperature (°C)
BOL	2.32	1.56	56.1	379.42	940
EOC1	2.32	1.56	72.2	379.42	944
EOC2	2.32	1.56	93.4	379.42	1045
EOC3	2.32	1.56	108.1	379.42	1133

SECTION 16.4.1 - TABLE D 8: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – RESULTS OBTAINED FOR INDICATOR RODS

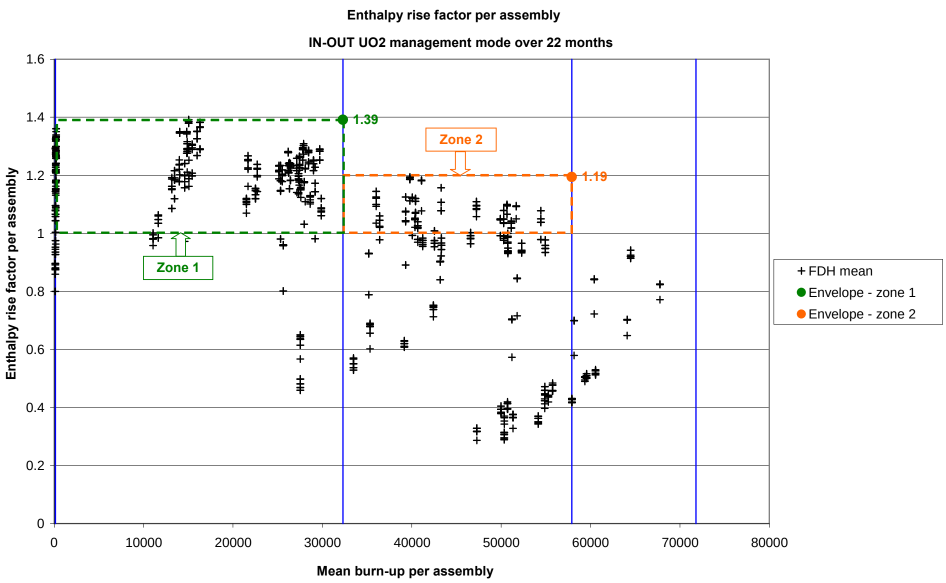
Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Rupture
BOL	693	612	541	0.34	NO
EOC1	690	616	544	0.33	NO
EOC2	731	637	553	0.36	NO
EOC3	755	660	566	0.43	NO

SECTION 16.4.1 - TABLE D 9: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – RESULTS OBTAINED FOR INDICATOR RODS

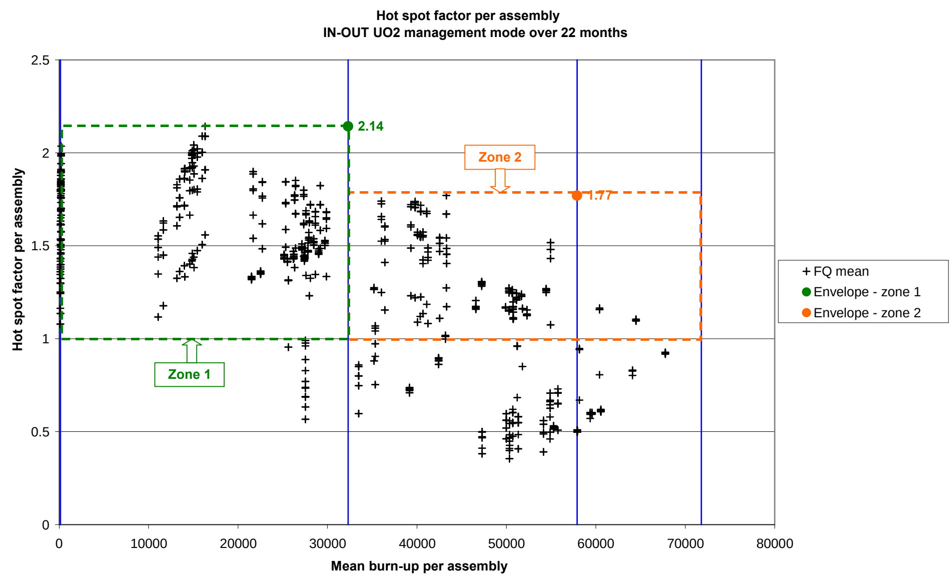
Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Rupture
BOL	683	600	575	0.34	NO
EOC1	682	605	578	0.33	NO
EOC2	724	625	589	0.36	NO
EOC3	749	647	603	0.40	NO



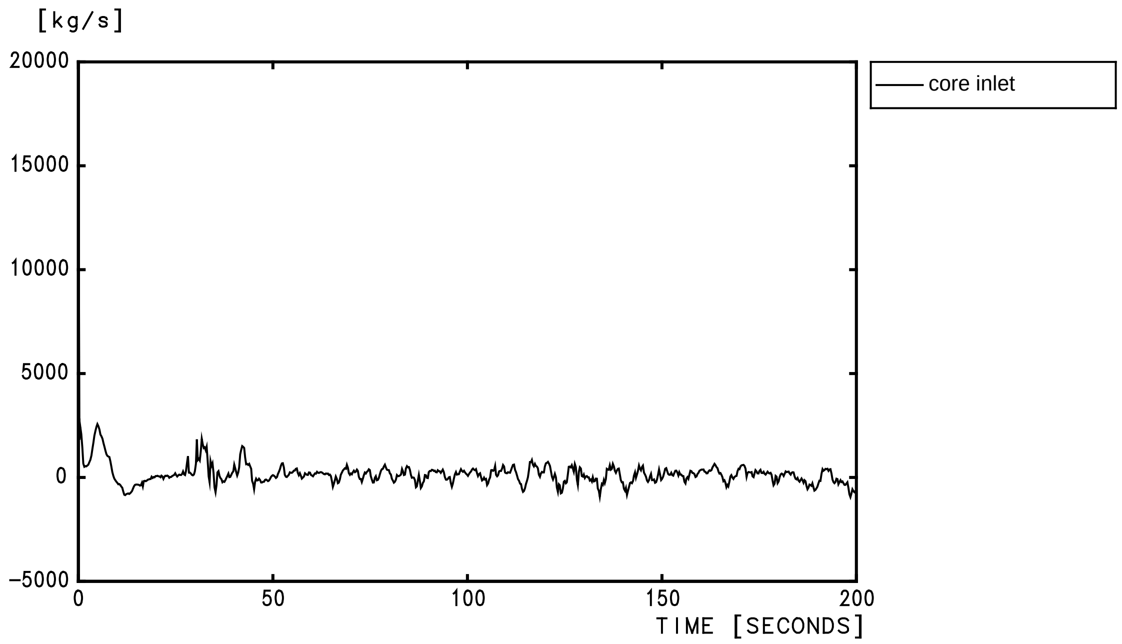
SECTION 16.4.1 - FIGURE D 1: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – POWER HISTORY OF UO_2 RODS



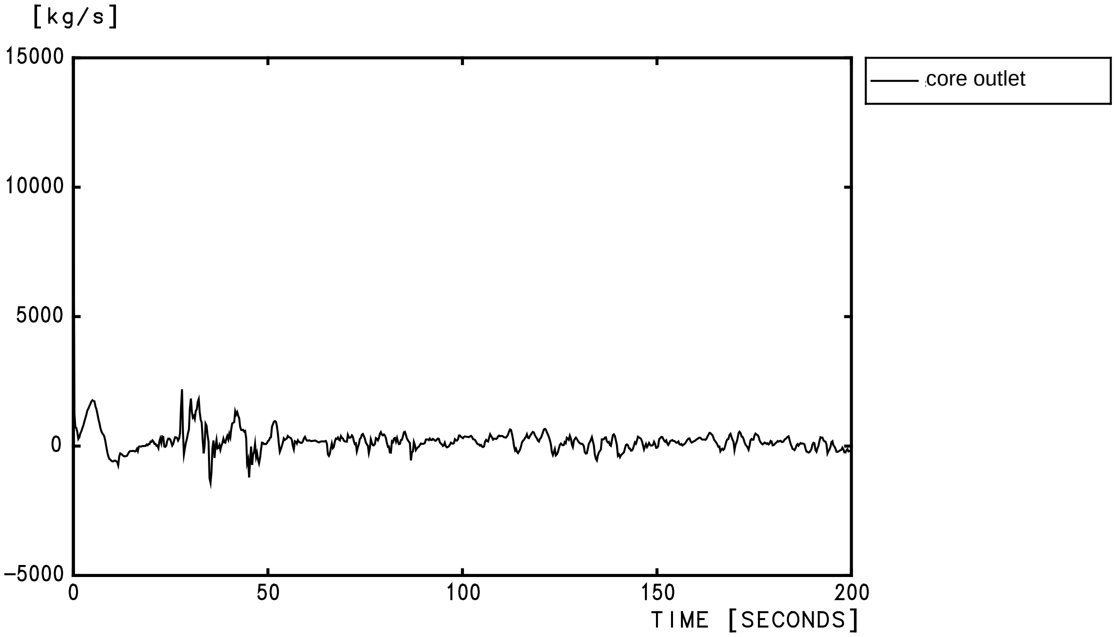
SECTION 16.4.1 - FIGURE D 2: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – FΔH FOR ASSEMBLIES AS A FUNCTION OF BURN-UP



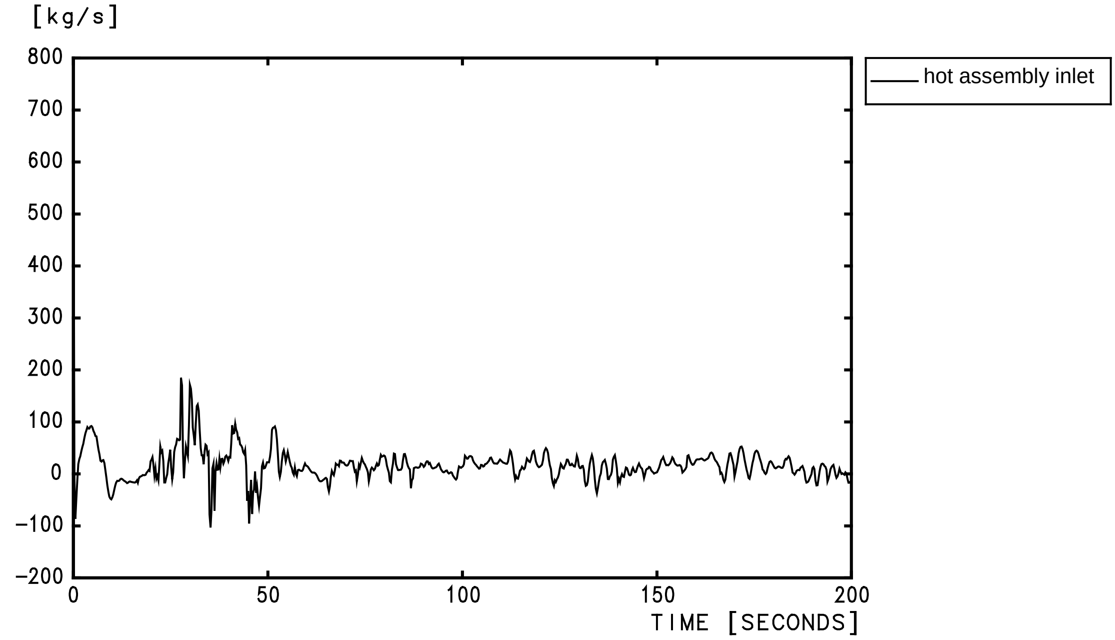
SECTION 16.4.1 - FIGURE D 3: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – FQ FOR ASSEMBLIES AS A FUNCTION OF BURN-UP



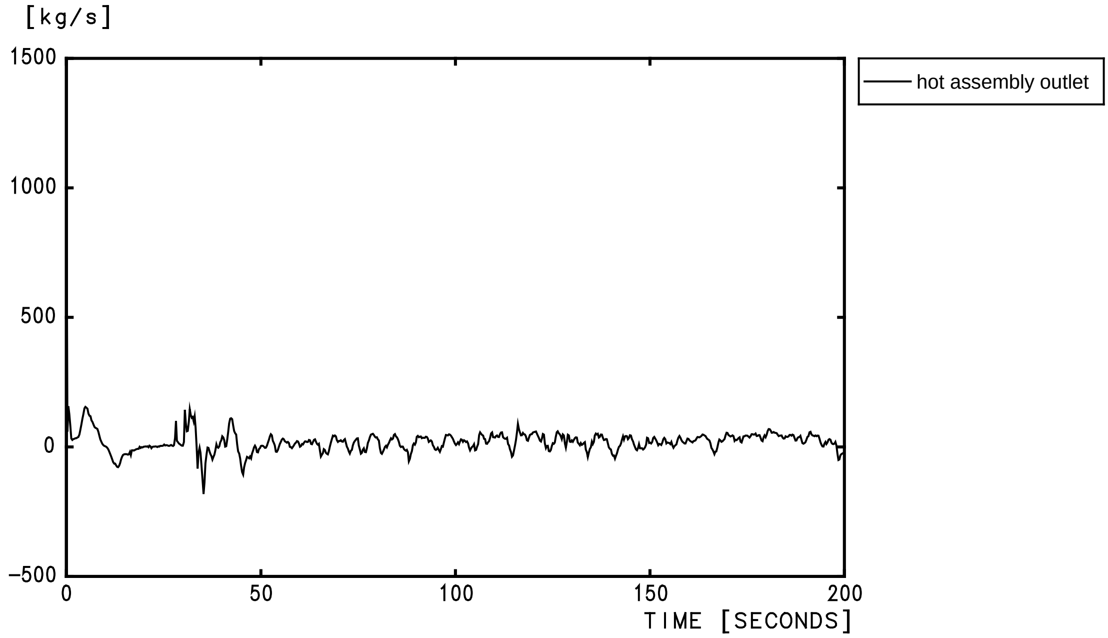
SECTION 16.4.1 - FIGURE D 4: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE CORE INLET



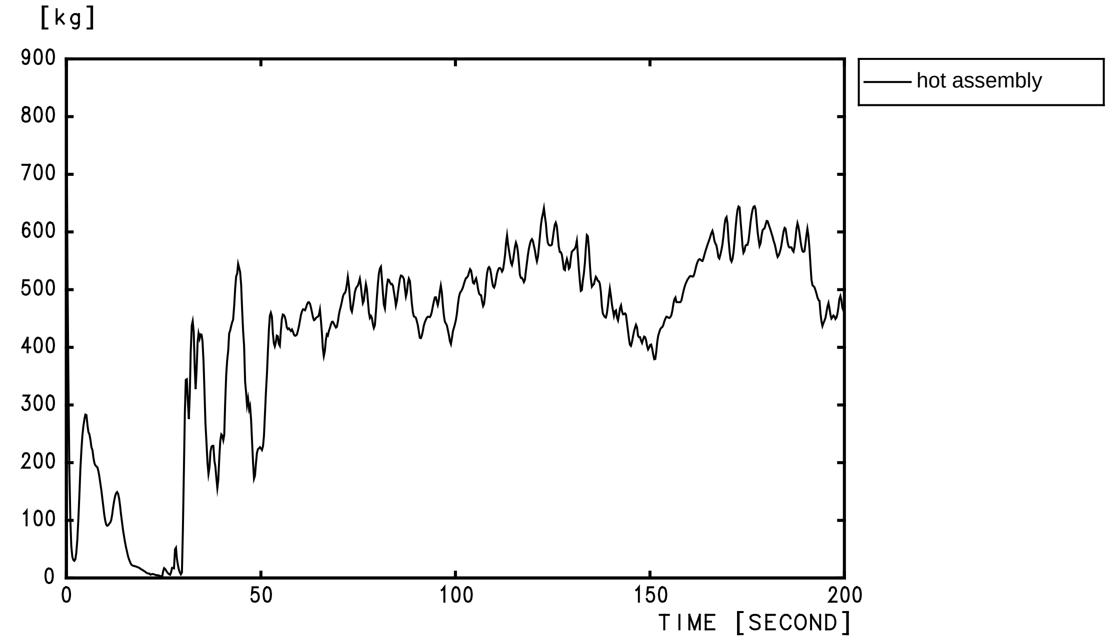
SECTION 16.4.1 - FIGURE D 5: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE CORE OUTLET



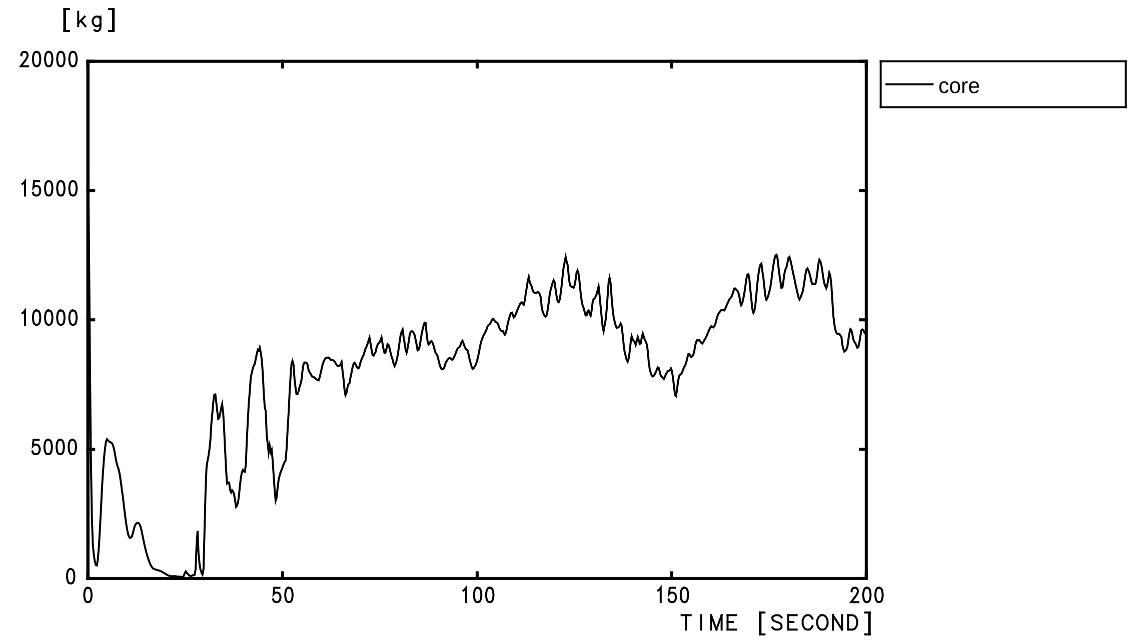
SECTION 16.4.1 - FIGURE D 6: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE HOT ASSEMBLY INLET



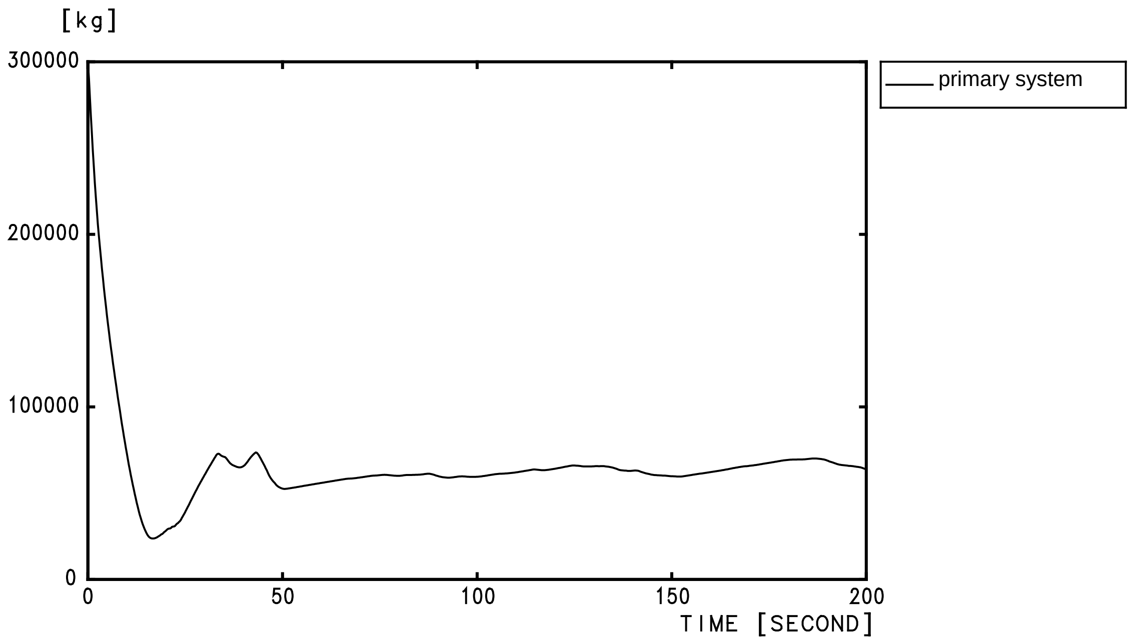
SECTION 16.4.1 - FIGURE D 7: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE HOT ASSEMBLY OUTLET



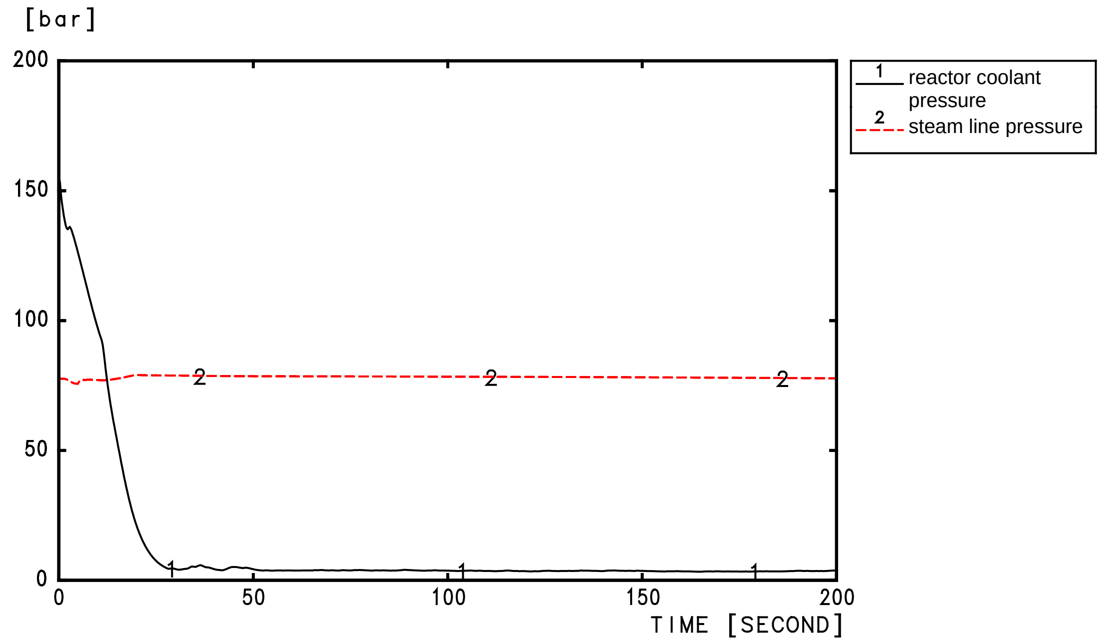
SECTION 16.4.1 - FIGURE D 8: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – LIQUID MASS IN THE HOT ASSEMBLY



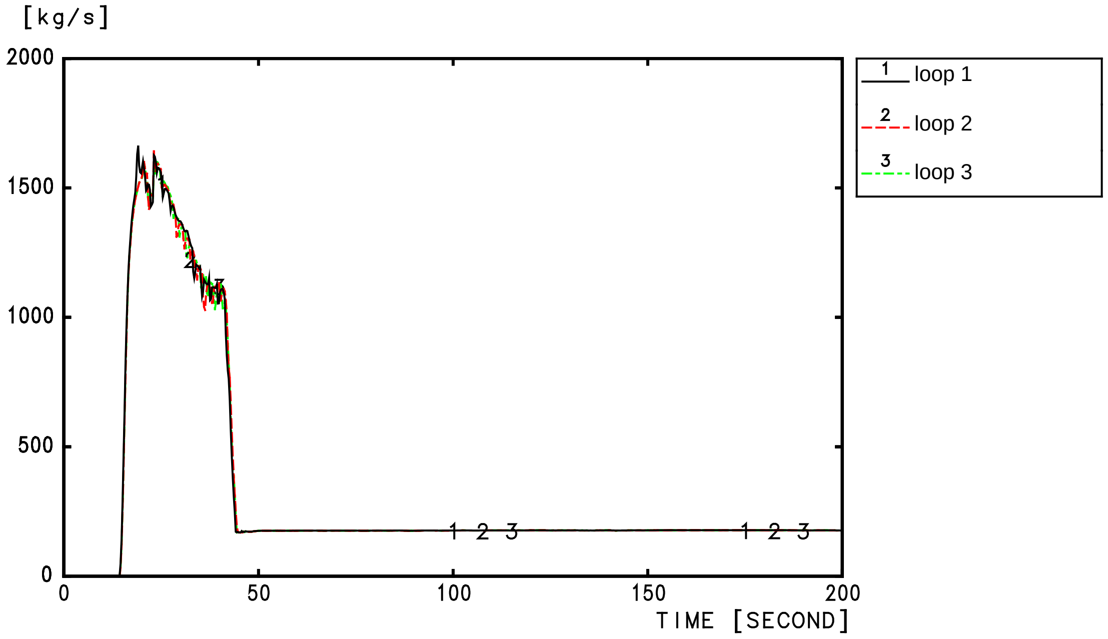
SECTION 16.4.1 - FIGURE D 9: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – LIQUID MASS IN THE CORE



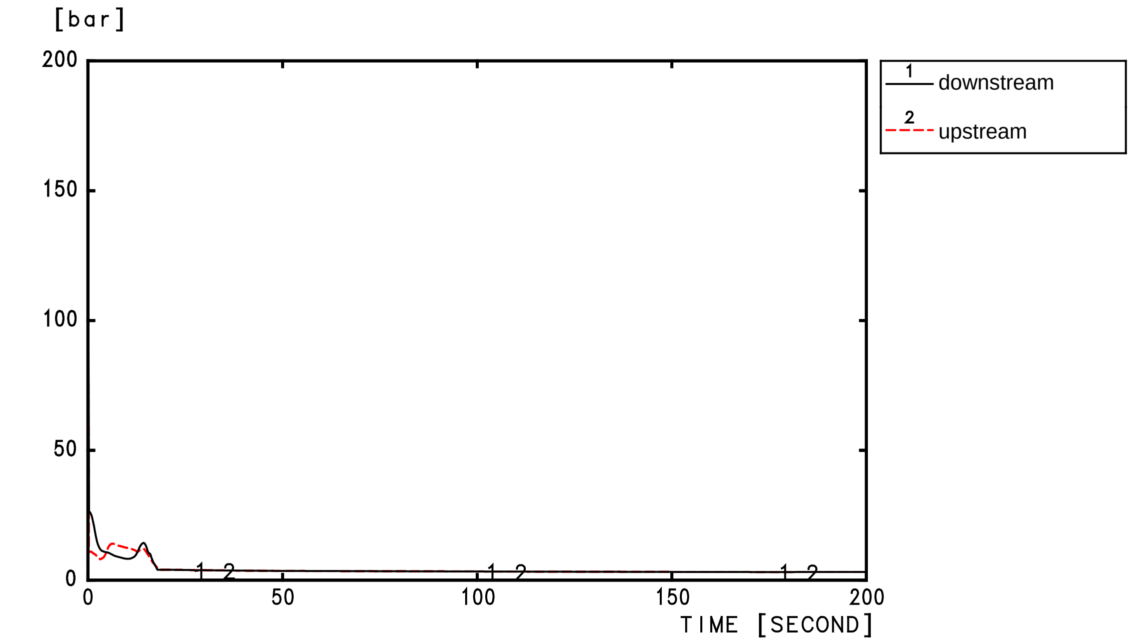
SECTION 16.4.1 - FIGURE D 10: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL MASS IN THE PRIMARY SYSTEM



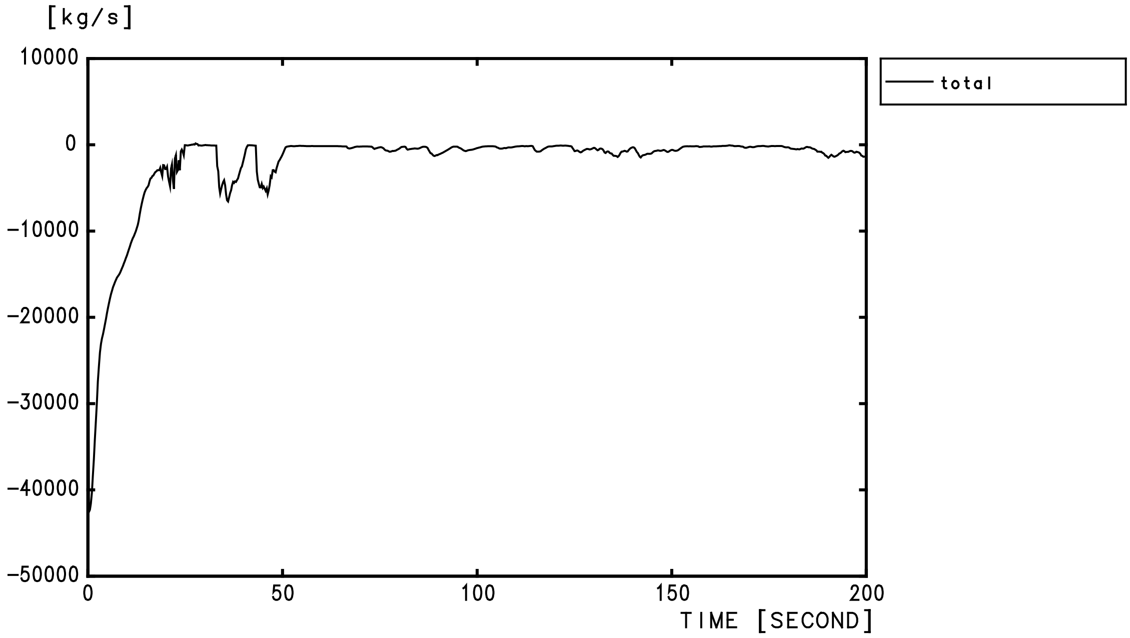
SECTION 16.4.1 - FIGURE D 11: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – REACTOR COOLANT PRESSURE AND STEAM LINE PRESSURE



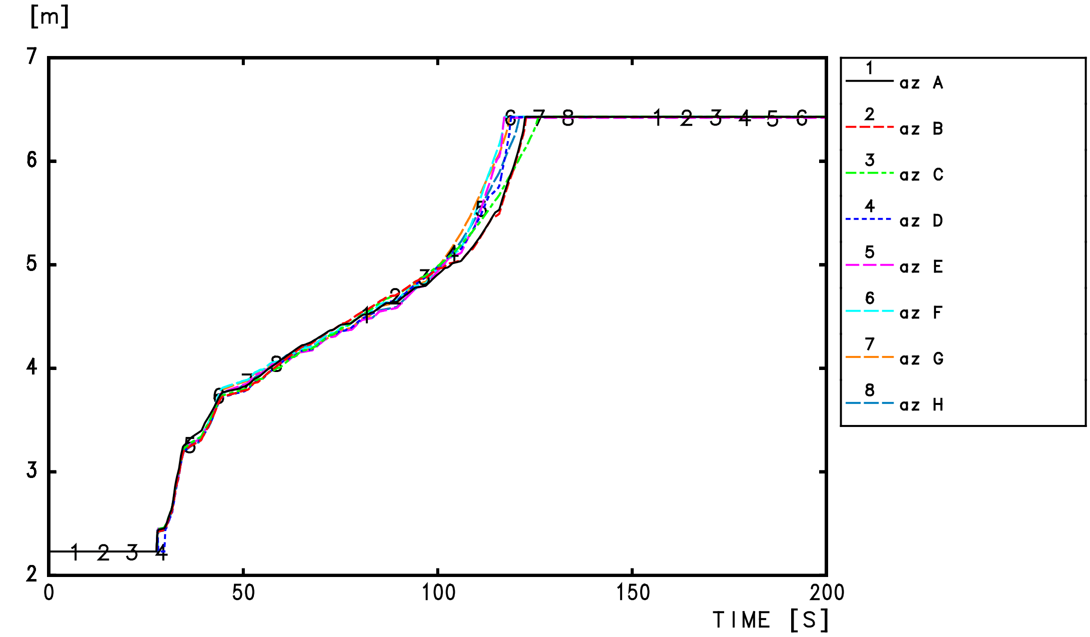
SECTION 16.4.1 - FIGURE D 12: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL RIS [SIS] FLOW RATE (ACCUMULATORS AND SI)



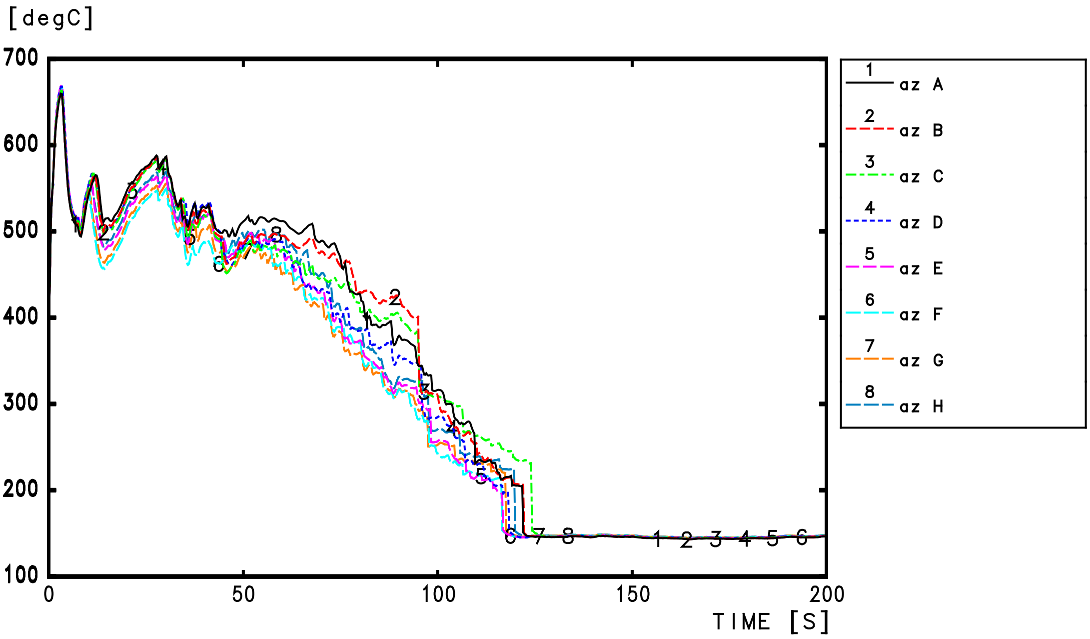
SECTION 16.4.1 - FIGURE D 13: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – PRESSURE AT THE BREAK



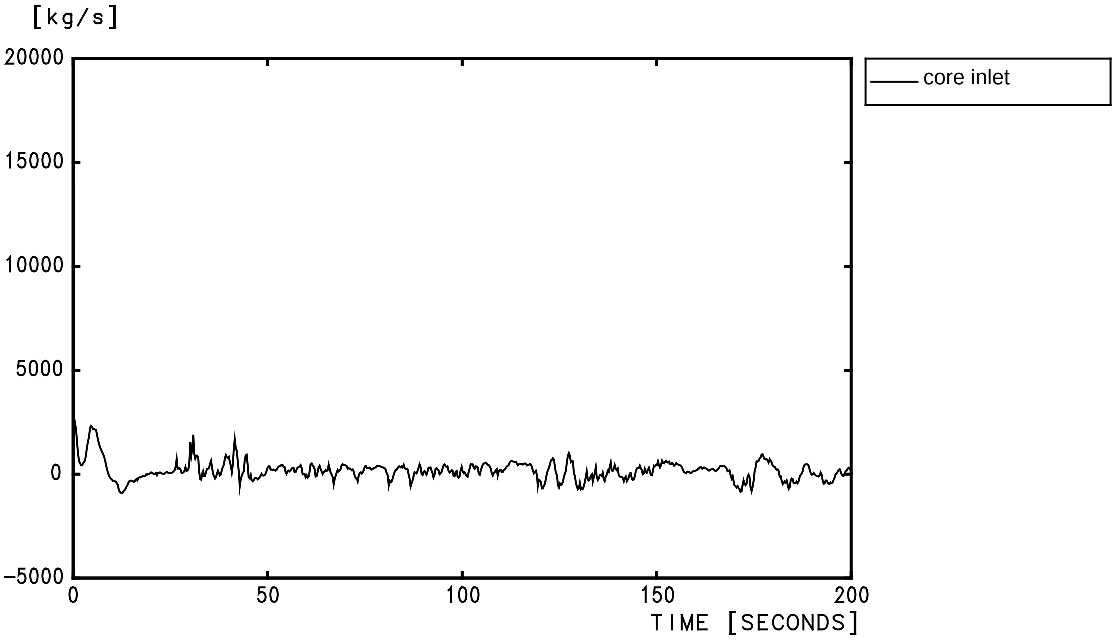
SECTION 16.4.1 - FIGURE D 14: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE BREAK



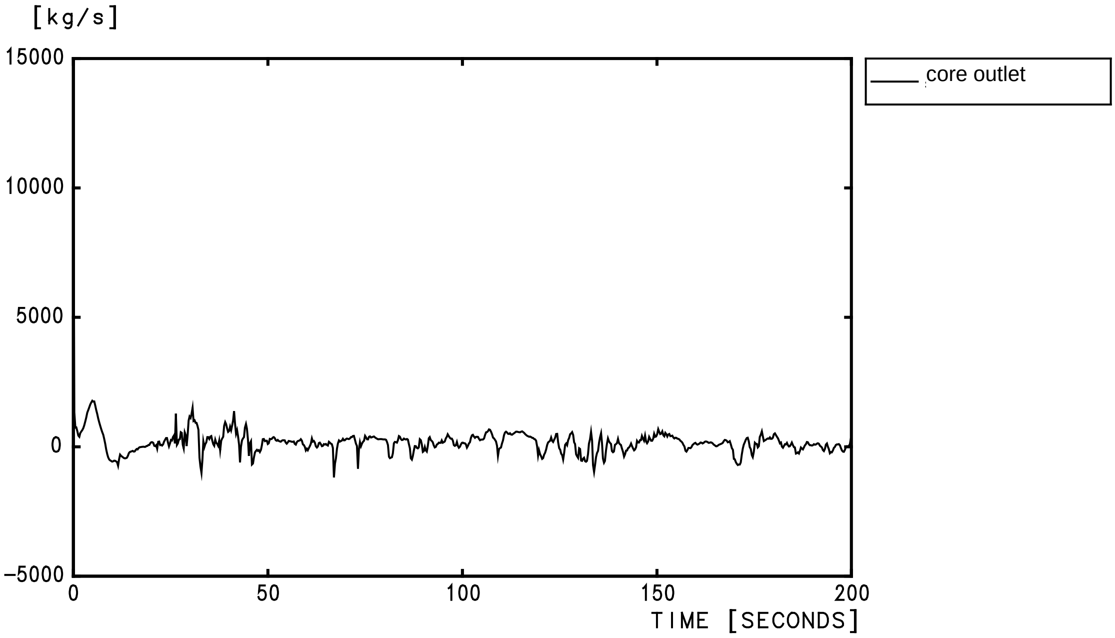
SECTION 16.4.1 - FIGURE D 15: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – QUENCH FRONT LEVEL IN HOT ASSEMBLIES



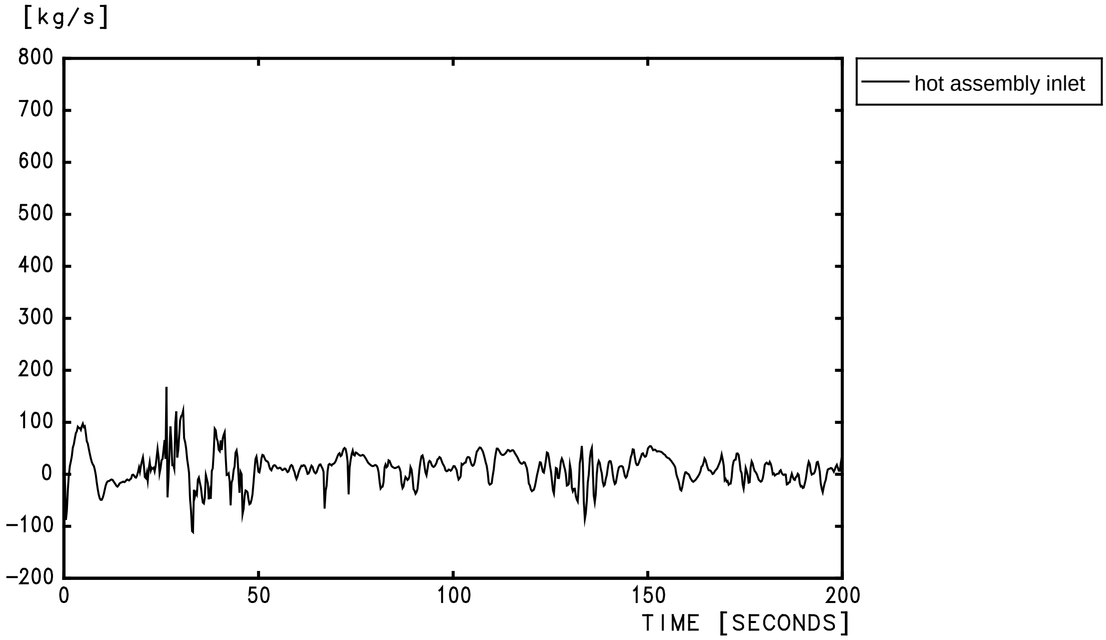
SECTION 16.4.1 - FIGURE D 16: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – MAXIMUM CLAD TEMPERATURES FOR HOT ASSEMBLIES



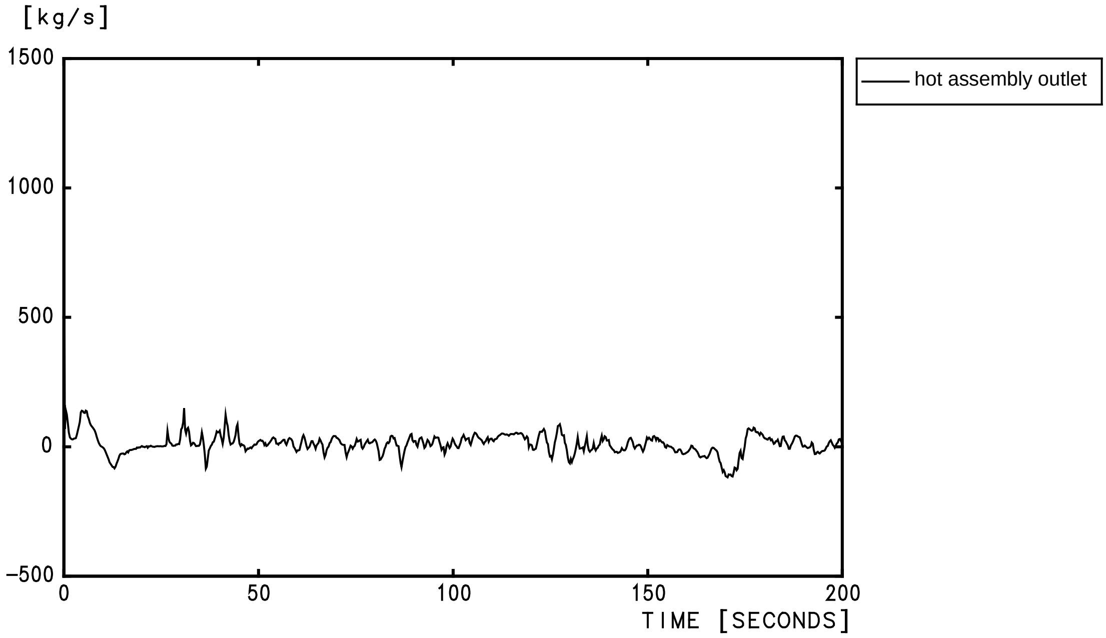
SECTION 16.4.1 - FIGURE D 17: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE CORE INLET



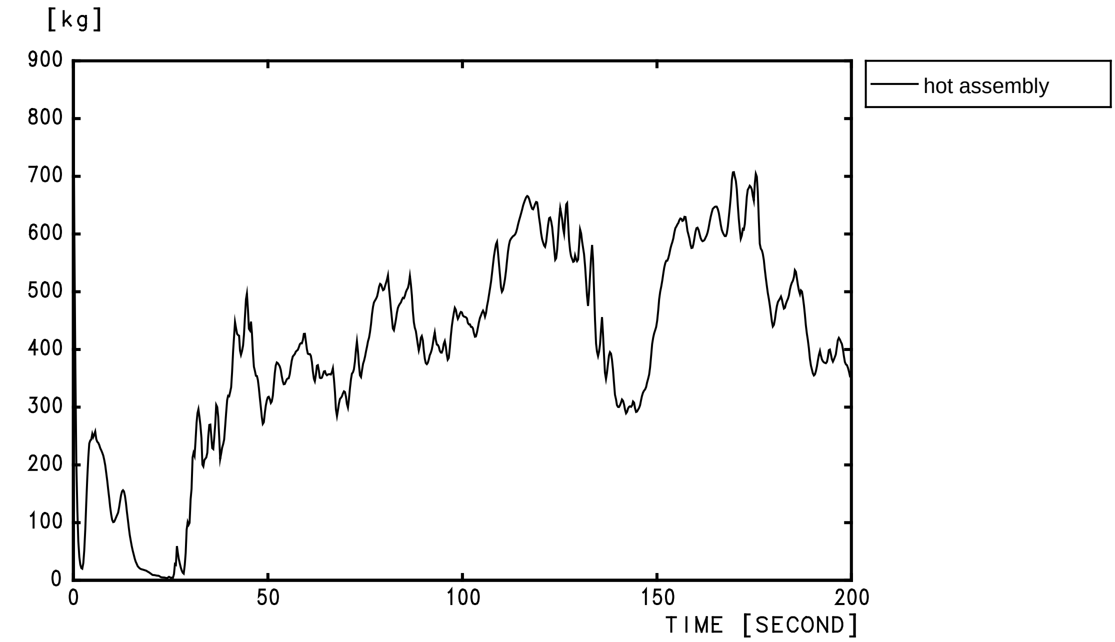
SECTION 16.4.1 - FIGURE D 18: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE CORE OUTLET



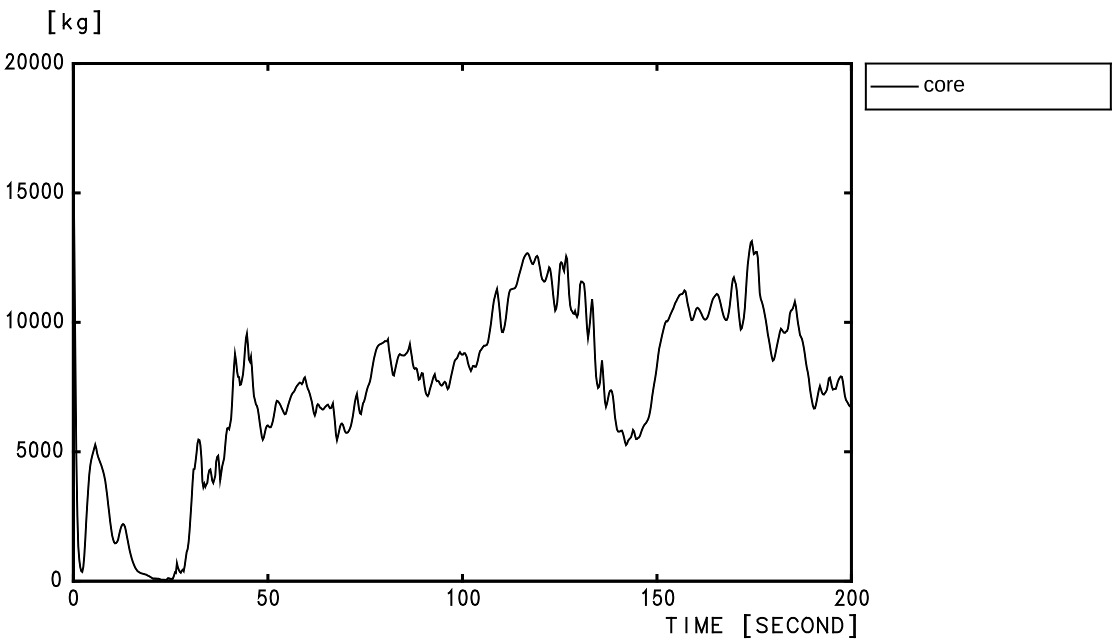
SECTION 16.4.1 - FIGURE D 19: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE HOT ASSEMBLY INLET



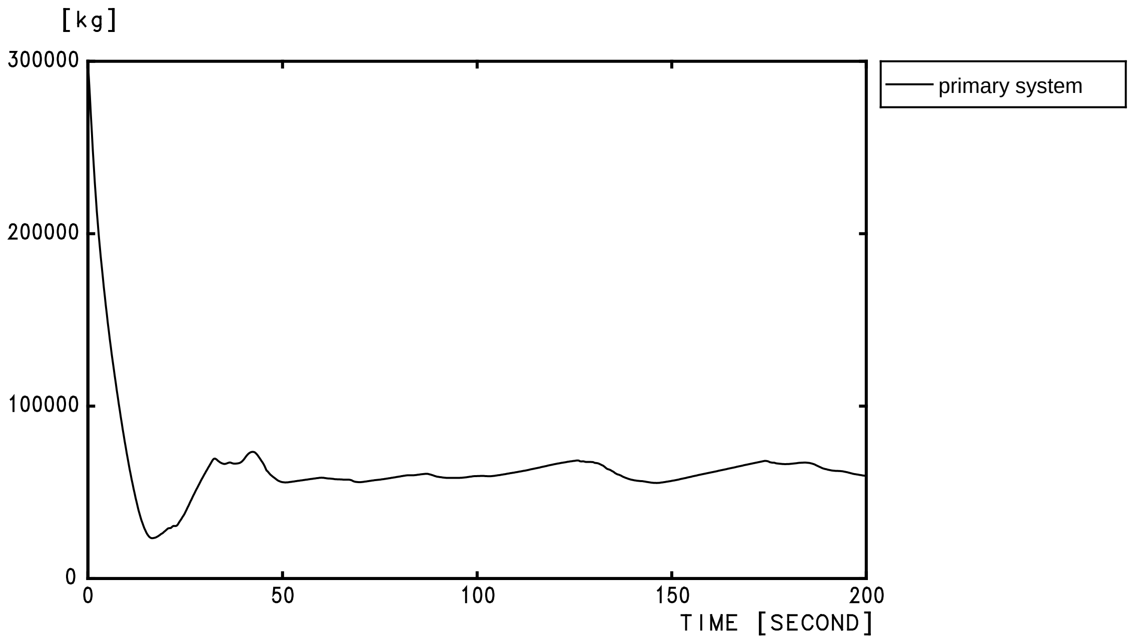
SECTION 16.4.1 - FIGURE D 20: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE HOT ASSEMBLY OUTLET



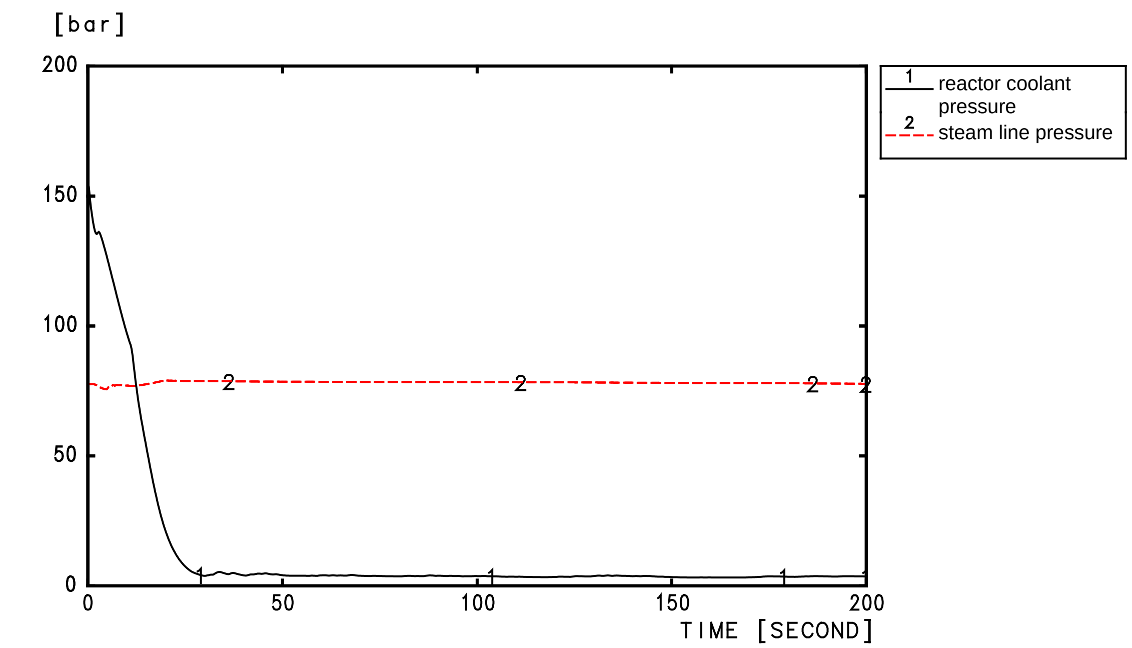
SECTION 16.4.1 - FIGURE D 21: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – LIQUID MASS IN THE HOT ASSEMBLY



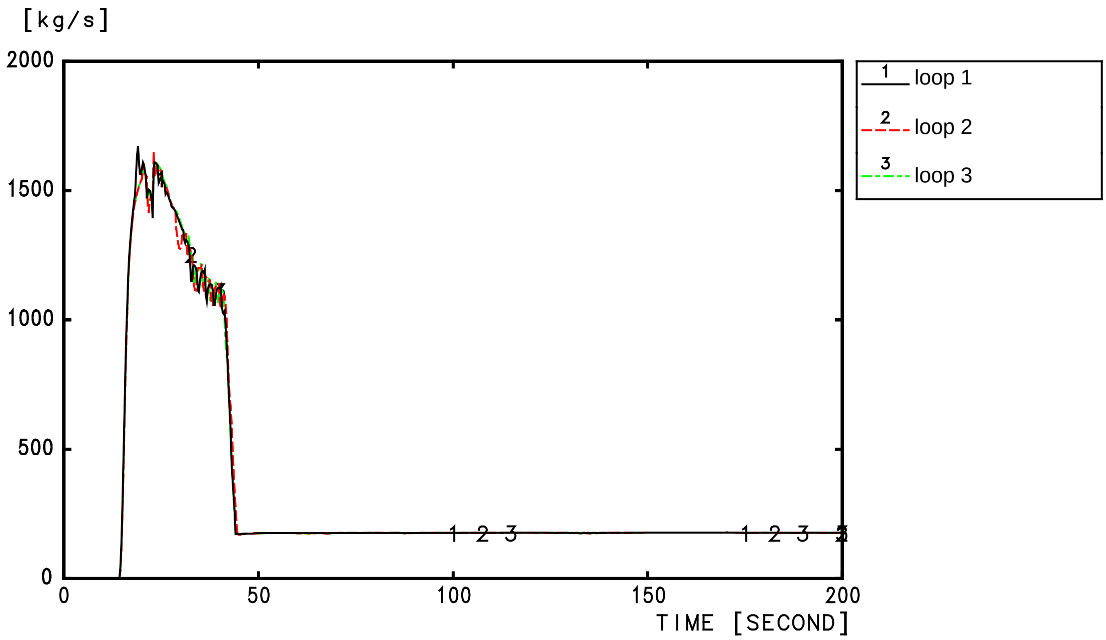
SECTION 16.4.1 - FIGURE D 22: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – LIQUID MASS IN THE CORE



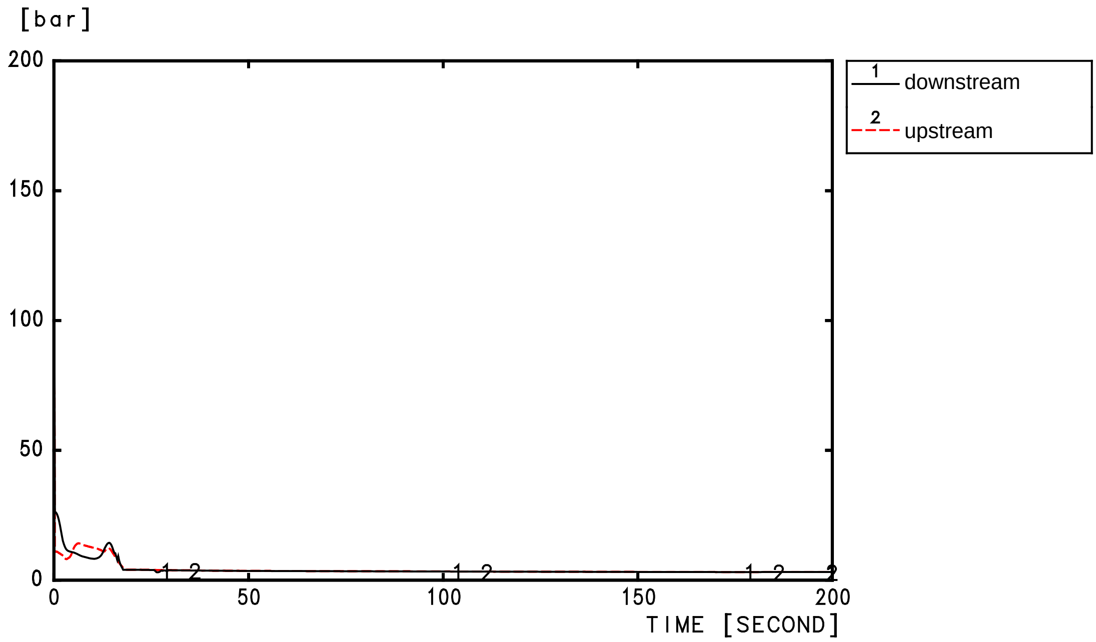
SECTION 16.4.1 - FIGURE D 23: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL MASS IN THE PRIMARY SYSTEM



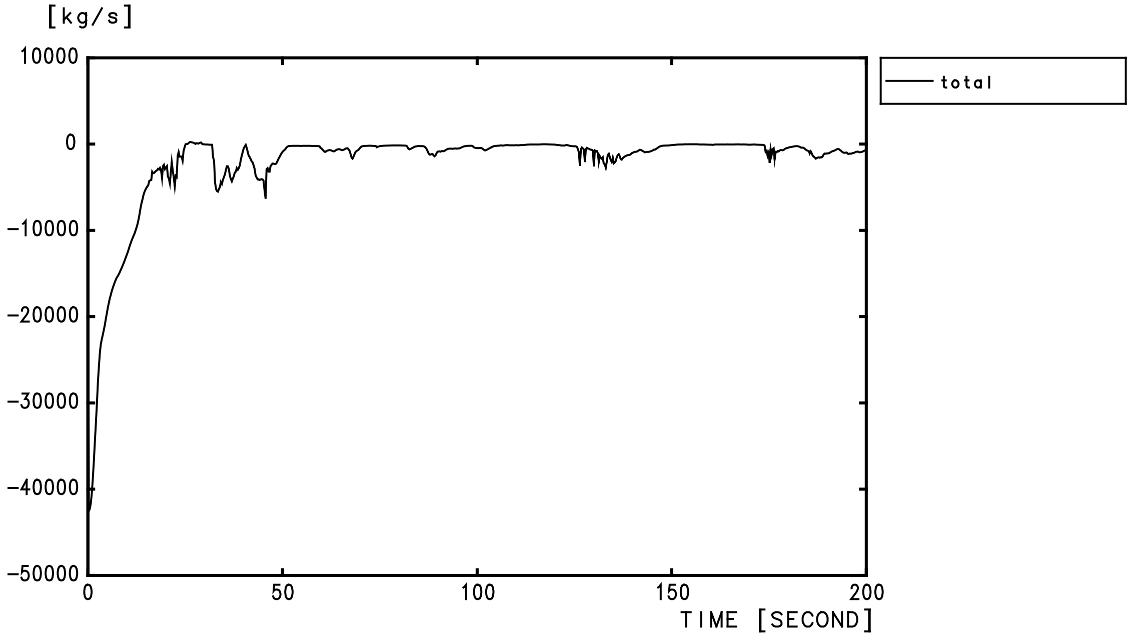
SECTION 16.4.1 - FIGURE D 24: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – REACTOR COOLANT PRESSURE AND STEAM LINE PRESSURE



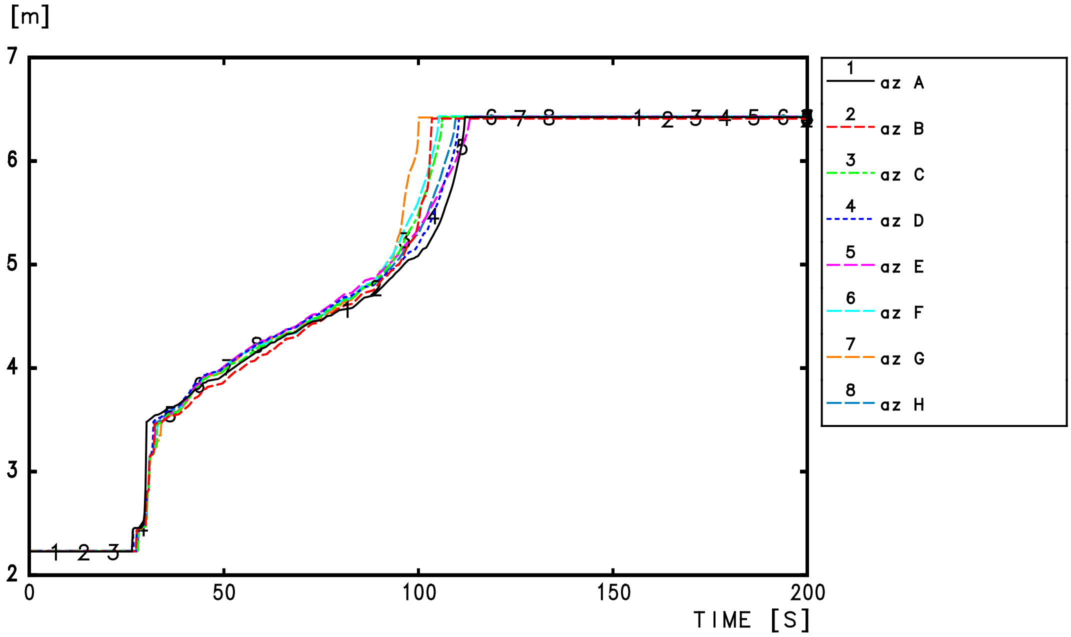
SECTION 16.4.1 - FIGURE D 25: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL RIS [SIS] FLOW RATE (ACCUMULATORS AND SI)



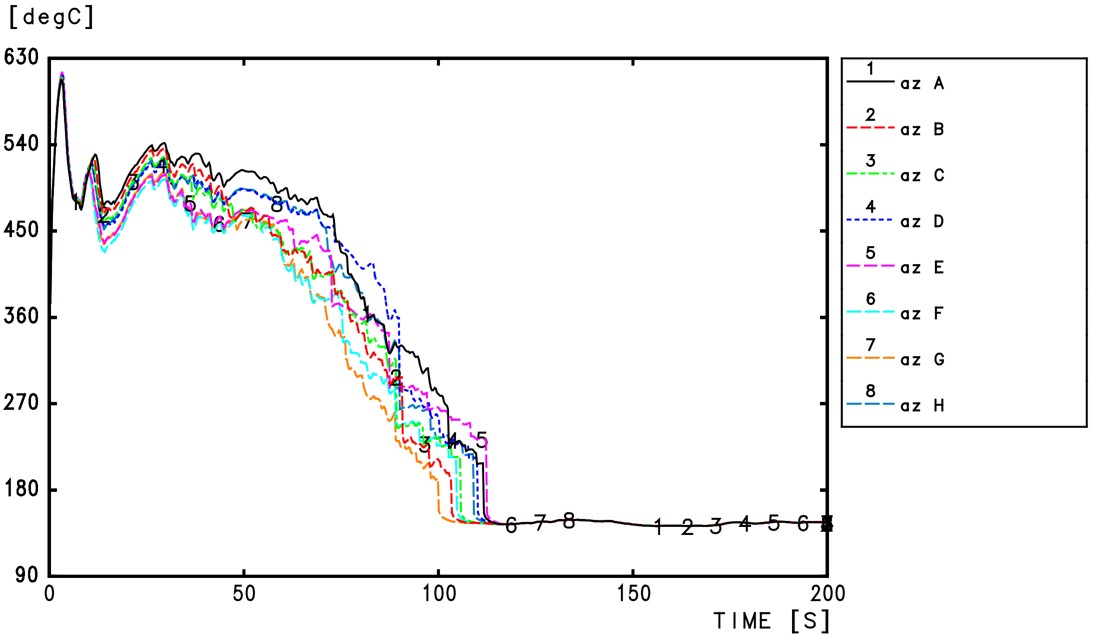
SECTION 16.4.1 - FIGURE D 26: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – PRESSURE AT THE BREAK



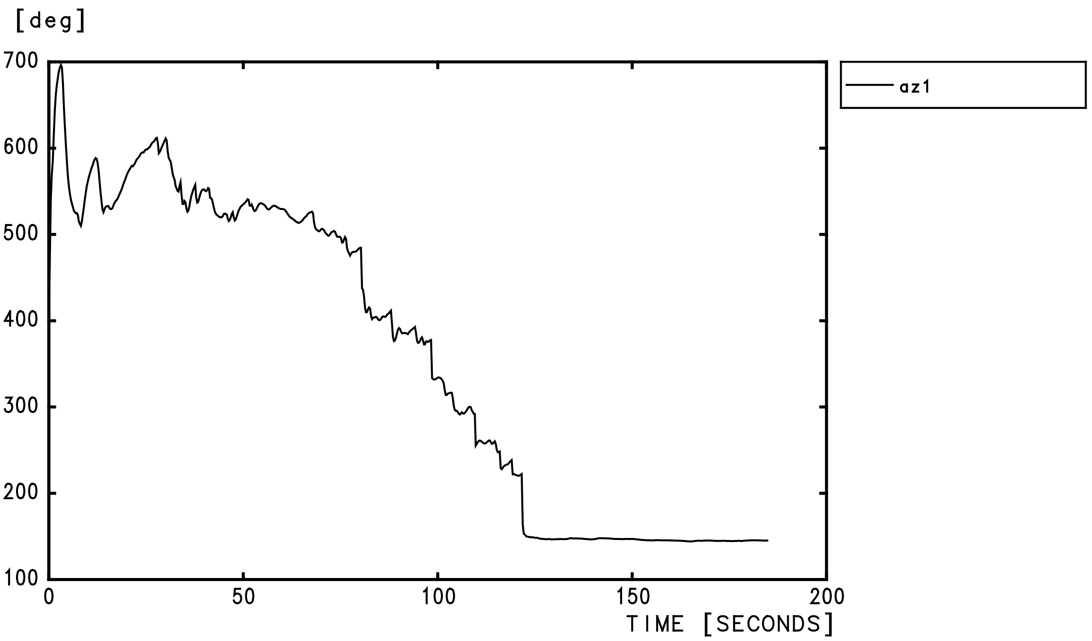
SECTION 16.4.1 - FIGURE D 27: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE BREAK



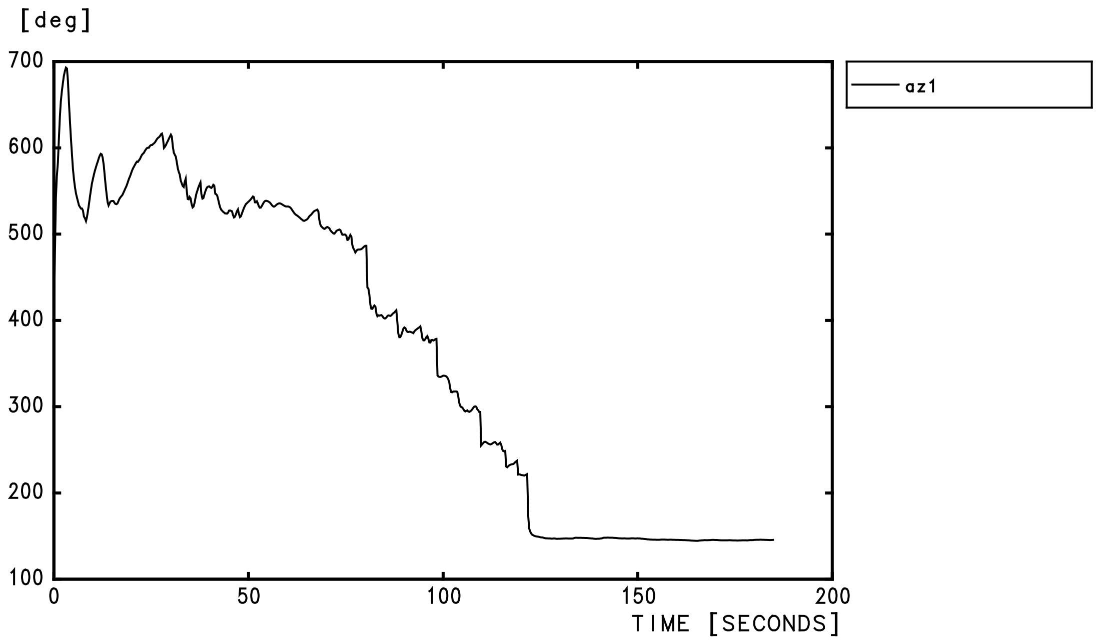
SECTION 16.4.1 - FIGURE D 28: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – QUENCH FRONT LEVEL IN HOT ASSEMBLIES



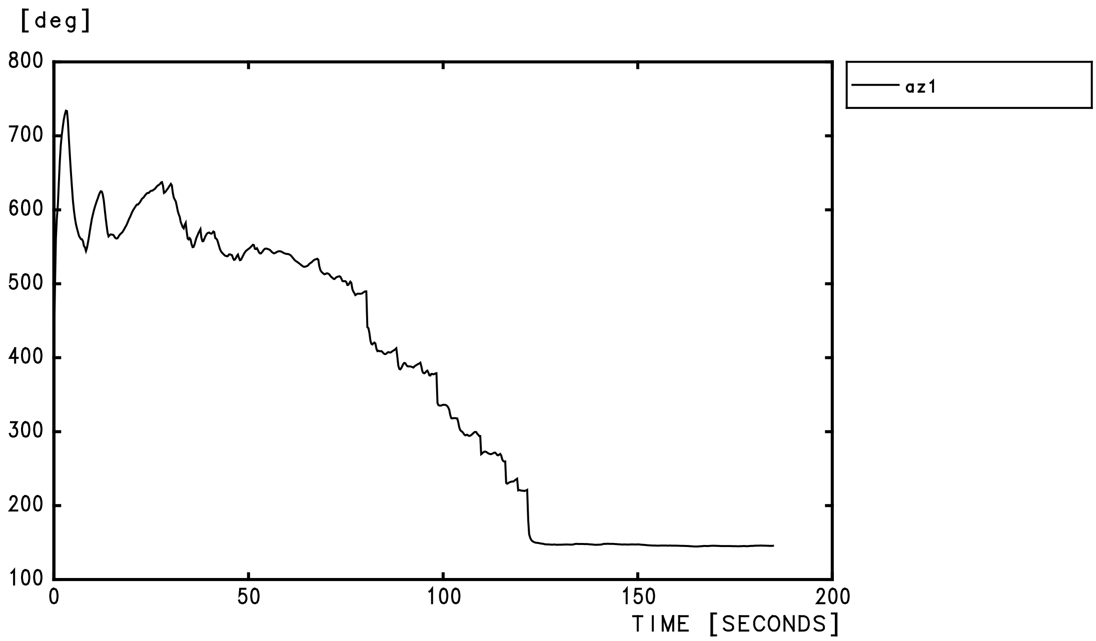
SECTION 16.4.1 - FIGURE D 29: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – MAXIMUM CLAD TEMPERATURES FOR HOT ASSEMBLIES



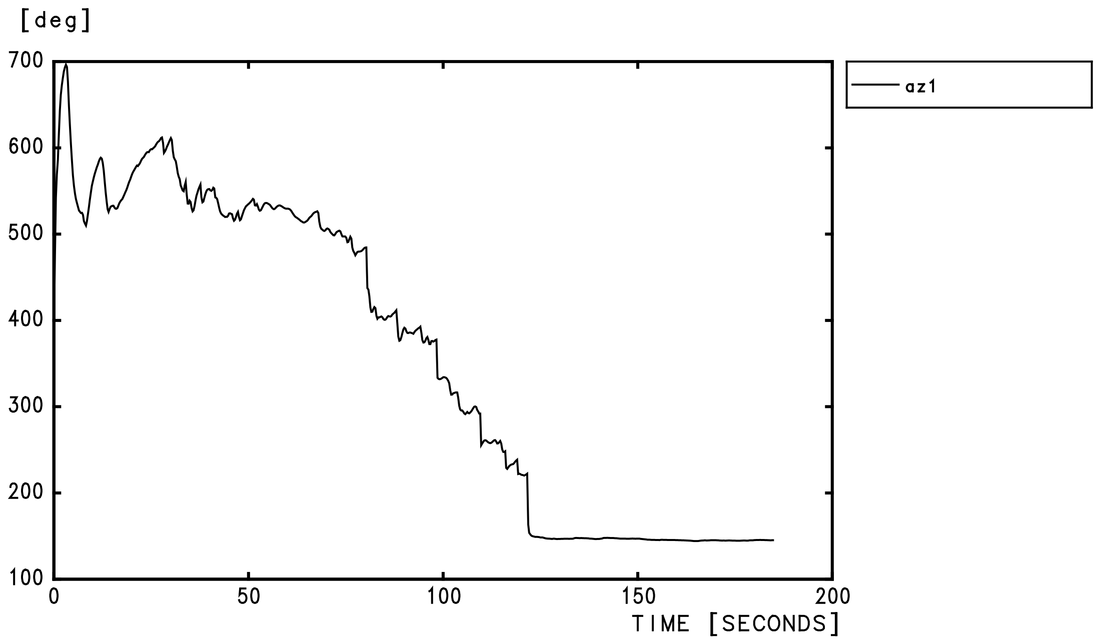
SECTION 16.4.1 - FIGURE D 30: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – BOL INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



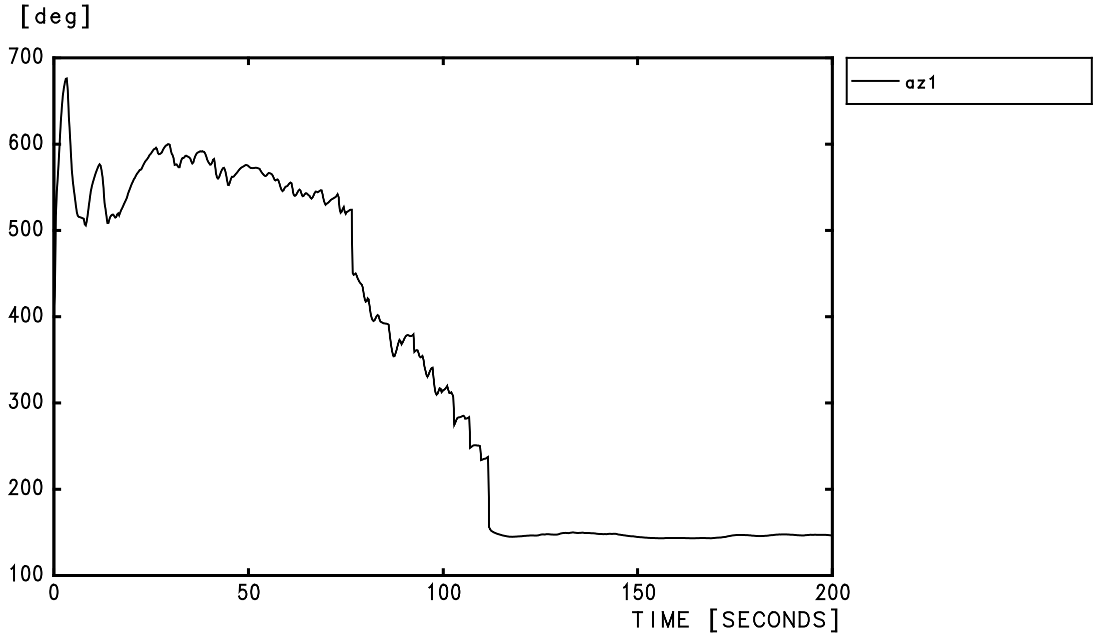
SECTION 16.4.1 - FIGURE D 31: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC1 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



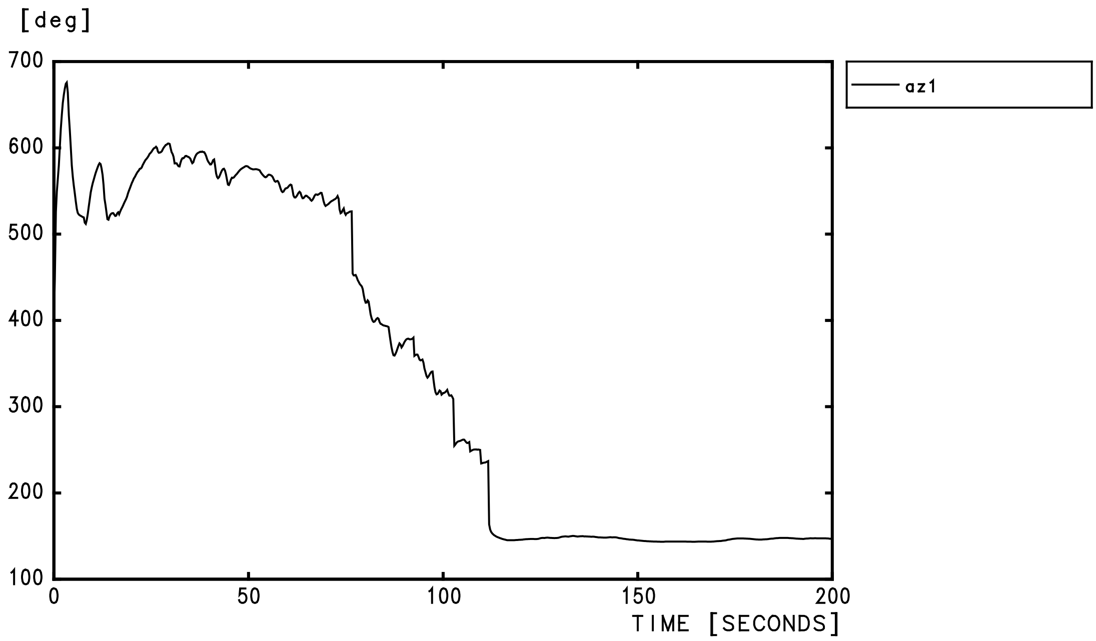
SECTION 16.4.1 - FIGURE D 32: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC2 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



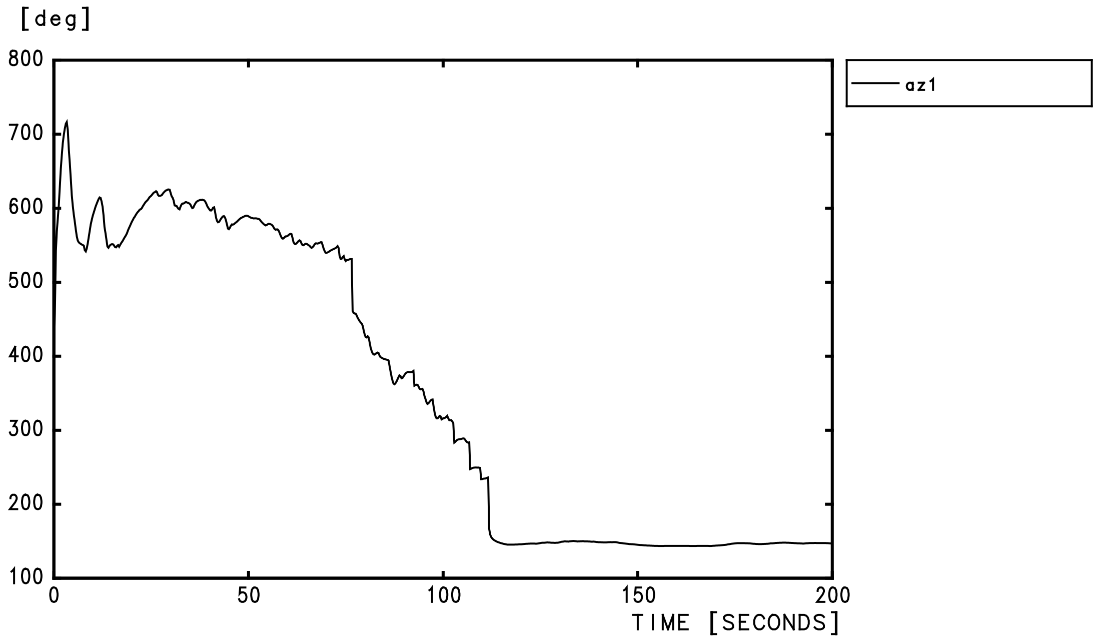
SECTION 16.4.1 - FIGURE D 33: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC3 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



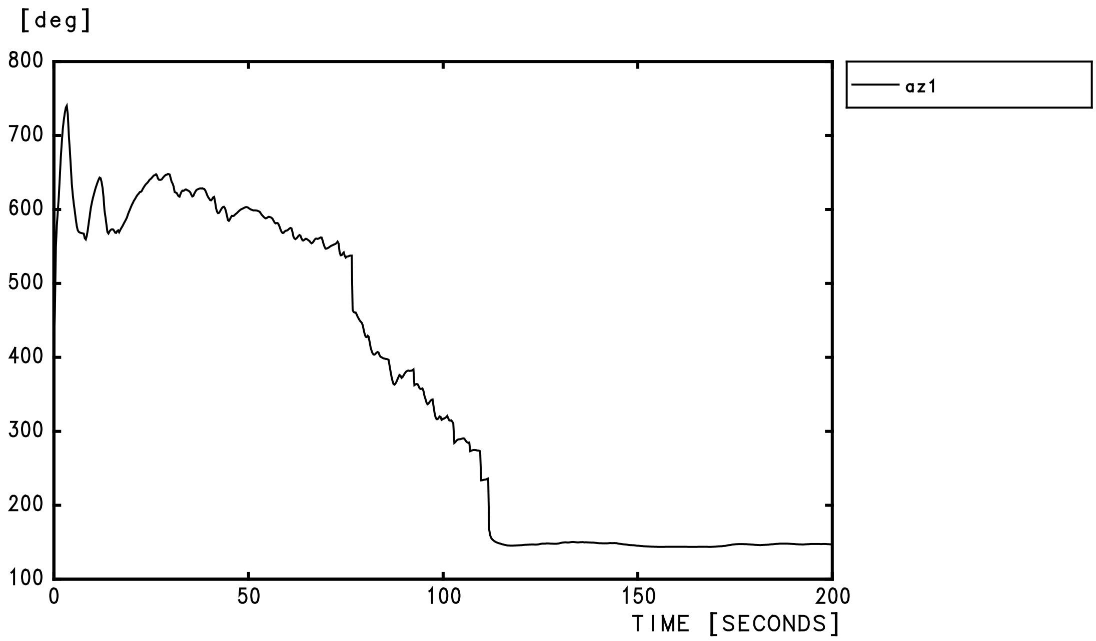
SECTION 16.4.1 - FIGURE D 34: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – BOL INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



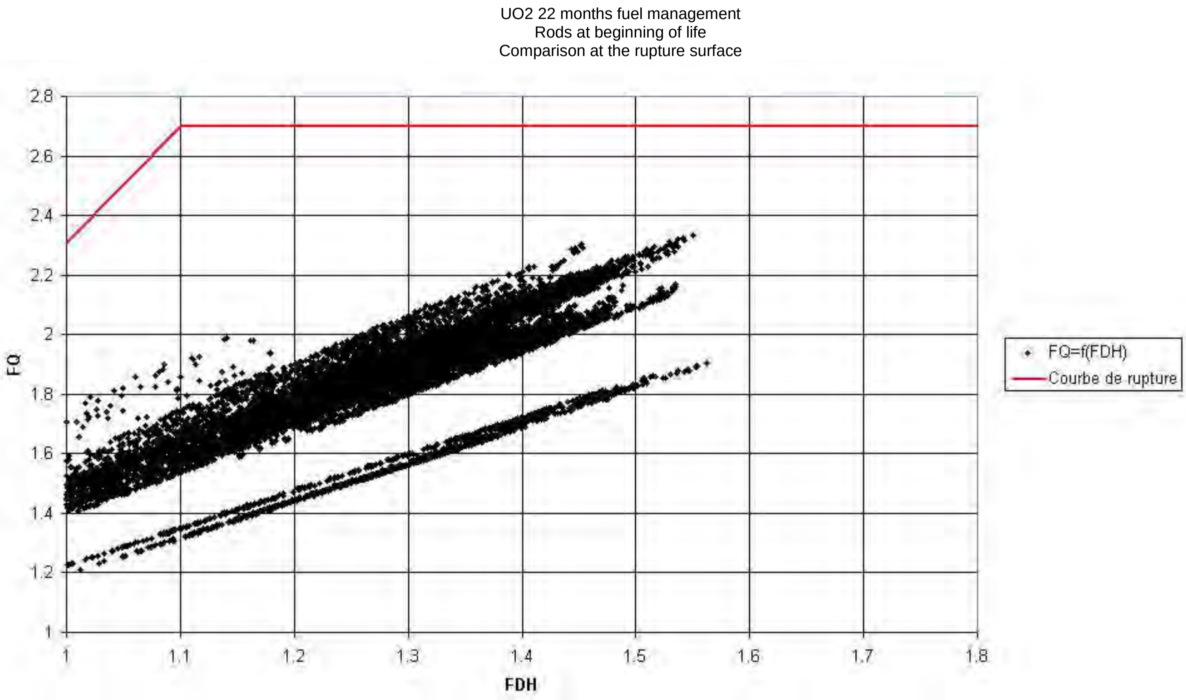
SECTION 16.4.1 - FIGURE D 35: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC1 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



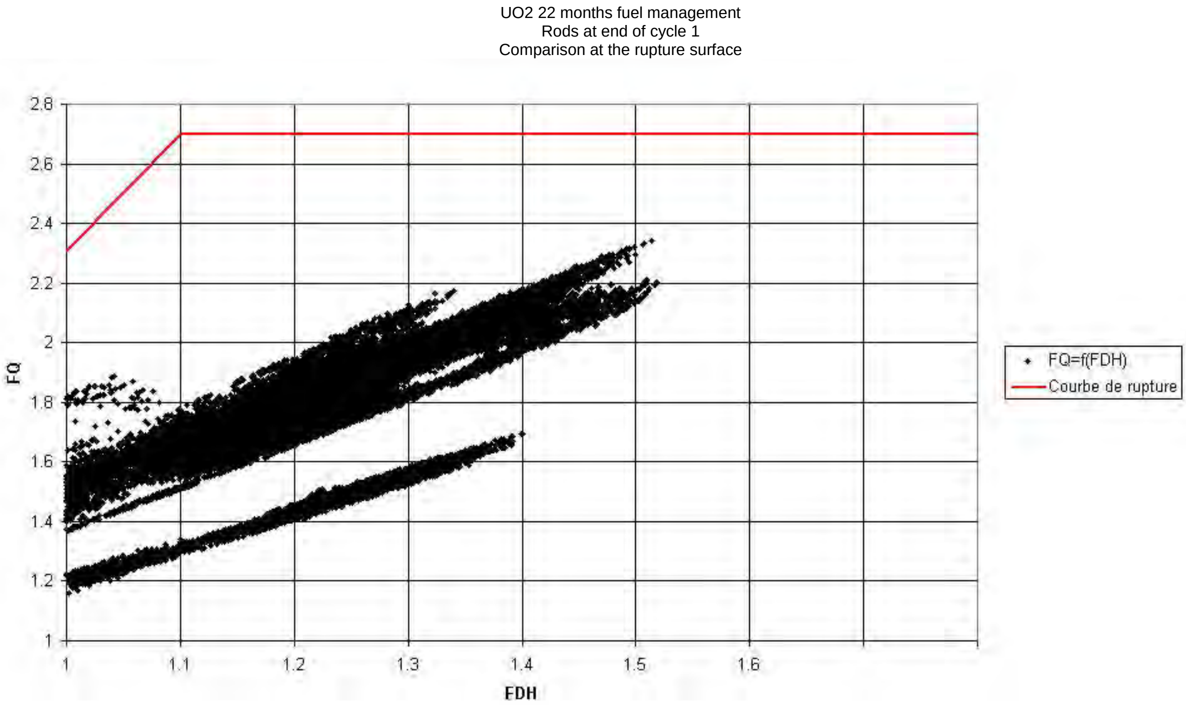
SECTION 16.4.1 - FIGURE D 36: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC2 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE



SECTION 16.4.1 - FIGURE D 37: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC3 INDICATOR ROD – MAXIMUM CLAD TEMPERATURE

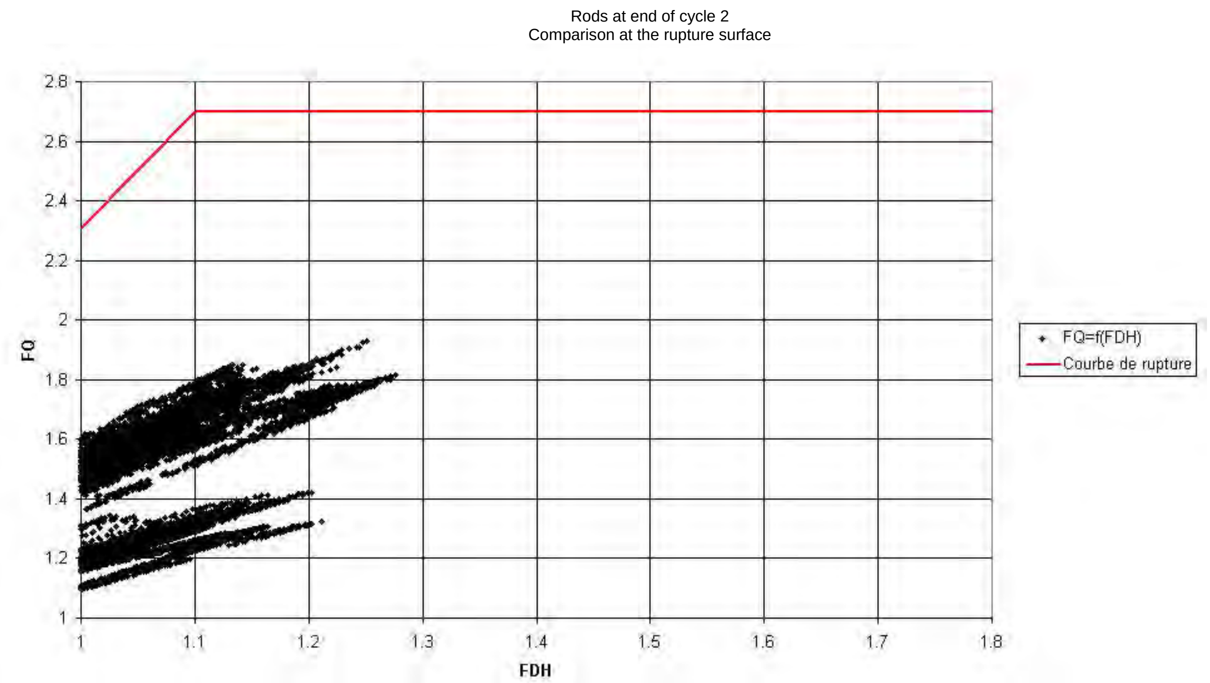


SECTION 16.4.1 - FIGURE D 38: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – RUPTURE CURVE – BOL



SECTION 16.4.1 - FIGURE D 39: IN-OUT UO_2 22 MONTHS FUEL MANAGEMENT – MC – RUPTURE CURVE – EOC1

UO2 22 months fuel management



SECTION 16.4.1 - FIGURE D 40: IN-OUT UO2 22 MONTHS FUEL MANAGEMENT – MC – RUPTURE CURVE – EOC2

SECTION 16.4.1 - TABLE E 1: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – PROPERTIES OF MOX RODS

	Content
Rods with an average Pu content	10.4% Pu total
Rods with a high Pu content	12.0% Pu total

SECTION 16.4.1 - TABLE E 2: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – FUEL CALCULATION POINTS – RODS WITH A HIGH PU CONTENT

Core cycle time	Rod burn-up
Beginning of life (BOL)	150 MWd/tU
End of Cycle 1 (EOC1)	18,500 MWd/tU
End of Cycle 2 (EOC2)	44,050 MWd/tU
Cycle 3 (C3)	65,000 MWd/tU
End of Cycle 4 (EOC4)	73,550 MWd/tU

SECTION 16.4.1 - TABLE E 3: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – FUEL CALCULATION POINTS – RODS WITH AN AVERAGE PU CONTENT

Core cycle time	Rod burn-up
Beginning of life (BOL)	150 MWd/tU
End of Cycle 1 (EOC1)	20,050 MWd/tU
End of Cycle 2 (EOC2)	45,700 MWd/tU
Cycle 3 (C3)	65,000 MWd/tU
End of Cycle 4 (EOC4)	77,250 MWd/tU

SECTION 16.4.1 - TABLE E 4: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – BOUNDING DATA USED TO SIMULATE A ROD AT THE END OF CYCLE 4 – ROD WITH A HIGH PU CONTENT

FΔH	FQ	Average pellet temperature (°C)	Pressure (bar)	External oxide thickness (μm)
1.381	2.059	930	183	33.20
1.273	1.674	1068	183	33.20
1.626	2.353	1181	183	33.20

SECTION 16.4.1 - TABLE E 5: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – BOUNDING DATA USED TO SIMULATE A ROD AT THE END OF CYCLE 4 – ROD WITH AN AVERAGE PU CONTENT

FΔH	FQ	Average pellet temperature (°C)	Pressure (bar)	External oxide thickness (μm)
1.626	2.353	1227	223	32.10

SECTION 16.4.1 - TABLE E 6: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – PROPERTIES OF HOT ASSEMBLIES

Hot assembly	FΔH	FQ	BU	Fuel
HA1	1.40	2.02	EOC1	MOX
HA2	1.33	1.87	C3	MOX

SECTION 16.4.1 - TABLE E 7: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – MAIN EVENTS OF THE TRANSIENT

Event	Time at which the event occurs
RT signal	3.6 s
SI signal	7.1 s
Start of accumulator injection	14.1 s
Start of safety injection system injection	23.0 s
Start of reflood	24.1 s
End of reflood	103.6 s
Duration of reflood	79.5 s
End of accumulator injection	44.2 s

SECTION 16.4.1 - TABLE E 8: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – PEAK CLAD TEMPERATURES

	T1	T2	T3
Time (s)	3.7	27.2	51.0
Mesh	11	11	11
Value (°C)	665	588	497

SECTION 16.4.1 - TABLE E 9: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – MAIN EVENTS OF THE TRANSIENT

Event	Time at which the event occurs
RT signal	3.6 s
SI signal	7.1 s
Start of accumulator injection	14.2 s
Start of safety injection system injection	23.0 s
Start of reflood	24.1 s
End of reflood	104.7 s
Duration of reflood	80.9 s
End of accumulator injection	44.0 s

SECTION 16.4.1 - TABLE E 10: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – PEAK CLAD TEMPERATURES

	T1	T2	T3
Time (s)	3.7	31.1	50.4
Mesh	11	11	11
Value (°C)	686	609	501

SECTION 16.4.1 - TABLE E 11: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – PROPERTIES OF THE INDICATOR RODS WITH A HIGH PU CONTENT

Burn-up	FQ	FΔH	Pressure in the gap (bar)	Maximum linear power density (W/cm)	Maximum initial pellet temperature (°C)
BOL	2.353	1.626	56.7	394.8	926
EOC1	2.353	1.626	63.2	394.8	877
EOC2	2.353	1.626	98.6	394.8	979
C3	2.353	1.626	164.5	394.8	1111
EOC4	2.353	1.626	182.9	394.8	1181

SECTION 16.4.1 - TABLE E 12: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – PROPERTIES OF THE INDICATOR RODS WITH AN AVERAGE PU CONTENT

Burn-up	FQ	F∆H	Pressure in the gap (bar)	Maximum linear power density (W/cm)	Maximum initial pellet temperature (°C)
BOL	2.353	1.626	56.6	394.8	927
EOC1	2.353	1.626	56.0	394.8	936
EOC2	2.353	1.626	117.1	394.8	997
C3	2.353	1.626	183.7	394.8	1124
EOC4	2.353	1.626	223.0	394.8	1227

SECTION 16.4.1 - TABLE E 13: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – RESULTS OBTAINED FOR INDICATOR RODS WITH A HIGH PU CONTENT

Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Average initial pellet temperature at peak (°C)	Rupture
BOL	693	621	531	0.4%	926	NO
EOC1	678	608	526	0.3%	877	NO
EOC2	714	629	537	0.4%	979	NO
C3	752	663	551	1.3%	1111	NO
EOC4	771	683	556	4.0%	1181	NO

SECTION 16.4.1 - TABLE E 14: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – RESULTS OBTAINED FOR INDICATOR RODS WITH AN AVERAGE PU CONTENT

Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Average initial pellet temperature at peak (°C)	Rupture
BOL	694	621	532	0.4%	927	NO
EOC1	697	624	533	0.4%	936	NO
EOC2	718	633	538	0.4%	997	NO
C3	755	666	550	2.5%	1124	NO
EOC4	782	687	541	16.9%	1227	NO

SECTION 16.4.1 - TABLE E 15: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – RESULTS OBTAINED FOR INDICATOR RODS WITH A HIGH PU CONTENT

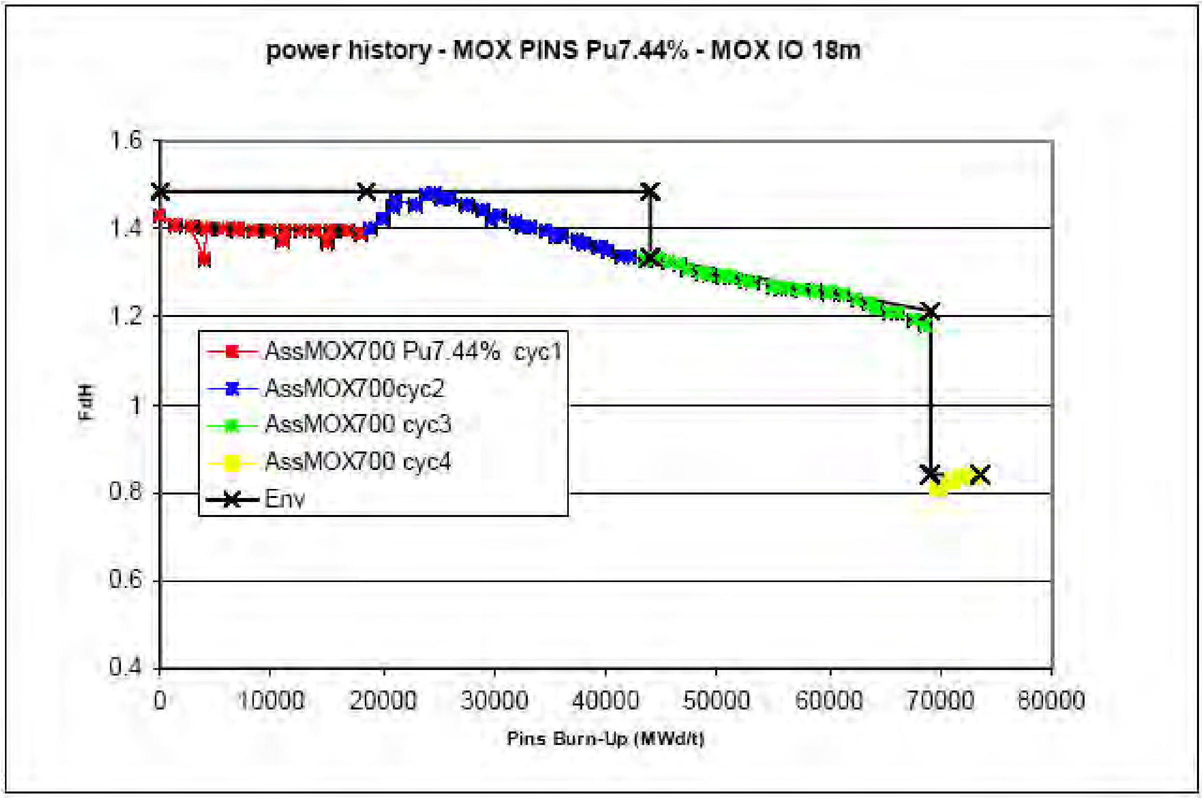
Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Average initial pellet temperature at peak (°C)	Rupture
BOL	697	642	537	0.37%	926	NO
EOC1	681	629	532	0.35%	877	NO
EOC2	717	649	543	0.37%	979	NO
C3	756	681	559	2.05%	1111	NO
EOC4	775	698	563	6.34%	1181	NO

SECTION 16.4.1 - TABLE E 16: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – RESULTS OBTAINED FOR INDICATOR RODS WITH AN AVERAGE PU CONTENT

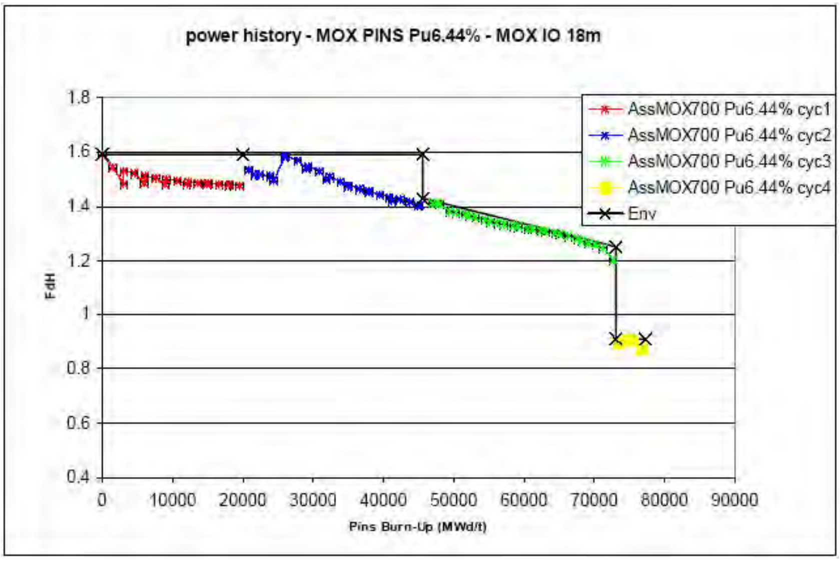
Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Average initial pellet temperature at peak (°C)	Rupture
BOL	697	642	537	0.37%	927	NO
EOC1	700	644	539	0.37%	936	NO
EOC2	722	653	545	0.43%	997	NO
C3	758	683	557	3.96%	1124	NO
EOC4	786	698	547	27.56%	1227	NO

SECTION 16.4.1 - TABLE E 17: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – RESULTS OBTAINED FOR ENVELOPE UO_2-TYPE INDICATOR RODS

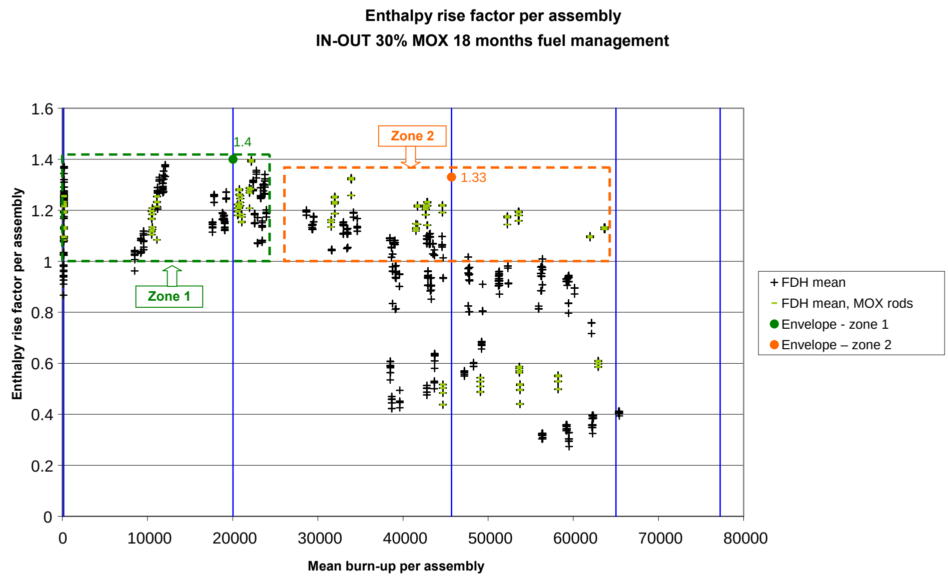
Hot residence assembly	FQ	FΔH	BU	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Average initial pellet temperature at peak (°C)	Rupture
HA1	2.353	1.626	EOC4	747	661	551	0.40%	750	NO
HA2	2.353	1.626	EOC4	750	678	557	0.48%	750	NO



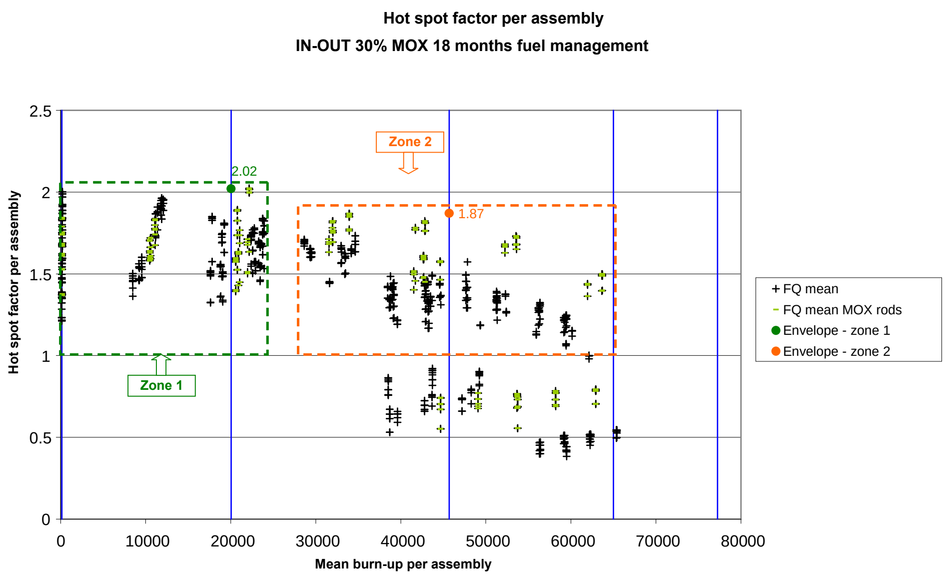
SECTION 16.4.1 - FIGURE E 1: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – POWER HISTORY OF MOX RODS WITH A HIGH PU CONTENT



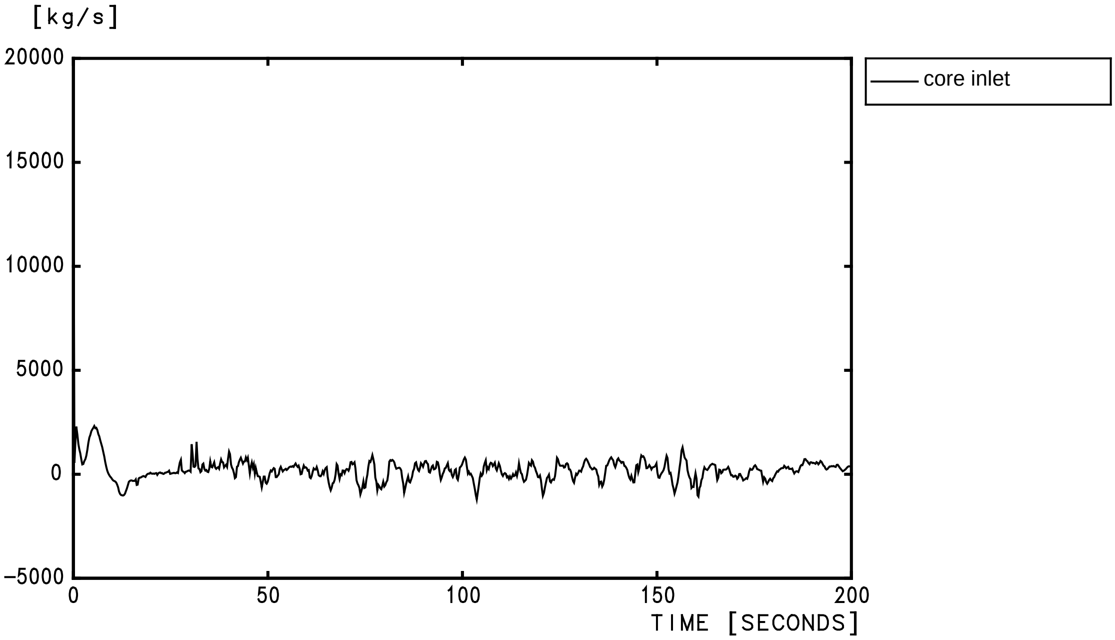
SECTION 16.4.1 - FIGURE E 2: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – POWER HISTORY OF MOX RODS WITH AN AVERAGE PU CONTENT



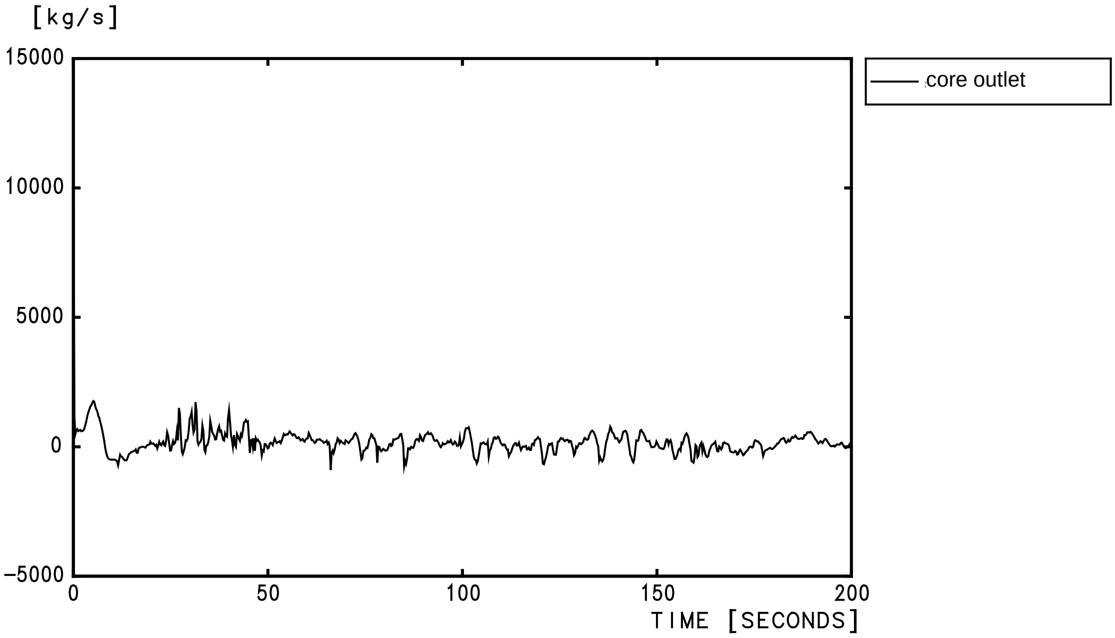
SECTION 16.4.1 - FIGURE E 3: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – F∆H FOR ASSEMBLIES AS A FUNCTION OF BURN-UP



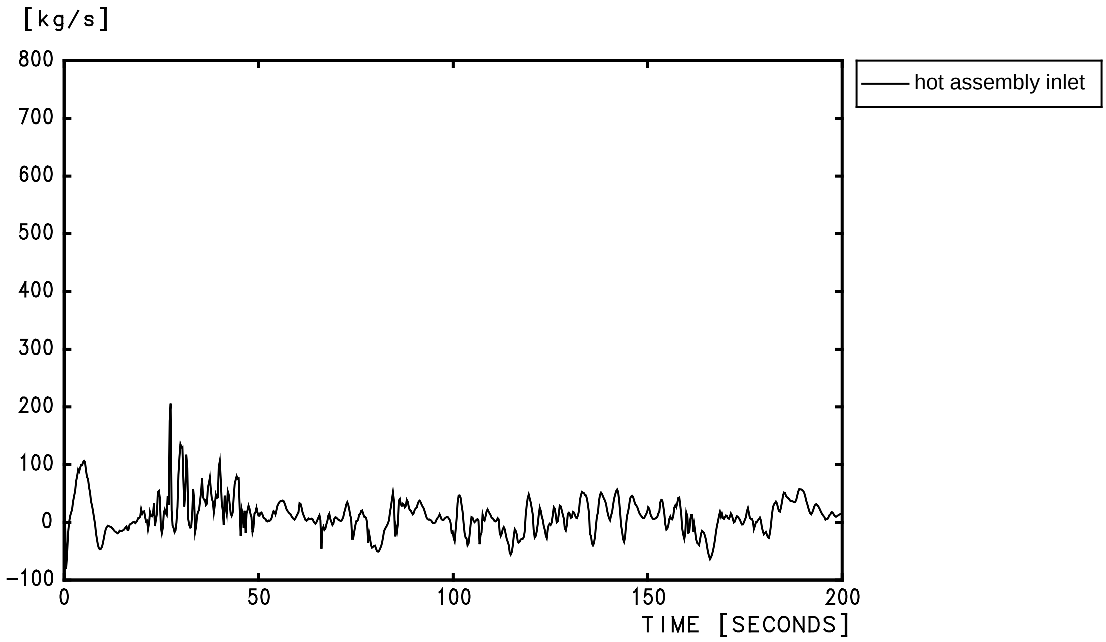
SECTION 16.4.1 - FIGURE E 4: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – FQ FOR ASSEMBLIES AS A FUNCTION OF BURN-UP



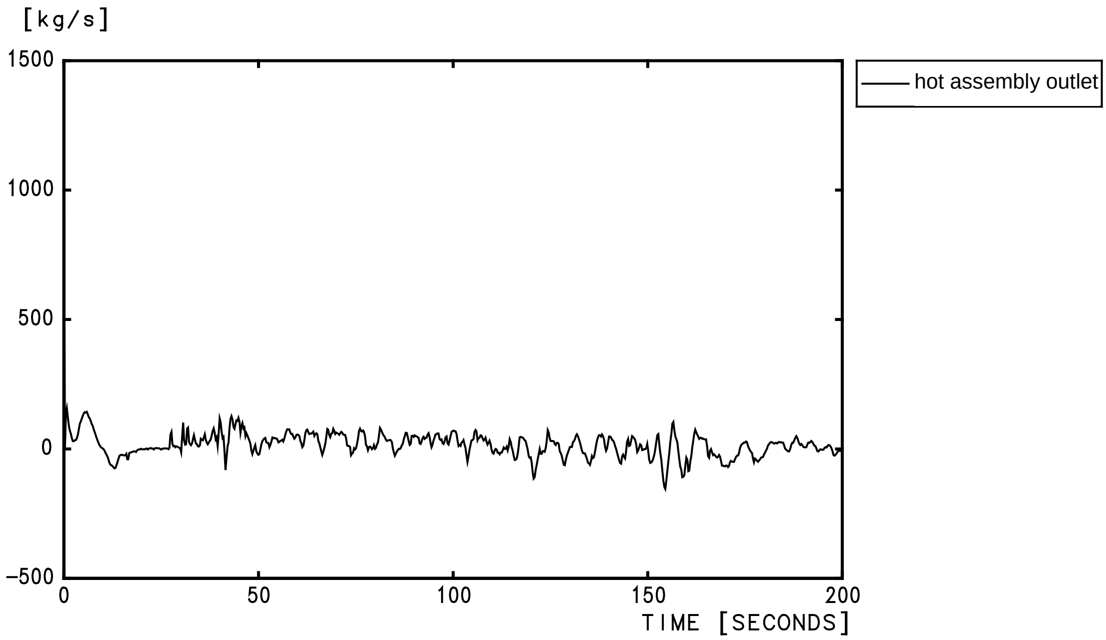
SECTION 16.4.1 - FIGURE E 5: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE CORE INLET



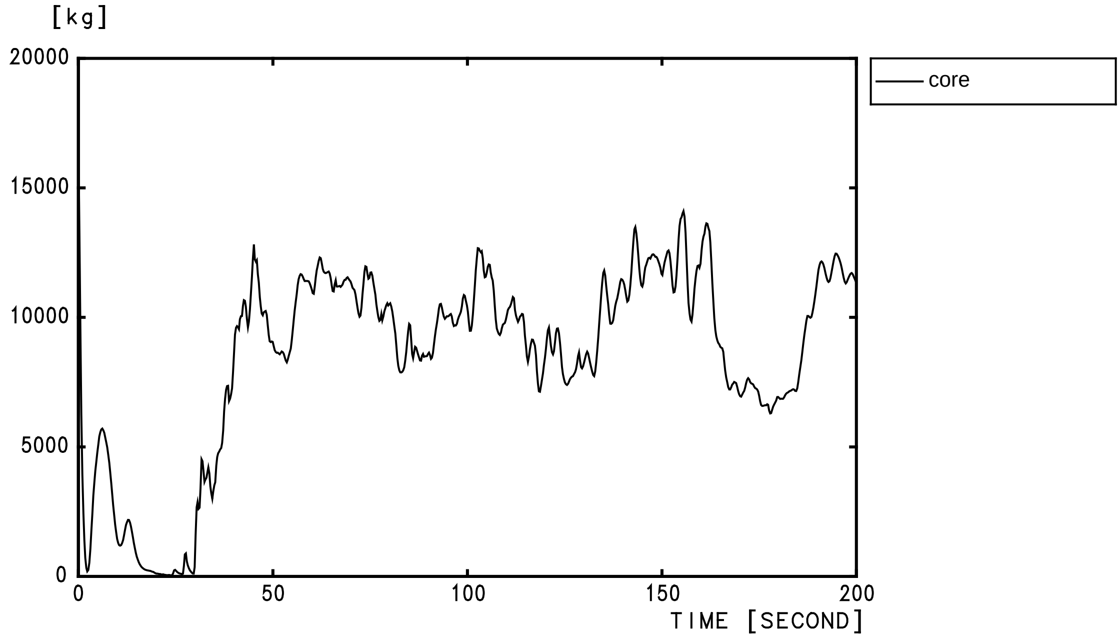
SECTION 16.4.1 - FIGURE E 6: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE CORE OUTLET



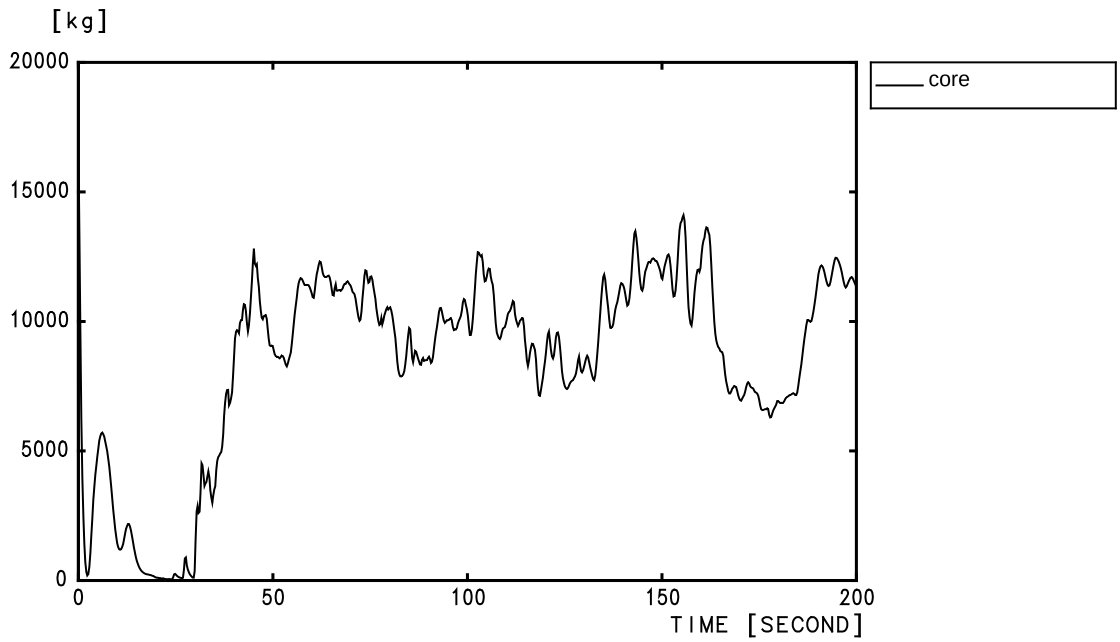
SECTION 16.4.1 - FIGURE E 7: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE HOT ASSEMBLY INLET



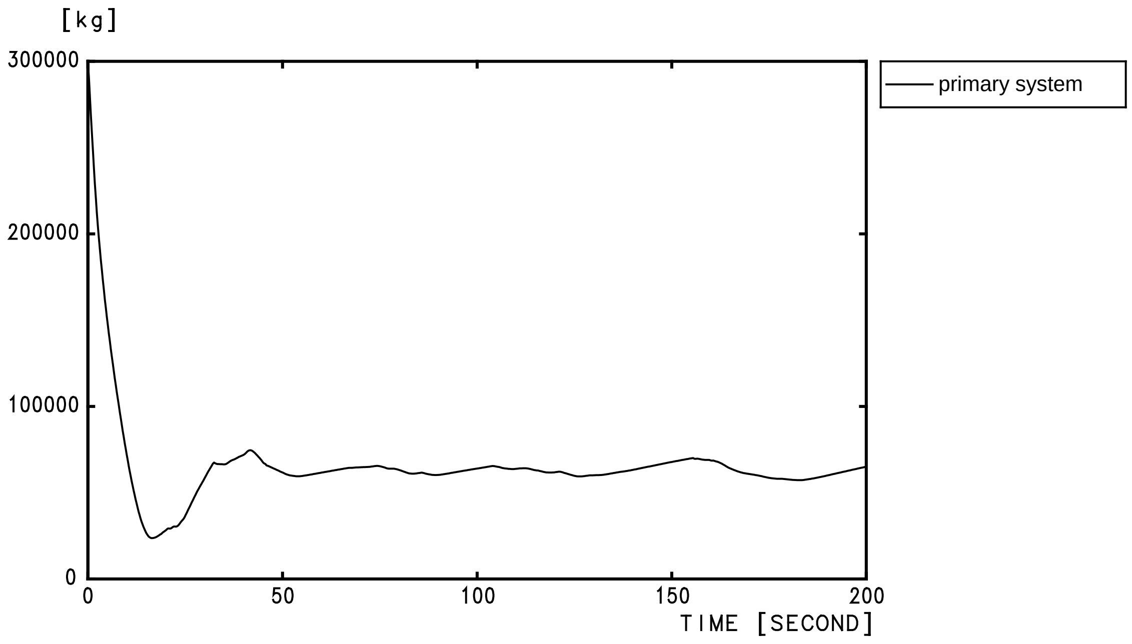
SECTION 16.4.1 - FIGURE E 8: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE HOT ASSEMBLY OUTLET



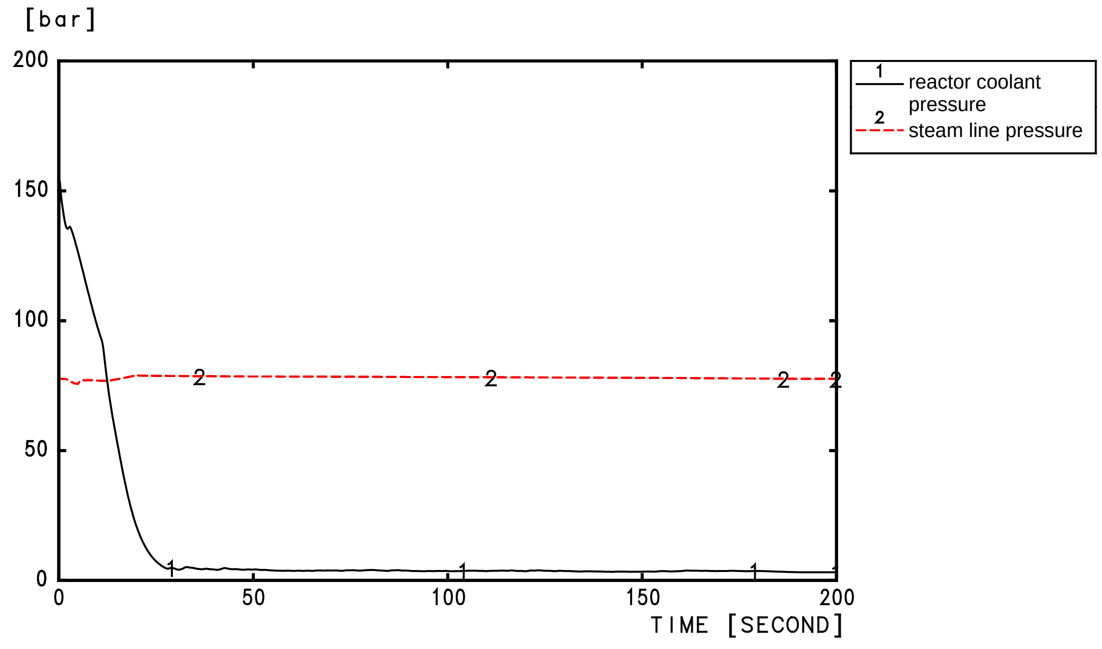
SECTION 16.4.1 - FIGURE E 9: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – LIQUID MASS IN THE HOT ASSEMBLY



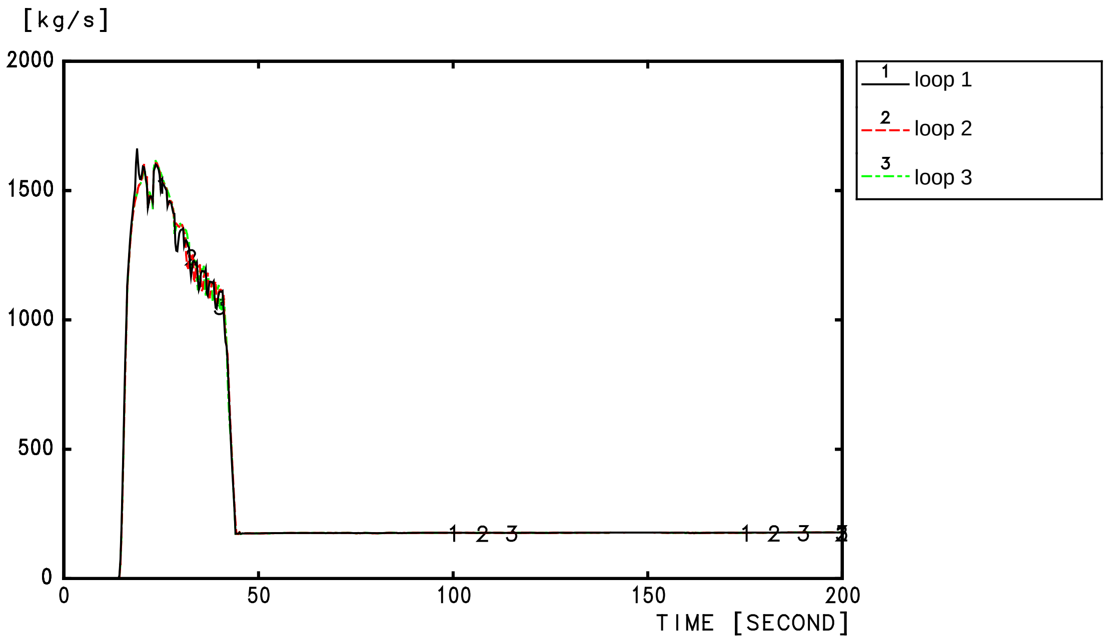
SECTION 16.4.1 - FIGURE E 10: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – LIQUID MASS IN THE CORE



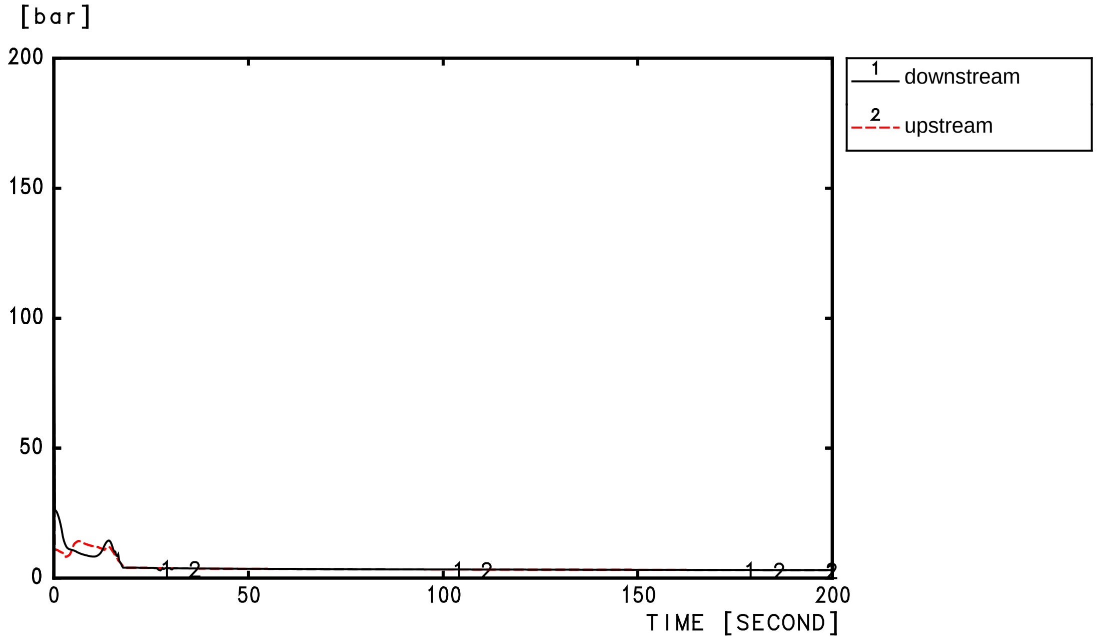
SECTION 16.4.1 - FIGURE E 11: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL MASS IN THE PRIMARY SYSTEM



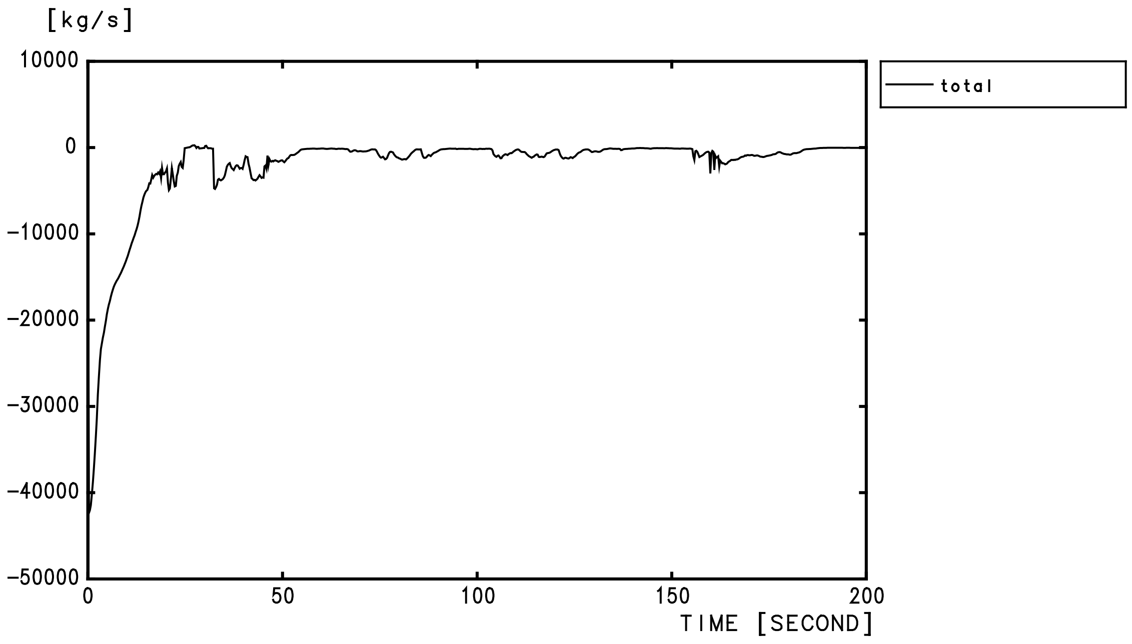
SECTION 16.4.1 - FIGURE E 12: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – REACTOR COOLANT PRESSURE AND STEAM LINE PRESSURE



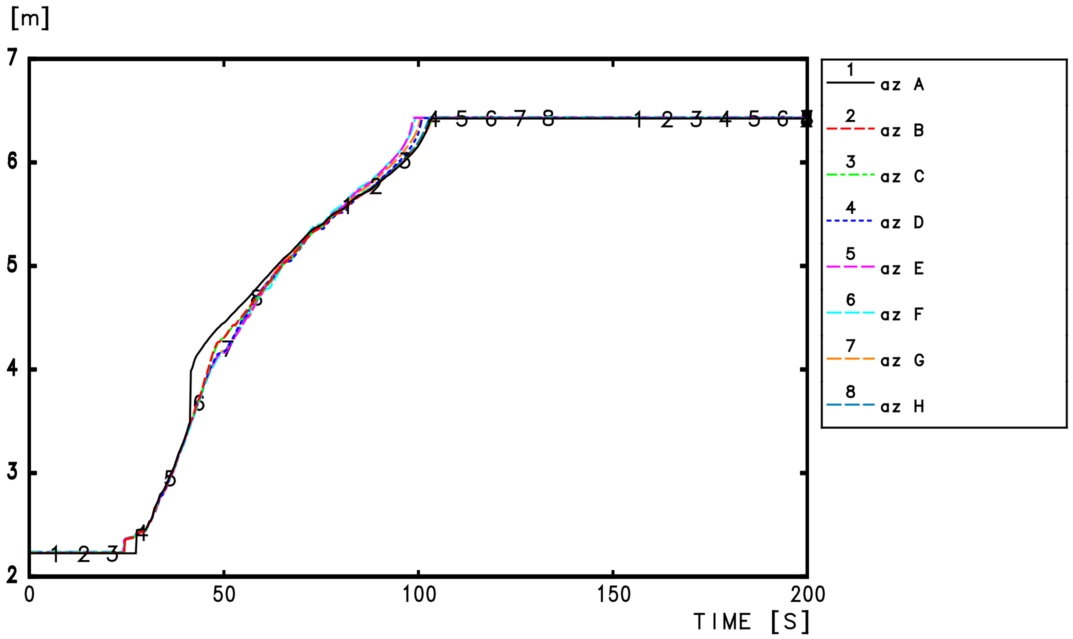
SECTION 16.4.1 - FIGURE E 13: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL RIS [SIS] FLOW RATE (ACCUMULATORS AND SI)



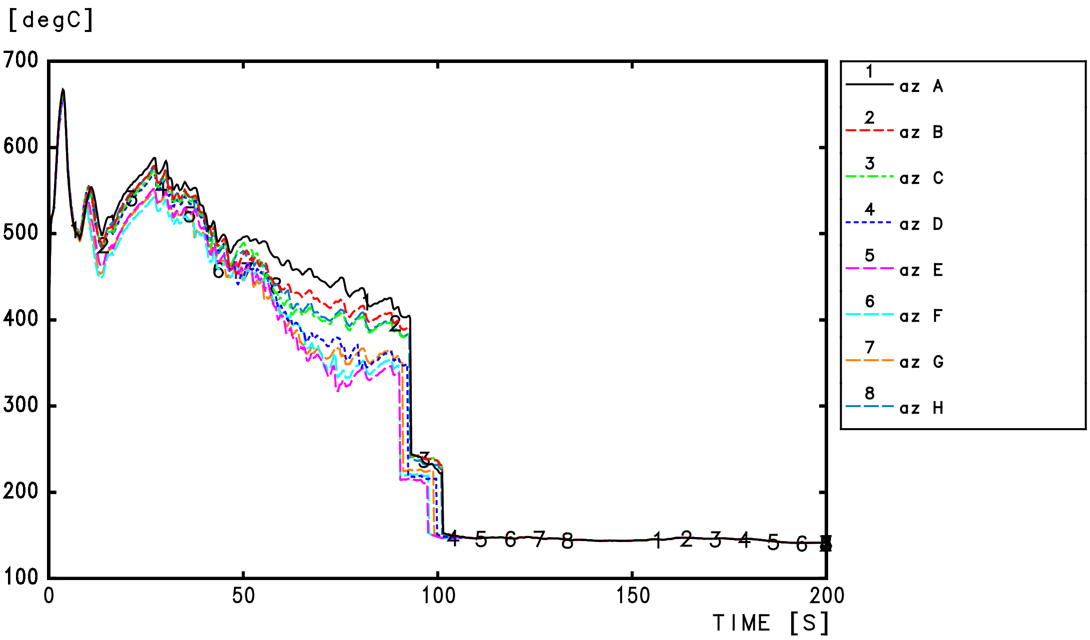
SECTION 16.4.1 - FIGURE E 14: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – PRESSURE AT THE BREAK



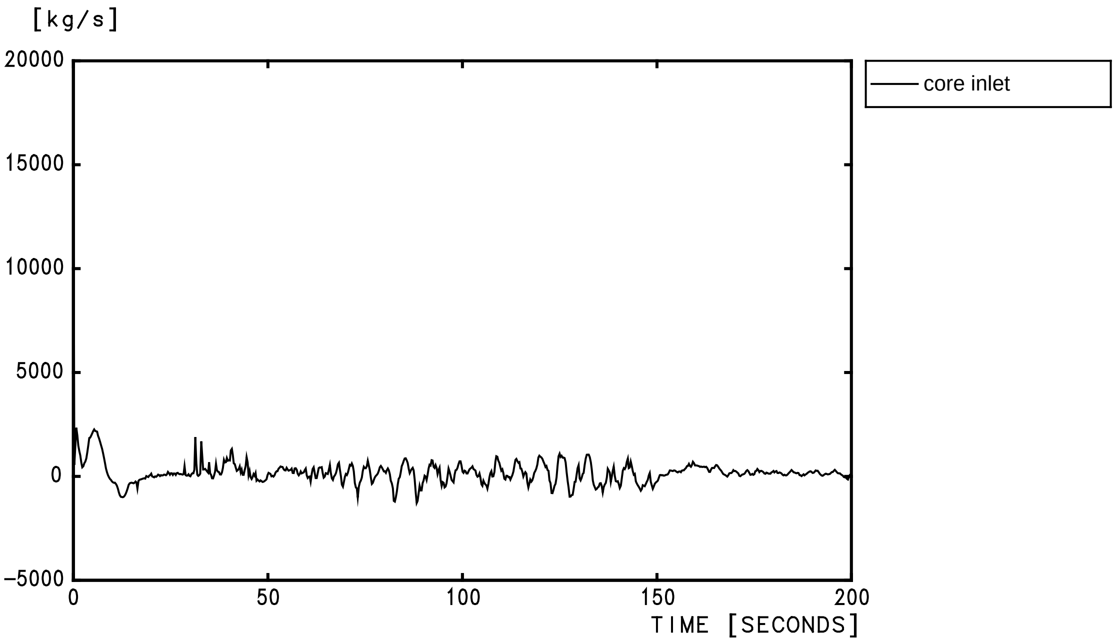
SECTION 16.4.1 - FIGURE E 15: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – TOTAL FLOW RATE AT THE BREAK



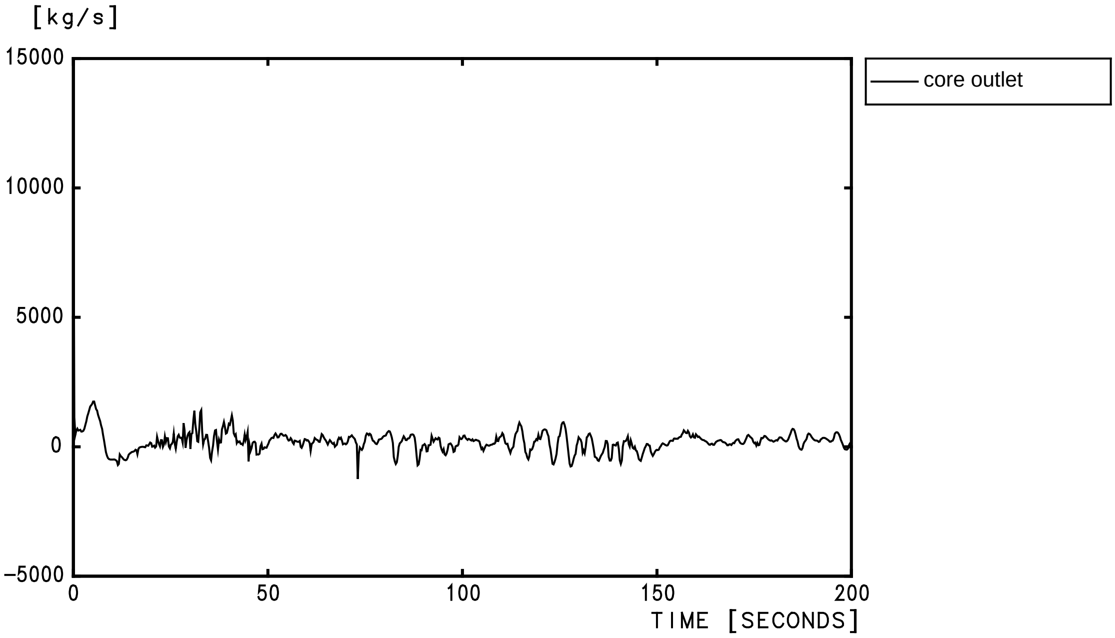
SECTION 16.4.1 - FIGURE E 16: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – QUENCH FRONT LEVEL IN HOT ASSEMBLIES



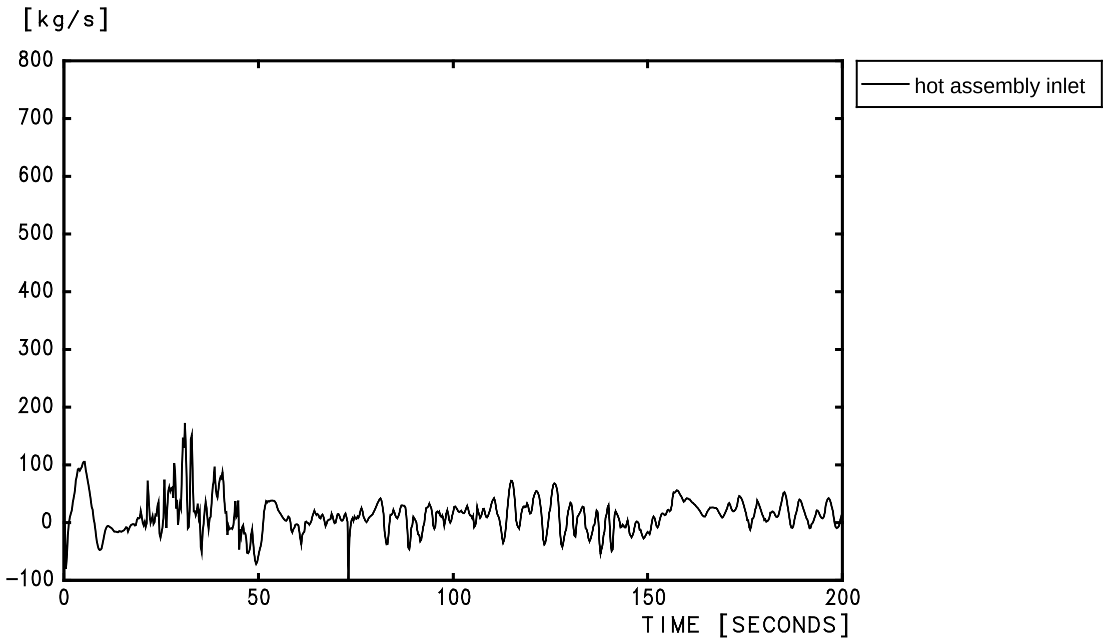
SECTION 16.4.1 - FIGURE E 17: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – MAXIMUM CLAD TEMPERATURES FOR HOT ASSEMBLIES



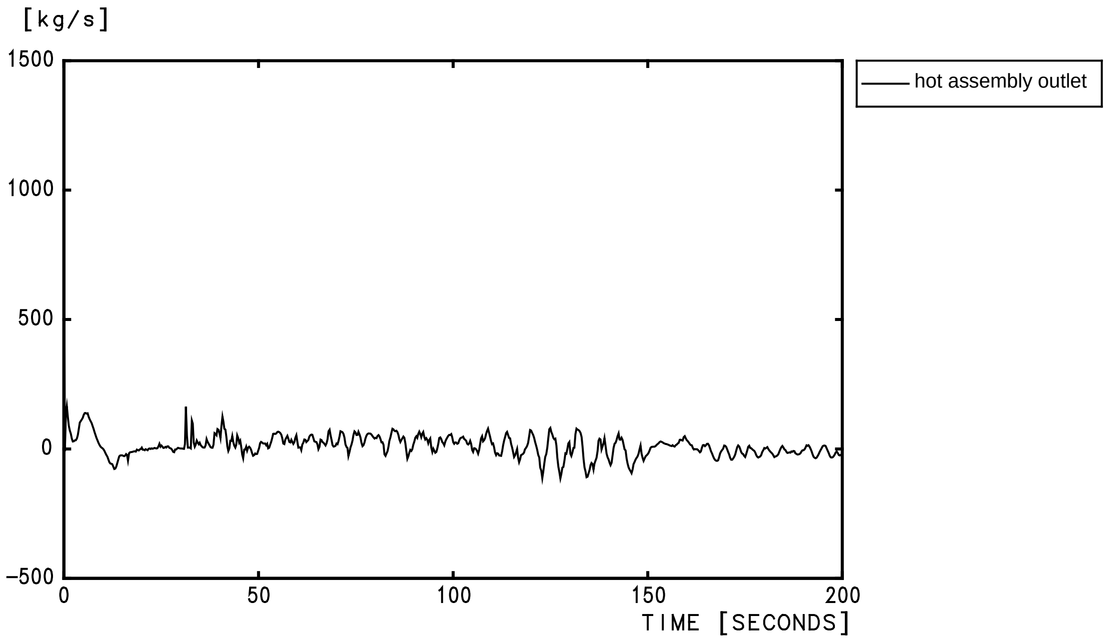
SECTION 16.4.1 - FIGURE E 18: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE CORE INLET



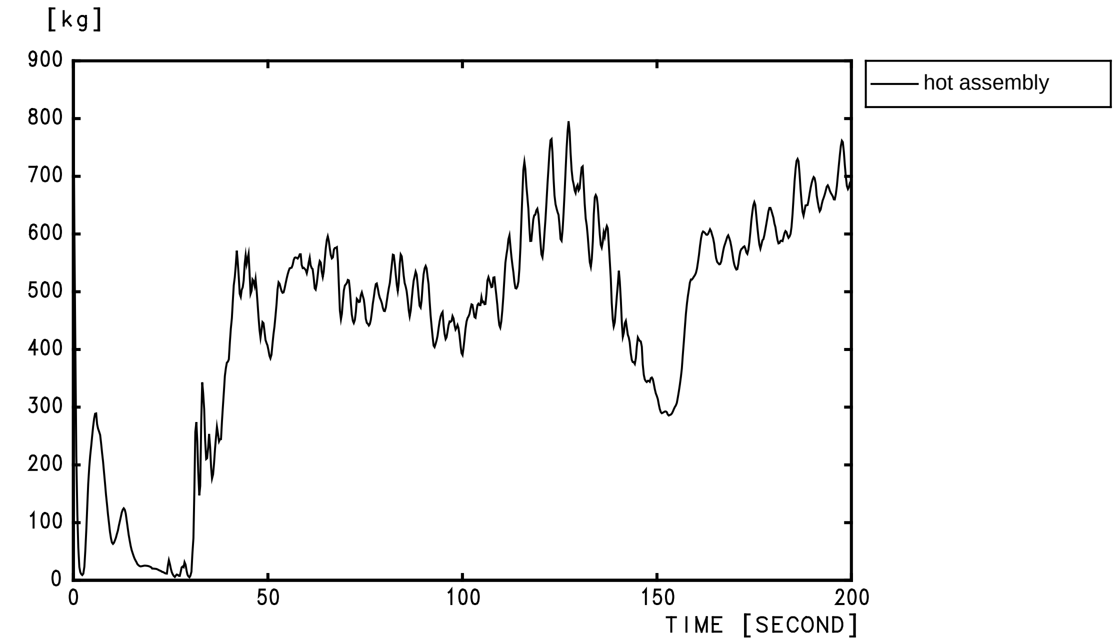
SECTION 16.4.1 - FIGURE E 19: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE CORE OUTLET



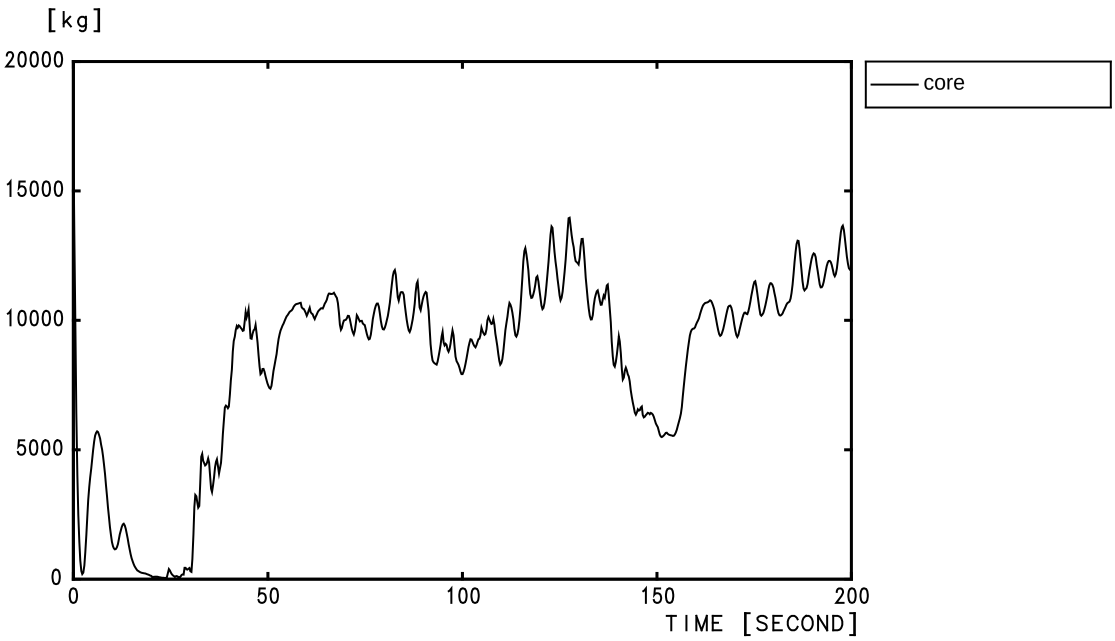
SECTION 16.4.1 - FIGURE E 20: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE HOT ASSEMBLY INLET



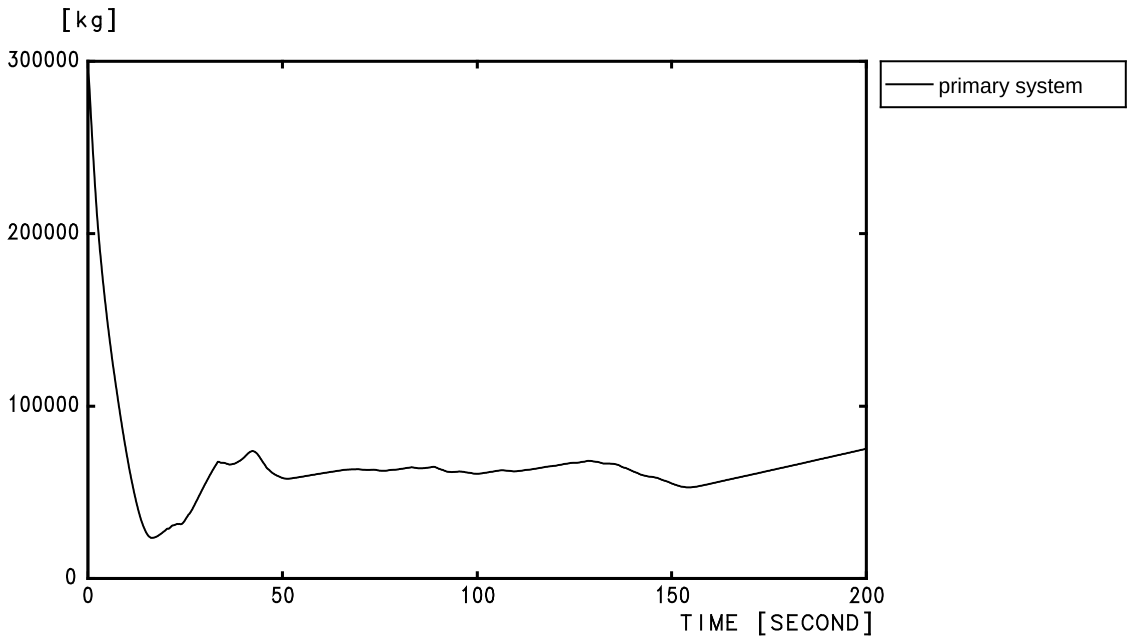
SECTION 16.4.1 - FIGURE E 21: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE HOT ASSEMBLY OUTLET



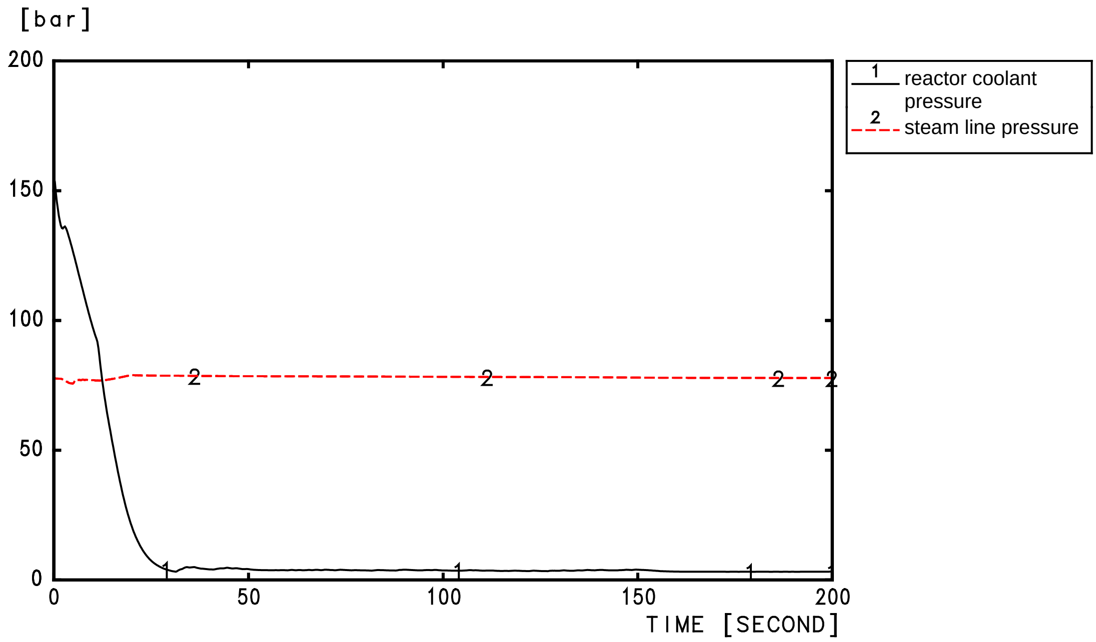
SECTION 16.4.1 - FIGURE E 22: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – LIQUID MASS IN THE HOT ASSEMBLY



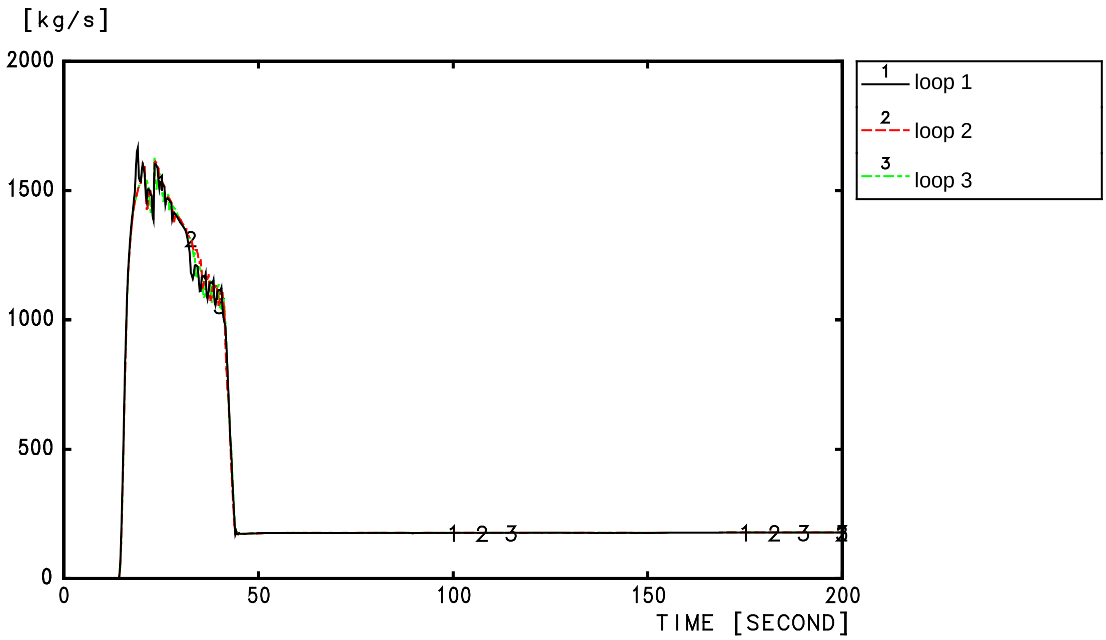
SECTION 16.4.1 - FIGURE E 23: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – LIQUID MASS IN THE CORE



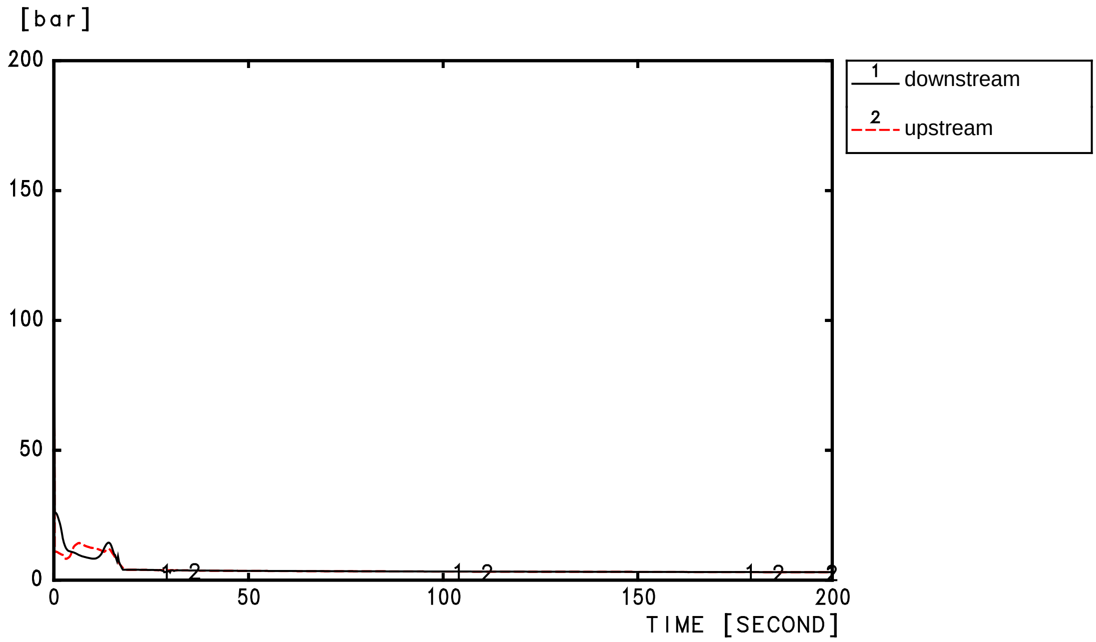
SECTION 16.4.1 - FIGURE E 24: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL MASS IN THE PRIMARY SYSTEM



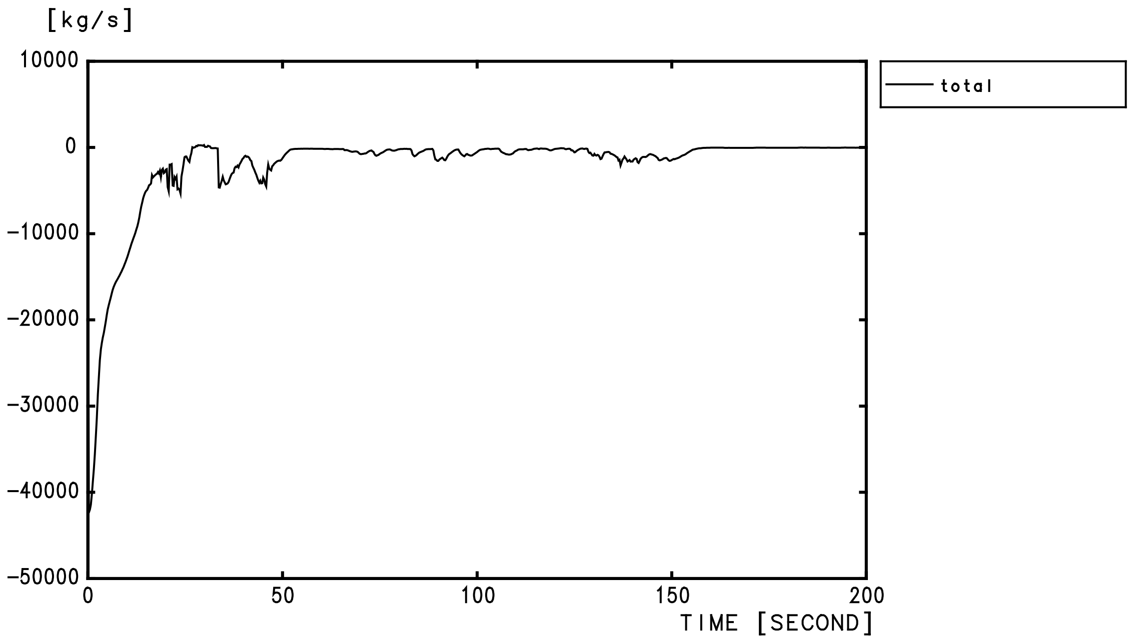
SECTION 16.4.1 - FIGURE E 25: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – REACTOR COOLANT PRESSURE AND STEAM LINE PRESSURE



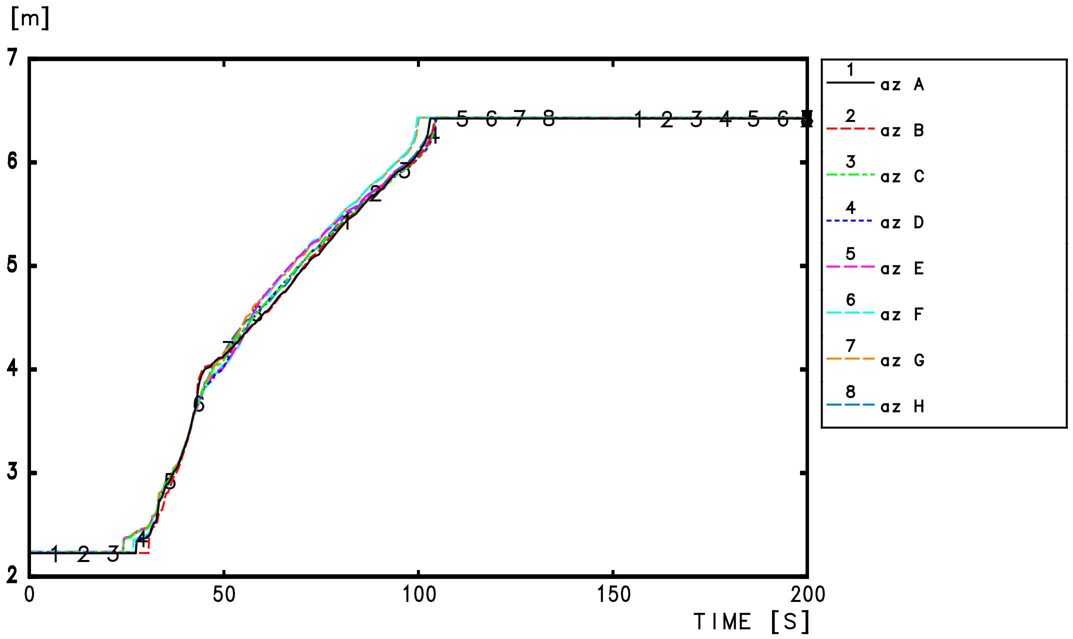
SECTION 16.4.1 - FIGURE E 26: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL RIS [SIS] FLOW RATE (ACCUMULATORS AND SI)



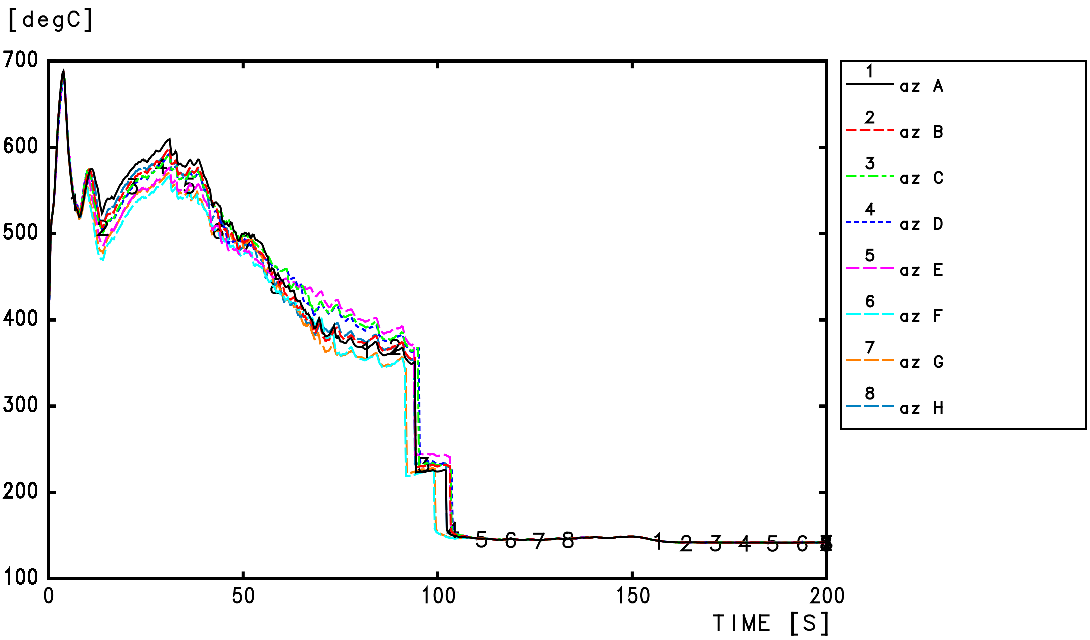
SECTION 16.4.1 - FIGURE E 27: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – PRESSURE AT THE BREAK



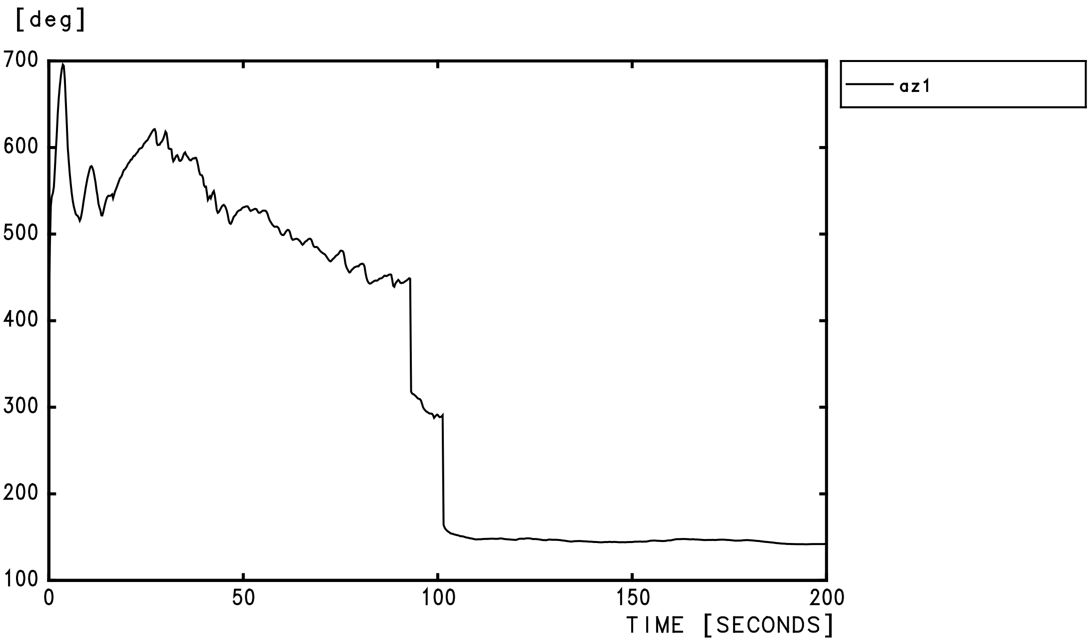
SECTION 16.4.1 - FIGURE E 28: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – TOTAL FLOW RATE AT THE BREAK



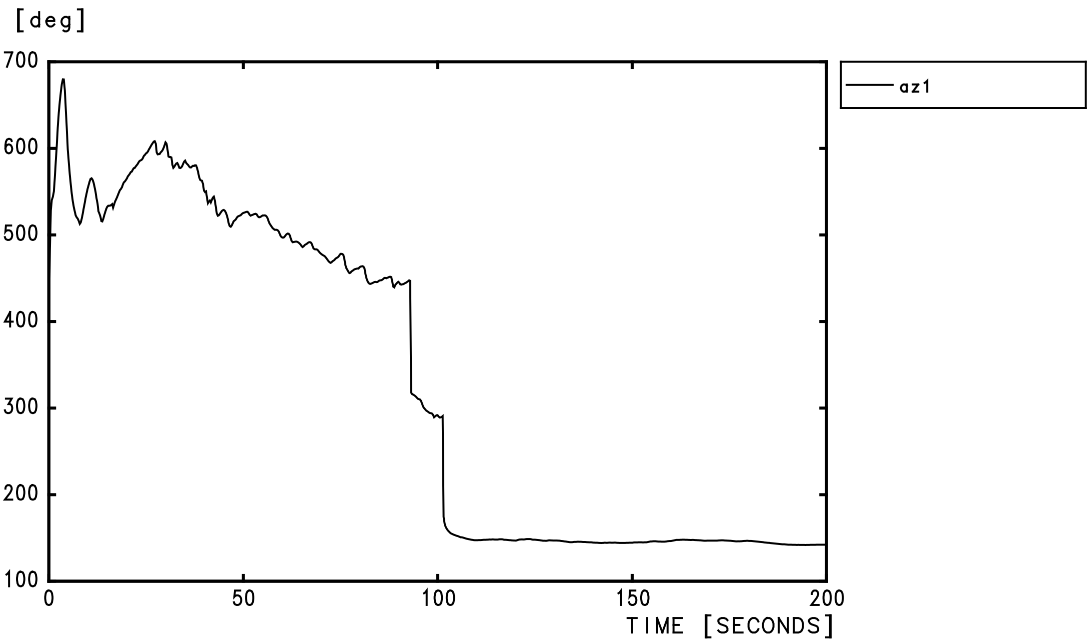
SECTION 16.4.1 - FIGURE E 29: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – QUENCH FRONT LEVEL IN HOT ASSEMBLIES



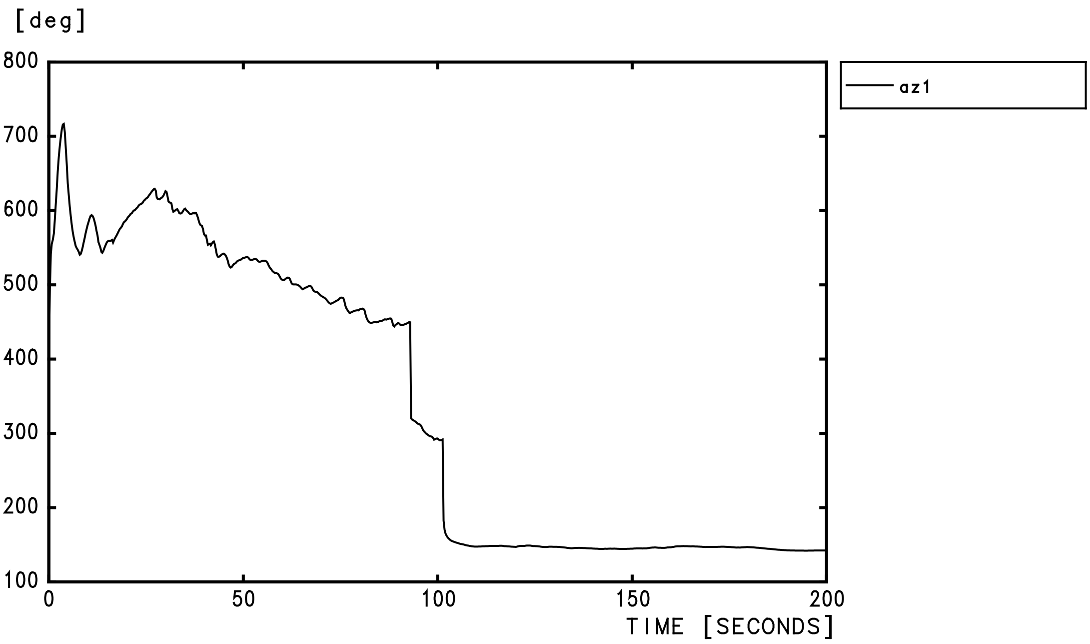
SECTION 16.4.1 - FIGURE E 30: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – MAXIMUM CLAD TEMPERATURES FOR HOT ASSEMBLIES



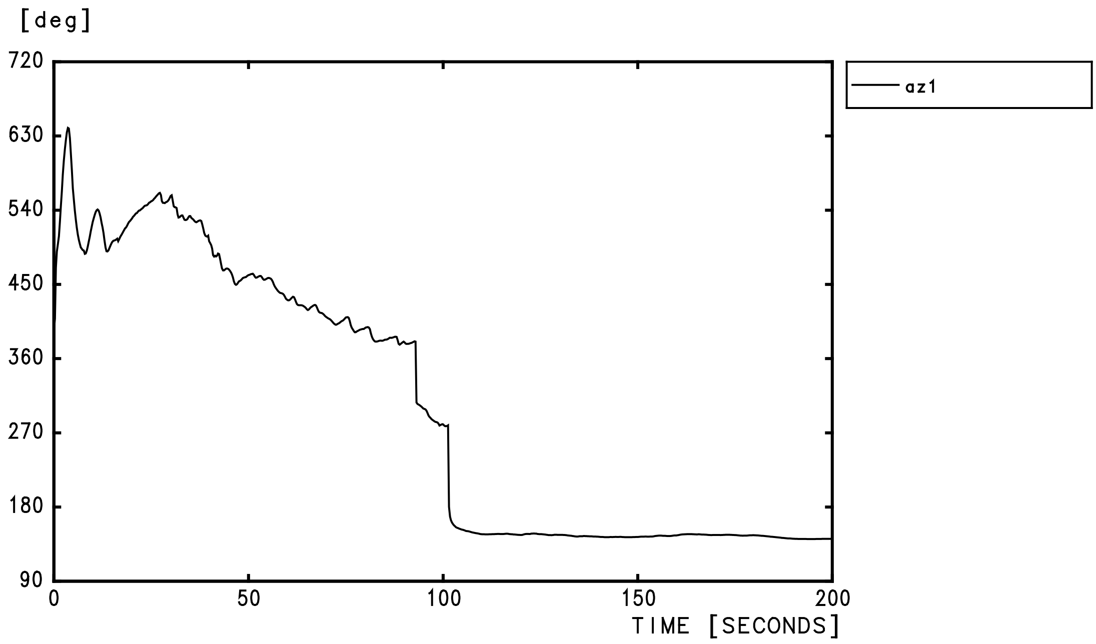
SECTION 16.4.1 - FIGURE E 31: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – BOL INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



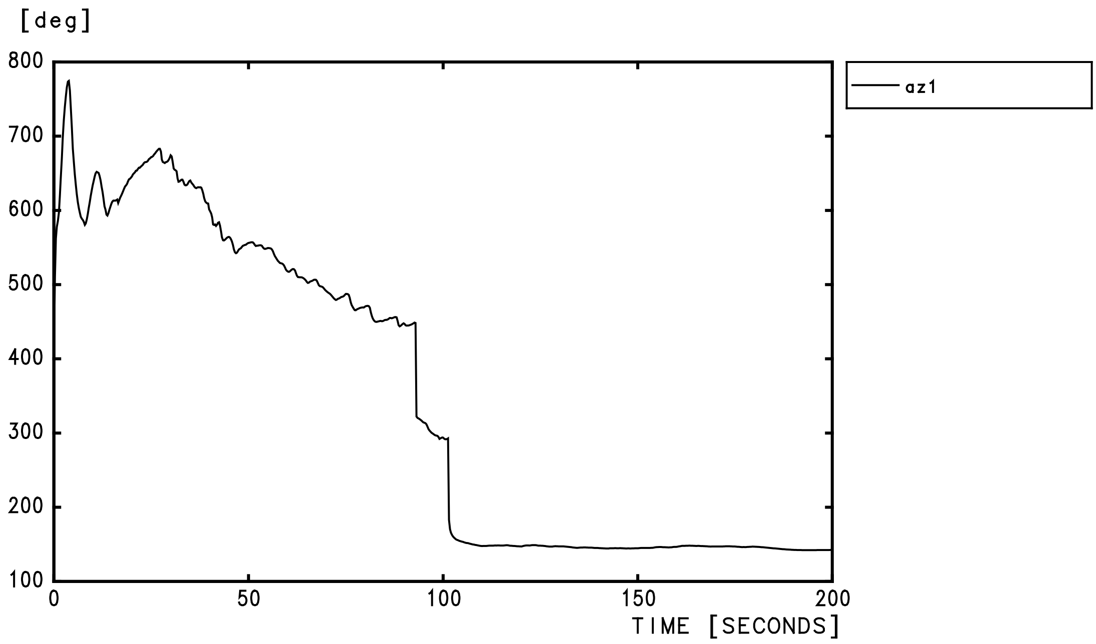
SECTION 16.4.1 - FIGURE E 32: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC1 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



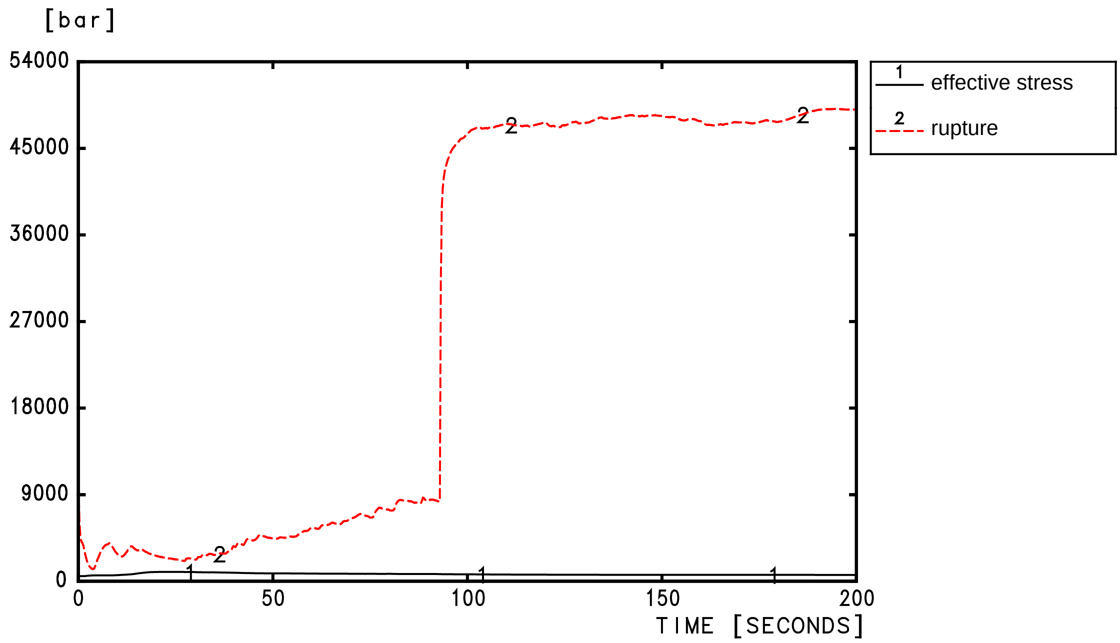
SECTION 16.4.1 - FIGURE E 33: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC2 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



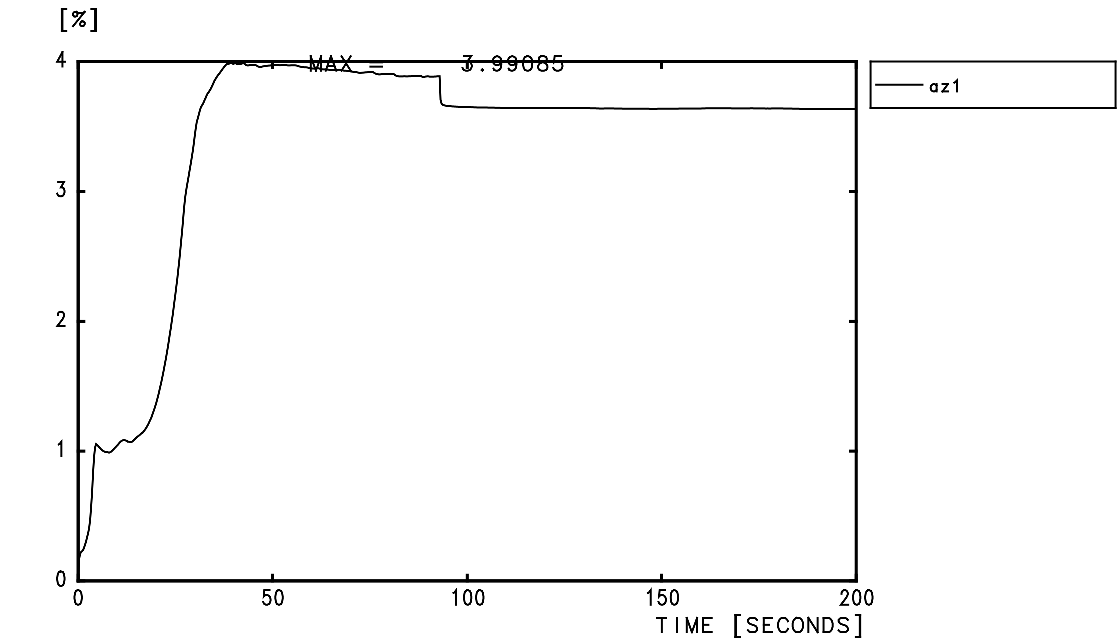
SECTION 16.4.1 - FIGURE E 34: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – C3 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



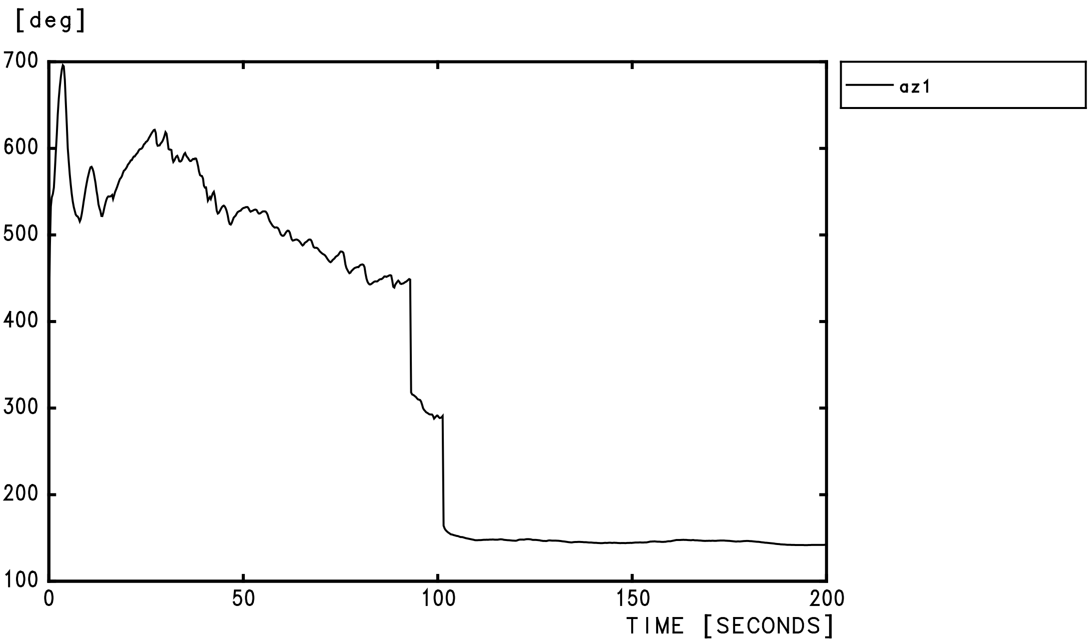
SECTION 16.4.1 - FIGURE E 35: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC4 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



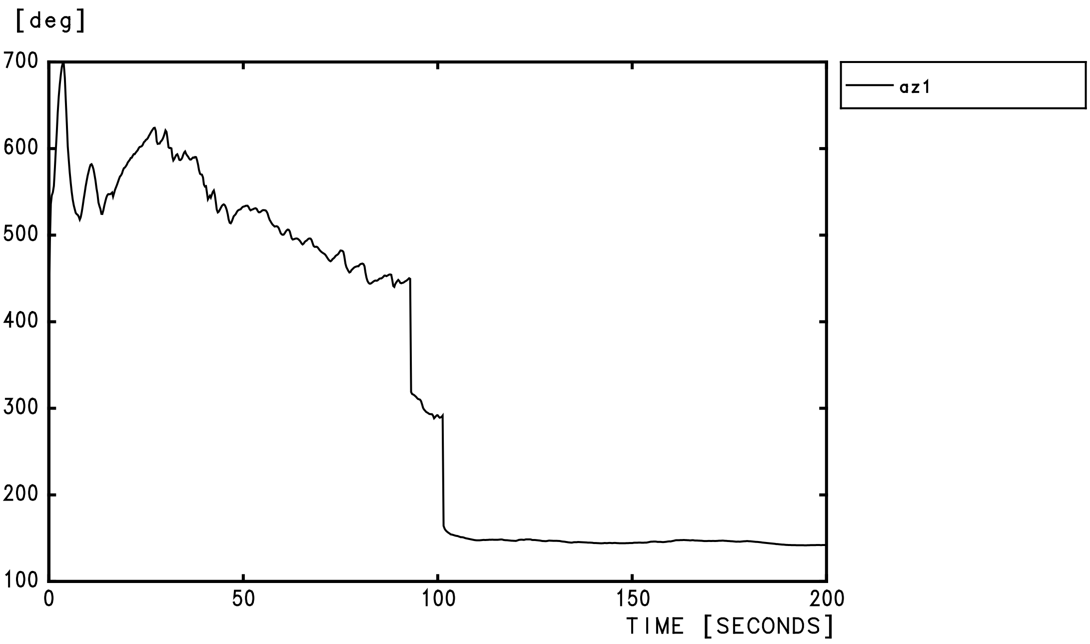
SECTION 16.4.1 - FIGURE E 36: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC4 INDICATOR ROD WITH A HIGH PU CONTENT – TRUE STRESS AND RUPTURE STRESS AT THE PEAK POWER HEIGHT



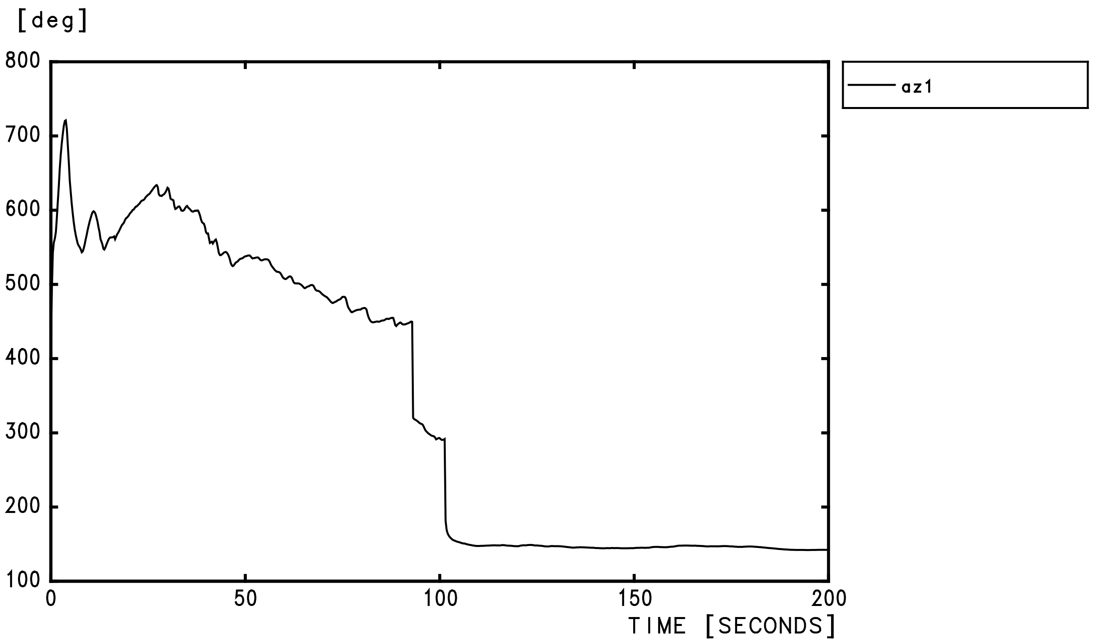
SECTION 16.4.1 - FIGURE E 37: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC4 INDICATOR ROD WITH A HIGH PU CONTENT – DEFORMATION AT THE PEAK POWER HEIGHT



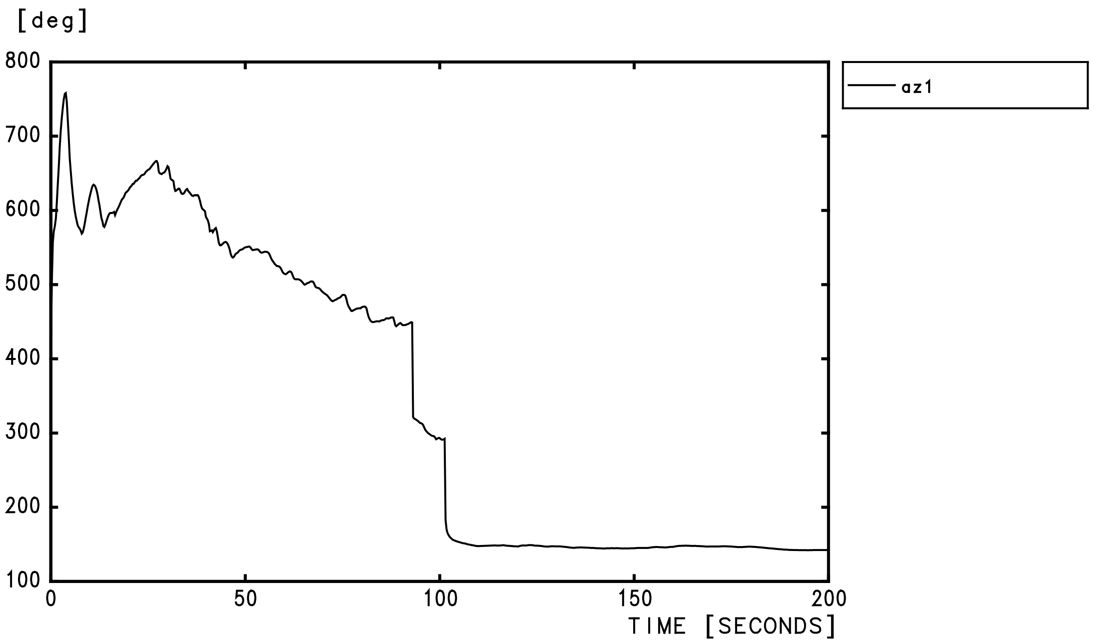
SECTION 16.4.1 - FIGURE E 38: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – BOL INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



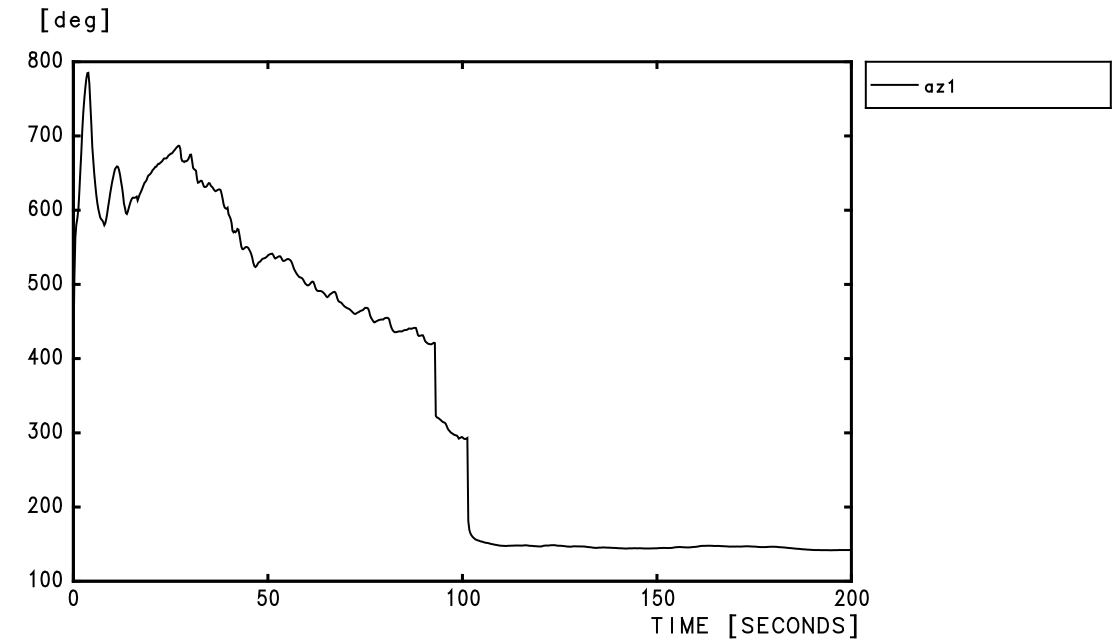
SECTION 16.4.1 - FIGURE E 39: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC1 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



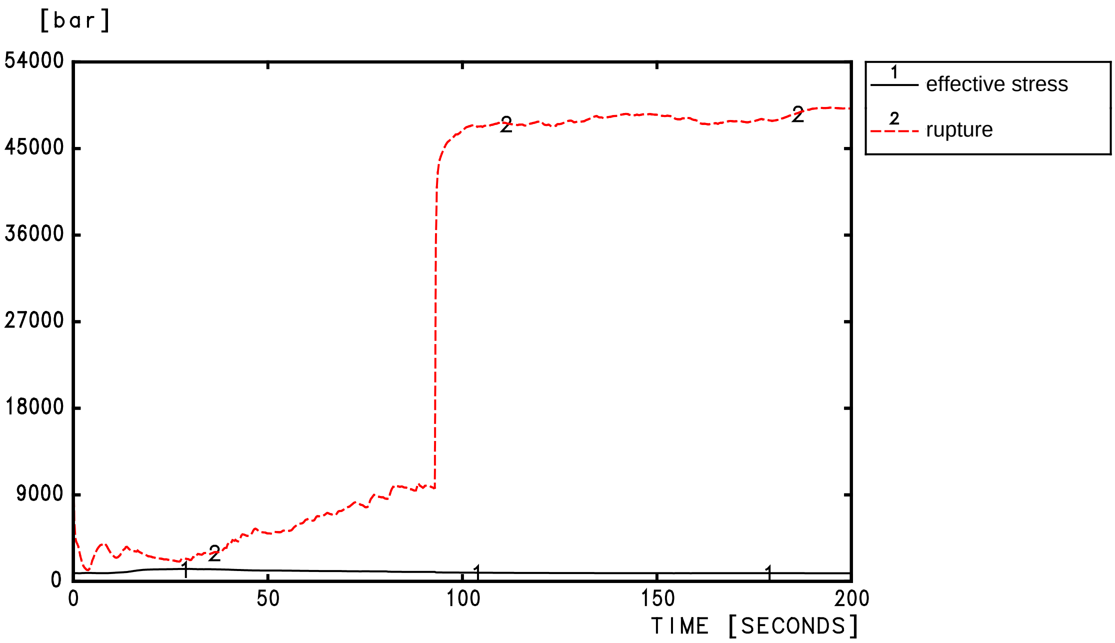
SECTION 16.4.1 - FIGURE E 40: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC2 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



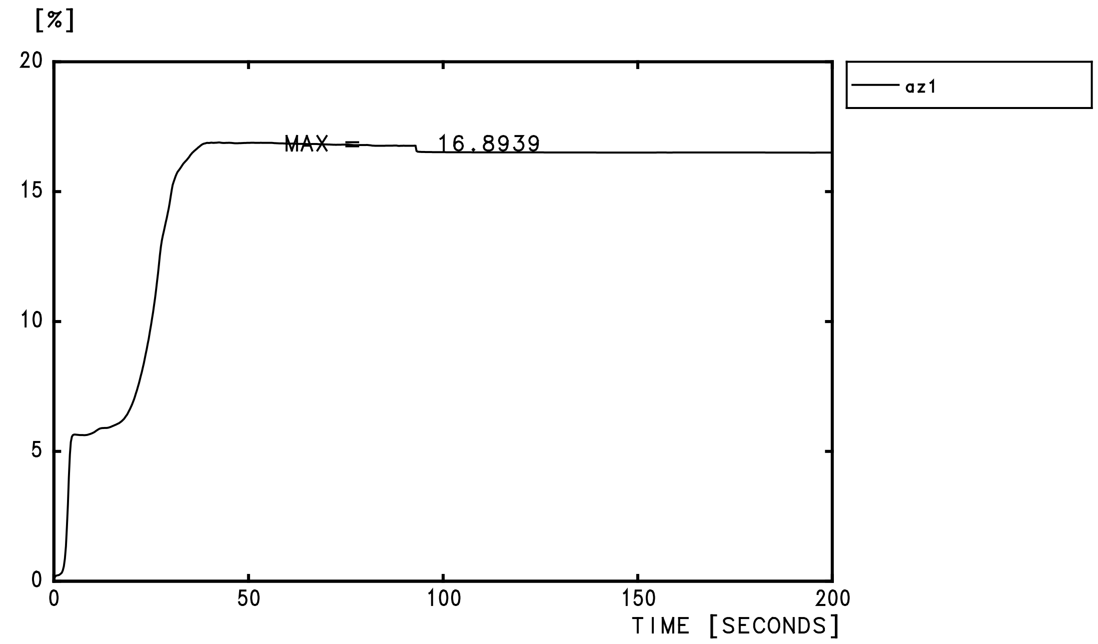
SECTION 16.4.1 - FIGURE E 41: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – C3 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



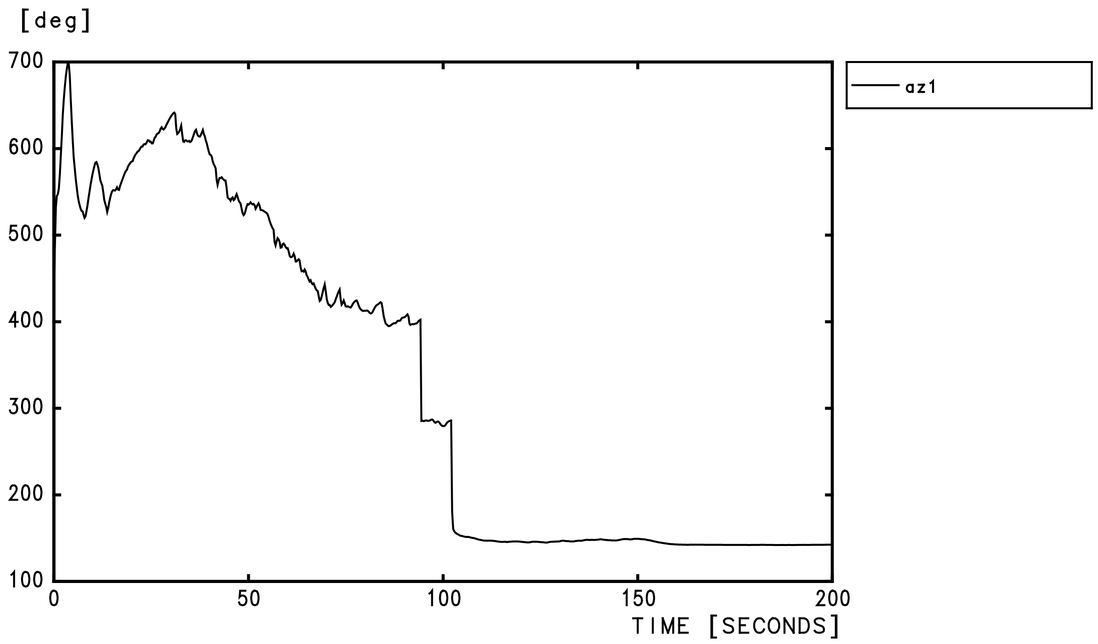
SECTION 16.4.1 - FIGURE E 42: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC4 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



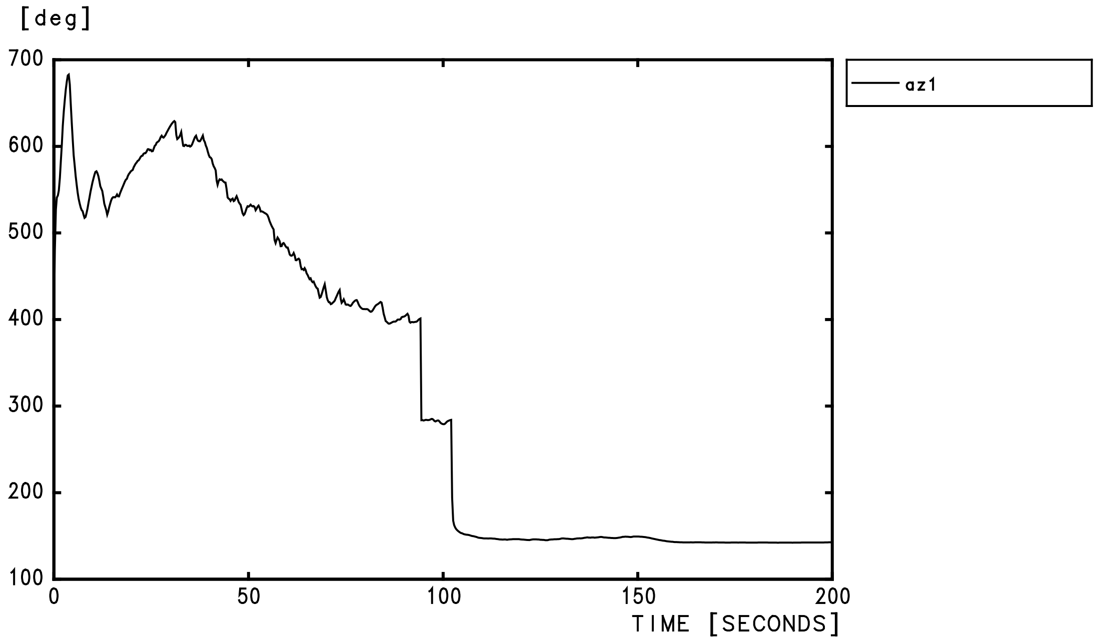
SECTION 16.4.1 - FIGURE E 43: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC4 INDICATOR ROD WITH AN AVERAGE PU CONTENT – TRUE STRESS AND RUPTURE STRESS AT THE PEAK POWER HEIGHT



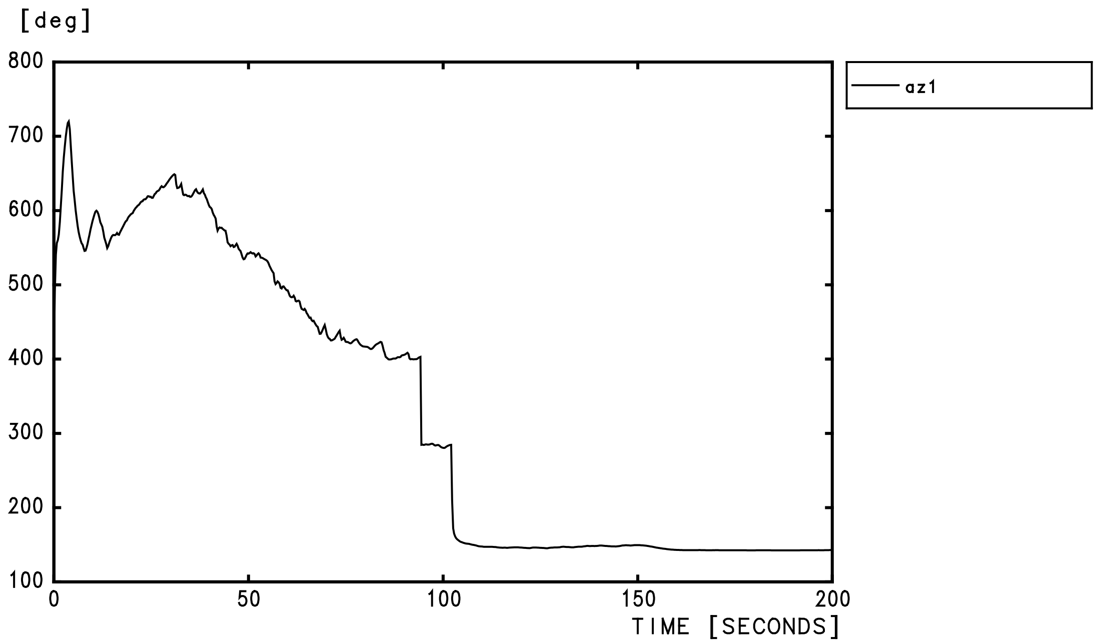
SECTION 16.4.1 - FIGURE E 44: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA1 – EOC4 INDICATOR ROD WITH AN AVERAGE PU CONTENT – DEFORMATION AT THE PEAK POWER HEIGHT



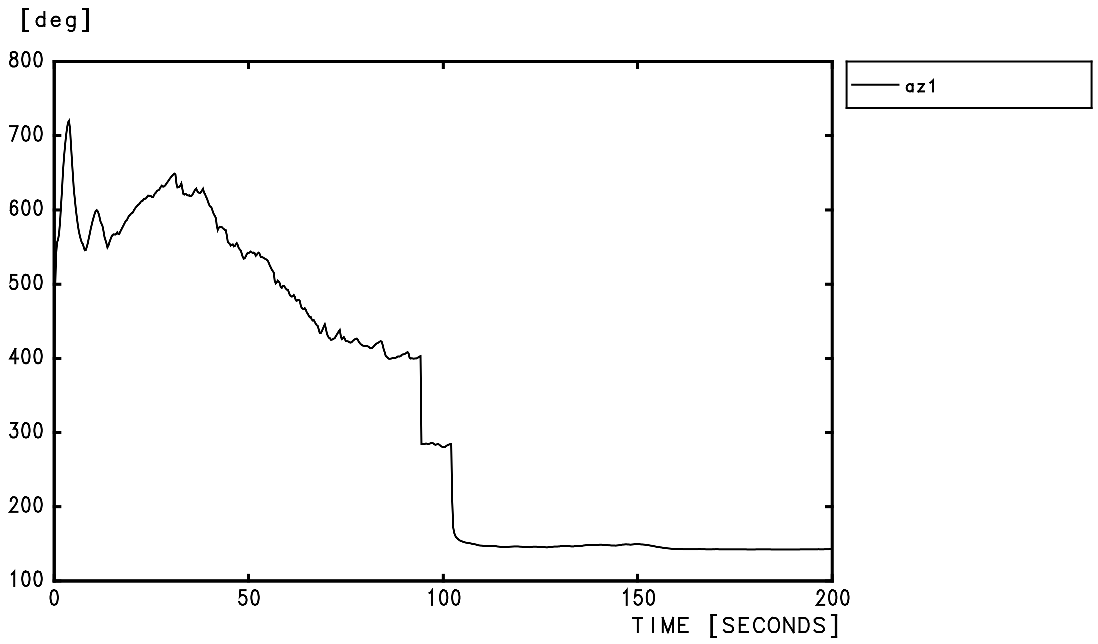
SECTION 16.4.1 - FIGURE E 45: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – BOL INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



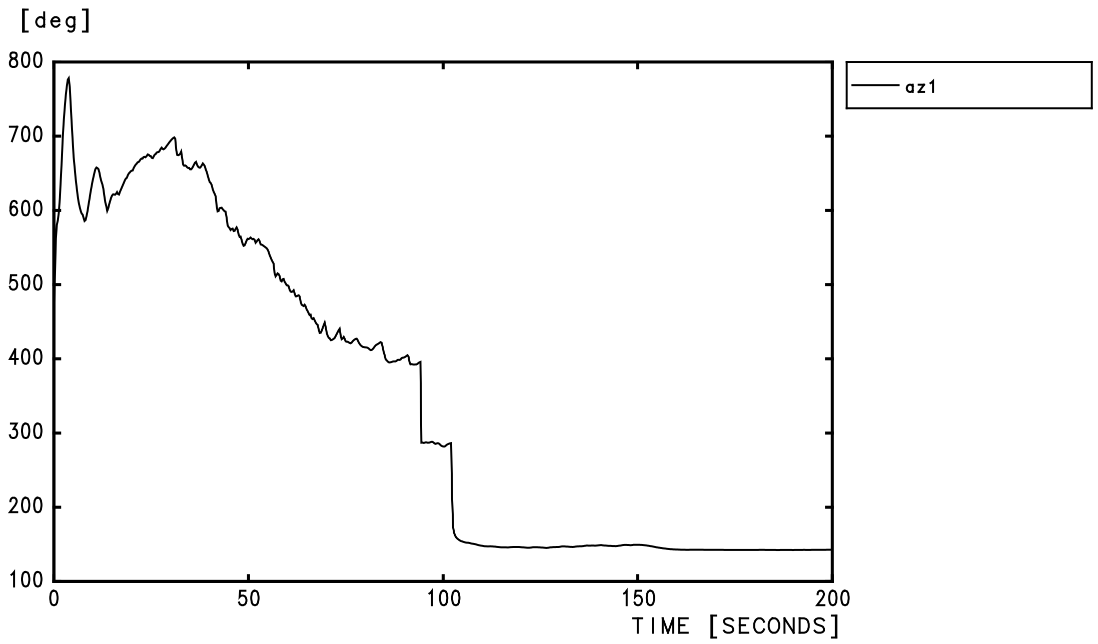
SECTION 16.4.1 - FIGURE E 46: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC1 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



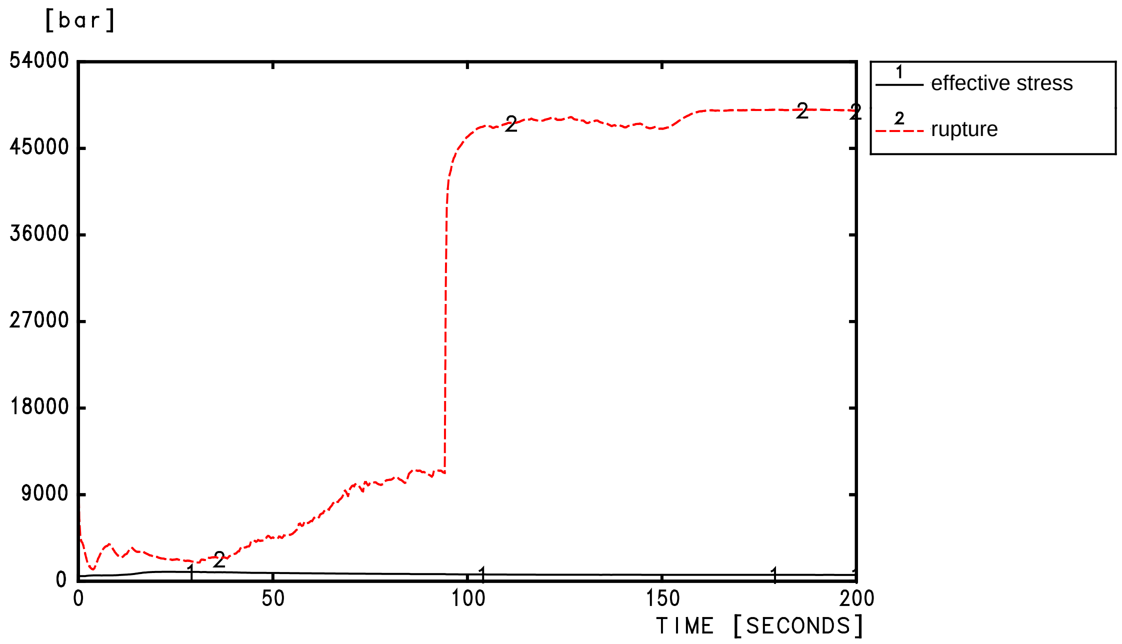
SECTION 16.4.1 - FIGURE E 47: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC2 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



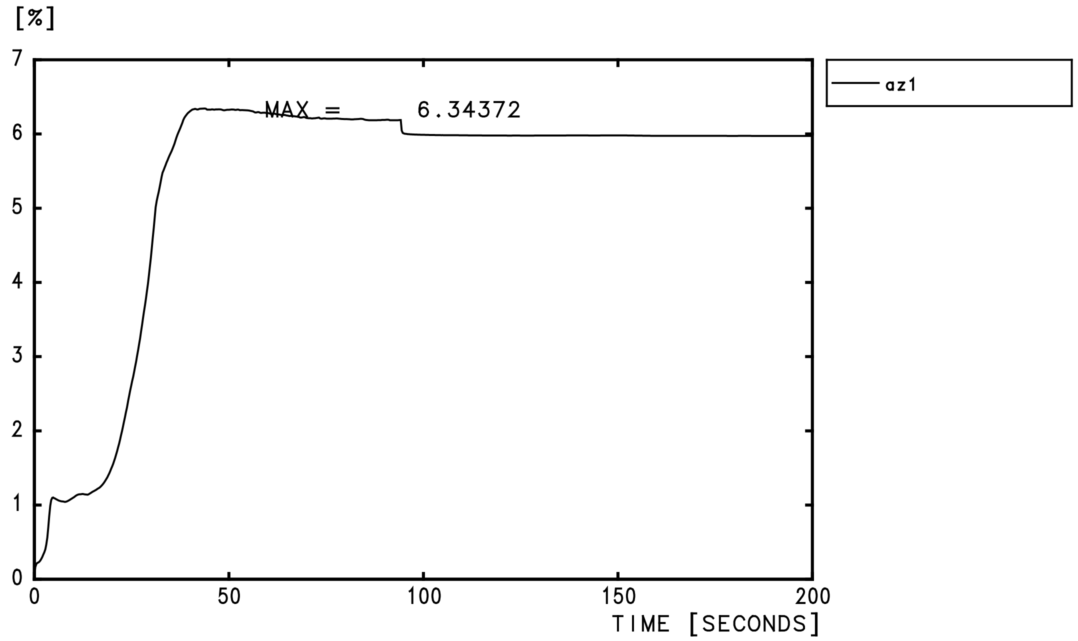
SECTION 16.4.1 - FIGURE E 48: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – C3 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



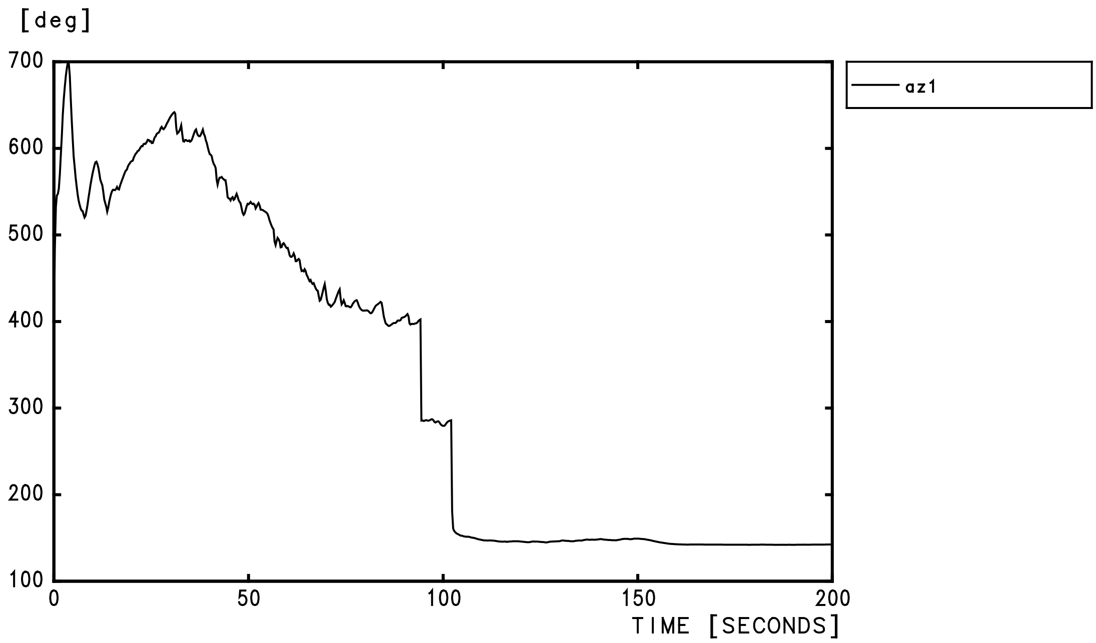
SECTION 16.4.1 - FIGURE E 49: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC4 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



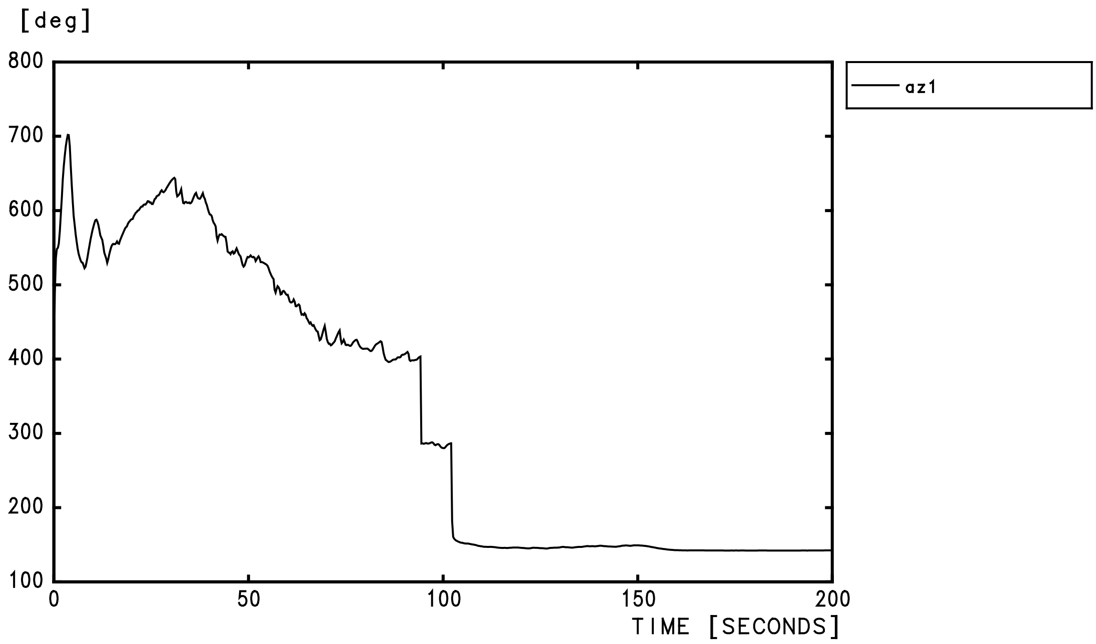
SECTION 16.4.1 - FIGURE E 50: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC4 INDICATOR ROD WITH A HIGH PU CONTENT – TRUE STRESS AND RUPTURE STRESS AT THE PEAK POWER HEIGHT



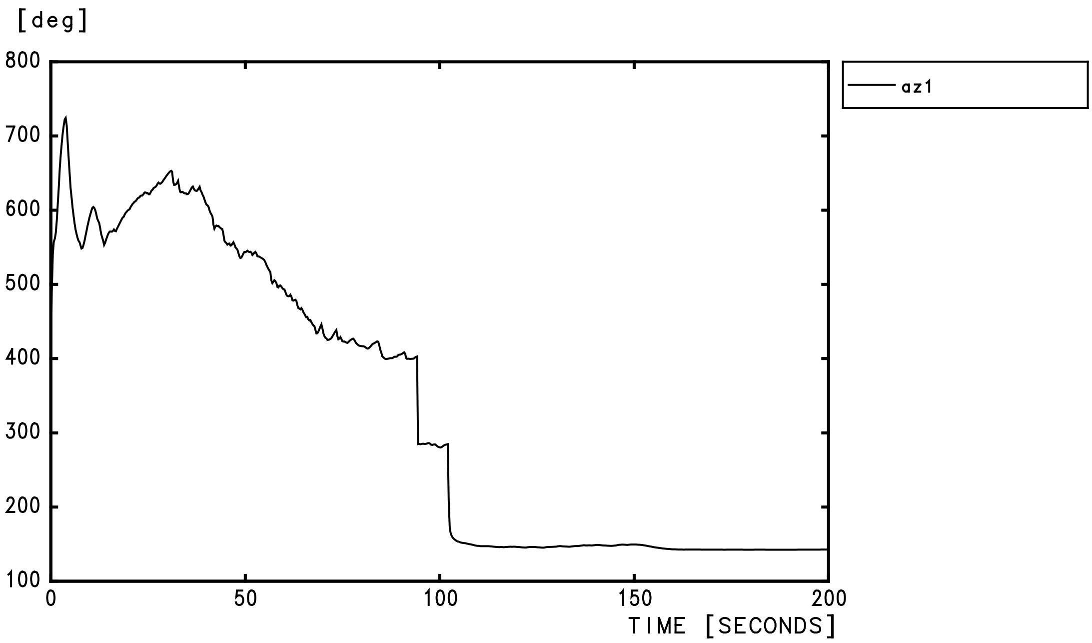
SECTION 16.4.1 - FIGURE E 51: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC4 INDICATOR ROD WITH A HIGH PU CONTENT – DEFORMATION AT THE PEAK POWER HEIGHT



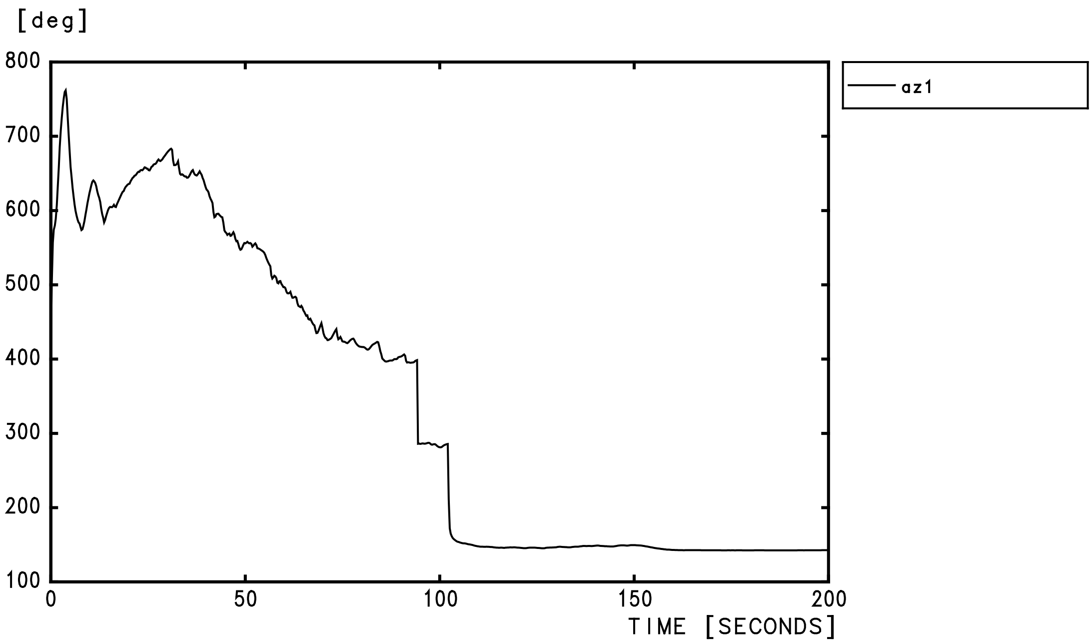
SECTION 16.4.1 - FIGURE E 52: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – BOL INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



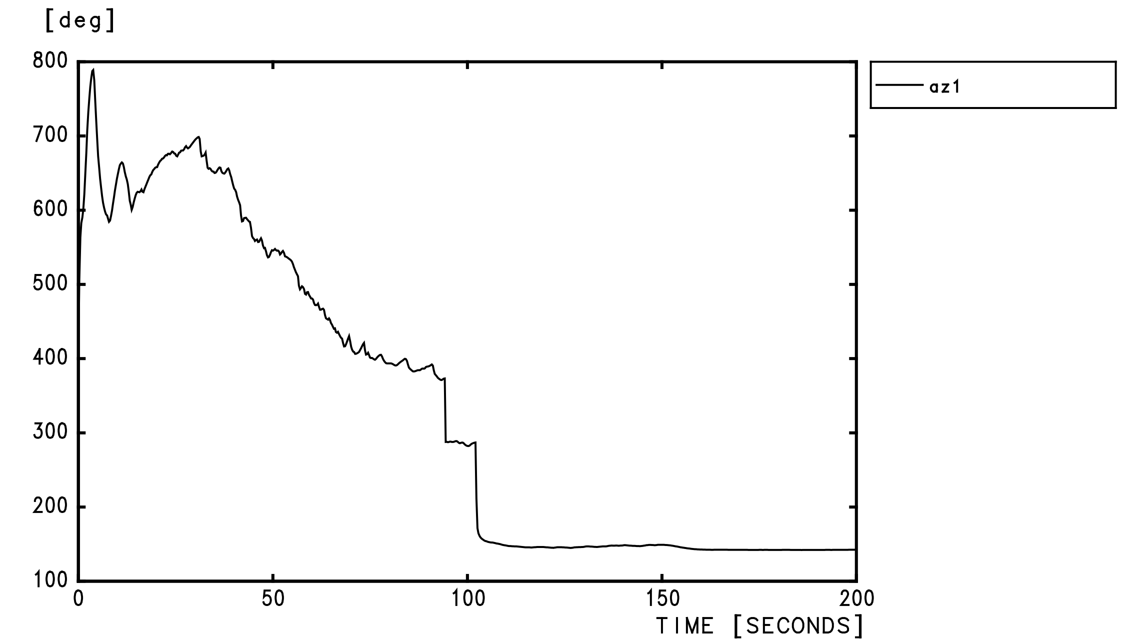
SECTION 16.4.1 - FIGURE E 53: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC1 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



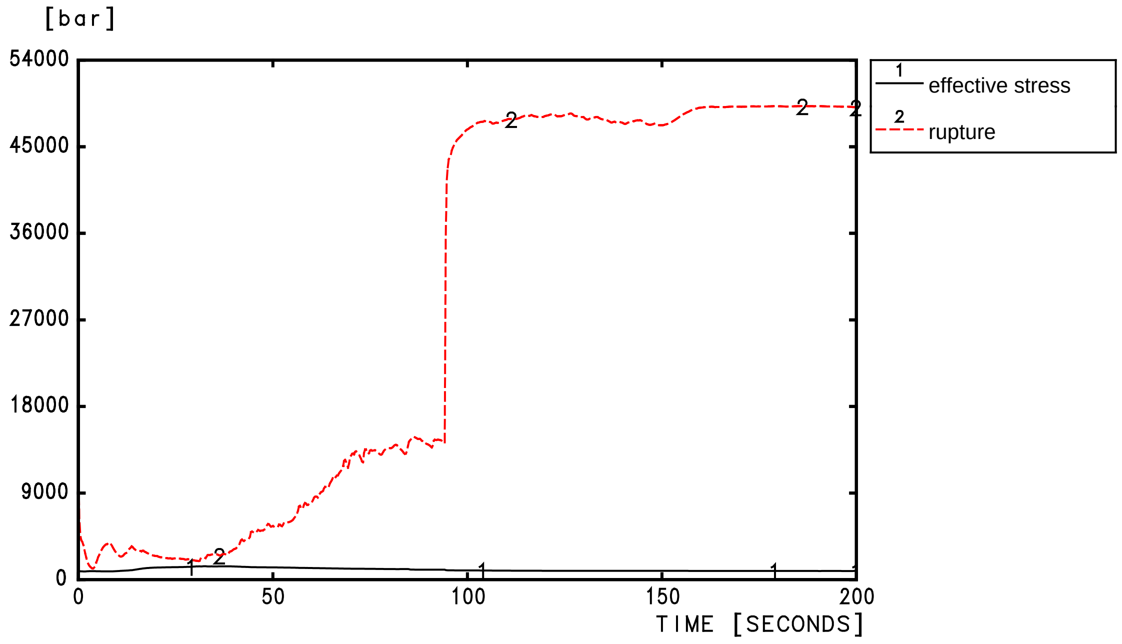
SECTION 16.4.1 - FIGURE E 54: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC2 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



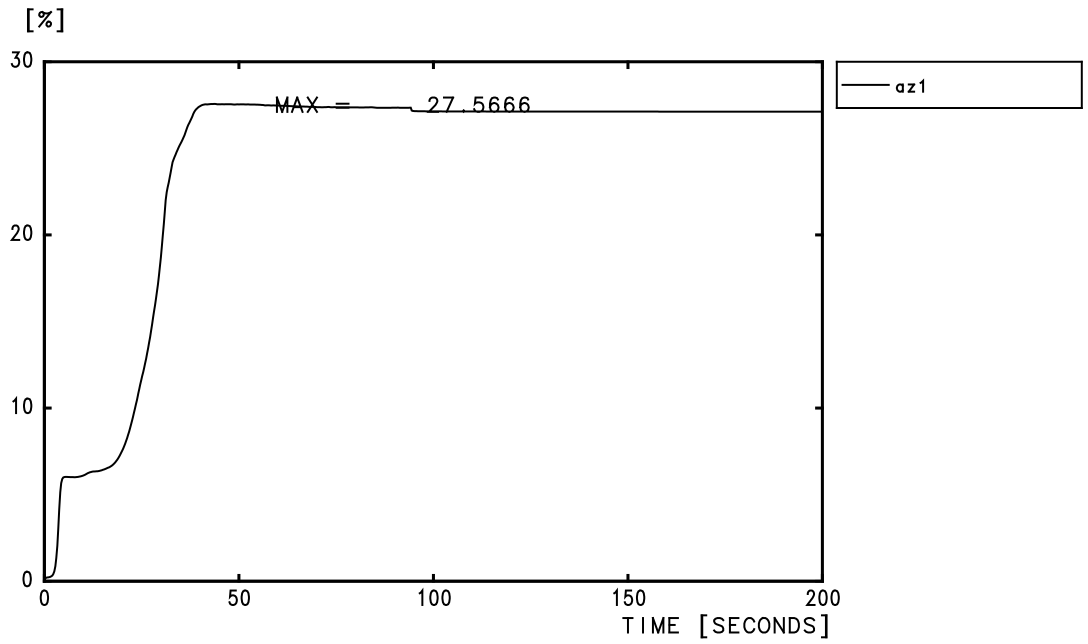
SECTION 16.4.1 - FIGURE E 55: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – C3 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



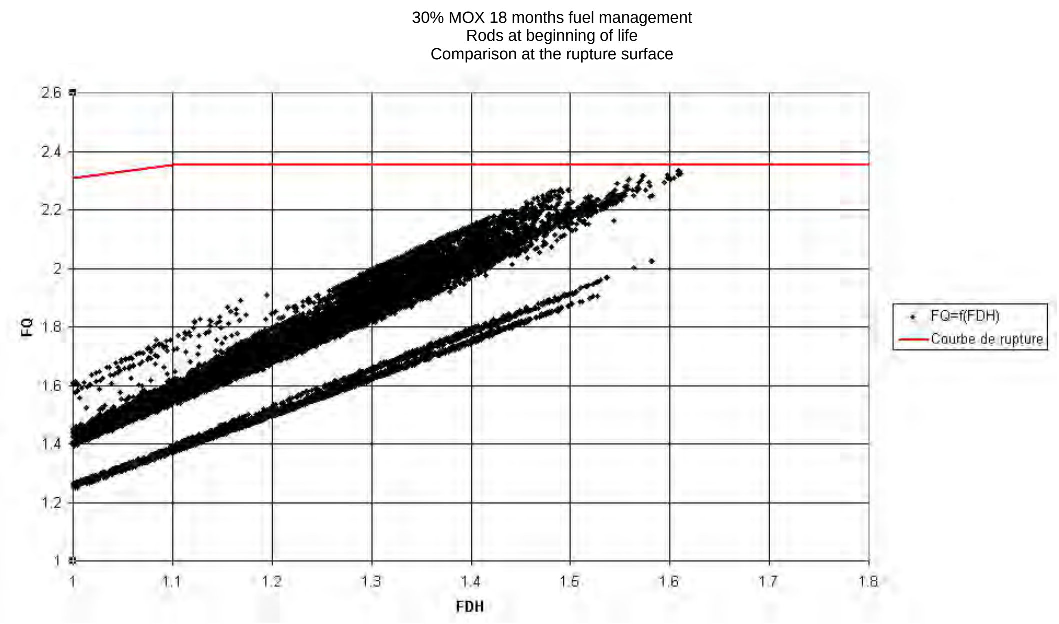
SECTION 16.4.1 - FIGURE E 56: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC4 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



SECTION 16.4.1 - FIGURE E 57: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC4 INDICATOR ROD WITH AN AVERAGE PU CONTENT – TRUE STRESS AND RUPTURE STRESS AT THE PEAK POWER HEIGHT

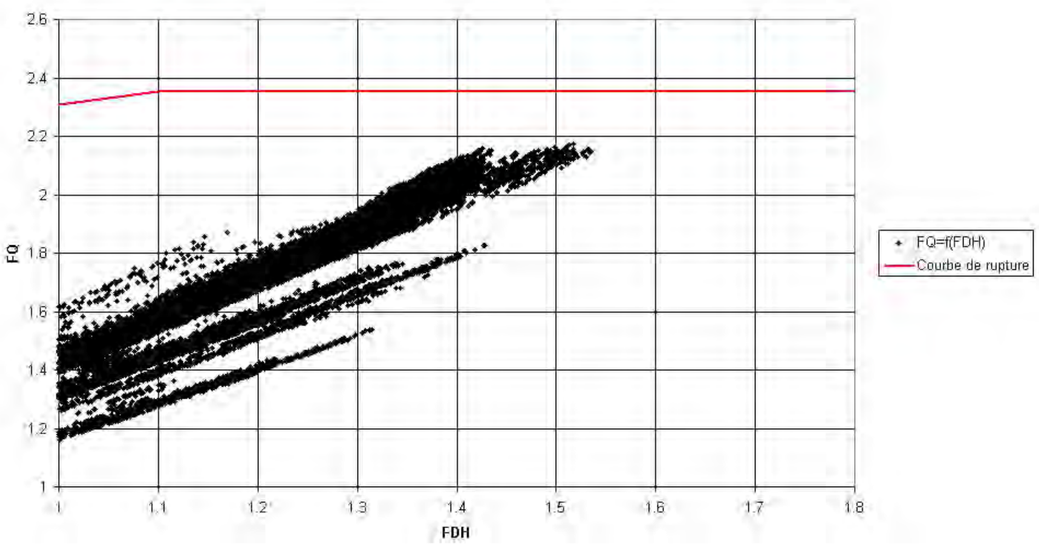


SECTION 16.4.1 - FIGURE E 58: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – HA2 – EOC4 INDICATOR ROD WITH AN AVERAGE PU CONTENT – DEFORMATION AT THE PEAK POWER HEIGHT



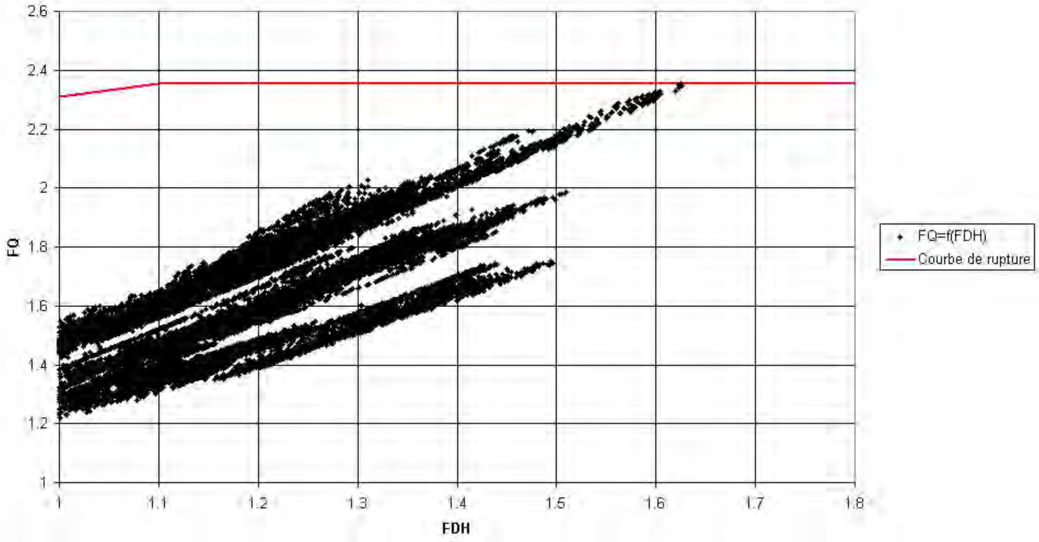
SECTION 16.4.1 - FIGURE E 59: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – RUPTURE CURVE – BOL

30% MOX 18 months fuel management
Rods at end of cycle 1
Comparison at the rupture surface

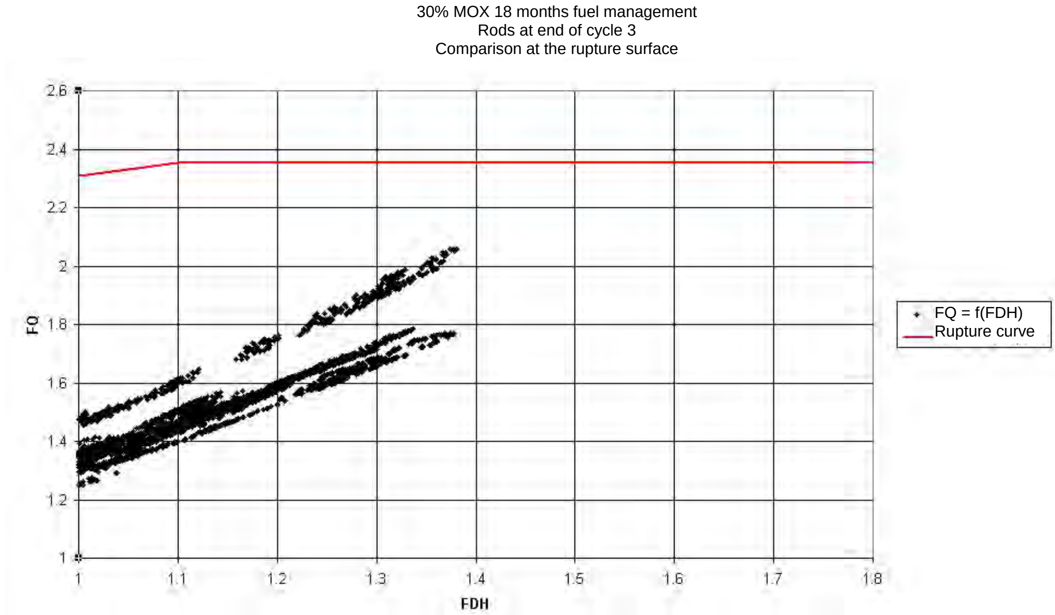


SECTION 16.4.1 - FIGURE E 60: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – RUPTURE CURVE – EOC1

30% MOX 18 months fuel management
Rods at end of cycle 2
Comparison at the rupture surface



SECTION 16.4.1 - FIGURE E 61: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – RUPTURE CURVE – EOC2



SECTION 16.4.1 - FIGURE E 62: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – TC – RUPTURE CURVE – C3

SECTION 16.4.1 - TABLE F 1: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – PROPERTIES OF MOX RODS

	Content
Rods with an average Pu content	10.4% Pu total
Rods with a high Pu content	12.0% Pu total

SECTION 16.4.1 - TABLE F 2: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – FUEL CALCULATION POINTS – RODS WITH A HIGH PU CONTENT

Core cycle time	**Rod burn-up**
Beginning of life (BOL)	150 MWd/tU
End of Cycle 1 (EOC1)	18,500 MWd/tU
End of Cycle 2 (EOC2)	44,050 MWd/tU
Cycle 3 (C3)	65,000 MWd/tU
End of Cycle 4 (EOC4)	73,550 MWd/tU

SECTION 16.4.1 - TABLE F 3: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – FUEL CALCULATION POINTS – RODS WITH AN AVERAGE PU CONTENT

Core cycle time	**Rod burn-up**
Beginning of life (BOL)	150 MWd/tU
End of Cycle 1 (EOC1)	20,050 MWd/tU
End of Cycle 2 (EOC2)	45,700 MWd/tU
Cycle 3 (C3)	65,000 MWd/tU
End of Cycle 4 (EOC4)	77,250 MWd/tU

SECTION 16.4.1 - TABLE F 4: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – BOUNDING DATA USED TO SIMULATE A ROD AT THE END OF CYCLE 4 – ROD WITH A HIGH PU CONTENT

FΔH	FQ	Average pellet temperature (°C)	Pressure (bar)	External oxide thickness (μm)
1.381	2.059	930	183	33.20
1.273	1.674	1068	183	33.20
1.626	2.353	1181	183	33.20

SECTION 16.4.1 - TABLE F 5: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – BOUNDING DATA USED TO SIMULATE A ROD AT THE END OF CYCLE 4 – ROD WITH AN AVERAGE PU CONTENT

FΔH	FQ	Average pellet temperature (°C)	Pressure (bar)	External oxide thickness (μm)
1.626	2.353	1227	223	32.10

SECTION 16.4.1 - TABLE F 6: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – PROPERTIES OF HOT ASSEMBLIES

Hot assembly	FΔH	FQ	BU	Fuel
HA1	1.4	1.94	EOC1	MOX
HA2	1.33	1.79	C3	MOX

SECTION 16.4.1 - TABLE F 7: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – MAIN EVENTS OF THE TRANSIENT

Event	Time at which the event occurs
RT signal	3.6 s
SI signal	7.1 s
Start of accumulator injection	14.2 s
Start of safety injection system injection	23.0 s
Start of reflood	24.1 s
End of reflood	119.6 s
Duration of reflood	95.5 s
End of accumulator injection	44.4 s

SECTION 16.4.1 - TABLE F 8: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – PEAK CLAD TEMPERATURES

	T1	T2	T3
Time (s)	3.1	30.7	49.2
Mesh	6	7	7
Value (°C)	642	579	471

SECTION 16.4.1 - TABLE F 9: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – MAIN EVENTS OF THE TRANSIENT

Event	Time at which the event occurs
RT signal	3.6 s
SI signal	7.1 s
Start of accumulator injection	14.2 s
Start of safety injection system injection	23.0 s
Start of reflood	24.1 s
End of reflood	105.1 s
Duration of reflood	81.0 s
End of accumulator injection	44.3 s

SECTION 16.4.1 - TABLE F 10: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – PEAK CLAD TEMPERATURES

	T1	T2	T3
Time (s)	3.2	30.1	51.3
Mesh	6	7	7
Value (°C)	660	596	475

SECTION 16.4.1 - TABLE F 11: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – PROPERTIES OF THE INDICATOR RODS WITH A HIGH PU CONTENT

Burn-up	FQ	F∆H	Pressure in the gap (bar)	Maximum linear power density (W/cm)	Maximum initial pellet temperature (°C)
BOL	2.257	1.635	56	378.6	913
EOC1	2.257	1.635	63	378.6	861
EOC2	2.257	1.635	99	378.6	963
C3	2.257	1.635	164	378.6	1114
EOC4	2.257	1.635	183	378.6	1177

SECTION 16.4.1 - TABLE F 12: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – PROPERTIES OF THE INDICATOR RODS WITH AN AVERAGE PU CONTENT

Burn-up	FQ	FΔH	Pressure in the gap (bar)	Maximum linear power density (W/cm)	Maximum initial pellet temperature (°C)
BOL	2.257	1.635	56	378.6	913
EOC1	2.257	1.635	56	378.6	916
EOC2	2.257	1.635	117	378.6	973
C3	2.257	1.635	183	378.6	1115
EOC4	2.257	1.635	223	378.6	1209

SECTION 16.4.1 - TABLE F 13: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – RESULTS OBTAINED FOR INDICATOR RODS WITH A HIGH PU CONTENT

Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Rupture
BOL	681	620	510	0.35%	NO
EOC1	665	608	506	0.34%	NO
EOC2	703	628	518	0.36%	NO
C3	748	666	536	1.44%	NO
EOC4	766	684	540	4.00%	NO

SECTION 16.4.1 - TABLE F 14: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – RESULTS OBTAINED FOR INDICATOR RODS WITH AN AVERAGE PU CONTENT

Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Rupture
BOL	681	620	510	0.35%	NO
EOC1	683	621	512	0.33%	NO
EOC2	706	631	519	0.37%	NO
C3	749	666	533	2.45%	NO
EOC4	774	686	525	13.20%	NO

SECTION 16.4.1 - TABLE F 15: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – RESULTS OBTAINED FOR INDICATOR RODS WITH A HIGH PU CONTENT

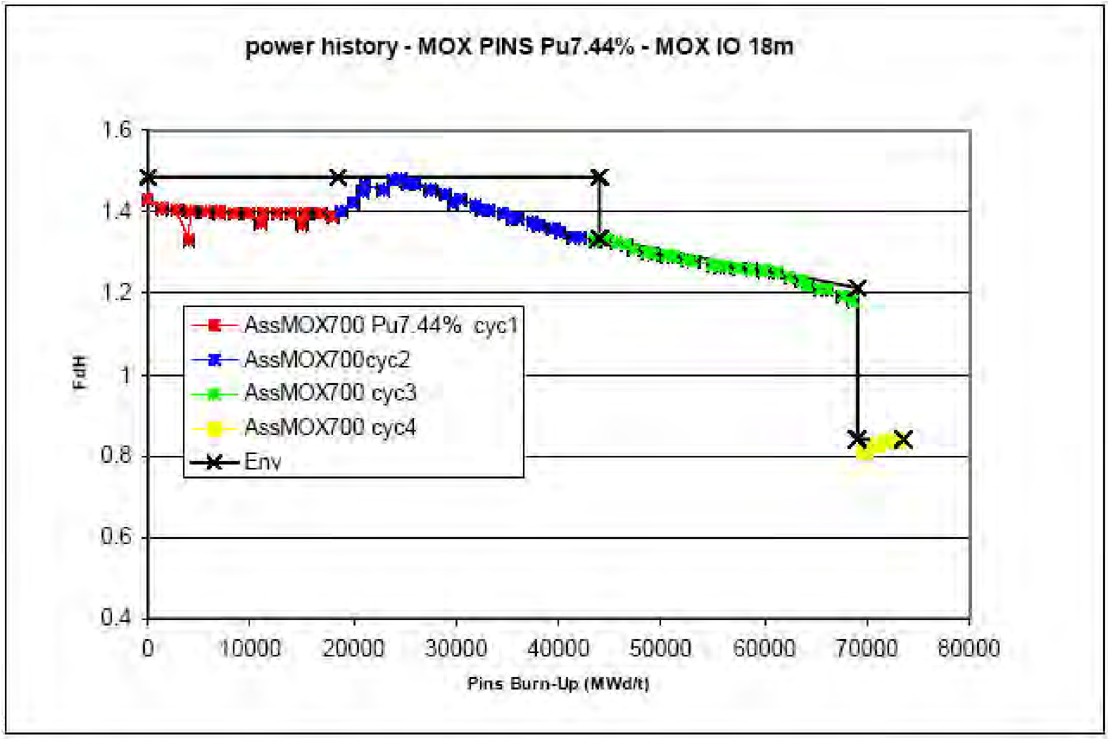
Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Rupture
BOL	678	624	511	0.36%	NO
EOC1	662	613	505	0.34%	NO
EOC2	701	633	514	0.36%	NO
C3	748	671	532	1.54%	NO
EOC4	764	687	536	4.26%	NO

SECTION 16.4.1 - TABLE F 16: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – RESULTS OBTAINED FOR INDICATOR RODS WITH AN AVERAGE PU CONTENT

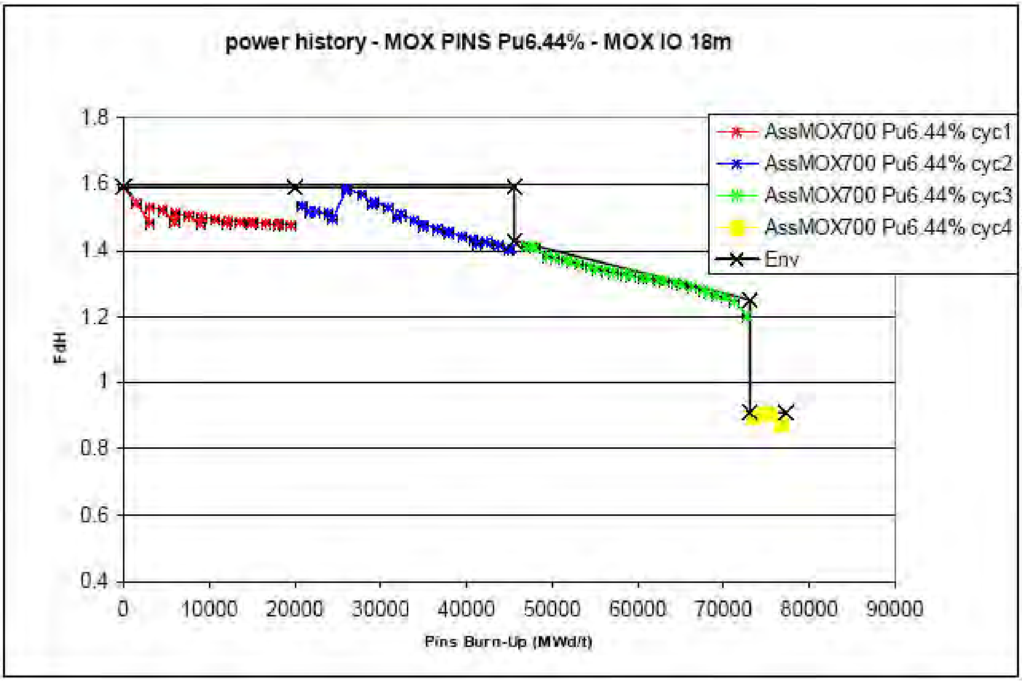
Burn-up	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Rupture
BOL	678	624	511	0.36%	NO
EOC1	680	626	512	0.36%	NO
EOC2	703	635	515	0.38%	NO
C3	747	671	529	2.59%	NO
EOC4	772	689	521	13.84%	NO

SECTION 16.4.1 - TABLE F 17: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – RESULTS OBTAINED FOR ENVELOPE UO_2-TYPE INDICATOR RODS

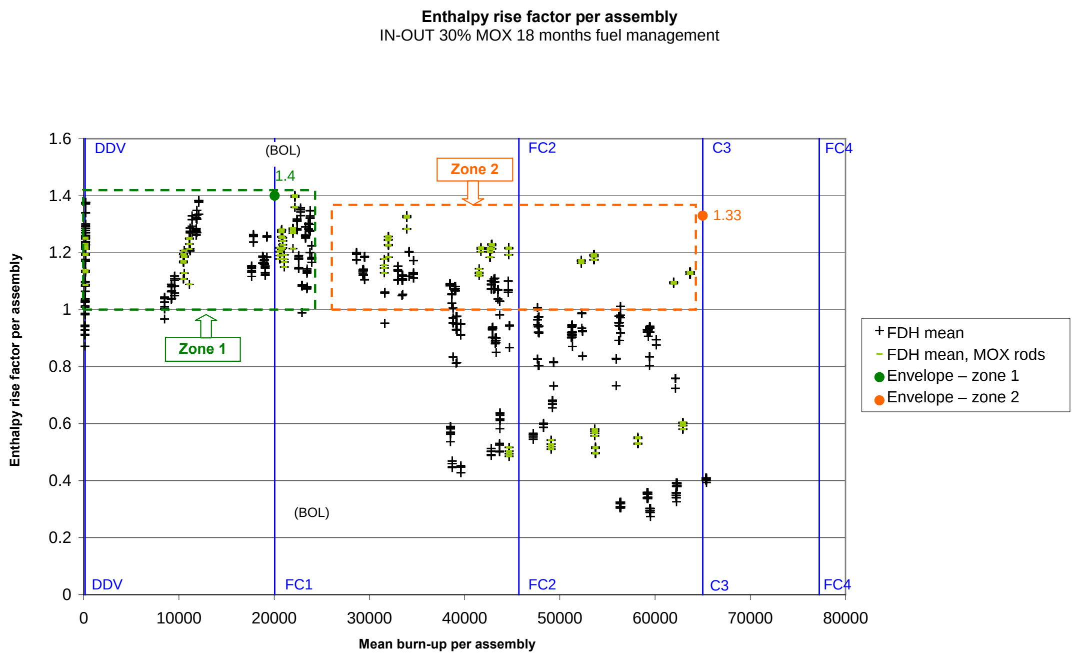
Hot residence assembly	FQ	F∆H	BU	T1 (°C)	T2 (°C)	T3 (°C)	Maximum deformation (%)	Average initial pellet temperature at peak (°C)	Rupture
HA1	2.257	1.635	EOC4	736	657	532	0.43%	750	NO
HA2	2.257	1.635	EOC4	735	663	529	0.44%	750	NO



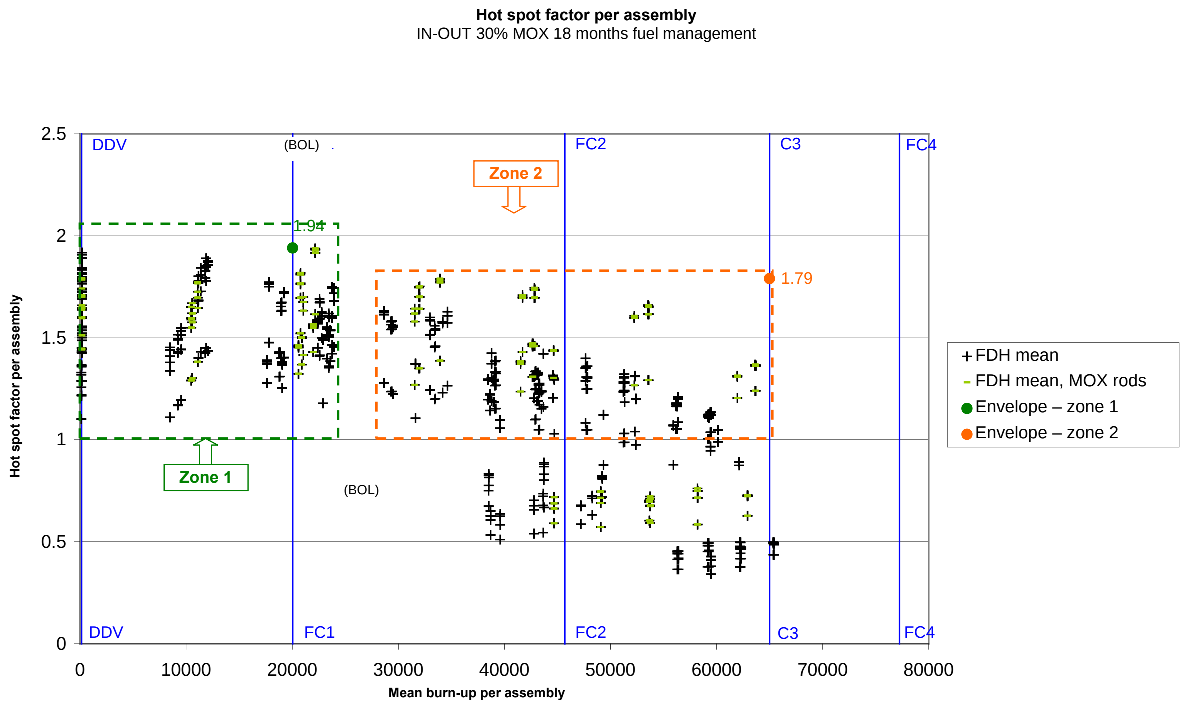
SECTION 16.4.1 - FIGURE F 1: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – POWER HISTORY OF MOX RODS WITH A HIGH PU CONTENT



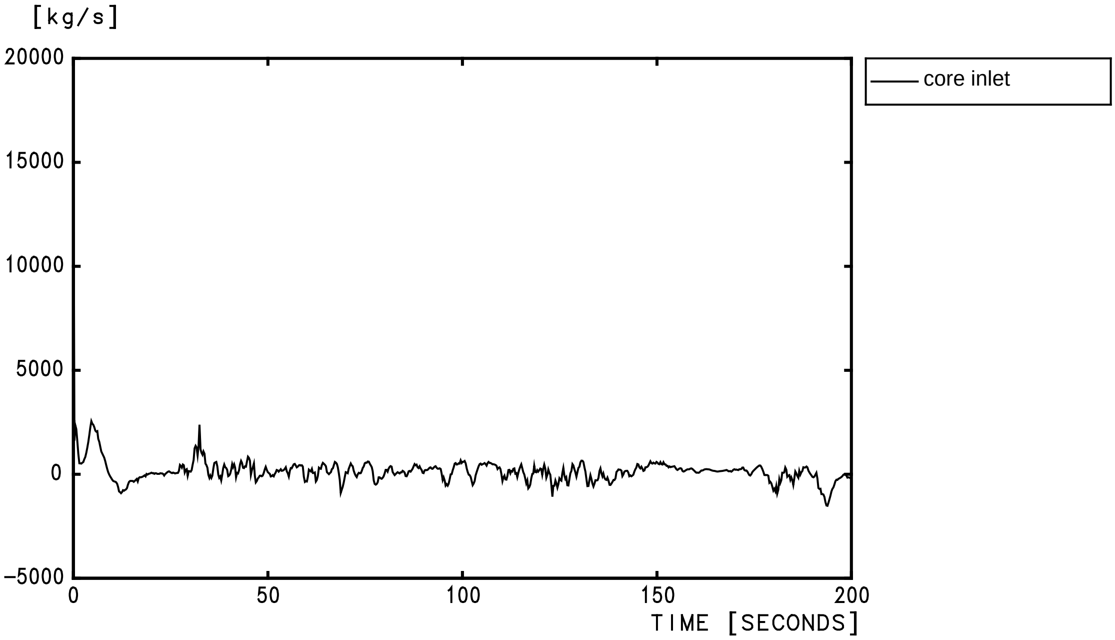
SECTION 16.4.1 - FIGURE F 2: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – POWER HISTORY OF MOX RODS WITH AN AVERAGE PU CONTENT



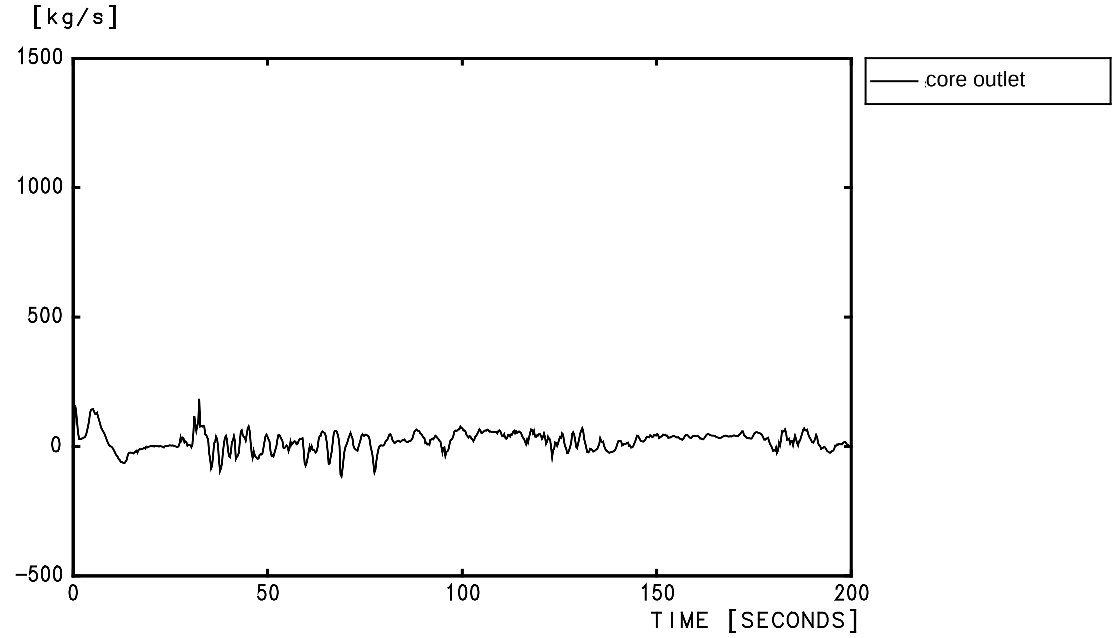
SECTION 16.4.1 - FIGURE F 3: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – F∆H FOR ASSEMBLIES AS A FUNCTION OF BURN-UP



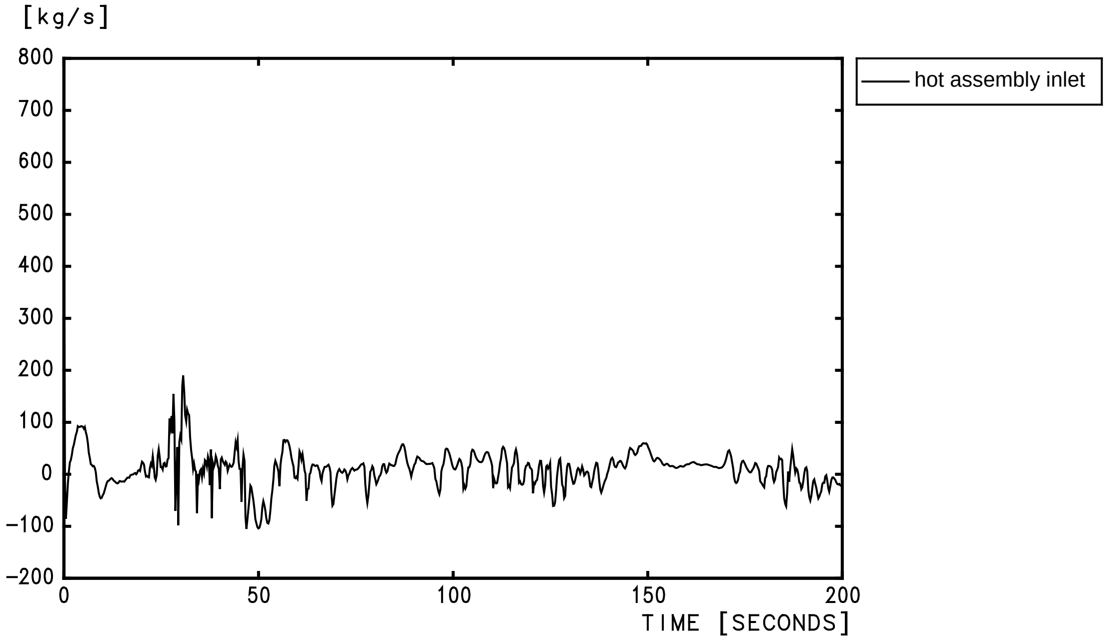
SECTION 16.4.1 - FIGURE F 4: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – FQ FOR ASSEMBLIES AS A FUNCTION OF BURN-UP



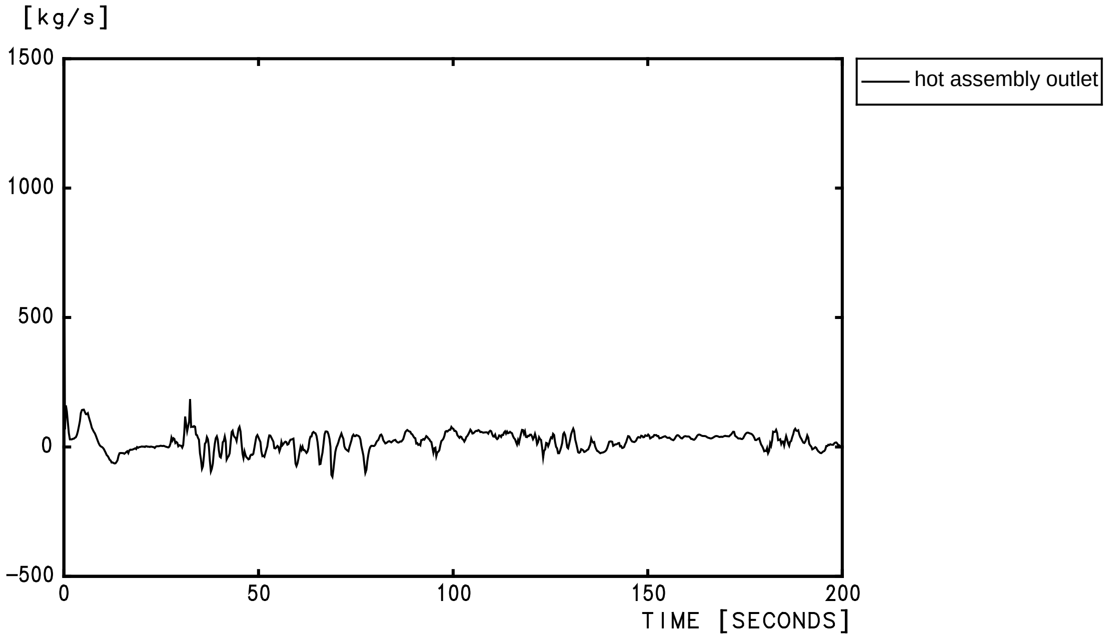
SECTION 16.4.1 - FIGURE F 5: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE CORE INLET



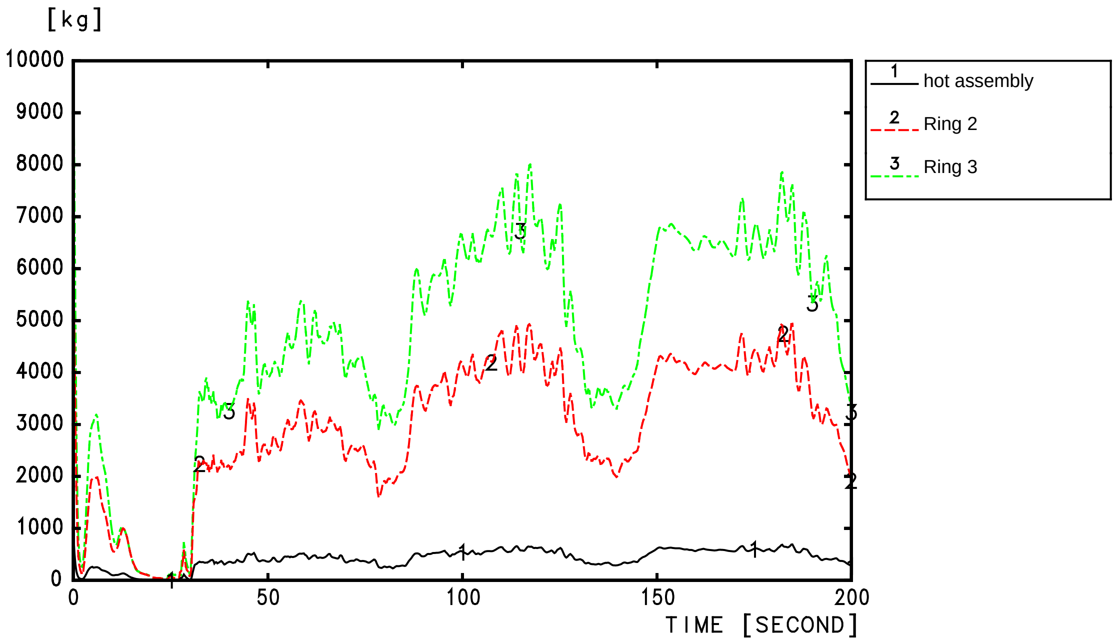
SECTION 16.4.1 - FIGURE F 6: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE CORE OUTLET



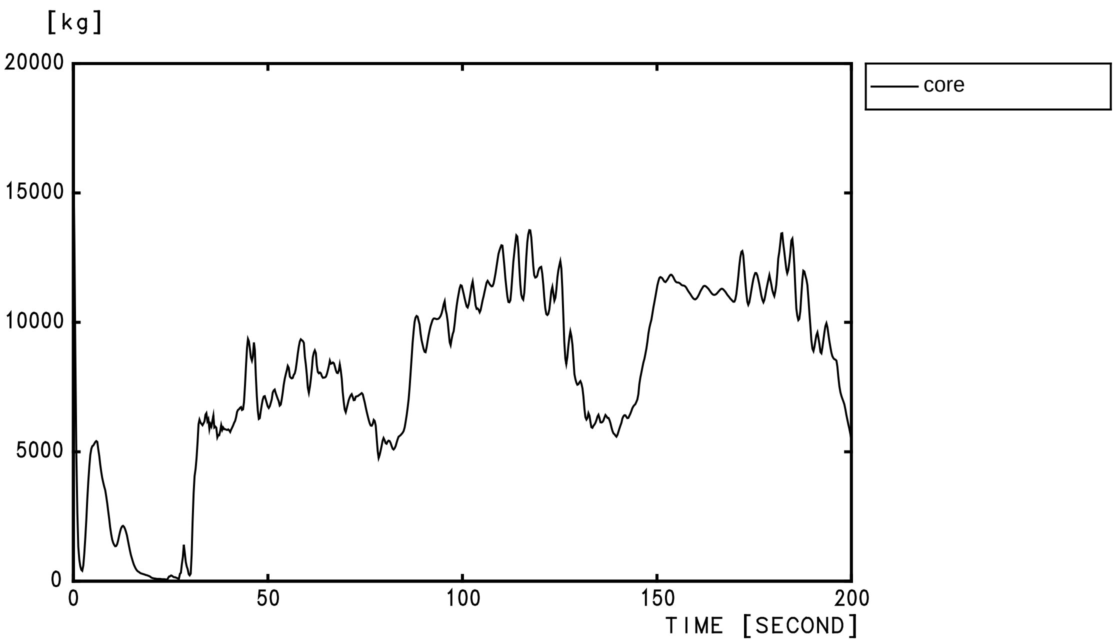
SECTION 16.4.1 - FIGURE F 7: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE HOT ASSEMBLY INLET



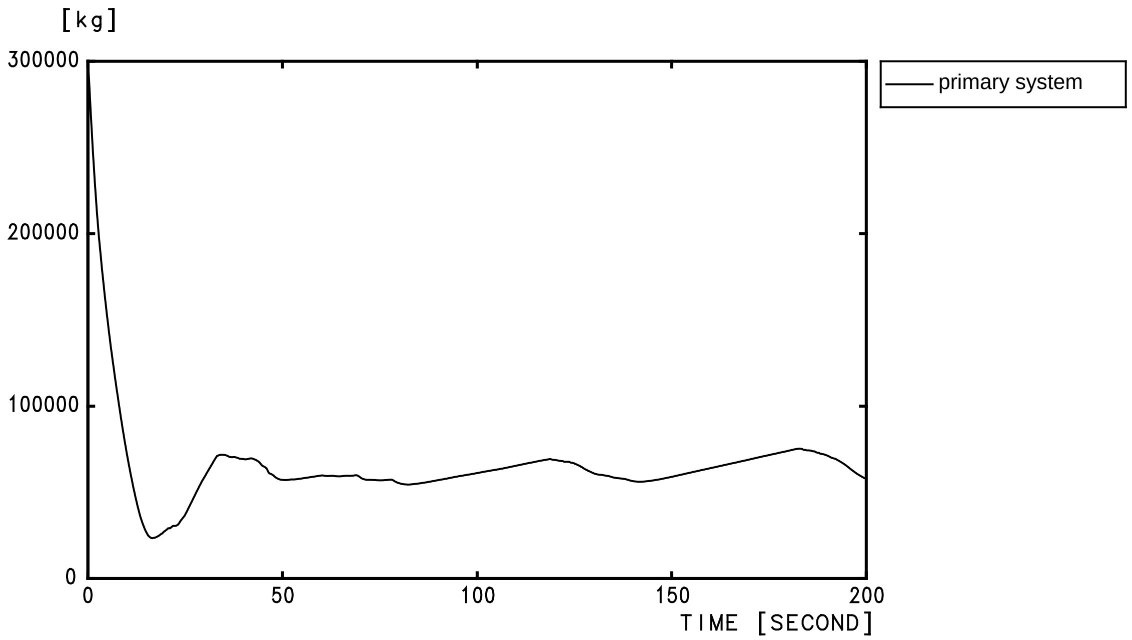
SECTION 16.4.1 - FIGURE F 8: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE HOT ASSEMBLY OUTLET



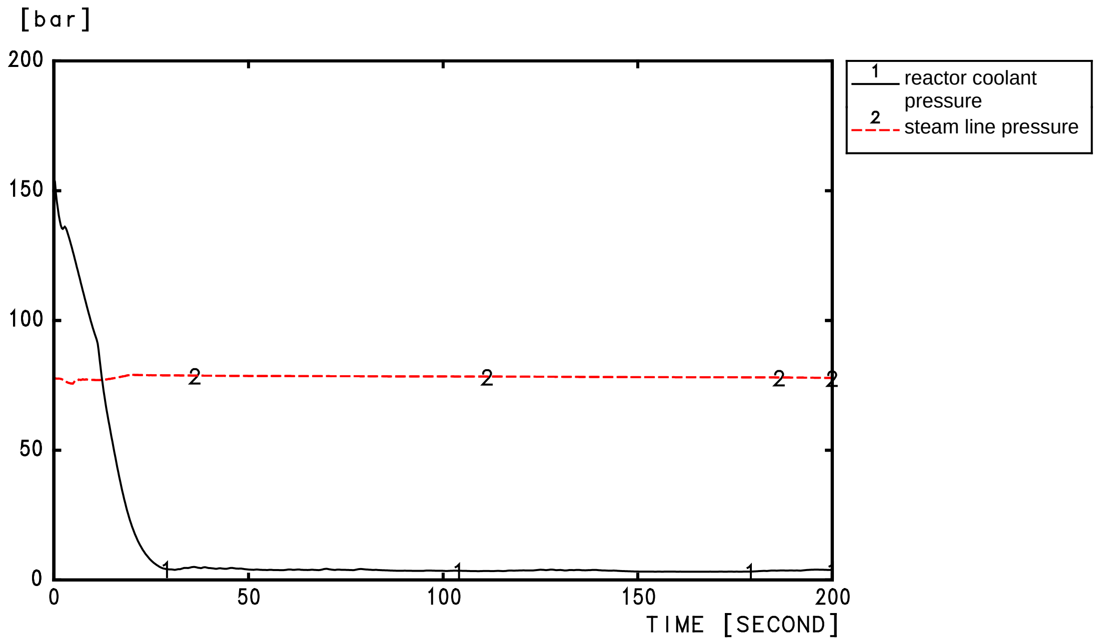
SECTION 16.4.1 - FIGURE F 9: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – LIQUID MASS IN THE HOT ASSEMBLY



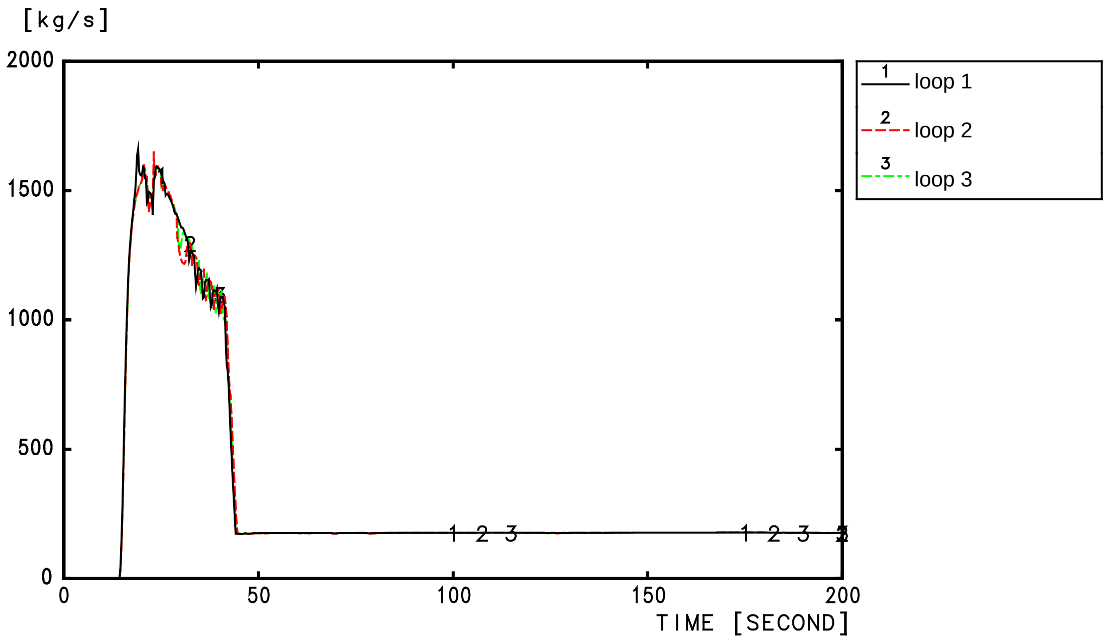
SECTION 16.4.1 - FIGURE F 10: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – LIQUID MASS IN THE CORE



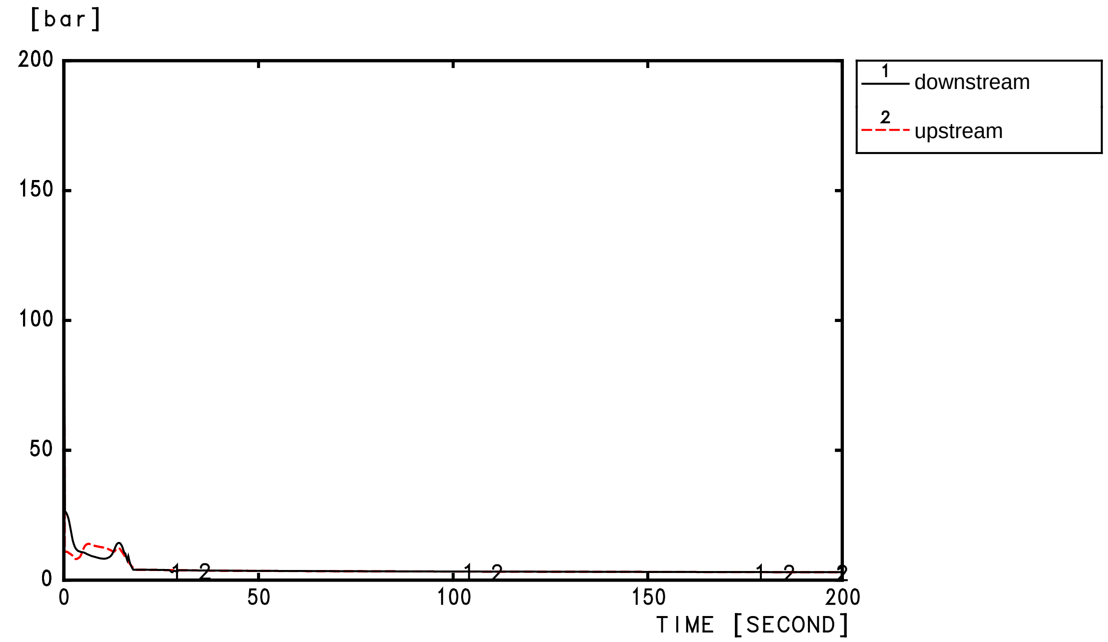
SECTION 16.4.1 - FIGURE F 11: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL MASS IN THE PRIMARY SYSTEM



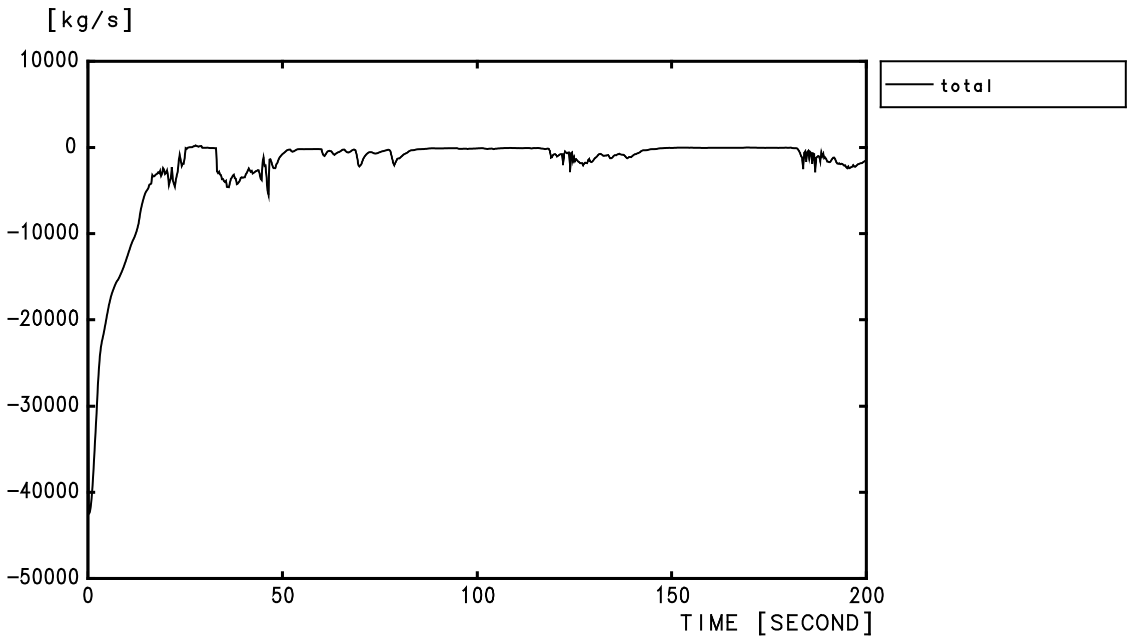
SECTION 16.4.1 - FIGURE F 12: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – REACTOR COOLANT PRESSURE AND STEAM LINE PRESSURE



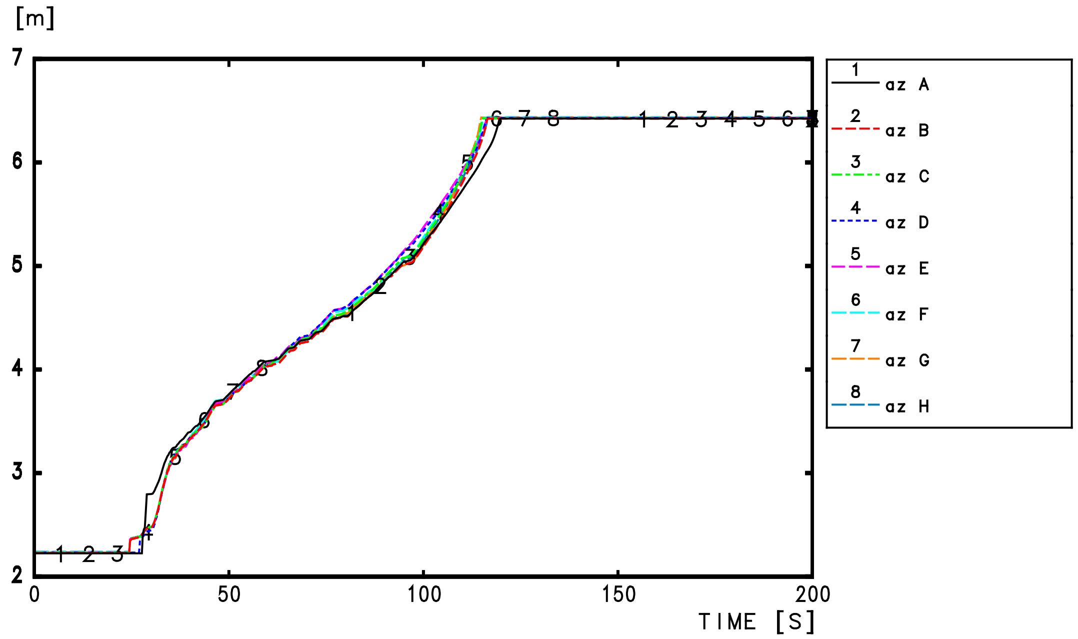
SECTION 16.4.1 - FIGURE F 13: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL RIS [SIS] FLOW RATE (ACCUMULATORS AND SI)



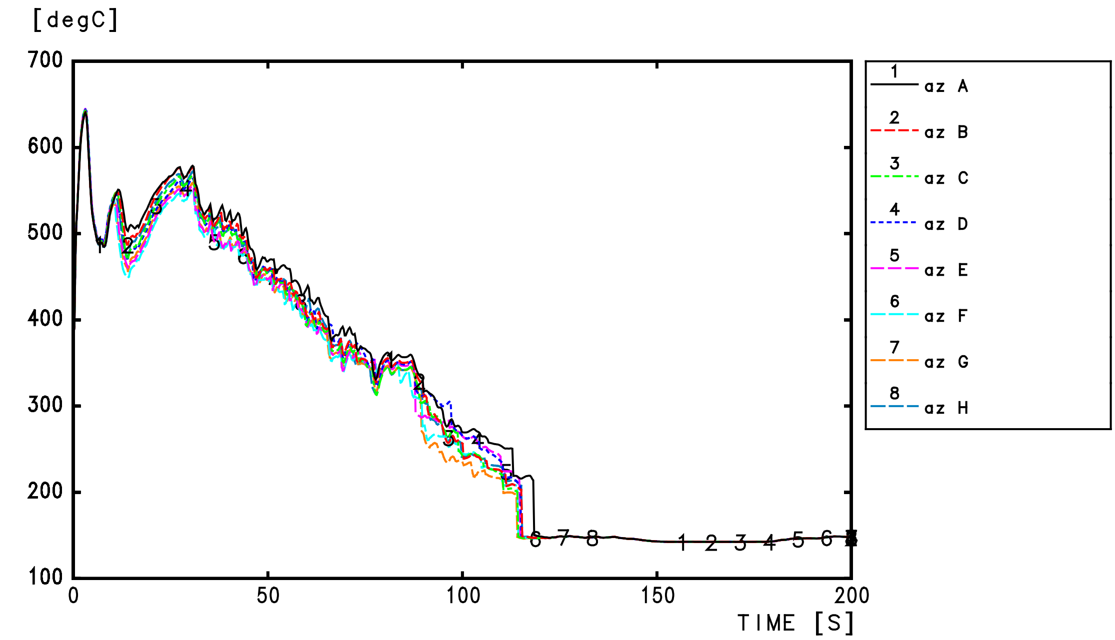
SECTION 16.4.1 - FIGURE F 14: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – PRESSURE AT THE BREAK



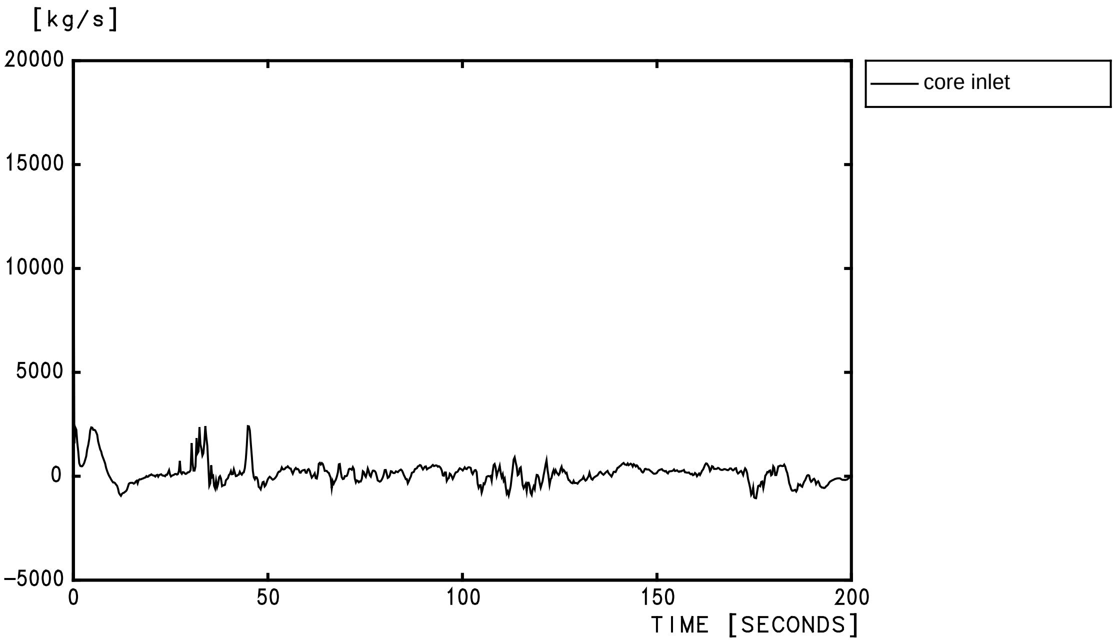
SECTION 16.4.1 - FIGURE F 15: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – TOTAL FLOW RATE AT THE BREAK



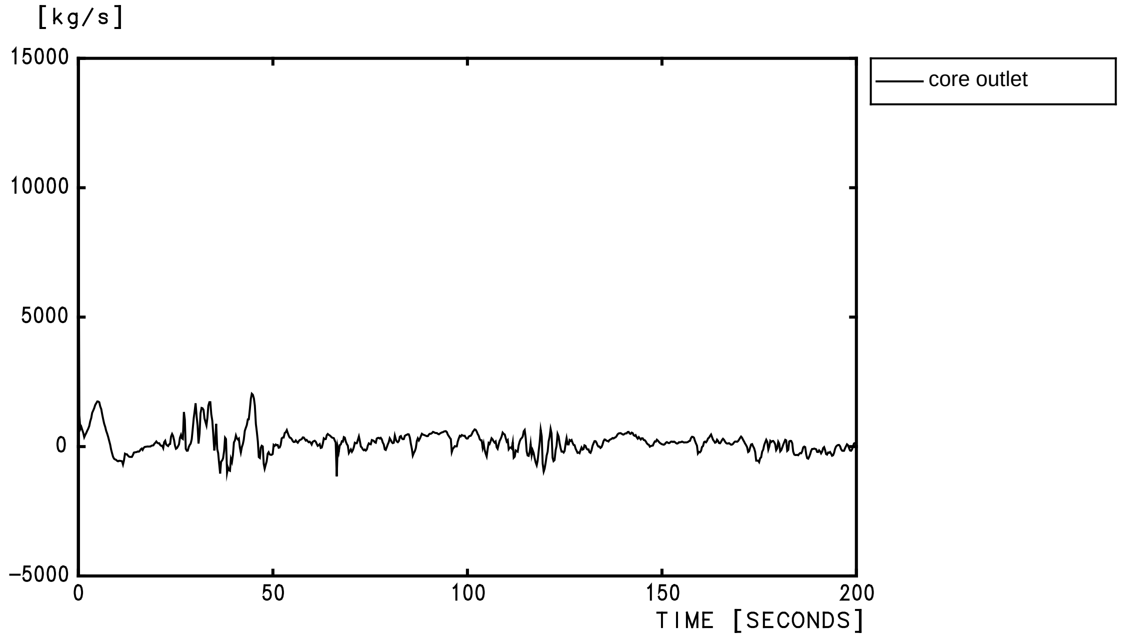
SECTION 16.4.1 - FIGURE F 16: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – QUENCH FRONT LEVEL IN HOT ASSEMBLIES



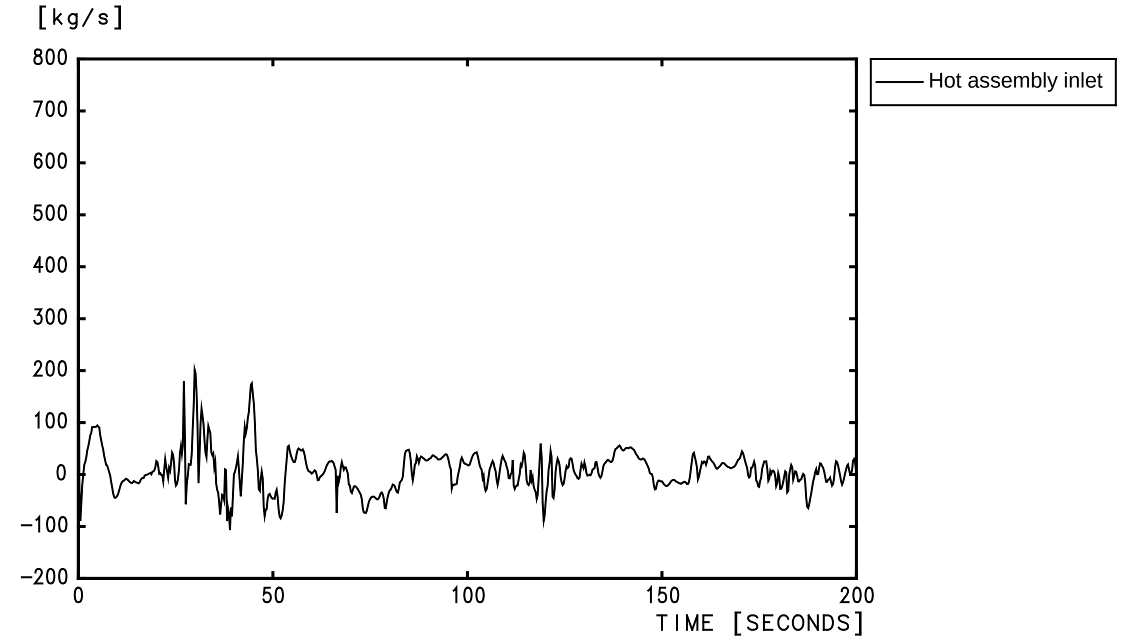
SECTION 16.4.1 - FIGURE F 17: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – MAXIMUM CLAD TEMPERATURES FOR HOT ASSEMBLIES



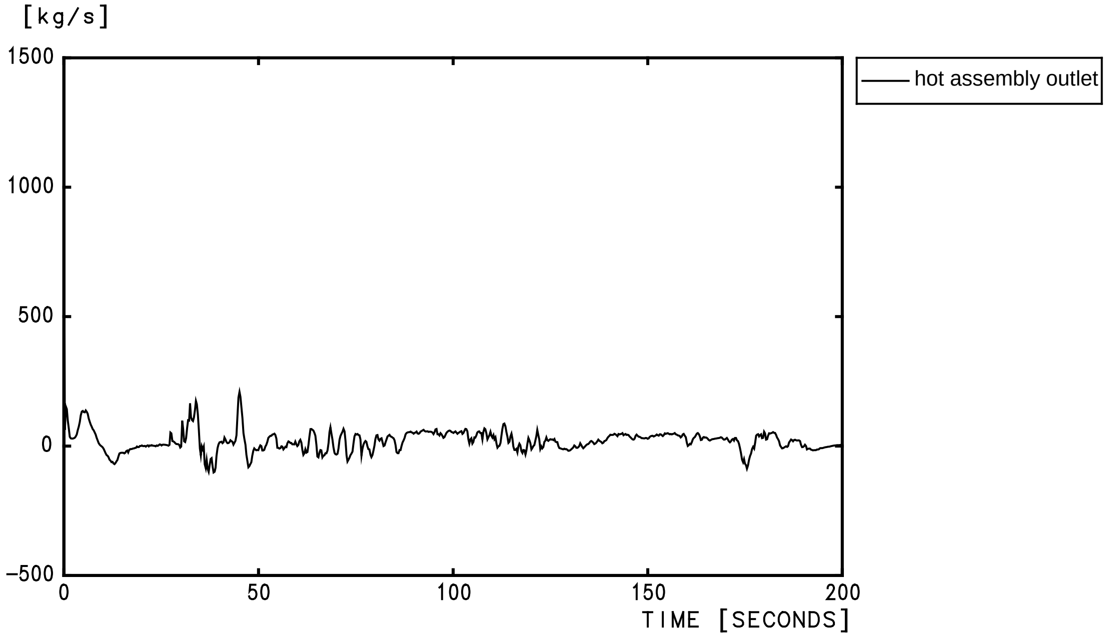
SECTION 16.4.1 - FIGURE F 18: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE CORE INLET



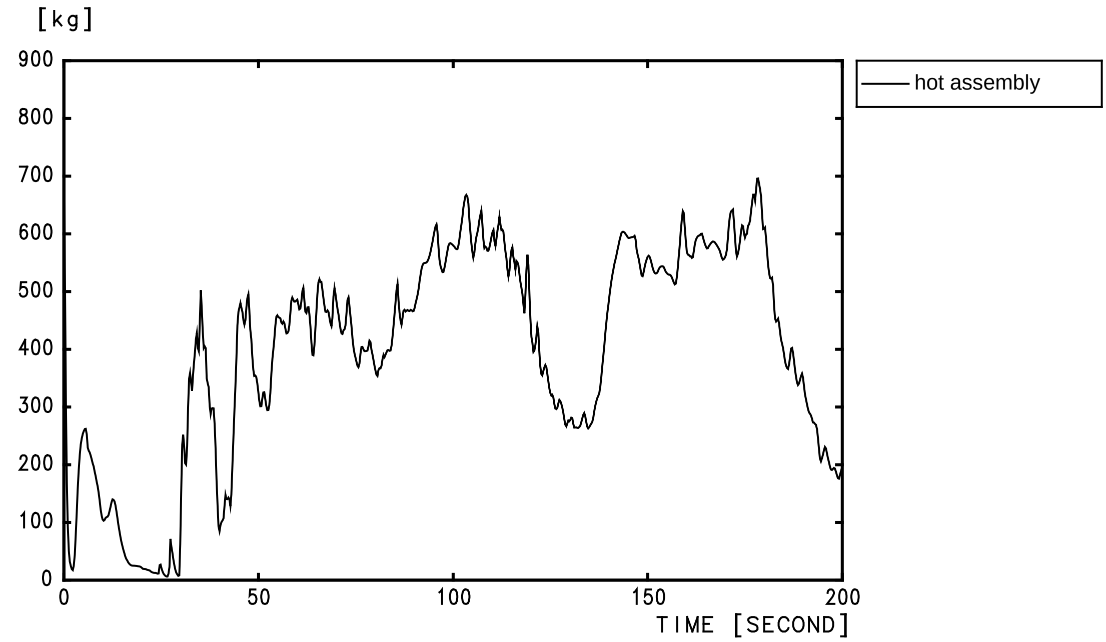
SECTION 16.4.1 - FIGURE F 19: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE CORE OUTLET



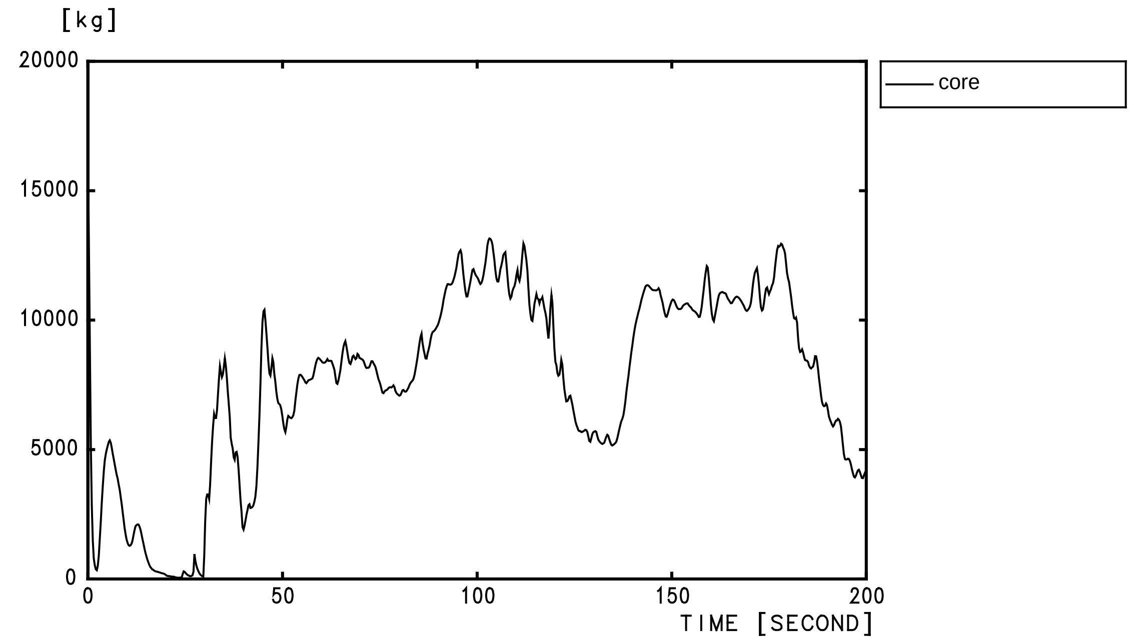
SECTION 16.4.1 - FIGURE F 20: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE HOT ASSEMBLY INLET



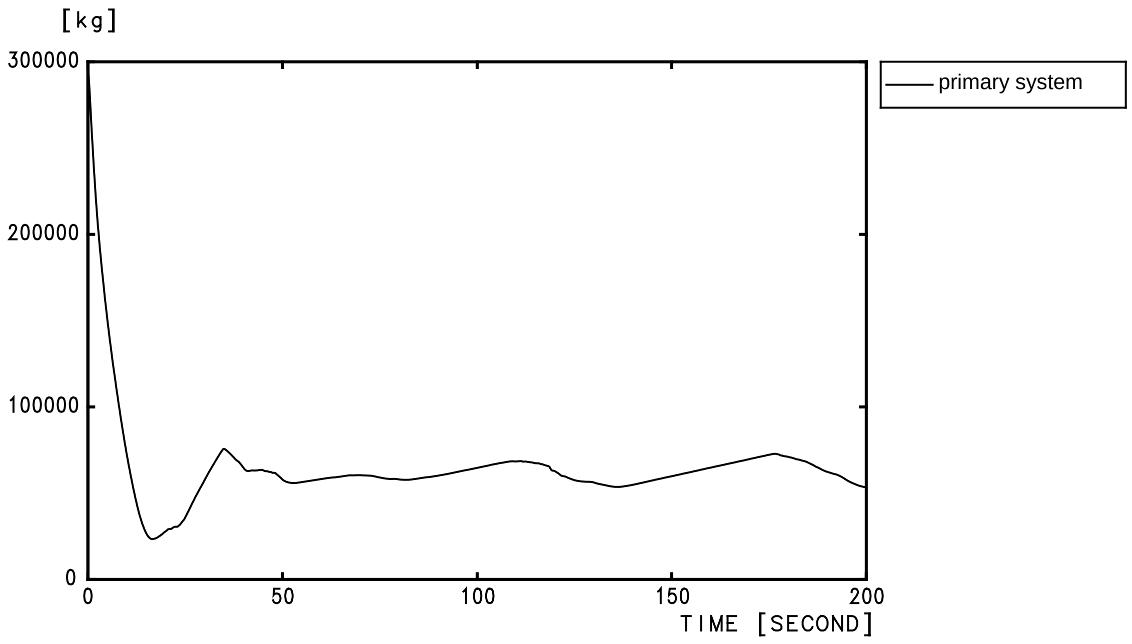
SECTION 16.4.1 - FIGURE F 21: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE HOT ASSEMBLY OUTLET



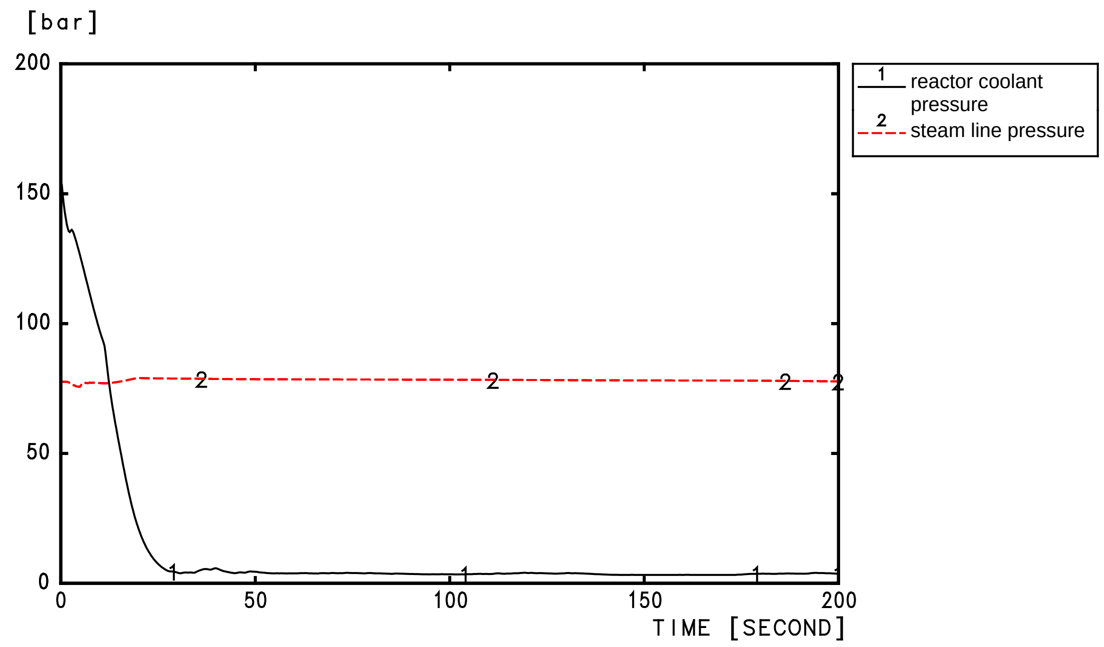
SECTION 16.4.1 - FIGURE F 22: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – LIQUID MASS IN THE HOT ASSEMBLY



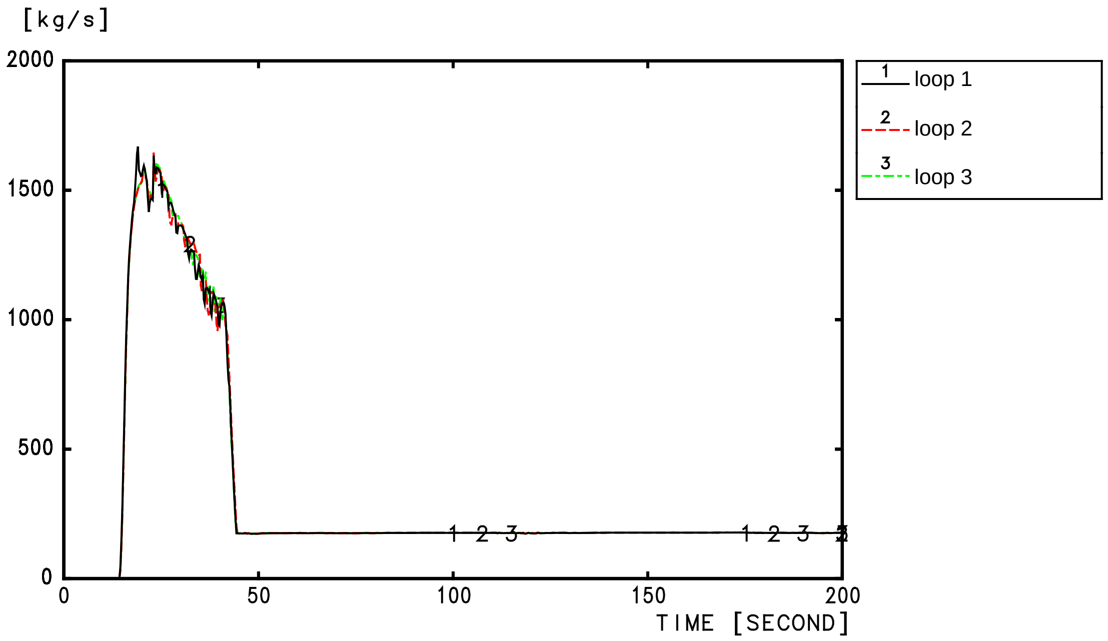
SECTION 16.4.1 - FIGURE F 23: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – LIQUID MASS IN THE CORE



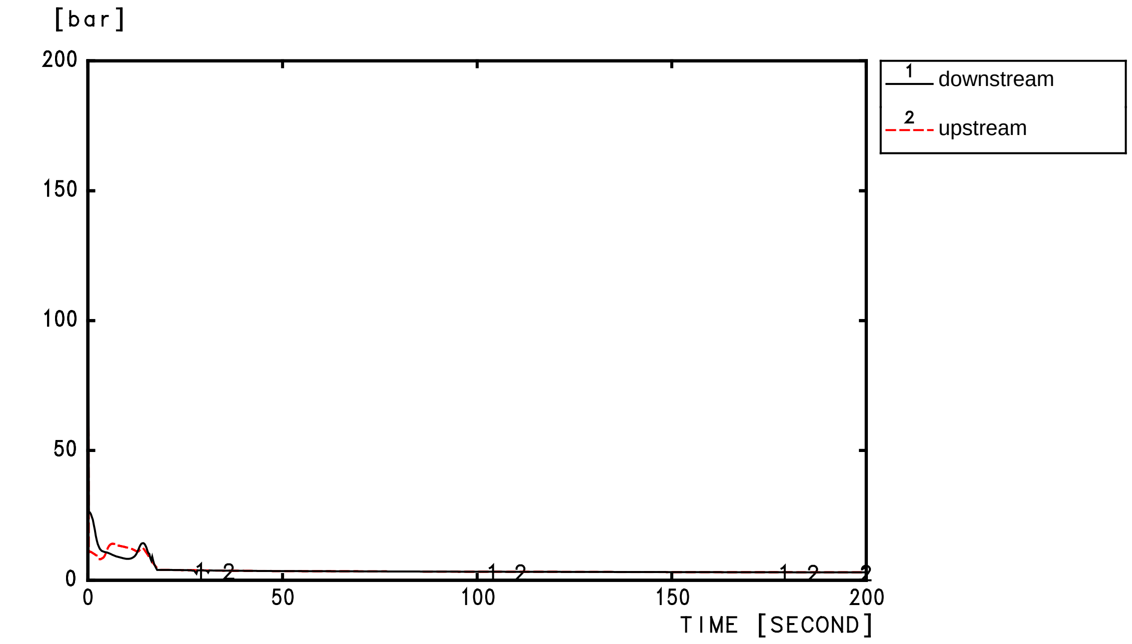
SECTION 16.4.1 - FIGURE F 24: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL MASS IN THE PRIMARY SYSTEM



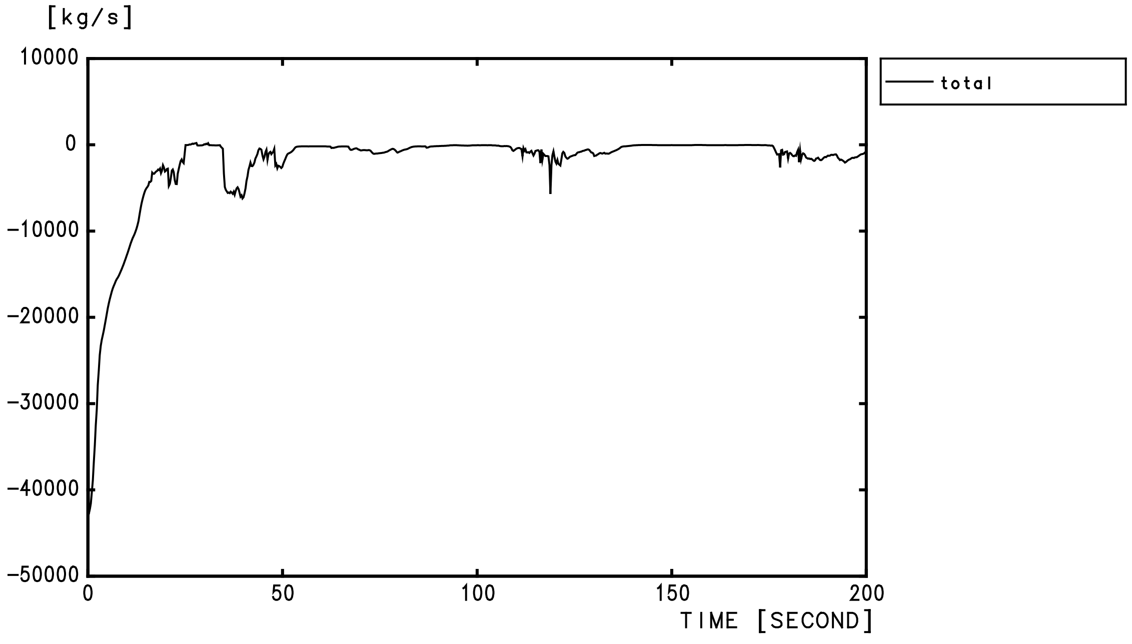
SECTION 16.4.1 - FIGURE F 25: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – REACTOR COOLANT PRESSURE AND STEAM LINE PRESSURE



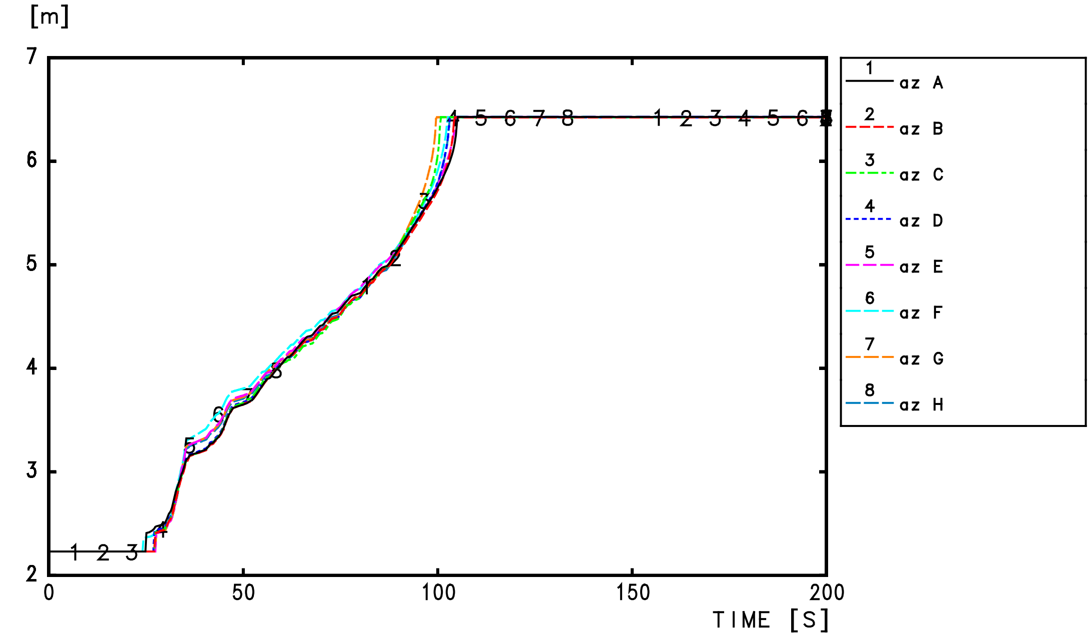
SECTION 16.4.1 - FIGURE F 26: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL RIS [SIS] FLOW RATE (ACCUMULATORS AND SI)



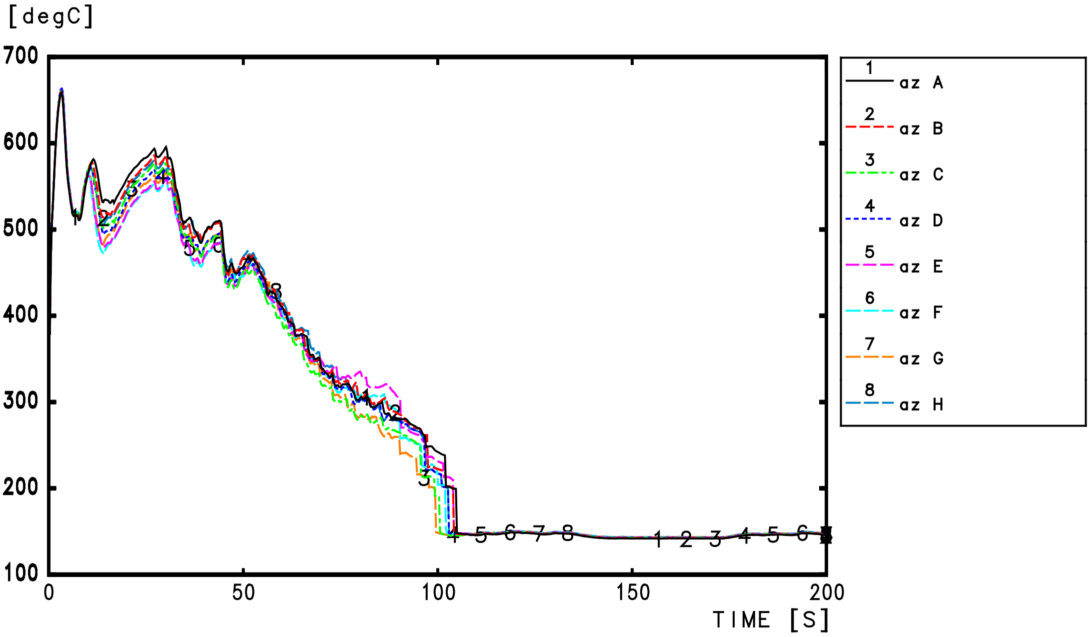
SECTION 16.4.1 - FIGURE F 27: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – PRESSURE AT THE BREAK



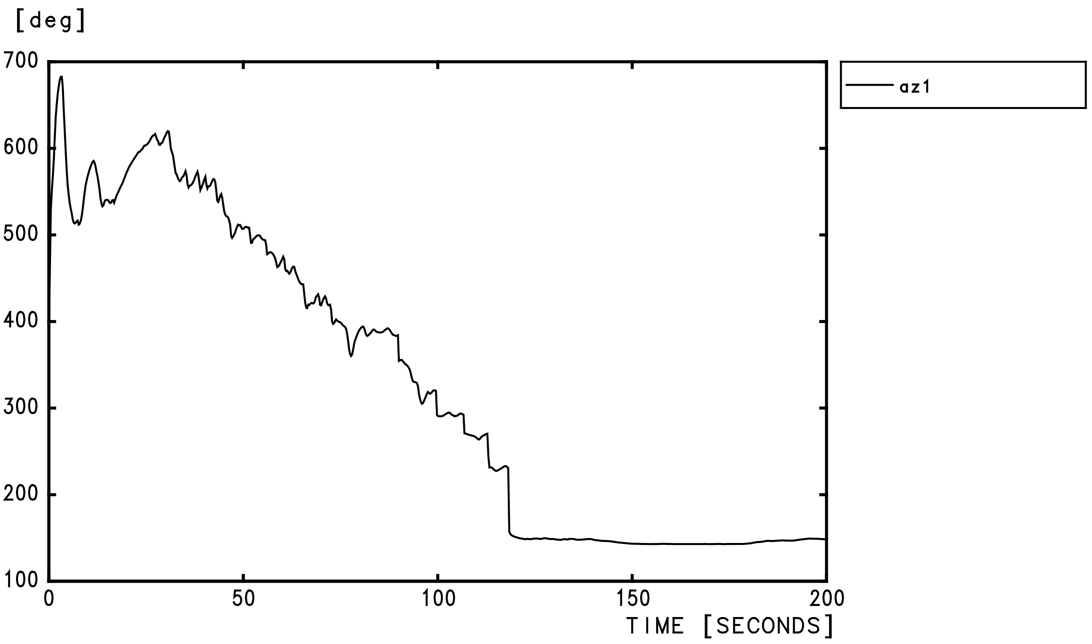
SECTION 16.4.1 - FIGURE F 28: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – TOTAL FLOW RATE AT THE BREAK



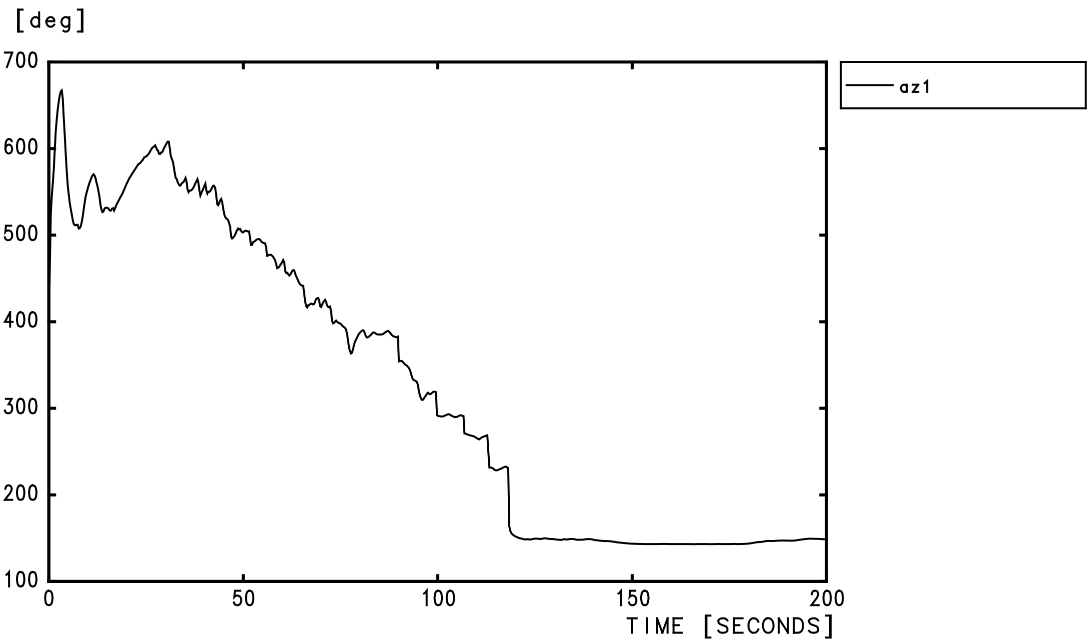
SECTION 16.4.1 - FIGURE F 29: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 –QUENCH FRONT LEVEL IN HOT ASSEMBLIES



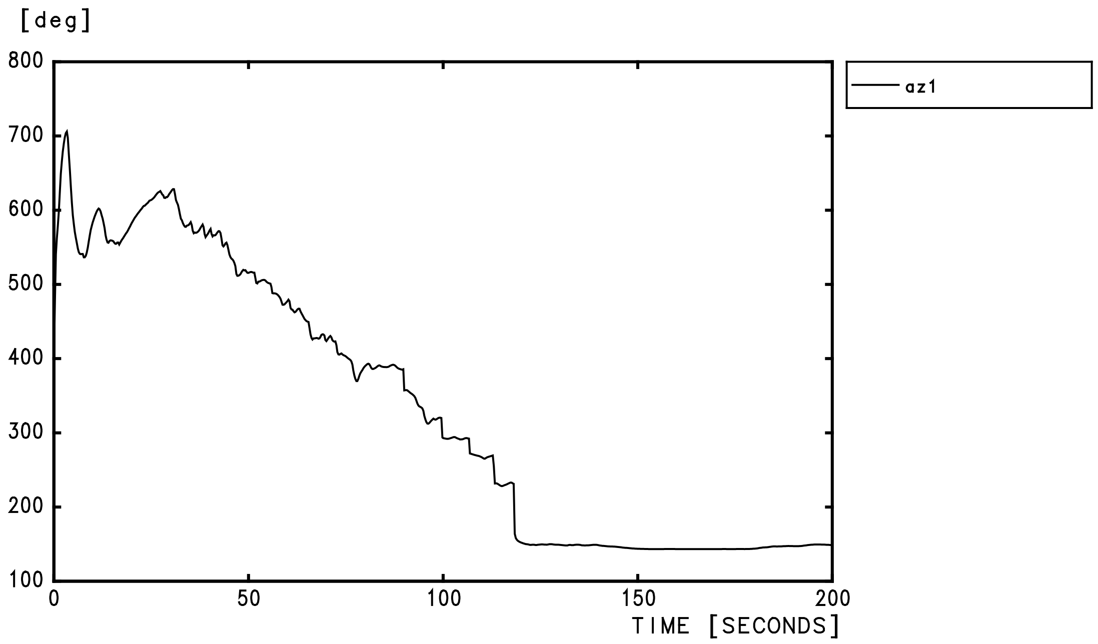
SECTION 16.4.1 - FIGURE F 30: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – MAXIMUM CLAD TEMPERATURES FOR HOT ASSEMBLIES



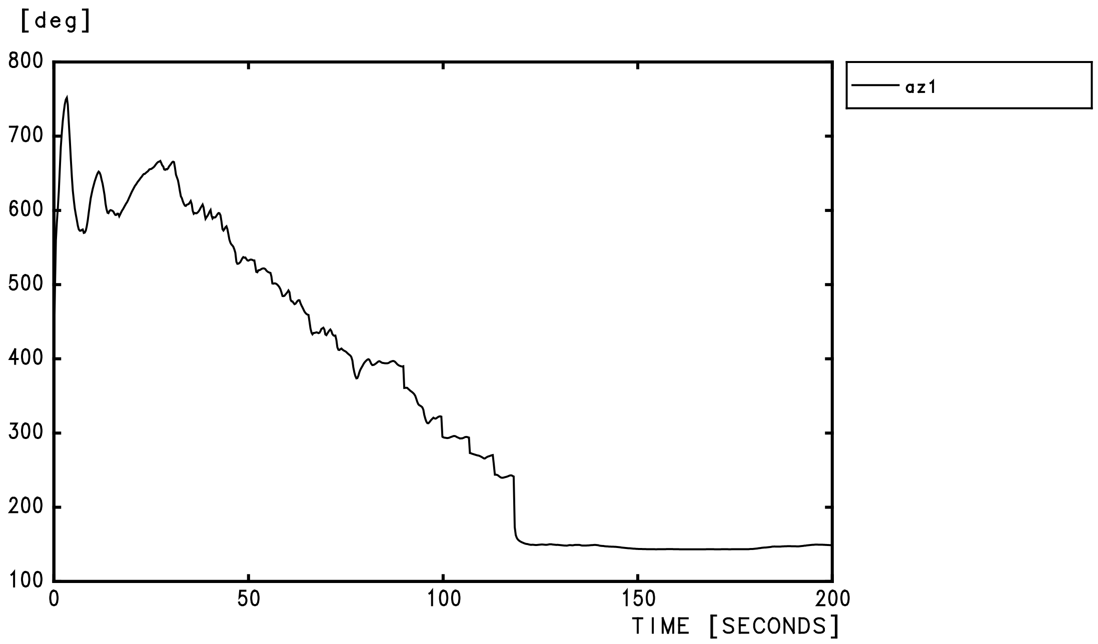
SECTION 16.4.1 - FIGURE F 31: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – BOL INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



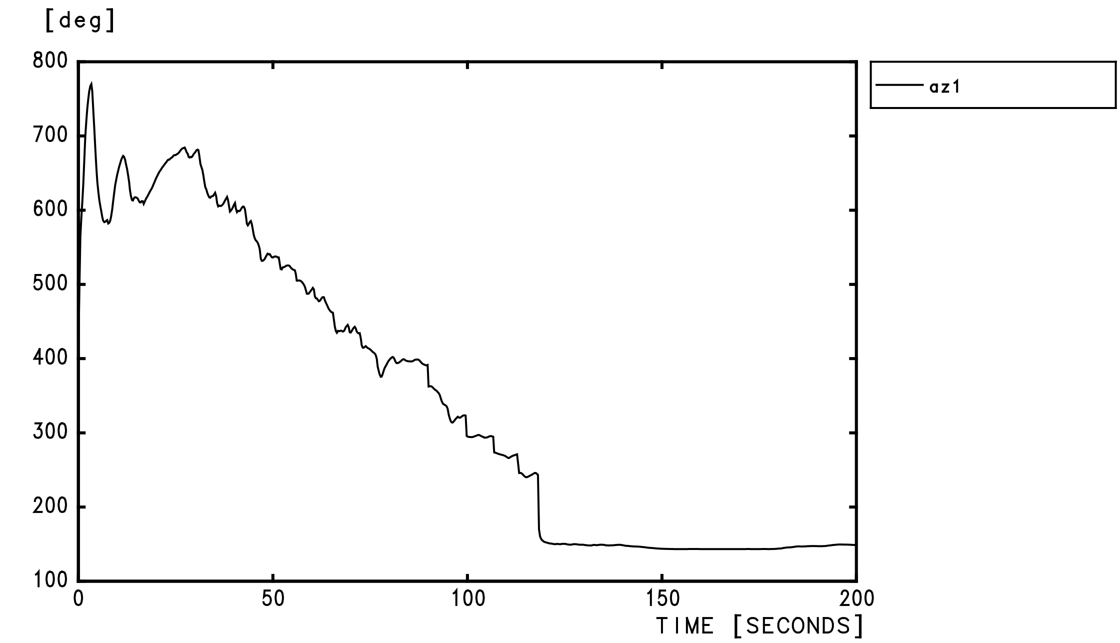
SECTION 16.4.1 - FIGURE F 32: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC1 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



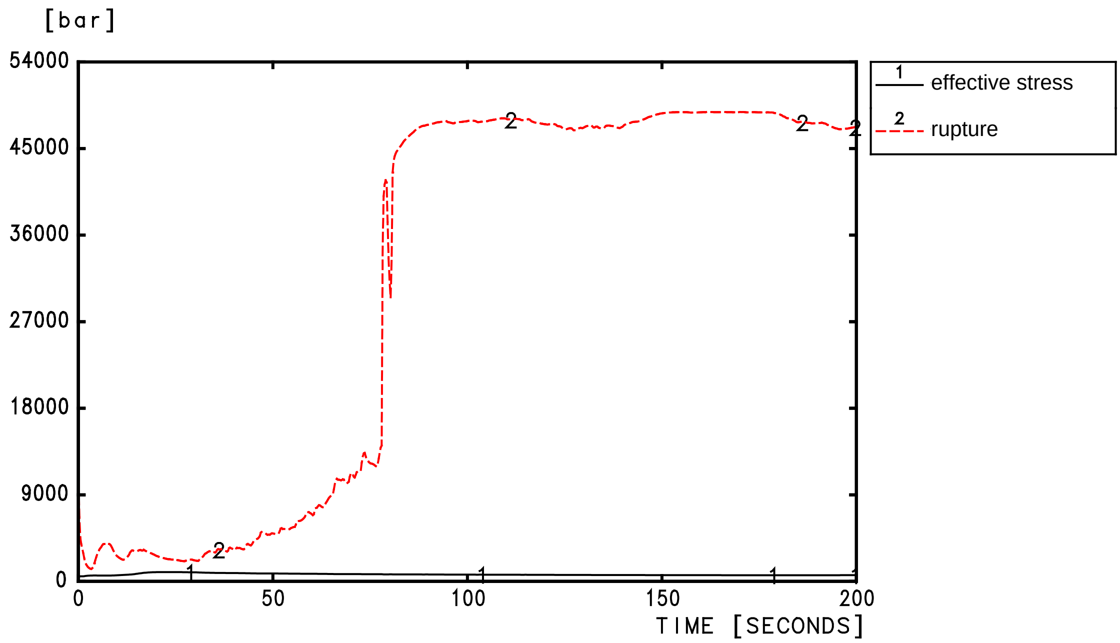
SECTION 16.4.1 - FIGURE F 33: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC2 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



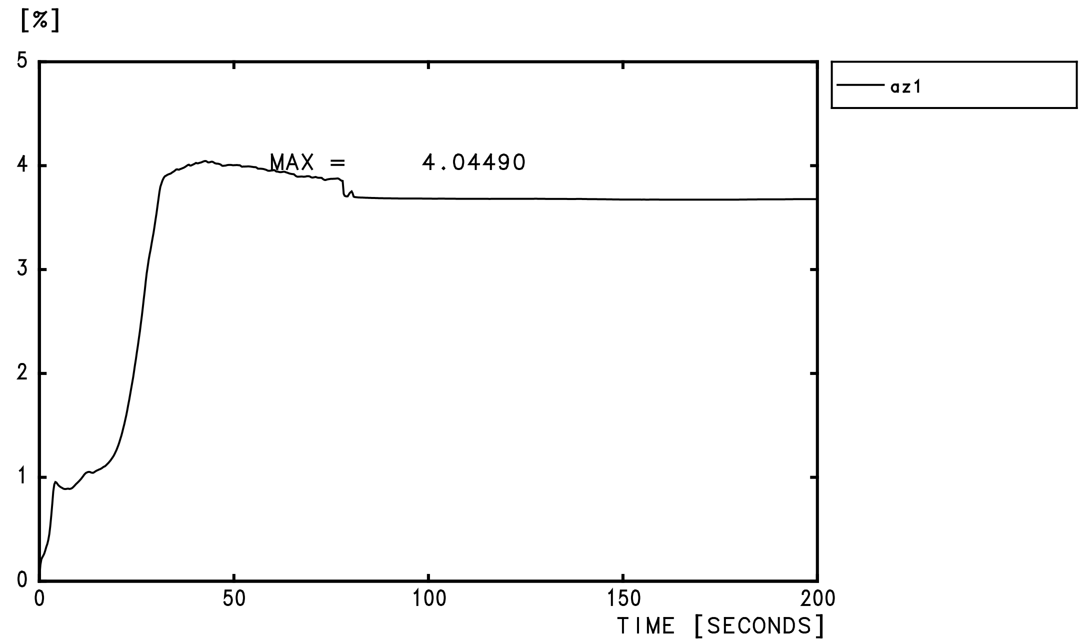
SECTION 16.4.1 - FIGURE F 34: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – C3 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



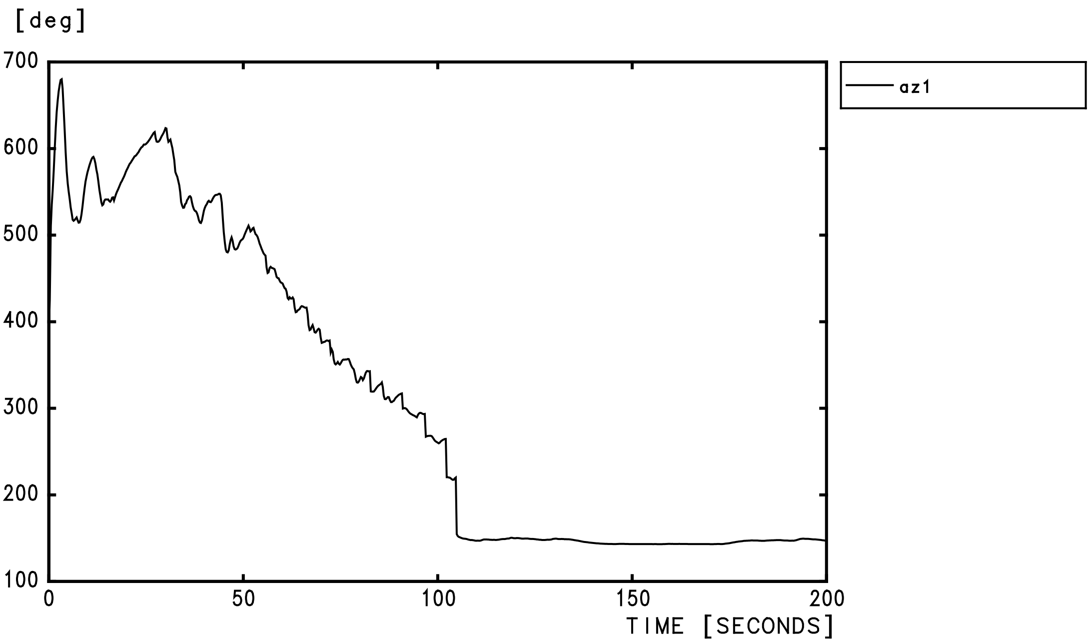
SECTION 16.4.1 - FIGURE F 35: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC4 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



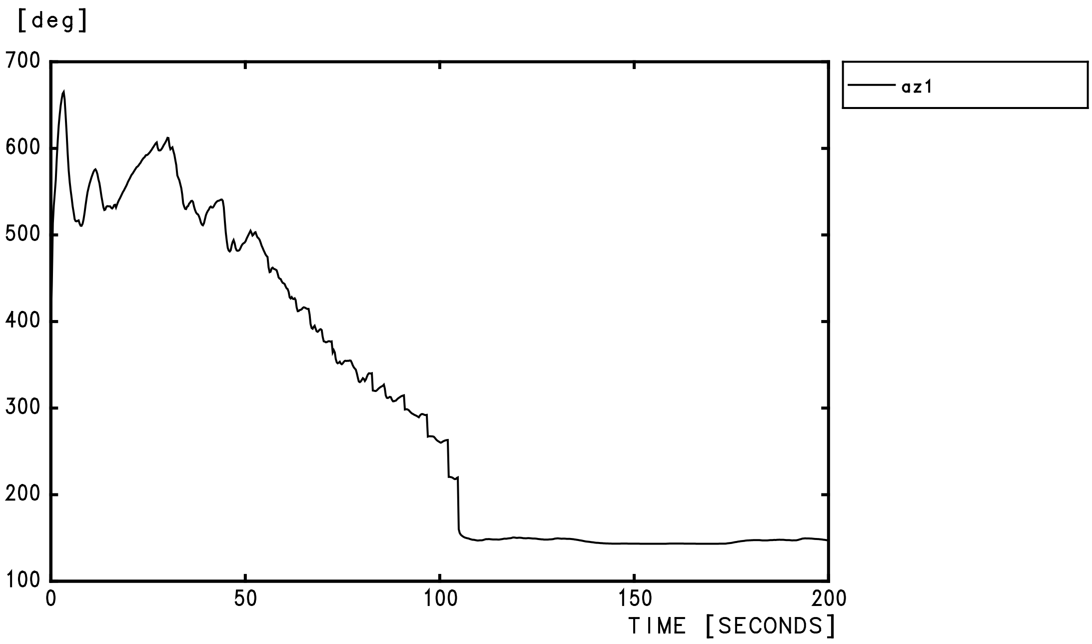
SECTION 16.4.1 - FIGURE F 36: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC4 INDICATOR ROD WITH A HIGH PU CONTENT – TRUE STRESS AND RUPTURE STRESS AT THE PEAK POWER HEIGHT



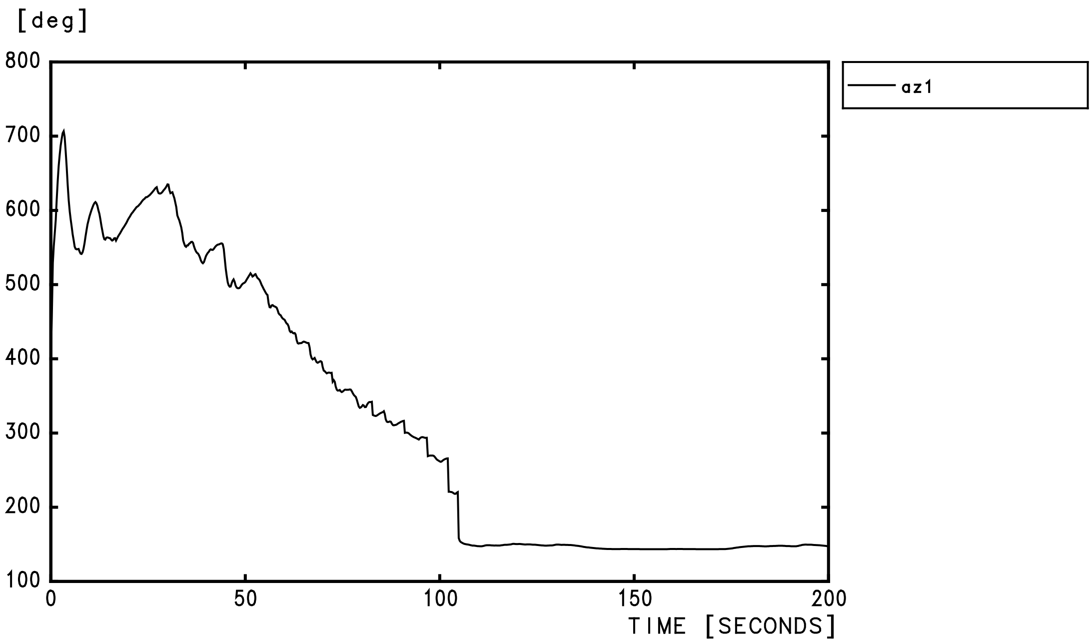
SECTION 16.4.1 - FIGURE F 37: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC4 INDICATOR ROD WITH A HIGH PU CONTENT – DEFORMATION AT THE PEAK POWER HEIGHT



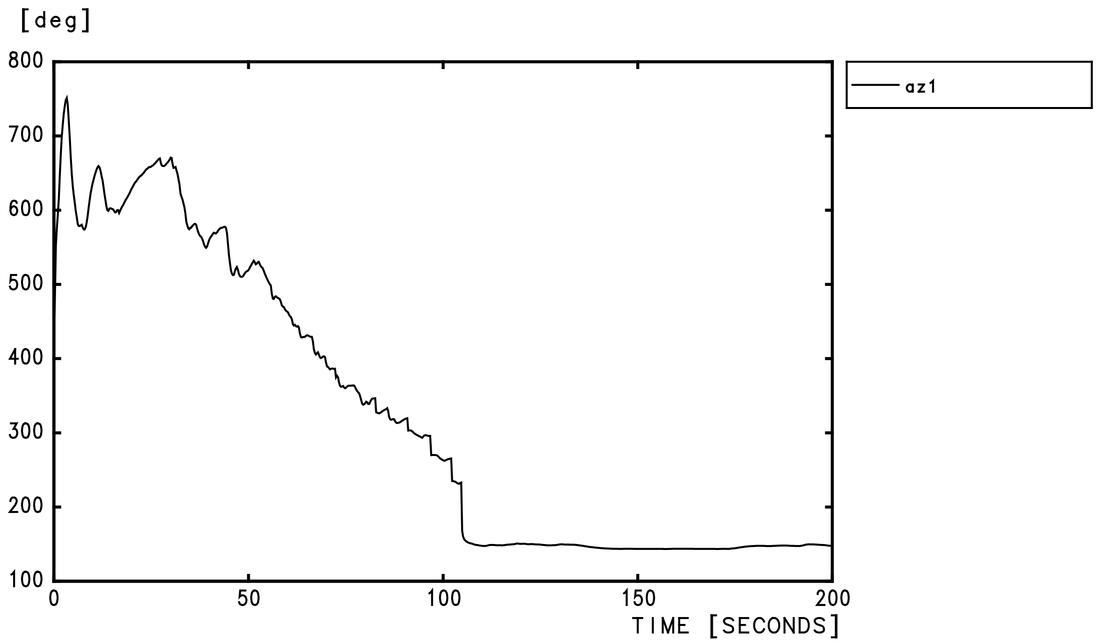
SECTION 16.4.1 - FIGURE F 38: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – BOL INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



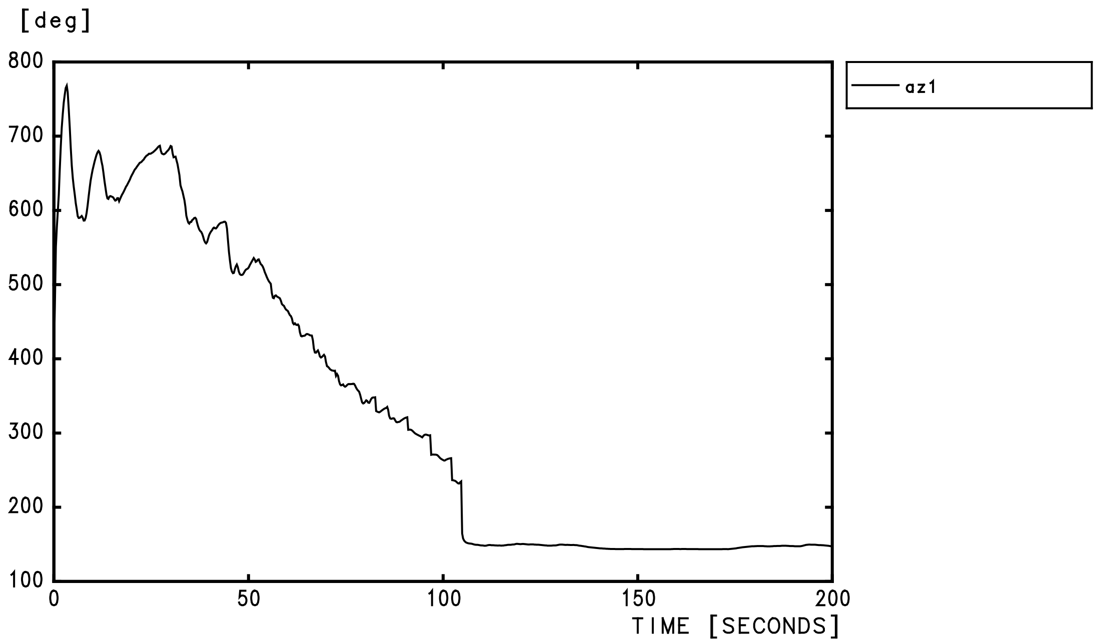
SECTION 16.4.1 - FIGURE F 39: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC1 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



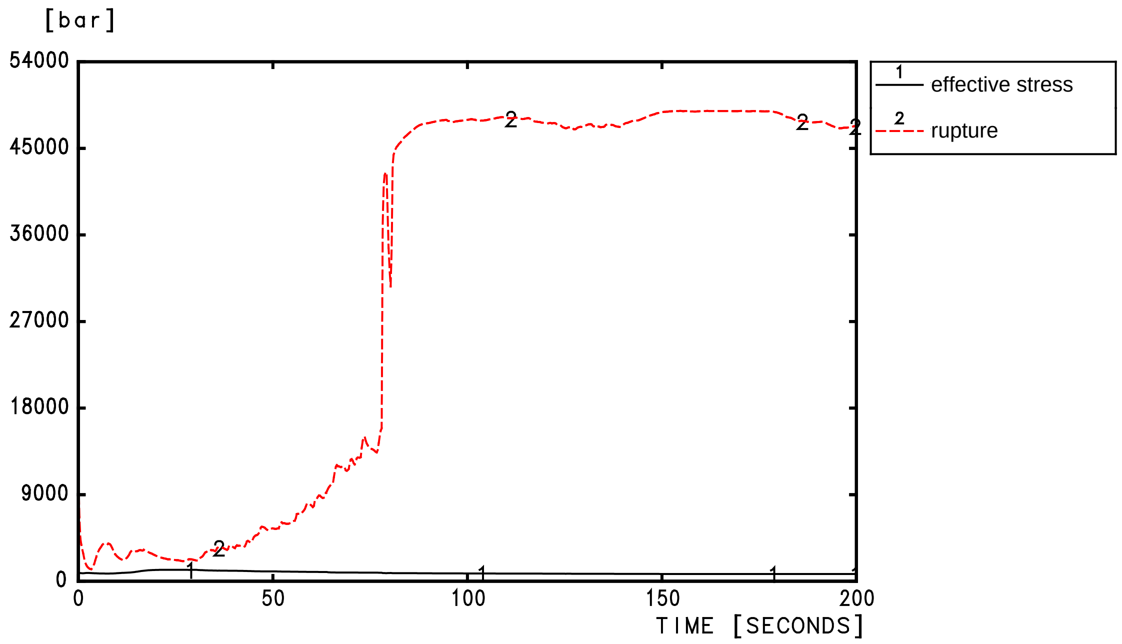
SECTION 16.4.1 - FIGURE F 40: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC2 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



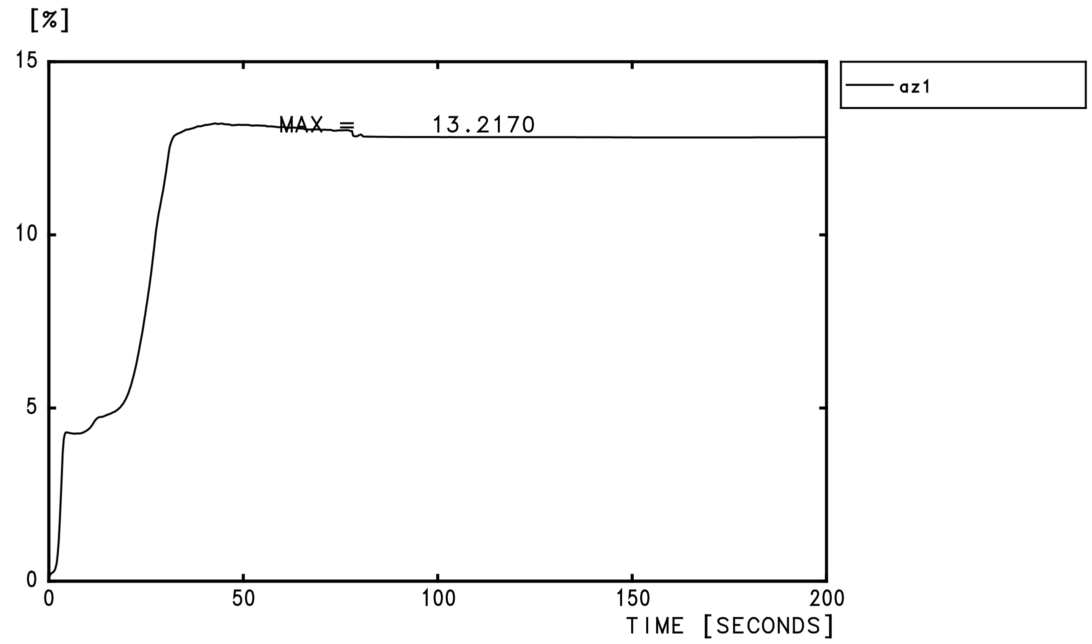
SECTION 16.4.1 - FIGURE F 41: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – C3 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



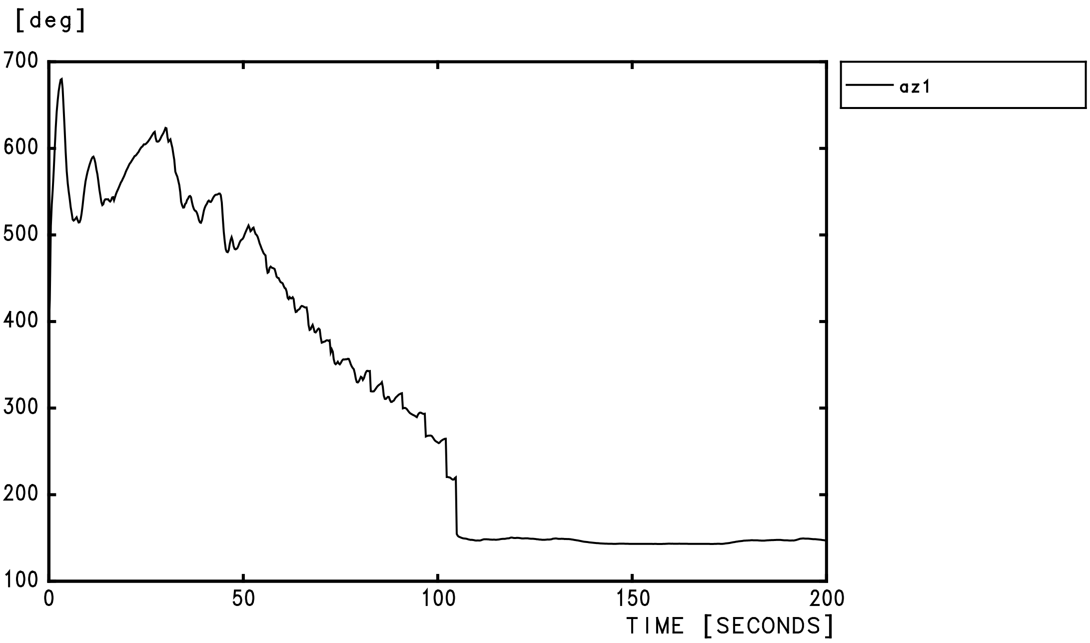
SECTION 16.4.1 - FIGURE F 42: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC4 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



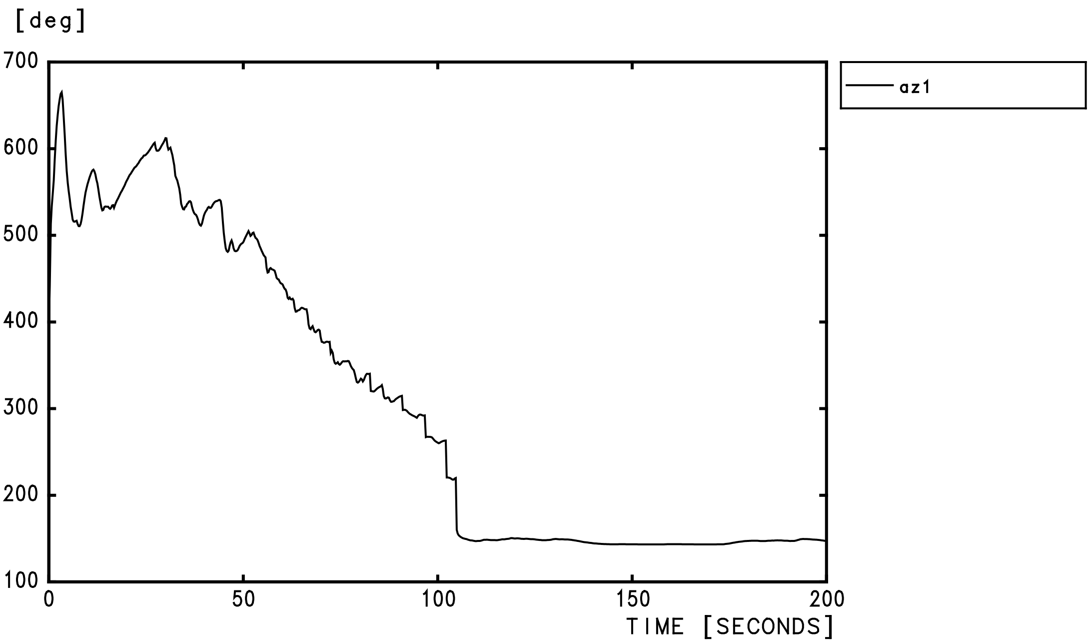
SECTION 16.4.1 - FIGURE F 43: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC4 INDICATOR ROD WITH AN AVERAGE PU CONTENT – TRUE STRESS AND RUPTURE STRESS AT THE PEAK POWER HEIGHT



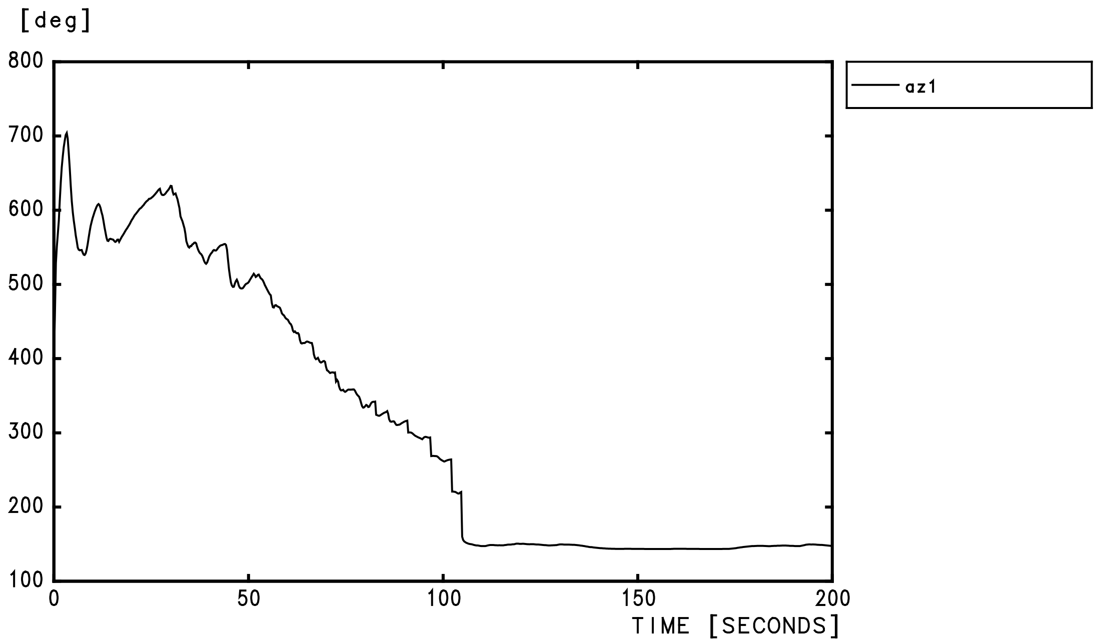
SECTION 16.4.1 - FIGURE F 44: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA1 – EOC4 INDICATOR ROD WITH AN AVERAGE PU CONTENT – DEFORMATION AT THE PEAK POWER HEIGHT



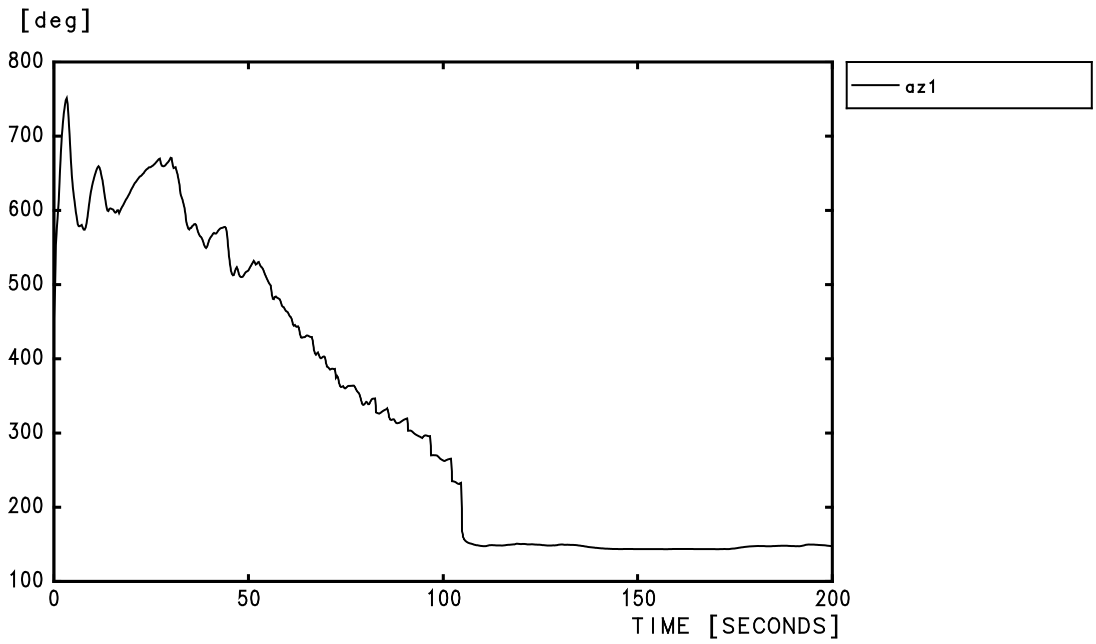
SECTION 16.4.1 - FIGURE F 45: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – BOL INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



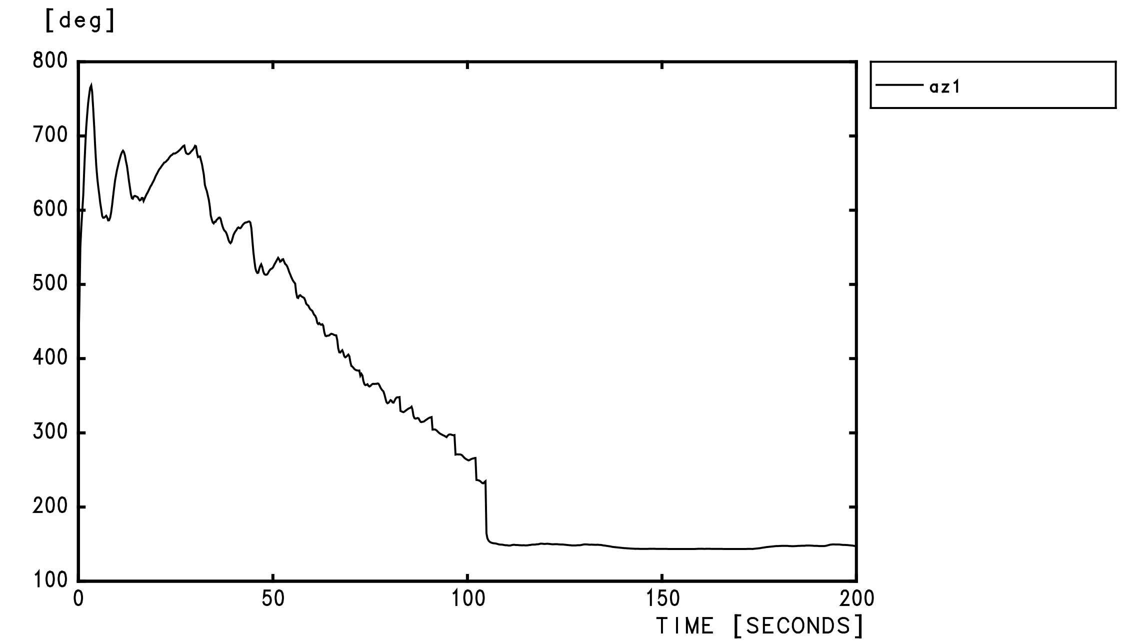
SECTION 16.4.1 - FIGURE F 46: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC1 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



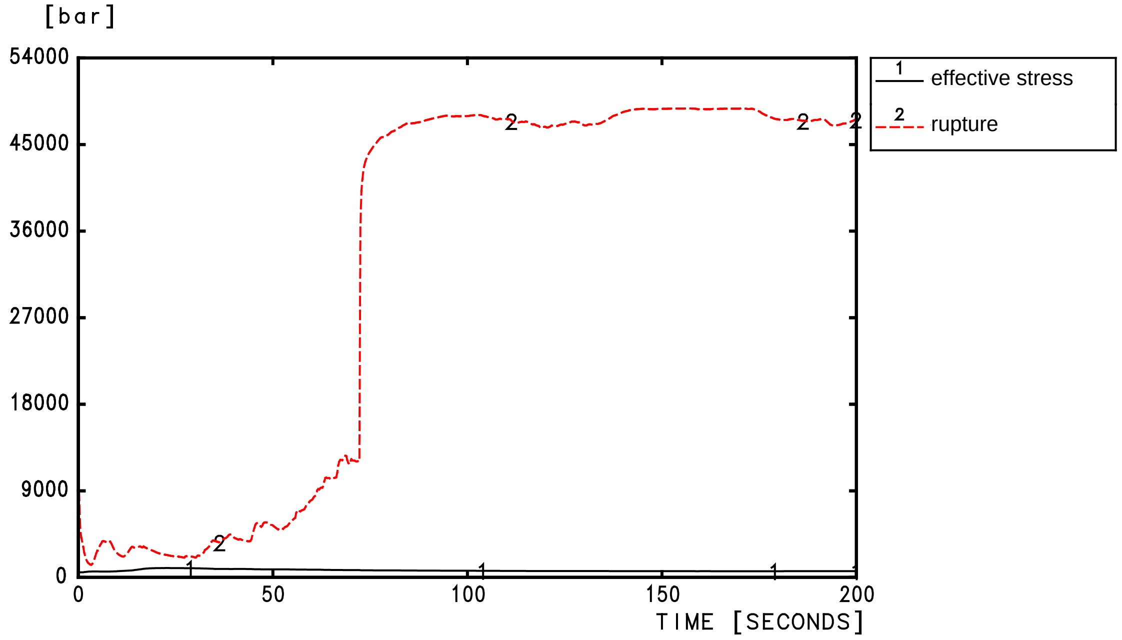
SECTION 16.4.1 - FIGURE F 47: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC2 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



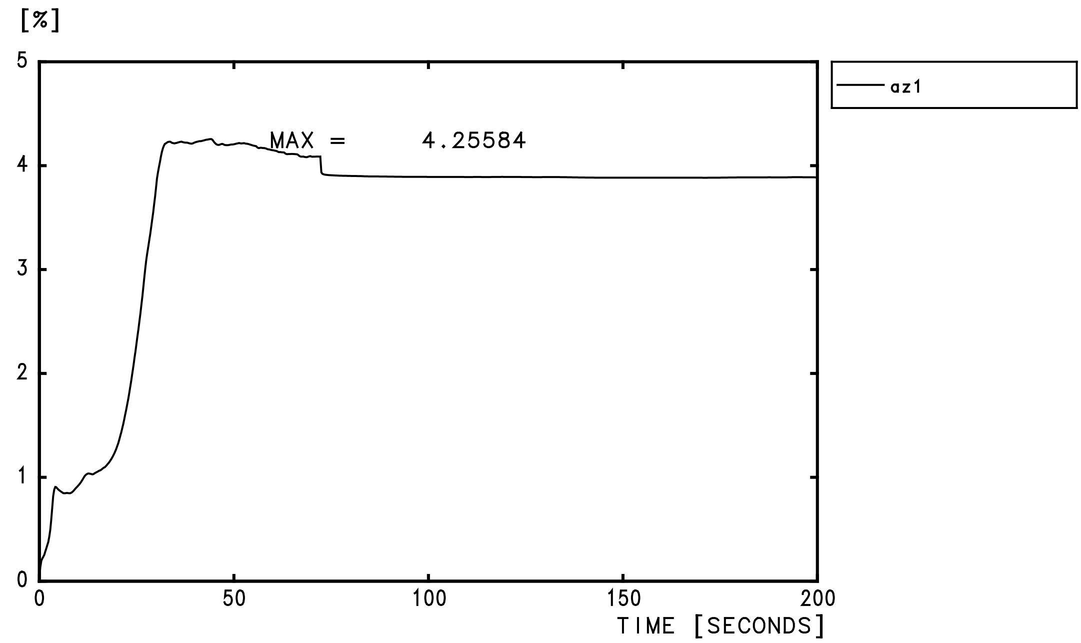
SECTION 16.4.1 - FIGURE F 48: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – C3 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



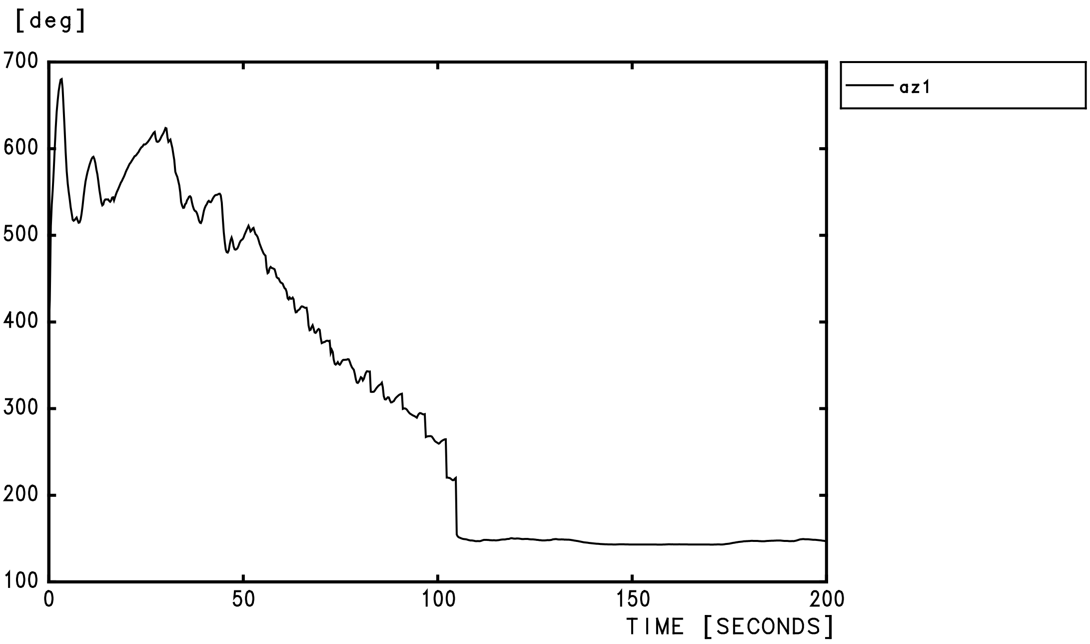
SECTION 16.4.1 - FIGURE F 49: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC4 INDICATOR ROD WITH A HIGH PU CONTENT – MAXIMUM CLAD TEMPERATURE



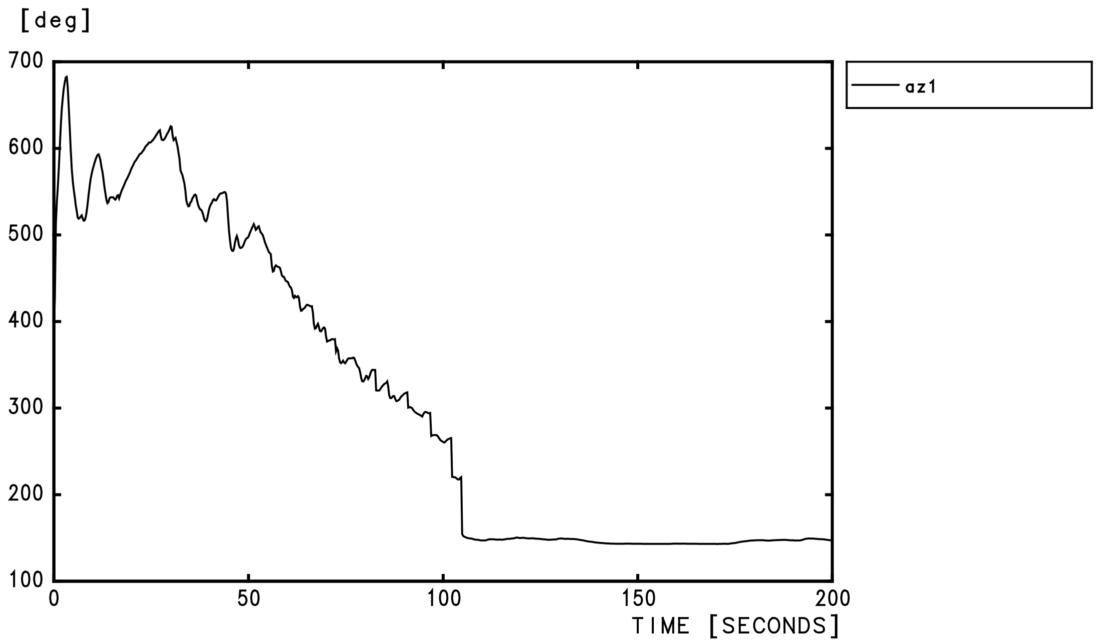
SECTION 16.4.1 - FIGURE F 50: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC4 INDICATOR ROD WITH A HIGH PU CONTENT – TRUE STRESS AND RUPTURE STRESS AT THE PEAK POWER HEIGHT



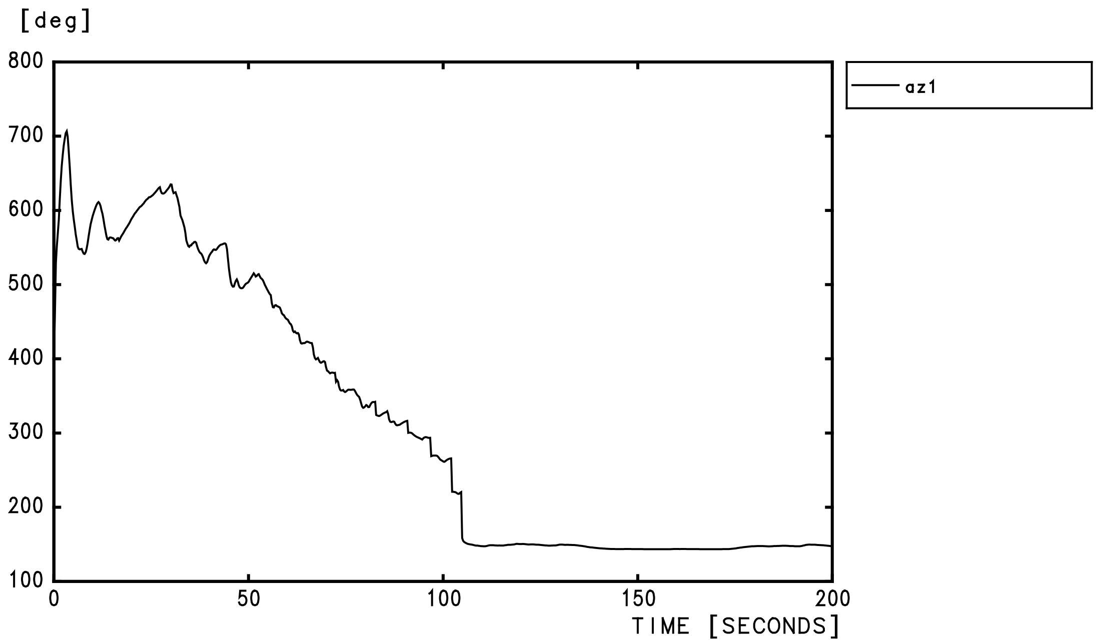
SECTION 16.4.1 - FIGURE F 51: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC4 INDICATOR ROD WITH A HIGH PU CONTENT – DEFORMATION AT THE PEAK POWER HEIGHT



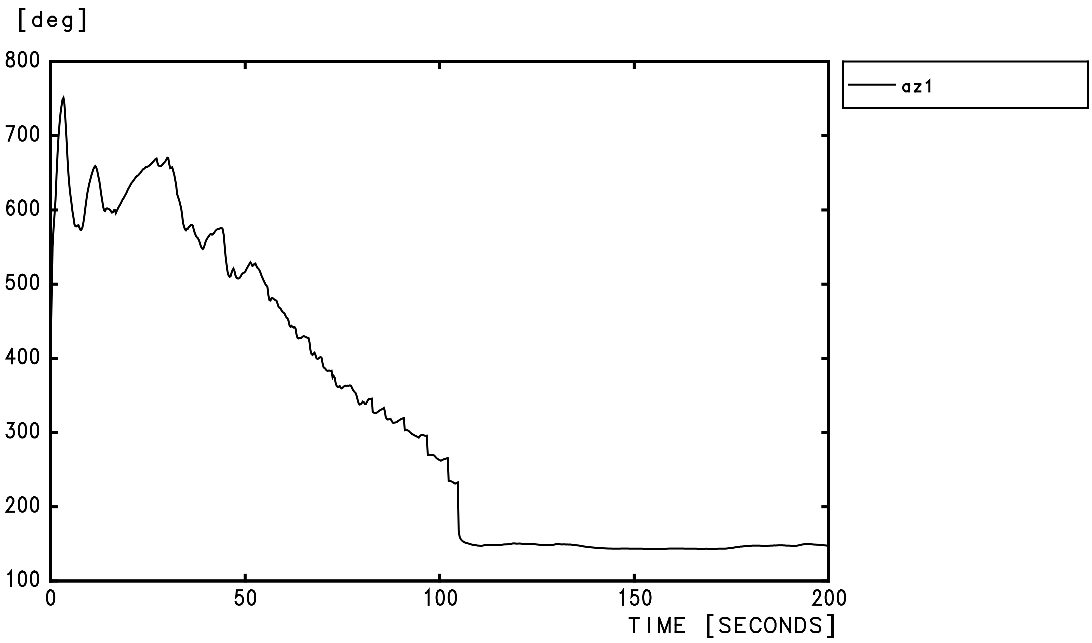
SECTION 16.4.1 - FIGURE F 52: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – BOL INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



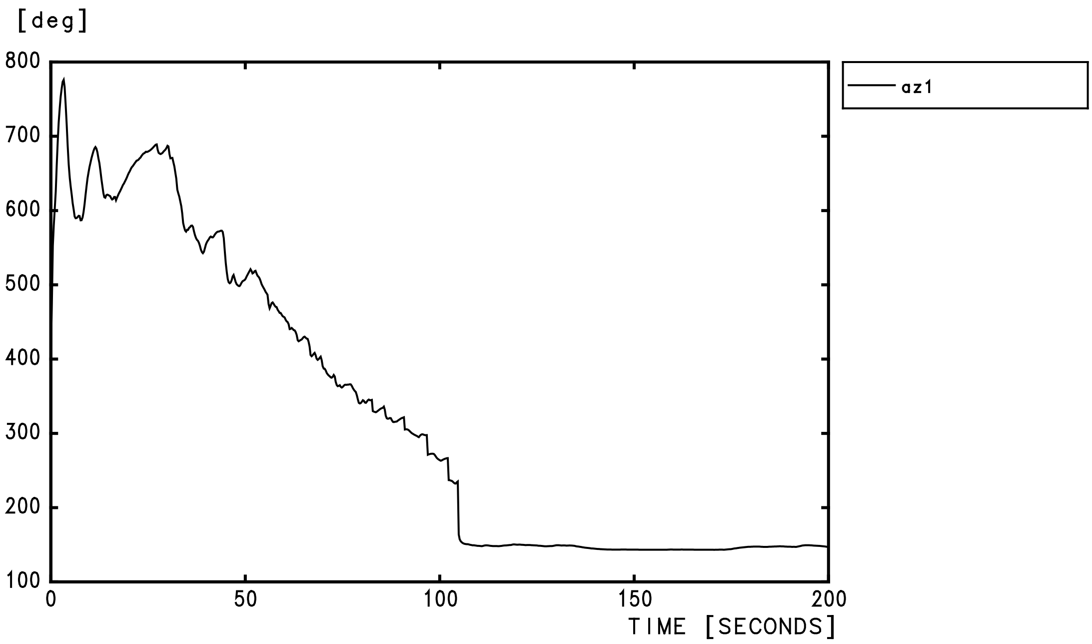
SECTION 16.4.1 - FIGURE F 53: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC1 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



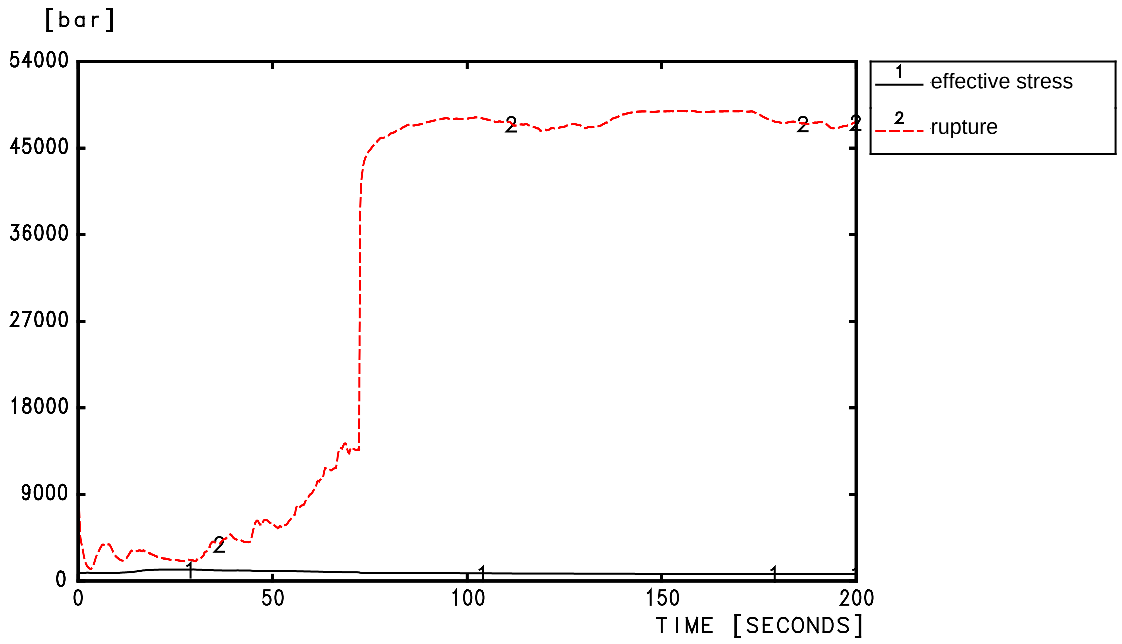
SECTION 16.4.1 - FIGURE F 54: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC2 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



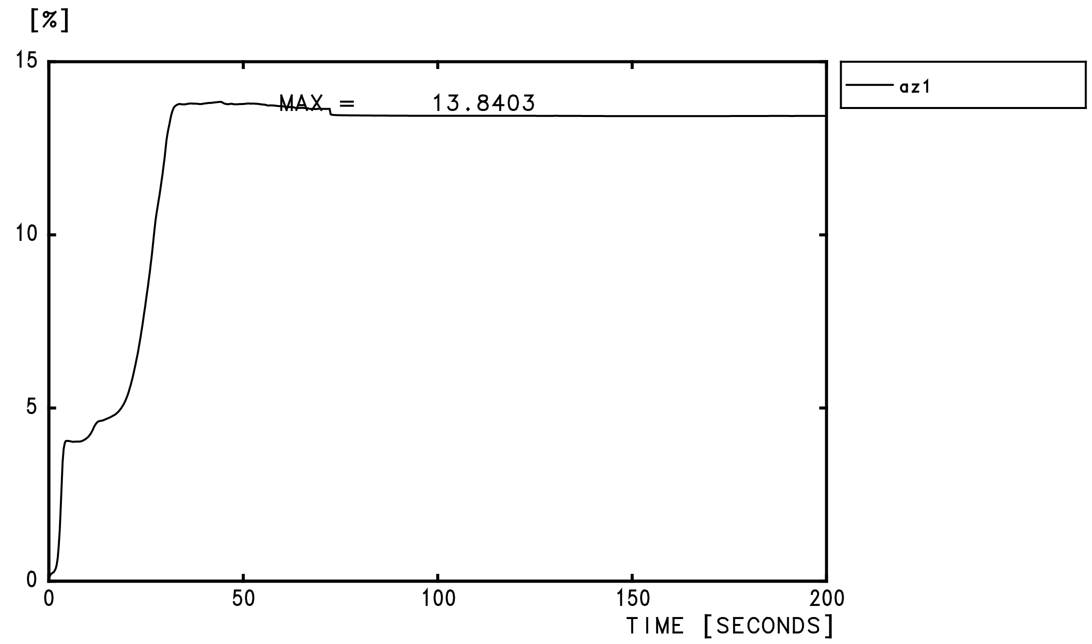
SECTION 16.4.1 - FIGURE F 55: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – C3 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE



SECTION 16.4.1 - FIGURE F 56: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC4 INDICATOR ROD WITH AN AVERAGE PU CONTENT – MAXIMUM CLAD TEMPERATURE

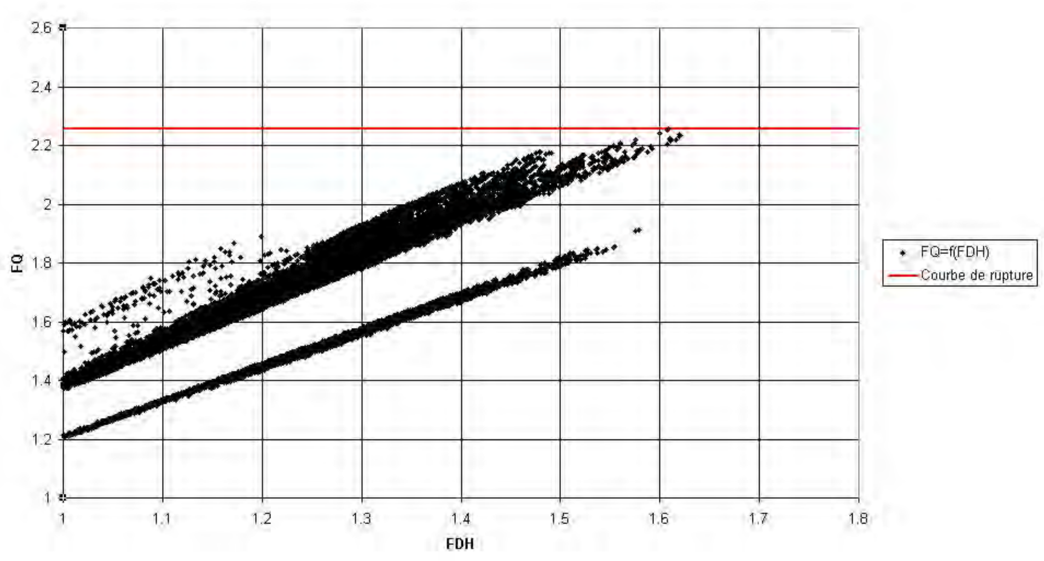


SECTION 16.4.1 - FIGURE F 57: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC4 INDICATOR ROD WITH AN AVERAGE PU CONTENT – TRUE STRESS AND RUPTURE STRESS AT THE PEAK POWER HEIGHT



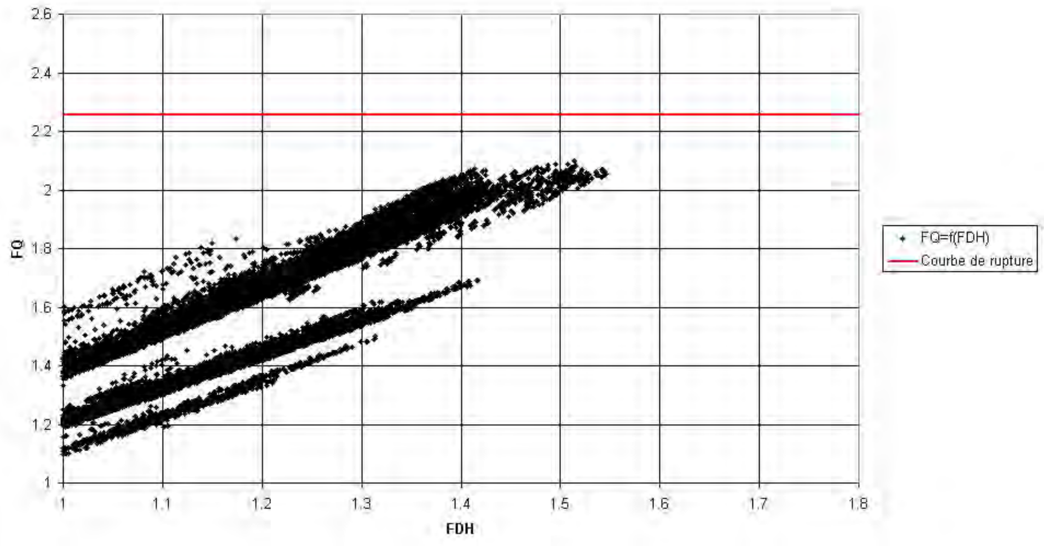
SECTION 16.4.1 - FIGURE F 58: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – HA2 – EOC4 INDICATOR ROD WITH AN AVERAGE PU CONTENT – DEFORMATION AT THE PEAK POWER HEIGHT

30% MOX 18 months fuel management
Rods at beginning of life
Comparison at the rupture surface



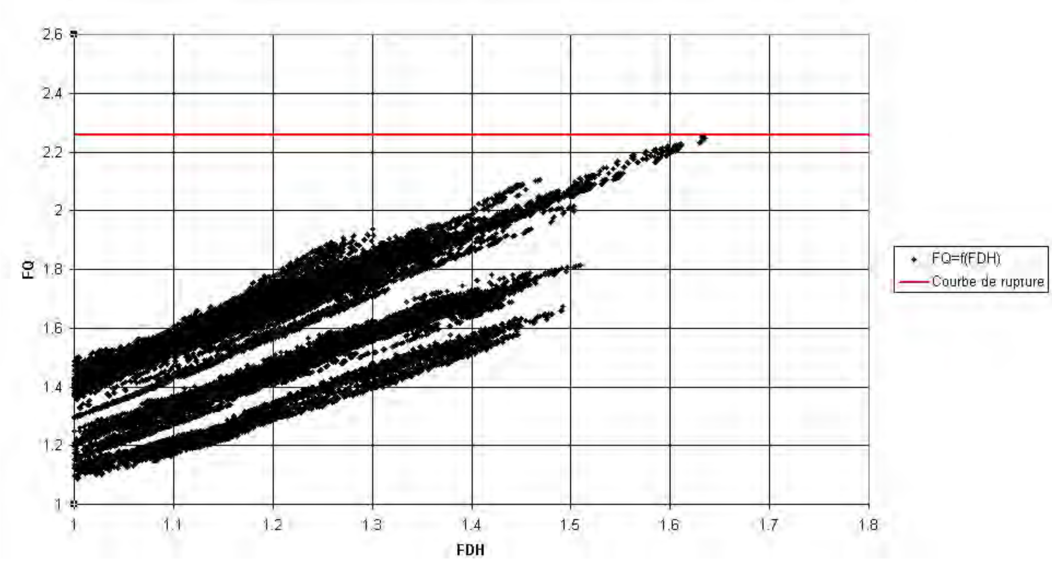
SECTION 16.4.1 - FIGURE F 59: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – RUPTURE CURVE – BOL

30% MOX 18 months fuel management
Rods at end of cycle 1
Comparison at the rupture surface



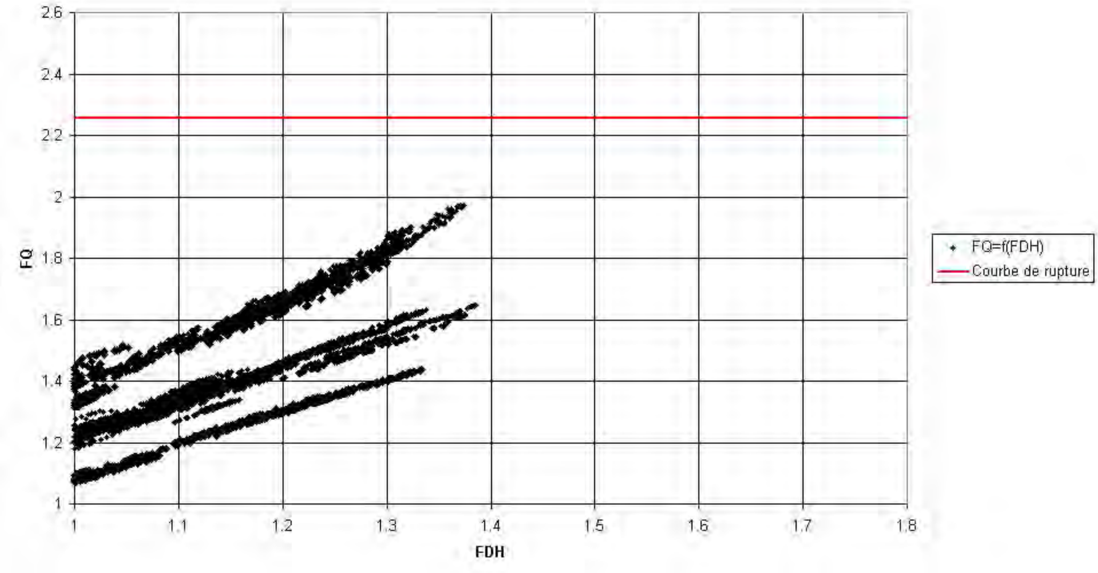
SECTION 16.4.1 - FIGURE F 60: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – RUPTURE CURVE – EOC1

30% MOX 18 months fuel management
Rods at end of cycle 2
Comparison at the rupture surface



SECTION 16.4.1 - FIGURE F 61: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – RUPTURE CURVE – EOC2

30% MOX 18 months fuel management
Rods at end of cycle 3
Comparison at the rupture surface



SECTION 16.4.1 - FIGURE F 62: IN-OUT 30% MOX 18 MONTHS FUEL MANAGEMENT – MC – RUPTURE CURVE – C3

2. DOUBLE ENDED BREAK OF THE MAIN STEAM LINE OUTSIDE THE CONTAINMENT

2.1. MAIN STEAM LINES INSIDE THE REACTOR BUILDING

Due to the implementation of the Break Preclusion (BP) concept on the Main Steam Lines (MSL) inside the Reactor Building, it is considered that the probability of a 2A – Steam Line Break (2A-SLB) inside the reactor building is extremely low.

As a defence-in-depth measure, the complete guillotine break of the MSL (2A-SLB) is however taken into account for:

- the design of the containment,
- the environmental qualification of equipment,
- the Steam Generator (SG) supports.

These aspects are treated respectively in:

- Section 1 of Sub-chapter 6.2.
- Sub-chapter 3.6.
- Section 9 of Sub-chapter 5.4.

Note: Even though the 2A steam line break upstream of the VIV [MSIV] is not postulated with respect to the assessment of core behaviour, due to the application of the Break Preclusion concept, this fault is considered in the PCC-4 analysis as a bounding case covering all PCC events (see section 2 of Sub-chapter 14.5).

2.2. MAIN STEAM LINES OUTSIDE THE REACTOR BUILDING

Due to the implementation of the BP concept on the MSL outside the reactor building, up to the Fixed Point downstream of the Main Steam Isolation Valve (FP_{MSIV}), it is considered that the probability of a 2A – steam line break outside the reactor building and upstream of the FP_{MISV} is extremely low.

Due to the implementation of the Break Preclusion (BP) concept on the three welds of the largest MSL nozzles (1 VDA [MSRT] nozzle, 2 Main Steam Safety Valve (MSSV) nozzles per line), the break of these branch connections (up to and including the complete guillotine break) can be "excluded".

This position is justified by the following main elements:

- The nozzles are extruded nozzles.
- The VDA [MSRT] and MSSV are welded directly on the extruded nozzles.

As a defence-in-depth measure, the complete guillotine break of the largest nozzle (2A-VDA [MSRT] nozzle) is however taken into account as a loading case to verify:

- the integrity of the MSL,
- the integrity of the reactor building penetration.

These aspects are treated in the section entitled "Break preclusion implementation for main steam lines inside and outside containment" and presented in Sub-chapter 10.5.

2.2.1. Definition of studied cases

As a defence-in-depth measure, the following two breaks located in the BP area are however taken into account. These two breaks are assessed to demonstrate the absence of a cliff-edge effect for the core behaviour:

- The break of the VDA [MSRT] branch connection. This break is postulated to result in the simultaneous break of all the branch connections of the MSL in the BP area outside the reactor building (1 VDA [MSRT], 2 MSSV, VIV [MSIV] Bypass-line, sensors). The bounding break size modelled is 1900 cm^2.
- The guillotine break of the MSL, this represents the largest potential break size in the BP area outside the reactor building: 2 x 3700 cm^2.

These two break sizes, 1900 cm^2 and 2 x 3700 cm^2, are analysed with the following set of conservative assumptions:

- The VIV [MSIV] assigned to the broken MSL does not close on demand. This failure to close is caused by the consequential effects resulting from the break.
- The VIV [MSIV] assigned to one of the three intact MSL does not close on demand. This failure to closure is as a result of the application of the Single Failure Criterion.

As a consequence of these two VIV [MSIV] failures, the RCP [RCS] experiences an uncontrolled cooldown with two SGs blowing down via the break. The differences between this assessment and the PCC-4 "bounding" case analysed in section 2 of Sub-chapter 14.5 (2A-SLB with single failure on 1 rod), are as follows:

- On the core side, all rods are inserted (instead of N-1)
- On the secondary side, blowdown of two SGs (compared to one SG). For simplification these two SGs will be referred to as the "affected SG" in the rest of the document.

The 2 cases are analysed below:

Break size	Consequential failure	Single failure	Cooldown / reactivity transient
1900 cm^2 (all branch connections)	VIV_a to close	VIV_u to close	2 SG discharging (1) N rods in
2 x 3700 cm^2 (2A-SLB)	VIV_a to close	VIV_u to close	2 SG discharging (1) N rods in

(1) Depressurisation of each SG is limited by the flow limiter (1300 cm^2)

The transient analyses are performed in accordance with the conservative rules and criteria specified for the PCC-4 analyses.

The study refers to the EPR definition presented in the BDR-99. The plant characteristics assumed for accident analyses are described in Appendix 14B. Differences between this definition and the UK EPR definition will not change the conclusion of the study. The results show that the DNBR criterion is met with a high margin and that the two cases are less onerous than the PCC-4 2A-SLB case.

2.2.2. Methods of analysis

The analysis of the 2 cases is carried out in two stages using the following codes, described in Appendix 14B:

- Stage 1: "system" calculation using the internal coupling of:
    - the MANTA code for the overall thermal-hydraulic behaviour of the main primary and secondary systems (RCP [RCS] and SG), accounting for F1 system operation,
    - the SMART code for neutronic and thermal-hydraulic behaviour of the core.
- Stage 2: SMART/FLICA "core" calculation giving DNBR transient
    - the DNBR calculation is carried out by SMART/FLICA in the form of a post-calculation after MANTA/SMART internal coupling.

2.2.3. Description of studied cases

The main characteristics of the accident are as follows:

- The steam release via the rupture in the 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 the coolant temperature and pressure. In the presence of a negative moderator coefficient, the cooldown leads to an insertion of positive reactivity. The core may become critical and return to power.

2.2.3.1. Initial state and specific assumptions

The initial conditions, identical for the two cases, are presented in Section 16.4.2 - Table 1.

This initial state is also a decoupling state since there are:

- hot shutdown conditions (including for the SG mass),
- ARE [MFWS] supply in the SG at full power (maximum feedwater flow is delivered in all SGs),
- starting of ASG [EFWS] in the two affected SGs from the beginning (conservative assumption).

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, since there is additional energy stored in the fuel. Moreover, since the initial mass of secondary fluid and steam generator pressure are greater at hot zero power, the amplitude and duration of the reactor coolant system cooldown are bigger.

A small initial nuclear power is conservative 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, which are included in the initial shutdown margin.

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

The specific assumptions (compared to PCC-4 SLB assumptions) are as follows:

- initial state: the SLB transients are started in the hot shutdown state (without any stuck rod: 89 rods fully inserted),
- single failure: the single failure is the non-closing of the VIV [MSIV] of one of the three SGs of the unbroken loops,
- sub-criticality: -3000 pcm corresponding to the shutdown margin criterion of -2500 pcm (N-1 rods), reduced by an allowance of 500 pcm, to which is added the minimum worth of the stuck rod (1000 pcm).

The additional assumptions are given below:

- fuel management: the study is carried out at the end of the life of the equilibrium cycle of UO_2 18 months in/out management (reference management). The information and analysis assumptions have been drawn up in an envelope type manner to cover all management options studied. This information is given below,
- power level: zero, representing 10^{-9} times the nominal power,
- pressure, temperature: 155 bar (nominal), 301°C (RCP [RCS] temperature at zero power),
- initial boron concentration: zero,
- Xenon: an axial Xenon distribution is used from the full power equilibrium Xenon situation, all rods withdrawn, plus an adjustment of the Xenon offset on the axial behaviour,

- Doppler effect: a minimum Doppler coefficient (in absolute value) is conservative since it maximises the power transient. At the initial conditions its value is -2.5 pcm/°C (including 20% uncertainty),

- moderator effect: a maximum moderator density coefficient (in absolute value) is conservative. At the initial conditions its value is -83.9 pcm/°C (including an uncertainty of 3.6 pcm/°C),

- kinetic parameters: a minimum effective delayed neutron fraction is conservative. The bounding value considered is 452 pcm, including 5% for uncertainties.

2.2.3.2. Protection signals

- Steam isolation on "Pressure Drop dP/dt > MAX1", for two SGs out of four.

- ARE [MFWS] isolation in the SGa on "Pressure Drop dP/dt > MAX2", SG by SG[1].

- RIS [SIS] on "Pressuriser Pressure < MIN3".

- ARE [MFWS] isolation in the unaffected SG on "SG level > MAX1" (narrow range), SG by SG.

- ASG [EFWS] isolation in the unaffected SG on "SG level > MAX1" (wide range), SG by SG.

- VDA [MSRT] opening on "SG pressure > MAX1", SG by SG.

2.2.3.3. Boundary conditions and protection actions

The characteristics of the fluid systems, associated with the signal setpoints, are chosen to maximise cooling of the reactor coolant during the transient. The assumptions related to RIS [SIS] are chosen to delay and minimise injection and are consistent with the non consideration of boron supply in the analysis.

Section 16.4.2 - Table 2 presents signal setpoints.

Section 16.4.2 - Table 3 gives the actuation delays for protection actions, which are taken into consideration after reaching the parameter setpoint.

<u>Main feedwater system (ARE [MFWS])</u>:

Maximum flows are considered: 900 kg/s for the two affected SGs and 500 kg/s for the two unaffected SGs.

For the two breaks studied, according to the PCC-4 SLB assumptions, the ARE [MFWS] high-load line to the two affected SG is not isolated on "SG level > MAX1" (narrow range).

A feed temperature of 120°C is assumed.

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[1] ARE [MFWS] high-load lines isolation in the SGa on "Pressure Drop dP/dt > MAX1" is not credited.

Emergency feedwater system (ASG [EFWS]):

Maximum flows of 200 te/h to the two affected SGs and 130 te/h to the unaffected SGs are assumed.

Conservatively, the ASG [EFWS] injects into the two affected SGs from the start of the accident, consistent with the PCC-4 SLB assumptions.

For the unaffected SGs, ASG [EFWS] injection occurs following a "SG level < MIN2" signal.

A feed temperature of 10°C (minimum) is assumed.

Safety injection system (RIS [SIS]):

The boron concentration of the RIS [SIS] is assumed to be zero. This is a conservative assumption, as in the PCC-4 analysis.

The minimum characteristic is assumed for the MHSI pumps (four MHSI), as in the PCC-4 analysis. A maximum delivery pressure for the MHSI of 80 bar is also assumed.

An injection temperature of 10°C (minimum) is assumed.

Accumulators:

The boron concentration of the accumulators is conservatively assumed to be zero, as in the PCC-4 analysis.

The accumulators discharge at the pressure of 45 bar. A minimum volume of water (30 m^3) is assumed.

Core inlet mixture matrix:

{CCI Removed}

b.

2.2.3.4. Specific assumptions related to the DNBR calculation

The following assumptions are only used for the DNBR calculation with the SMART/FLICA codes (see Appendix 14A):

- thermal-hydraulic mesh: the thermal-hydraulic calculation, which is carried out with FLICAIII-F (See Appendix 14A), uses a mesh including one channel per assembly, resulting in 241 channels for the core,
- inlet temperature zoning: the inlet temperature zoning by thermal-hydraulic channel in SMART/FLICA is presented in Section 16.4.2 - Figure 1. The choice of noding represents the most onerous core cooling,
- a penalty of 10% is applied to the local heat flux of the assembly with more detailed modelling. This pessimises both the F_q and the $F_{\Delta h}$ of all the hot assembly channels at the same time.

2.2.3.5. DNBR Criteria

The decoupling criterion retained is that of the PCC-4 SLB: a departure from nucleate boiling must be prevented, which ensures that the integrity of the first barrier to fission product release is maintained.

The minimum DNBR over the transient must be more than the following limit, based on the use of Doroschuk critical flux tables [Ref-1]:

- for high pressures (~120 bar), DNBR > 1.18,
- for low pressures (< 120 bar), DNBR > 1.2.

2.2.4. Results of the transients

The same calculation procedure is adopted for the two break sizes, as follows:

- The first calculation is carried out with internal coupling of the MANTA/SMART codes (see Appendix 14A). From this the evolution of power and of all the thermal-hydraulic parameters can be deduced,
- The second calculation uses the SMART/FLICA calculation for calculating the DNBR. Inlet information, core outlet pressure, temperature before mixture for both affected loops and for the unaffected loops, is taken from the previous MANTA/SMART calculation. A second power transient is therefore calculated. This is then compared with that obtained from the MANTA/SMART calculation. The DNBR is calculated during the transient.

The results obtained for the two break sizes (1900 cm^2, 2A) are given below.

2.2.4.1. 1900 cm^2 break

2.2.4.1.1. Sequence of events

Section 16.4.2 – Table 4 gives the sequence of events from the MANTA/SMART calculation.

2.2.4.1.2. Variation of power during the transient

The variation of all thermal-hydraulic parameters from the MANTA/SMART calculation is presented in Section 16.4.2 - Figure 2 to Figure 10.

The power transients calculated by the MANTA/SMART and SMART/FLICA codes differ slightly. The trends are very similar with a maximum power of 23% for the MANTA/SMART calculation and 26% for SMART/FLICA calculation. The SMART/FLICA calculation is therefore slightly more conservative for the power excursion and consequently for DNBR.

2.2.4.1.3. Variation of DNBR during the transient

The minimum DNBR is reached at 250 seconds and is equal to 3.6, which is significantly above the criterion.

2.2.4.2. 2A break

2.2.4.2.1. Sequence of events

Section 16.4.2 - Table 5 gives the sequence of events from MANTA/SMART calculation.

2.2.4.2.2. Variation of power during the transient

The transient data for all thermal-hydraulic parameters from the MANTA/SMART calculation is presented in Section 16.4.2 – Figure 11 to Figure 19.

The power transient derived in the MANTA/SMART and SMART/FLICA calculations differ slightly. The trends are very similar with a maximum power of 35%NP for the MANTA/SMART calculation and 36% NP for the SMART/FLICA calculation. The SMART/FLICA calculation is therefore slightly more conservative for the power transient, and consequently for DNBR.

2.2.4.2.3. Evolution of DNBR during the transient

The minimum DNBR is reached at 110 seconds and is calculated to be 3.1, which is significantly above the criterion.

2.2.5. CONCLUSION

The 2A-SLB in the BP area outside of the containment leads to two SGs discharging when the consequential failure of the associated VIV [MSIV] to close is assumed, with the single failure of another VIV [MSIV] to close on demand. Analysed with the same set of conservative assumptions as the PCC-4 SLB, the results show a large margin to the DNBR criterion with a minimum DNBR of 3.1 compared to a criterion of 1.2. The "BP SLB" is less onerous than the "PCC-4 SLB" case, where one SG discharges with one stuck rod, being the most conservative single failure in the PCC analysis. The minimum DNBR is 3.1 versus of a value of 2.2 in BDR-99 as shown in section 2.15 of Appendix 14B. This is despite a higher power excursion of 36% compared to approximately 13% in the BDR-99 analysis. The negative effect of a longer overcooling, two SGs discharging instead of one SG, is more than compensated for by the positive effect of the "no stuck rod" as the power is not concentrated in a low number of fuel assemblies.

The 1900 cm^2 break in the BP area outside of containment, corresponding to the failure of all branch connections, is bounded in terms of the power excursion and minimum DNBR by the 2A-SLB of larger size (2 x 3700 cm^2) in the BP area.

Note: These conclusions are also relevant for the UK EPR PCSR. The conservative assumptions used in the analyses are the same for the BDR-99 and UK-EPR PCC-4 SLB, even though all the numerical values may not be the same. Taking into account the difference between the minimum DNBR obtained for the BDR-99 and UK-EPR PCC-4 SLB, and the high DNBR margins in the analysed cases, minimum DNBR of 3.1 versus a criterion of 1.2, it is concluded that the DNBR criterion will still be met.

SECTION 16.4.2 - TABLE 1

Initial conditions

Parameters		Values used
Reactor power	(FP)	10^{-9}
Thermal-hydraulic loop flow rate	(m^3/h/loop)	26987
Pressuriser pressure	(bar a)	155
Secondary side pressure	(bar a)	87.1
Average RCP [RCS] temperature	(°C)	301
Pressuriser level	(% of MR)	28% + 5%
SG level	(% of NR)	56% (NOM)
ARE [MFWS] temperature	(°C)	120°C
RCP [RCS] boron concentration	(ppm)	0
Initial ARE [MFWS] flow rate in the affected SGs	(kg/s)	900
Initial ARE [MFWS] flow rate in the unaffected SGs	(kg/s)	500
ASG [EFWS] flow rate in the affected SG	(kg/s)	55.6

SECTION 16.4.2 - TABLE 2

Signal Setpoints

SIGNALS	THRESHOLDS
dP/dt > MAX1 VIV [MSIV] closure (two out of four)	Gradient of -2 bar/min Setpoint: 7 + 1.5 bar below the SG pressure
dP/dt > MAX2 ARE [MFWS] isolation in the two SGa	Gradient of -2 bar/min Setpoint: 17 + 1.5 bar below the SG pressure
Pressuriser pressure < MIN3 SI signal	115 - 3 = 112 bar
SG level > MAX1 ARE [MFWS] isolation (SG by SG) for the two unaffected SGs	71% + 5% = 76% narrow range

SECTION 16.4.2 - TABLE 3

Signal and Action Delays

SIGNAL	ACTUATION TIME DELAY	ACTION
dP/dt > MAX1	+ 5.9 s (0.9 s signal delay and 5 s valve closure)	VIV [MSIV] closure
dP/dt > MAX2	+ 10.9 s (10 s valve closure and 0.9 s signal delay)	SGa ARE [MFWS] total isolation
Pressuriser Pressure < MIN3	+ 15.9 s (10 s for starting the pumps, 5 s for obtaining full flow and 0.9 s signal delay)	RIS [SIS] injection

SECTION 16.4.2 - TABLE 4

Sequence of events: 1900 cm² break

EVENT	TIME (seconds)
"SG pressure drop > MAX1"	9
Steam lines isolation	15
SGa ARE [MFWS] isolation	32
RIS [SIS] signal	33
SGa ARE [MFWS] isolation (SG undergoing second emptying)	38
Unaffected SG ARE [MFWS] isolation	72
ASG [EFWS] isolation	no
Unaffected SG ASG [EFWS] start-up	no

SECTION 16.4.2 - TABLE 5

Sequence of events – 2A break

EVENT	TIME (seconds)
"SG pressure drop > MAX1"	1
Steam lines isolation	11
SGa ARE [MFWS] isolation	13
RIS [SIS] signal	25
SGa ARE [MFWS] isolation (SG undergoing second emptying)	26
Unaffected SG ARE [MFWS] isolation	72
ASG [EFWS] isolation	no
Unaffected SG ASG [EFWS] start-up	no

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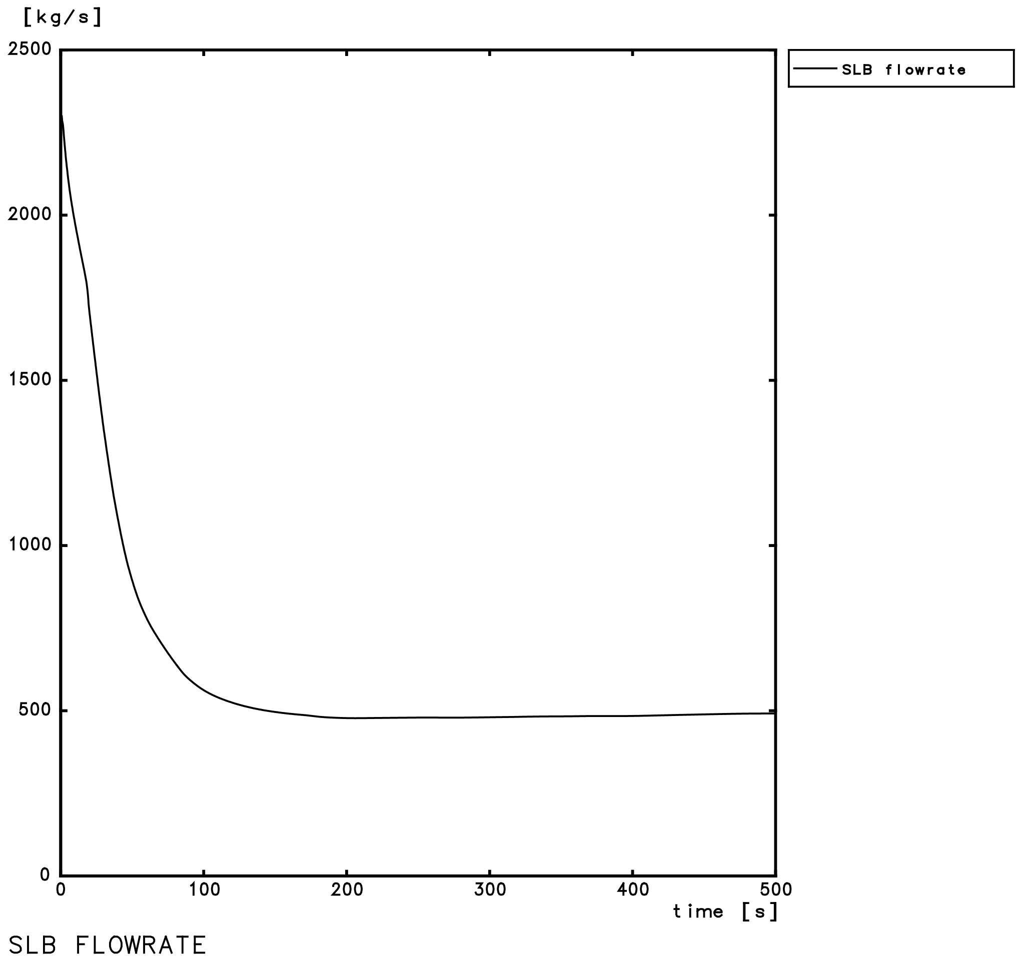
b

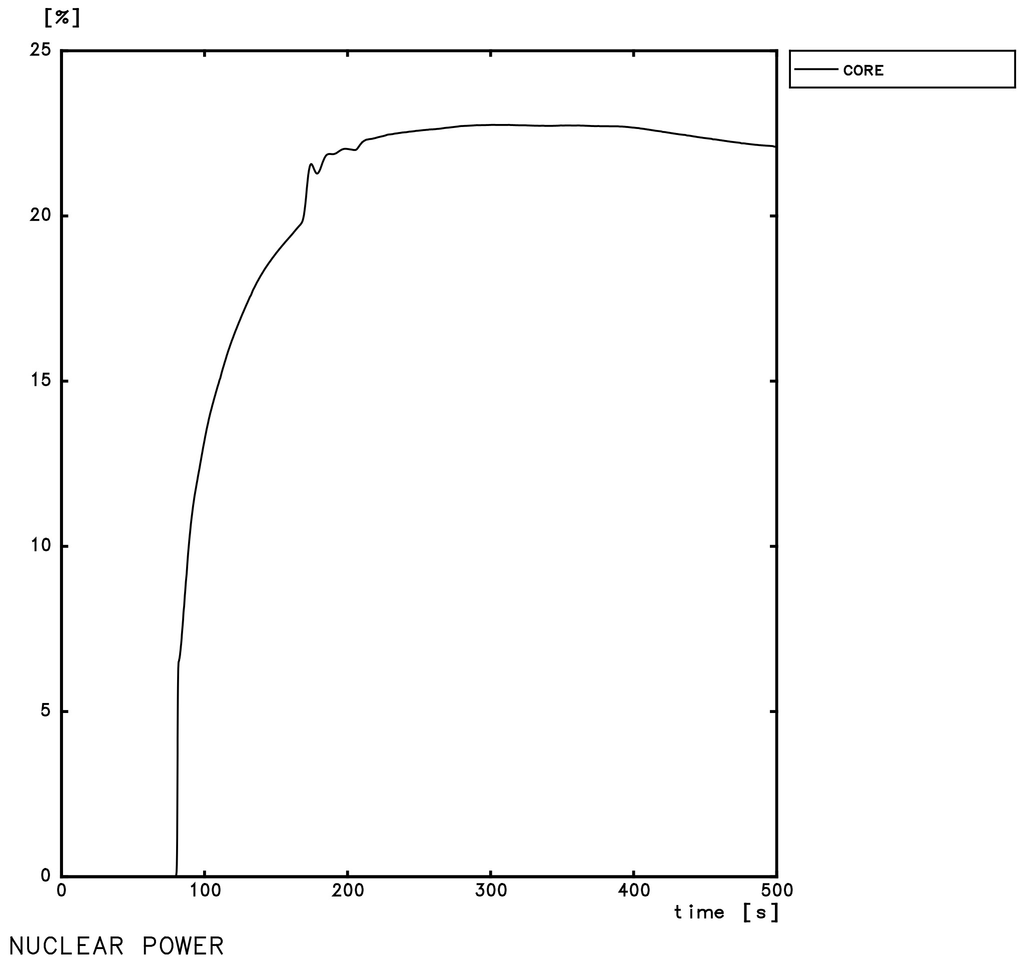
SECTION 16.4.2 - FIGURE 2

1900 cm^2 break – SLB flow rate



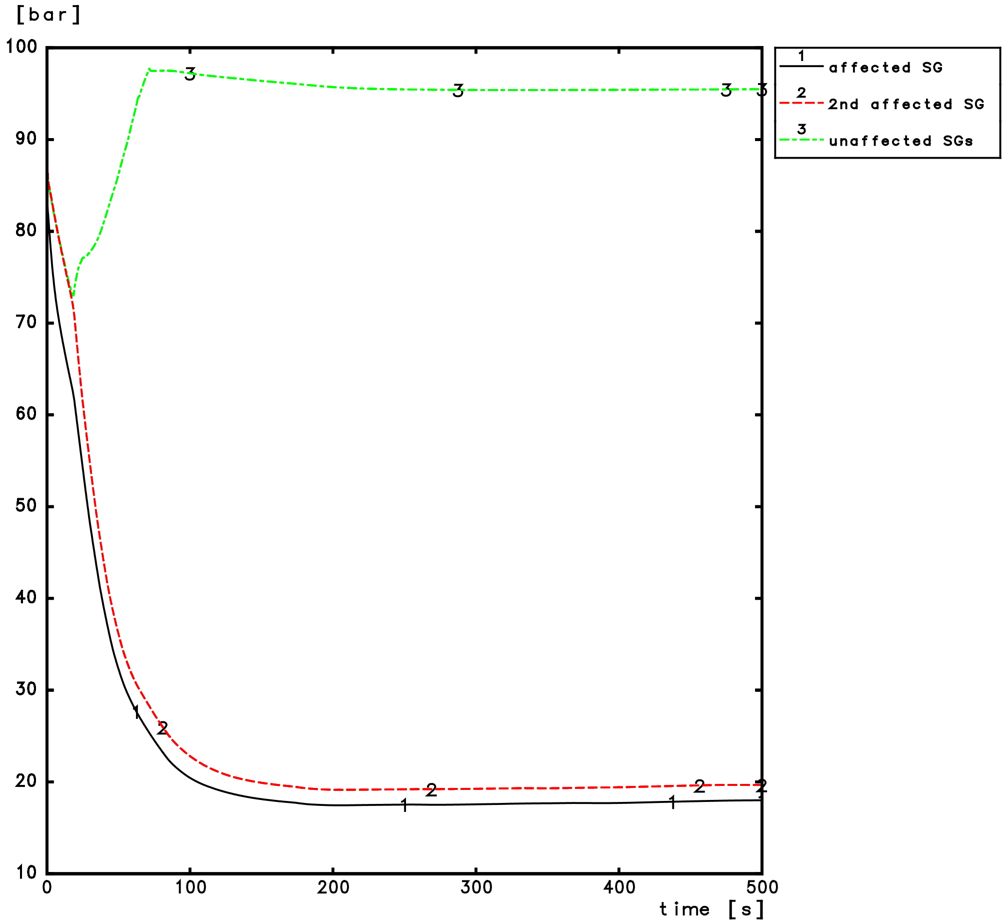
SECTION 16.4.2 - FIGURE 3

1900 cm^2 break – Nuclear power



SECTION 16.4.2 - FIGURE 4

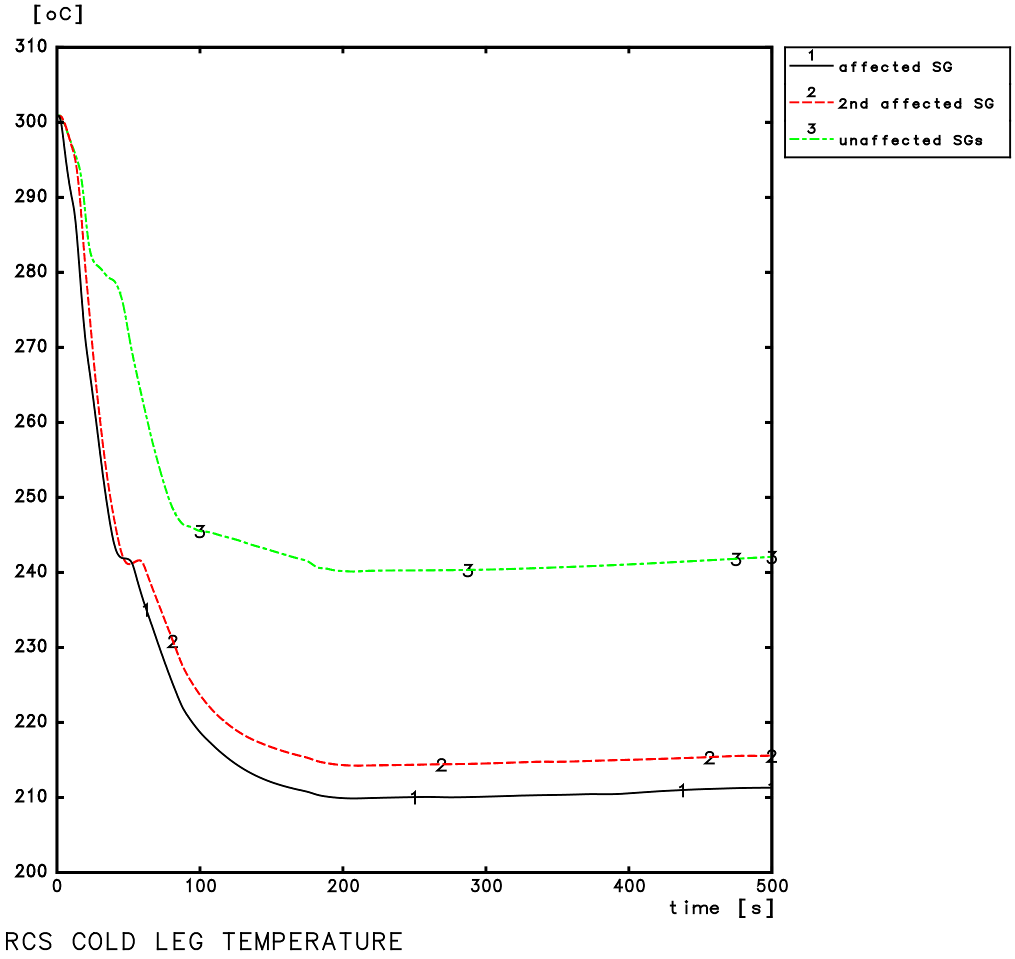
1900 cm² break – Secondary side pressure



SECONDARY SIDE PRESSURE

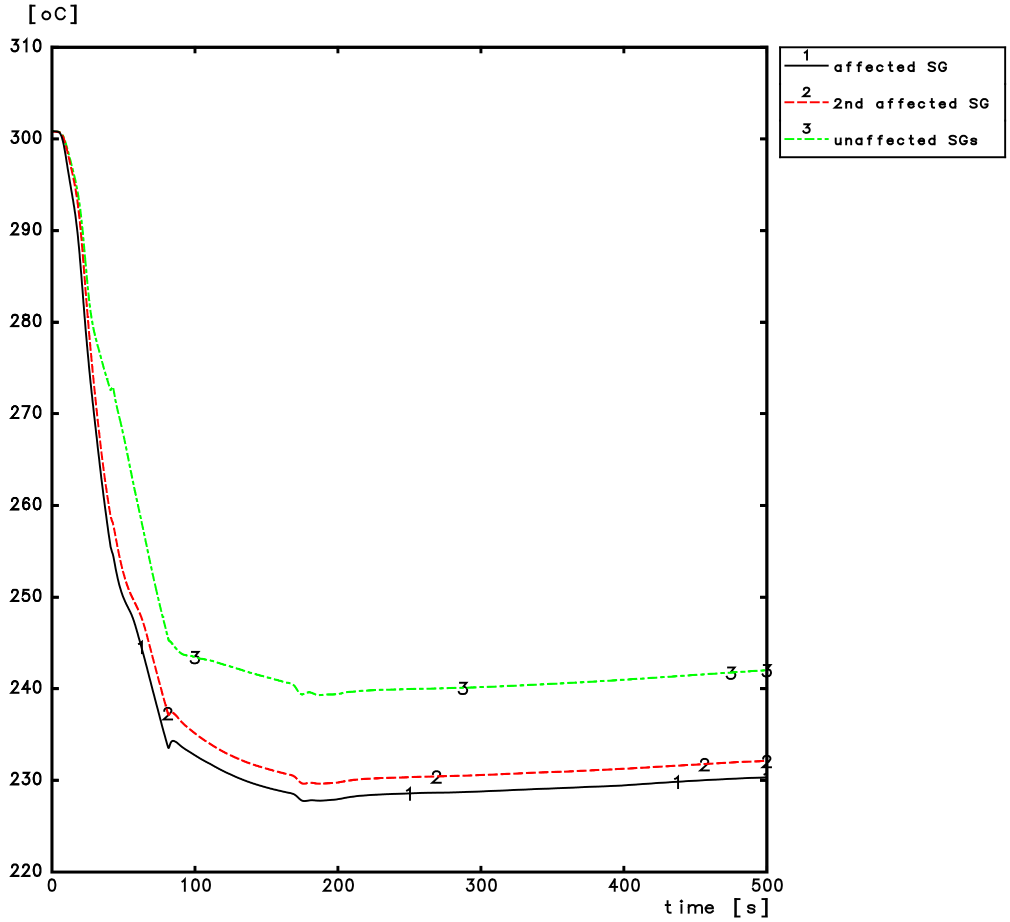
SECTION 16.4.2 - FIGURE 5

1900 cm^2 break – Cold leg temperature



SECTION 16.4.2 - FIGURE 6

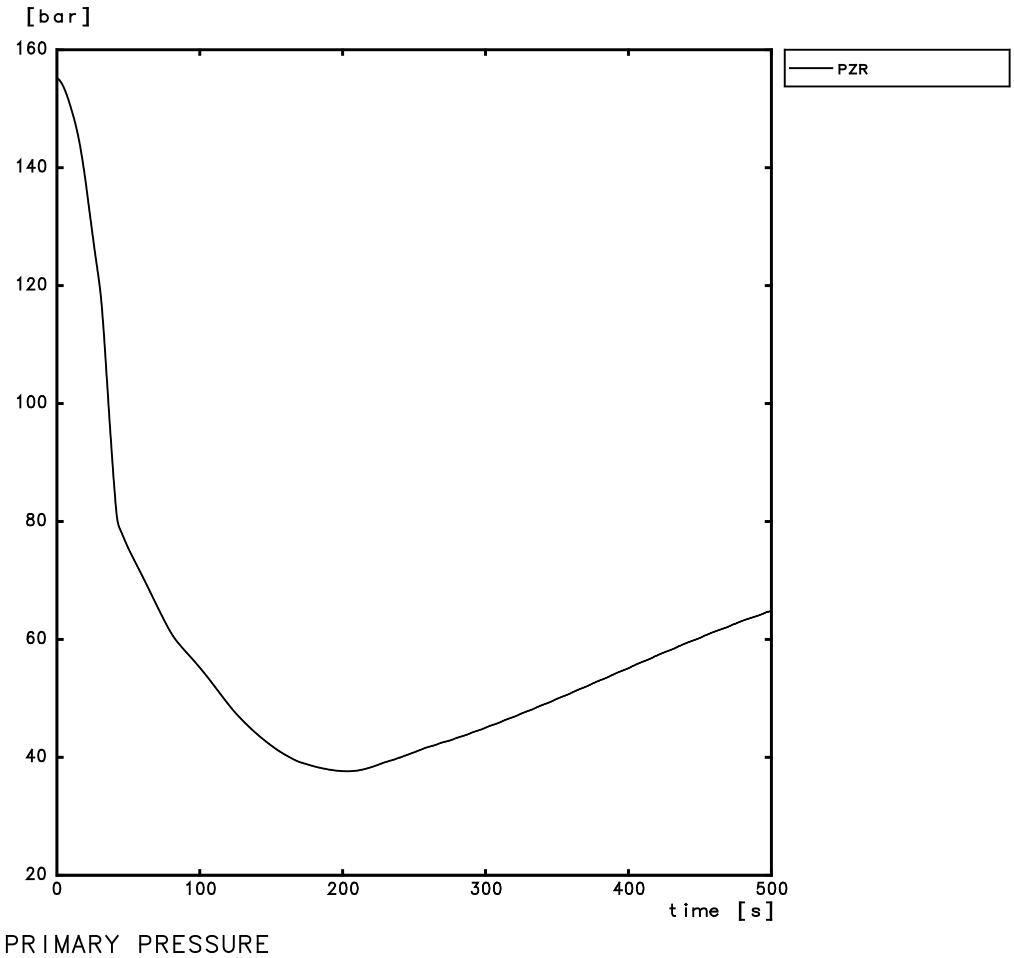
1900 cm^2 break – Hot leg temperature



RCS HOT LEG TEMPERATURE

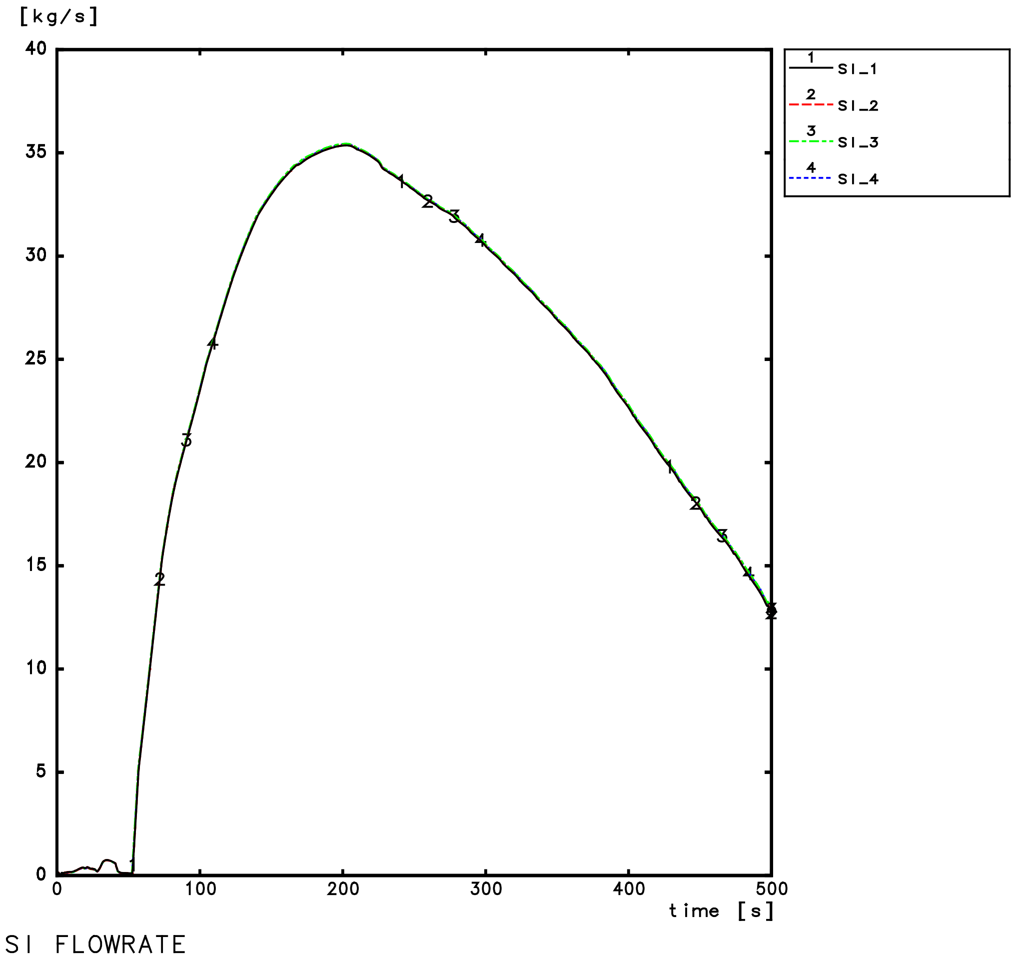
SECTION 16.4.2 - FIGURE 7

1900 cm^2 break – Pressuriser pressure



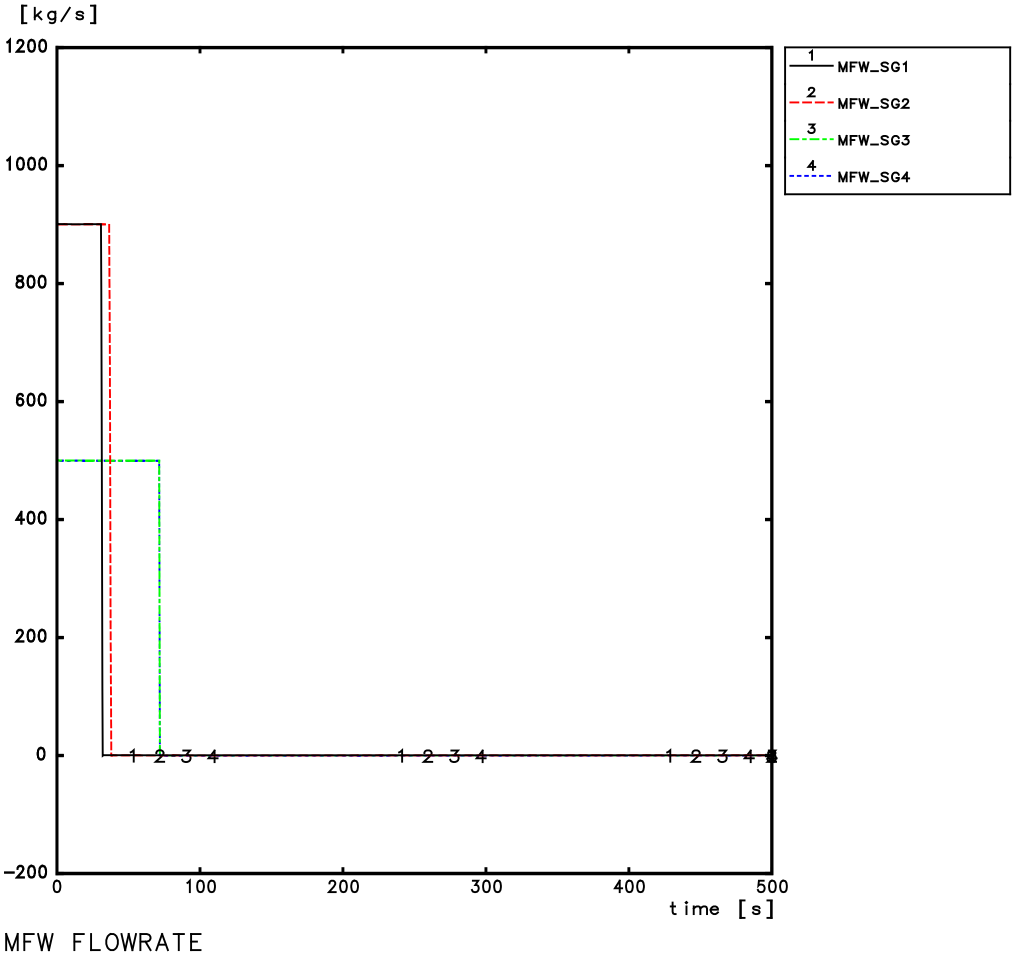
SECTION 16.4.2 - FIGURE 8

1900 cm² break – RIS [SIS] injection



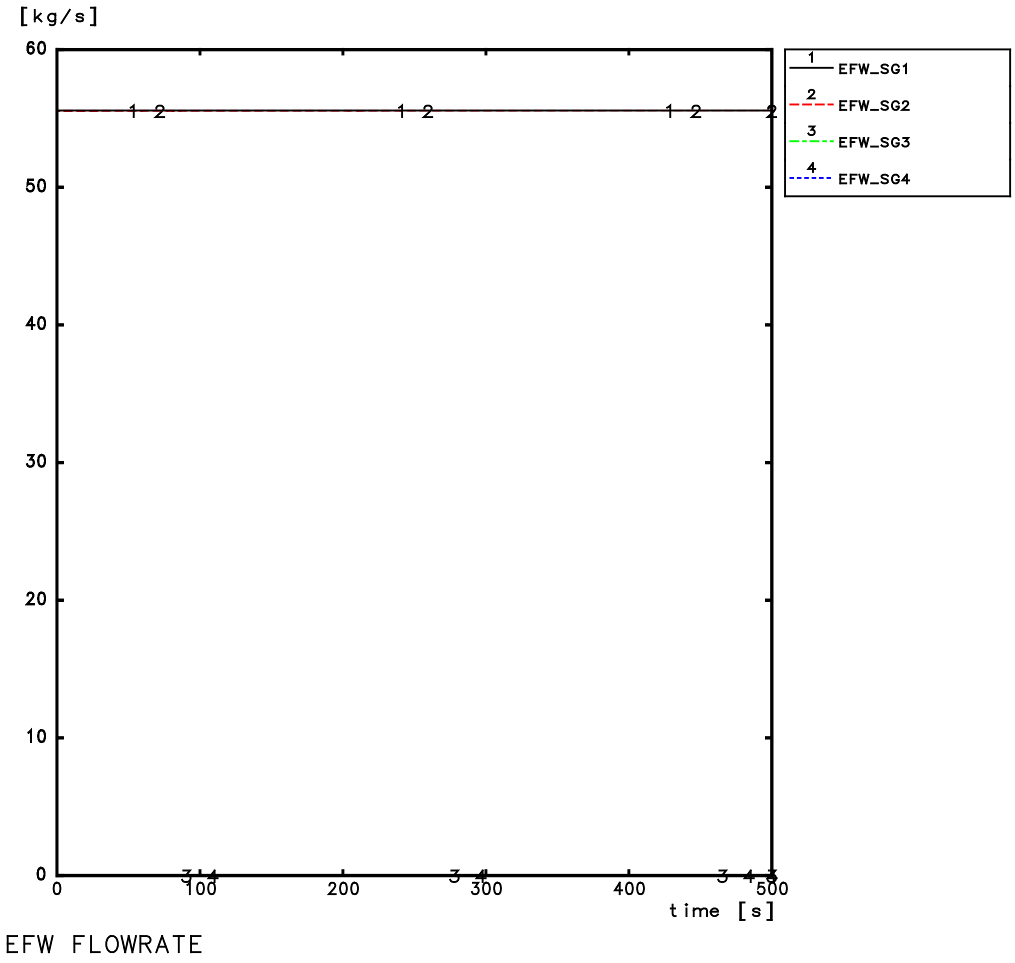
SECTION 16.4.2 - FIGURE 9

1900 cm² break – Main feedwater flow rate



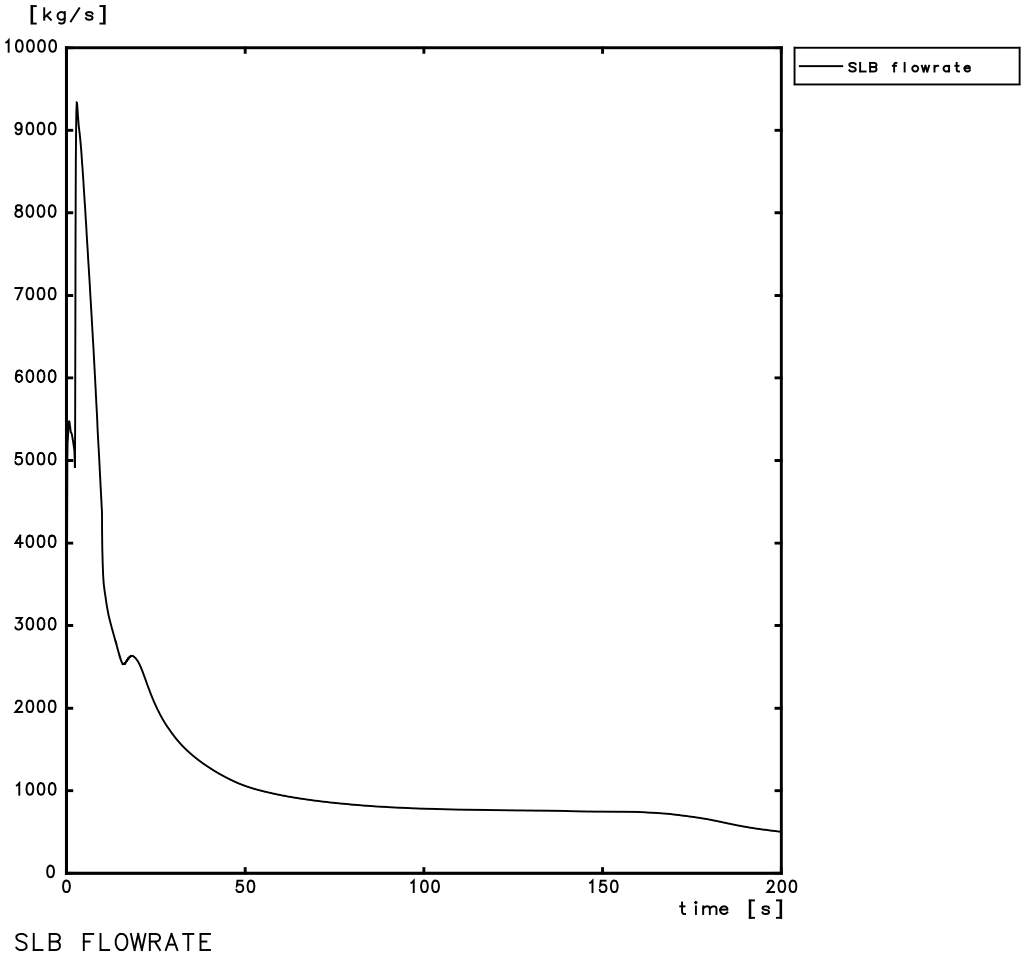
SECTION 16.4.2 - FIGURE 10

1900 cm² break – Emergency feedwater flow rate



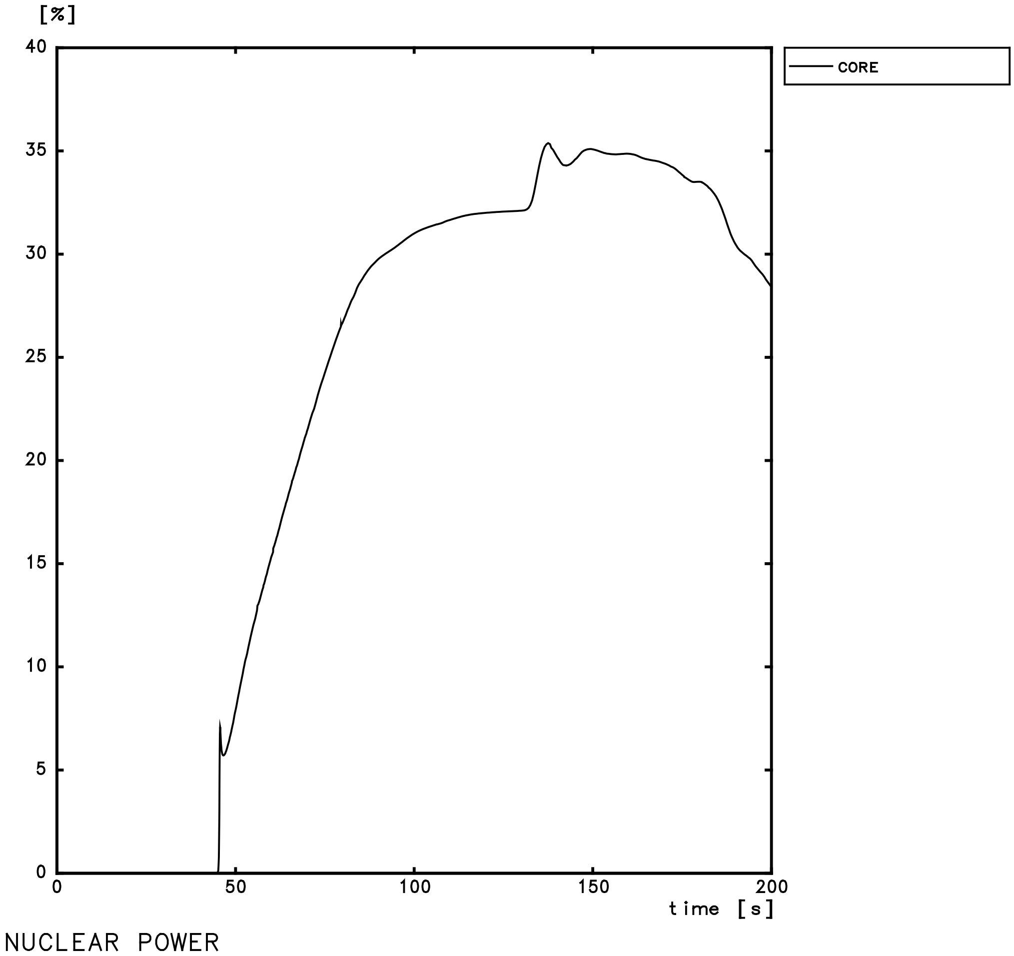
SECTION 16.4.2 - FIGURE 11

2A break – Steam line break flow rate



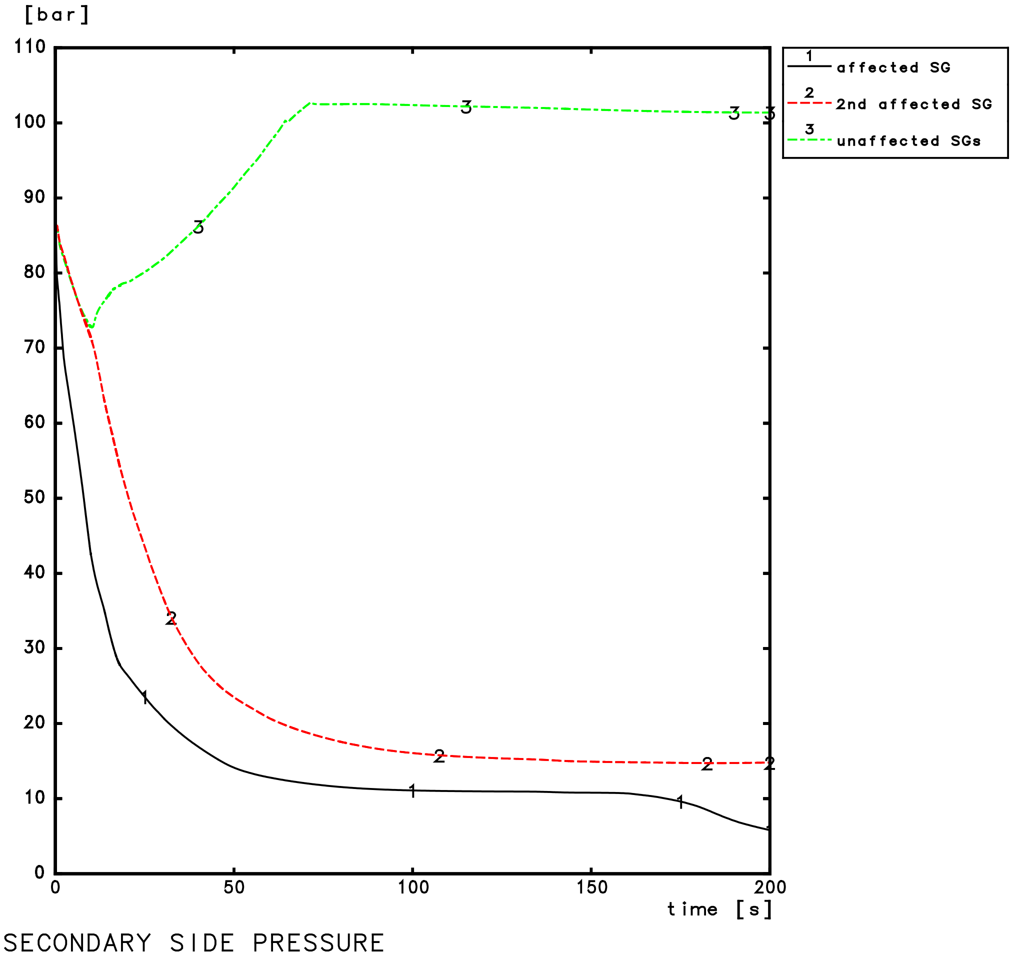
SECTION 16.4.2 - FIGURE 12

2A break – Nuclear power



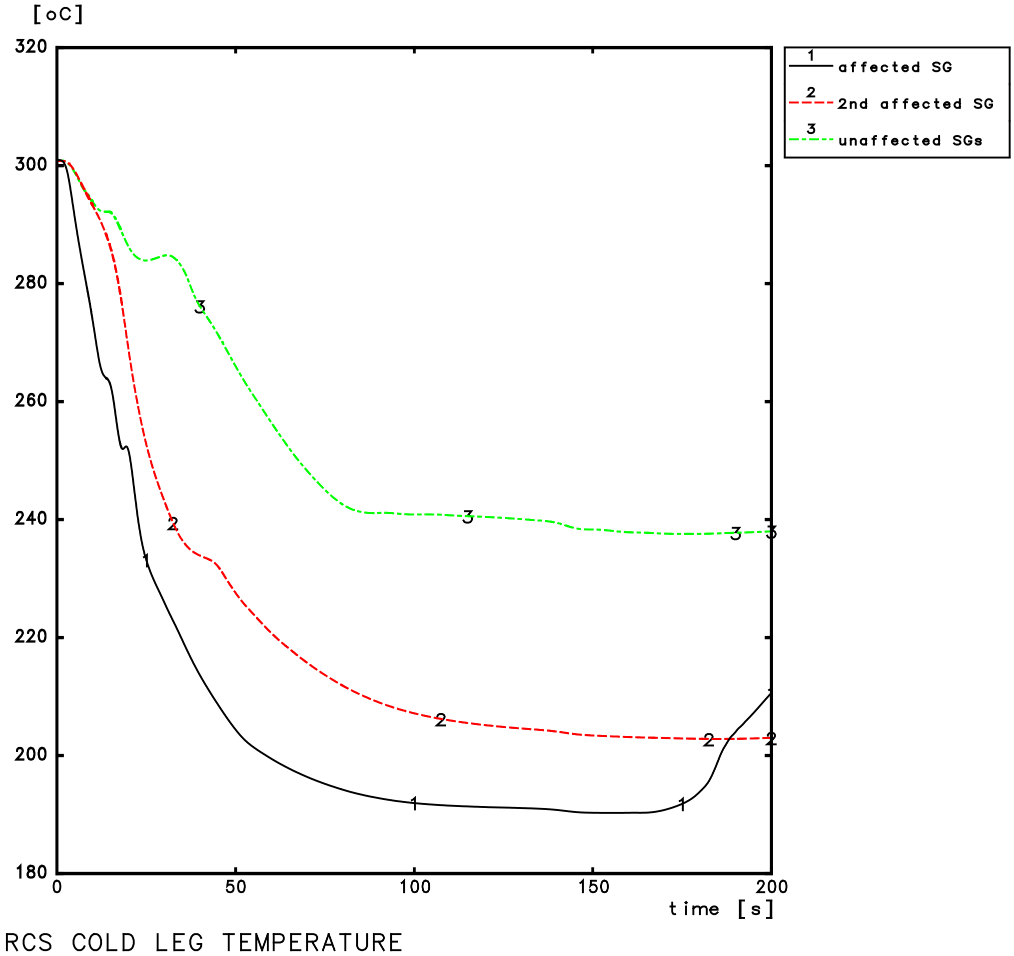
SECTION 16.4.2 - FIGURE 13

2A break – Secondary side pressure



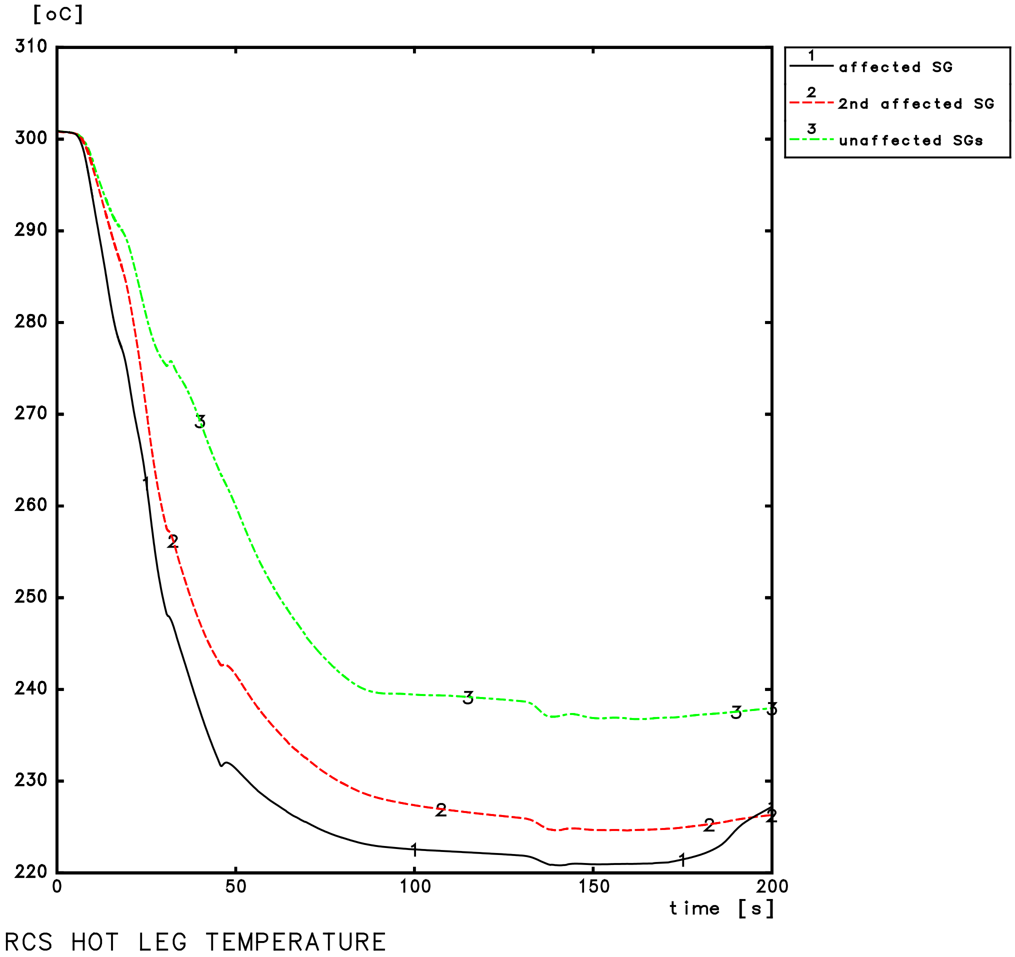
SECTION 16.4.2 - FIGURE 14

2A break – Cold leg temperature



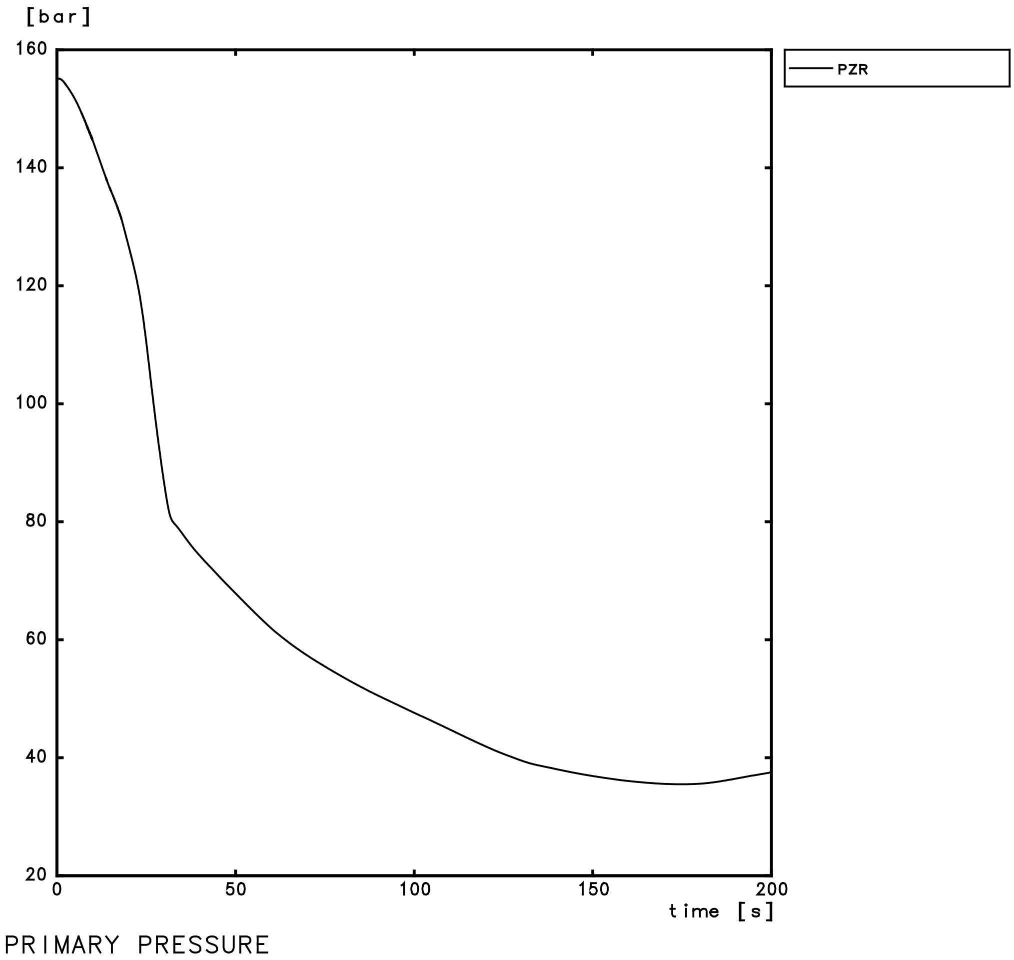
SECTION 16.4.2 - FIGURE 15

2A break – Hot leg temperature



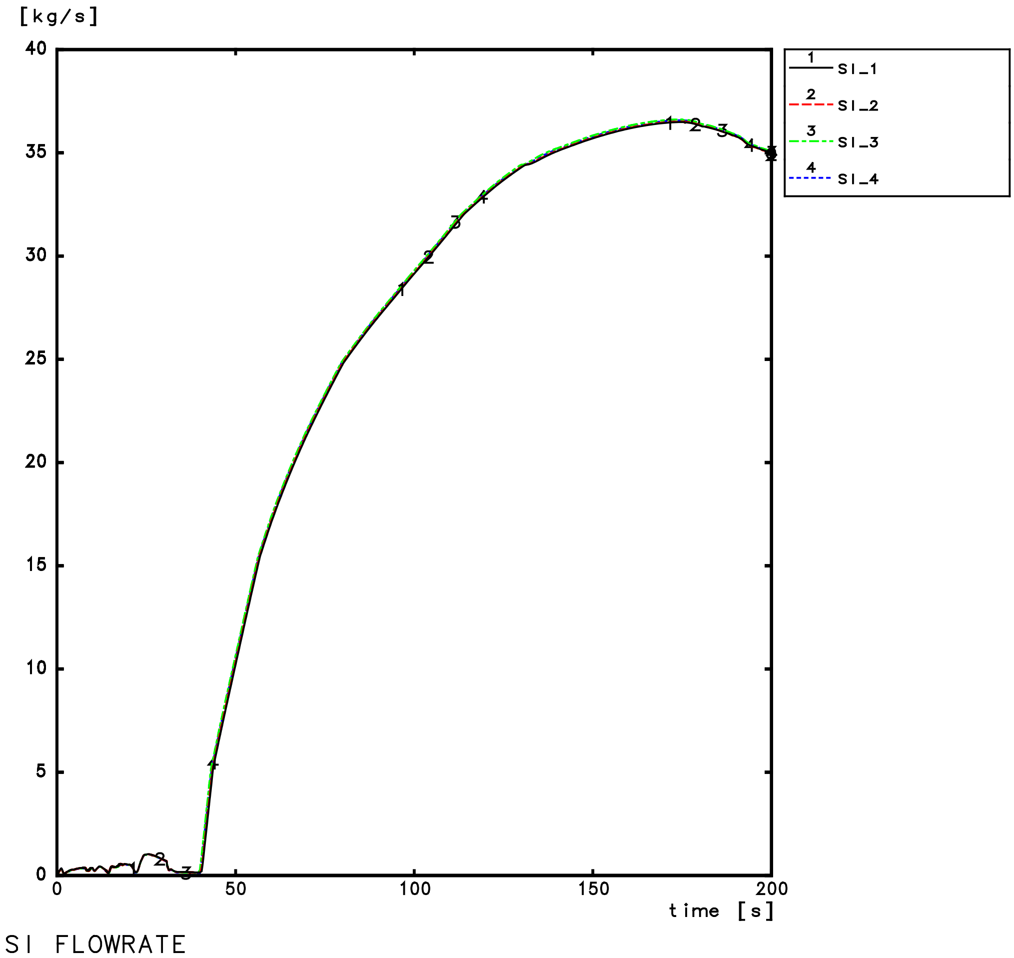
SECTION 16.4.2 - FIGURE 16

2A break – Pressuriser pressure



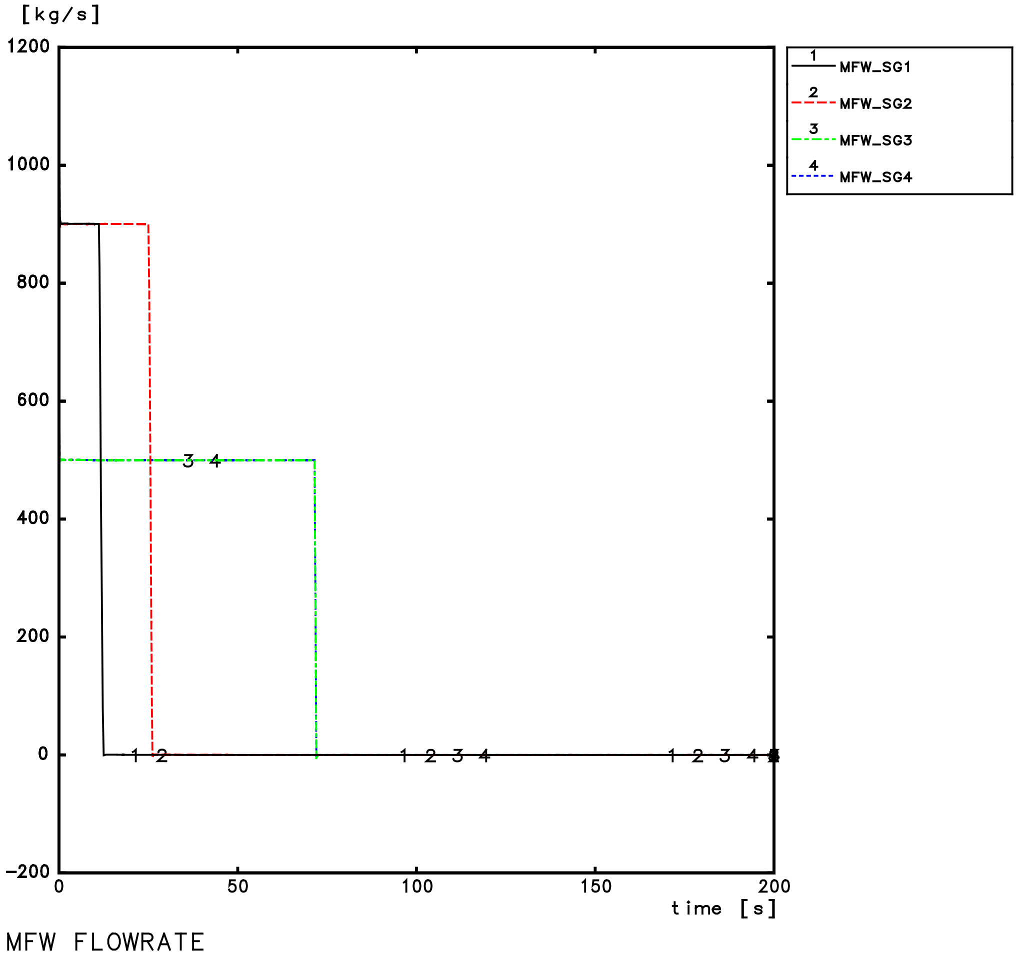
SECTION 16.4.2 - FIGURE 17

2A break – RIS [SIS] injection



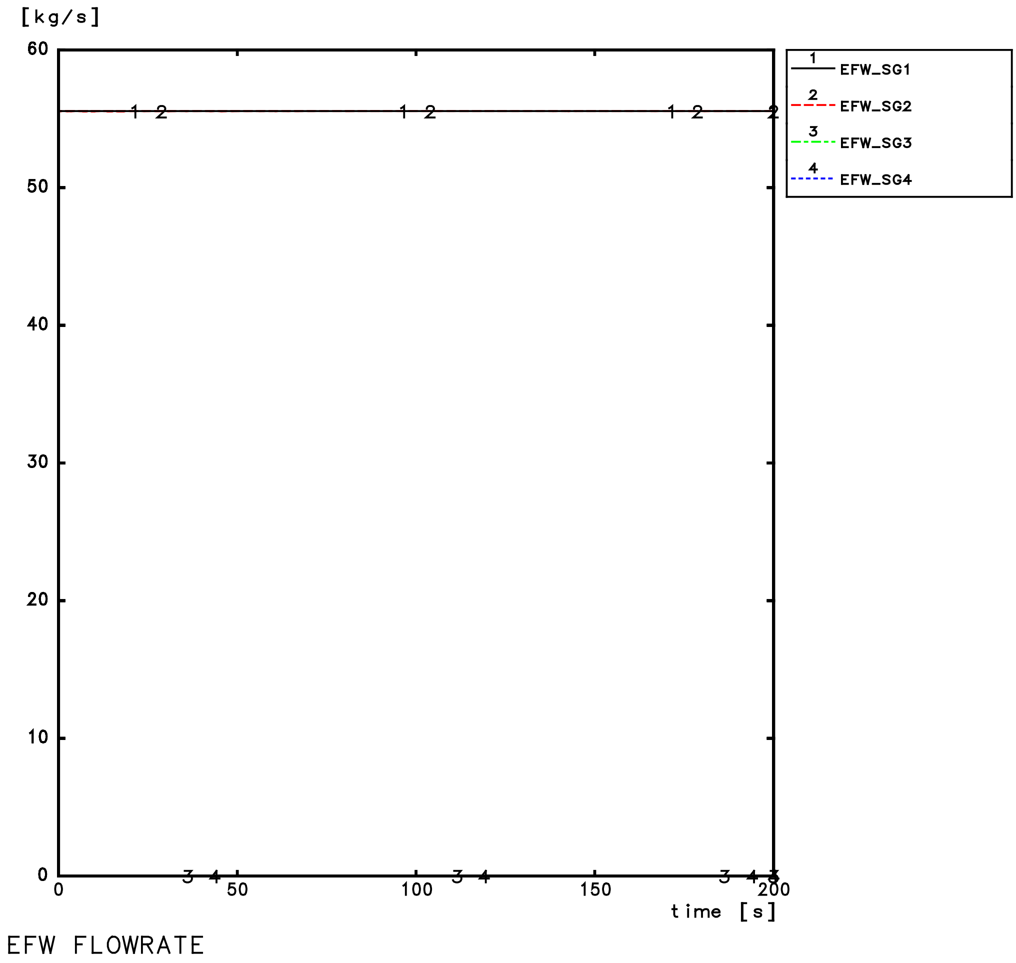
SECTION 16.4.2 - FIGURE 18

2A break – Main feedwater flow rate



SECTION 16.4.2 - FIGURE 19

2A break – Emergency feedwater flow rate



3. STEAM GENERATOR TUBE RUPTURE (1 TUBE) WITH MAIN STEAM LINE BREAK

3.1. INTRODUCTION

3.1.1. Accident description

A steam system piping break leads to a heat removal increase from the core by the secondary side. It is an overcooling event principally leading to a reactor power rise in power operation because of the negative moderator reactivity feedback. It might, therefore, lead to fuel and cladding failure.

The steam generator tube rupture (SGTR) leads to a loss of primary coolant, which is transferred to the steam generator affected by the rupture. The break induces a decrease in the primary pressure and contamination of the secondary side due to the tube leakage flow rate. The main consequence is a possible discharge of activity to the atmosphere via the main steam relief trains (primarily the affected steam generator). Primary coolant may be contaminated by corrosion and fission products but the activity is limited by the maximum activity level allowed by the technical specifications.

A steam line break located downstream of the main steam isolation valve combined with a tube rupture in the same steam generator might lead to a direct uncontrolled discharge of activity to the atmosphere.

The concurrent occurrence of both initiating events leads to overall plant behaviour which is identical to a steam line break event up until break isolation, and thereafter is identical to a steam generator tube rupture with an unavailable main condenser.

A steam line break with steam generator tube rupture in the same steam generator in State A is considered as a specific study in this section.

3.1.2. Identification of causes

The accident examined is the result of a postulated break on the main steam pipe downstream of the isolation valve accompanied by the rupture of one steam generator tube, with both events occurring in the same steam generator.

3.1.3. Precautions limiting accident occurrence

In order to limit the occurrence of a steam line break, piping from the steam generator outlet to the main steam isolation valve is conditioned according to the break preclusion concept (Sub-chapter 10.5):

- The main steam piping system is optimised with regard to thermal expansion stresses and thermal transients are monitored during plant operation.

- Fluid dynamics effects e.g. steam lines filling with water, or condensation in the steam lines, are prevented by the protection system, the large free volume of the steam generator steam cavity and continuous draining of the steam lines. Nevertheless, the loading due to complete filling of the steam lines with water has been taken into account.

- Loads on the steam lines outside the containment due to actuation of the main steam safety valves or relief valves have been considered in the design.
- Erosion and corrosion phenomena are prevented by stable system conditions and suitable materials.

The probability of a steam generator tube rupture is very low due to the following design features:

- The steam generator tube material is highly ductile.
- The blowdown system is located at the bottom of the SG tube bundle and is designed to prevent solid deposits on the tube plate.
- The secondary water is chemically conditioned, thereby protecting the SG tubes from corrosion phenomena.
- The steam generator support plates maintaining the tube bundle are designed (type of material, geometry of borings) to prevent denting of the tubes and, in the case of a double-ended guillotine tube rupture, to prevent the whipping phenomenon which could cause rupture of neighbouring tubes.
- The control of the activity in the secondary side steam generator water and steam allows for continuous monitoring of compliance with the established limits.
- On the mechanical side, the steam generators are designed to prevent projectiles, from the main feedwater, violently hitting one or several tubes.

The steam generator tubes are designed to withstand fluid and mechanical loads caused by a steam line break.

3.2. TYPICAL SEQUENCE OF EVENTS

The following section describes the typical sequence of events during a postulated break on the main steam pipe downstream of the isolation valve accompanied by the rupture of one steam generator tube. The tube rupture is located at the bottom of the cold side tube leg. This location is chosen to maximise the primary coolant discharge into the secondary side.

The steam line break induces a rapid depressurisation in the affected steam generator which initiates a reactor trip on “high pressure drop MAX1” signal. In response, the core power drops, the main steam lines are isolated, the feedwater full load valves close and turbine trip is activated.

After the main steam lines are isolated, the secondary pressure increases to the main steam relief train (VDA [MSRT]) setpoint pressure, the main steam relief valves open to remove the steam produced in the steam generators and the plant stabilises at hot shutdown conditions.

In the affected steam generator, the initial fall in water level is caused by the loss of inventory through the steam line break. After the leak is isolated, the liquid level increases steadily and the steam generator tube rupture (SGTR) is identified by monitoring the activity or by production of a high level signal. Later on, when the level reaches the threshold value, MAX0, the feedwater low load control valve will be closed.

In the unaffected steam generators the liquid levels are controlled by the feedwater low load control valves.

The decay heat is removed by the steam generators via the VDA [MSRT], the feedwater supply is ensured by the main and emergency feedwater systems (ARE [MFWS], ASG [EFWS]), the core is subcritical and the tube leak flow might be compensated by the chemical and volume control system (RCV [CVCS]) and/or the safety injection system (RIS [SIS]).

Thirty minutes after occurrence of the accident, at the earliest, the operator brings the plant to the final state by following the SGTR strategy (requested by automatic diagnosis). The final state is defined in the present analysis as the point at which the primary coolant system (RCP [RCS]) long term cooldown is ensured by the residual heat removal system (RIS [SIS] in RHR mode) and the release of activity is halted.

The SGTR emergency operating procedure (EOP) allows for the different phases of the transient, taking account of the actual plant conditions. Principally, the operator is instructed to carry out the following actions in order to reach the RIS [SIS] in RHR mode operating condition:

- Steam and water isolation of the affected steam generator

  With respect to the contaminated steam/water discharge the operator has to confirm the termination of the feedwater supply and the isolation of the steam line break before performing the RCP [RCS] cooldown. The VDA [MSRT] setpoint of the affected steam generator is increased to 99 bar.

- Reactor coolant system boration

  Boration is performed via the RCV [CVCS] or via the extra boration system (RBS [EBS]) if the RCV [CVCS] is not available. Since the RCV [CVCS] pumps can inject water up to the range of the RCP [RCS] design pressure, the charging injection will be isolated to avoid over-pressure in the primary and secondary systems (by the pressuriser or steam generator level high signal).

  The safety injection pumps (MHSI, LHSI) and accumulators also deliver borated water into the RCP [RCS] at low pressure levels.

- Reactor coolant system cooldown

  The reactor coolant system cooldown is performed via the unaffected steam generators by reducing the VDA [MSRT] pressure setpoint, as long as this is not limited by the VDA [MSRT] steam flow capacity.

- Reactor coolant system depressurisation

  At end of the RCP [RCS] cooldown, if the primary system pressure is still high, the operator depressurises the RCP [RCS] by activation of the pressuriser normal or auxiliary spray.

3.3. ACCEPTANCE CRITERIA

The acceptance criteria for this study are the radiological limits given in Sub-chapter 14.6. For the present analysis, the following relevant criteria also have to be met:

- It must be demonstrated that the plant can be brought to a final state (as defined in section 3.2 above).

- Radiological releases must be below those of PCC-4 events.

3.4. METHOD OF ANALYSIS

The analysis is performed using the CATHARE V2.5 computer code described in Appendix 14A.

This code is an advanced, two-fluid, thermal-hydraulic computer code designed for use in realistic studies of accident thermal-hydraulics in pressurised water reactors. It provides a detailed representation of the primary and secondary systems.

3.4.1. Principle of the selected method

The reactor pressure vessel and the four main coolant loops with steam generators are simulated separately. The pressuriser is connected to the hot leg of loop 3; the main steam line break and the steam generator tube rupture are located in loop 3.

The reactor core is modelled by one average channel and a conservative decay heat curve is applied to model the decay power history after reactor trip. On the secondary side, the main steam lines are equipped with the control and isolation valves. The main steam header, turbine and bypass station are also represented.

Plant auxiliary systems such as the feedwater systems, chemical and volume control system, pressuriser system, RIS [SIS], etc, are not geometrically simulated but their functional characteristics are known for the range of conditions modelled.

3.4.2. Choice of single failure and preventive maintenance

All plant systems not affected by the initiating event are assumed to be operating normally. The preventive maintenance and single failure assumptions are not taken into account for this specific study.

3.4.3. Initial state

For this specific study the nominal values of plant data are generally used, as documented in Section 16.4.3 - Table 1, which shows the relevant initial and boundary conditions applied in the analysis.

3.4.4. Specific assumptions

In the case of a MSLB + SGTR event, the automatic and operator actions are aimed at tripping the reactor, removing the residual heat, isolating the affected steam generator (both the feedwater supply and the steam line), limiting the contaminated water mass released to the atmosphere and bringing the plant to the final state.

3.4.4.1. Neutronic data

The initial state is at 102% nominal power. A conservative decay heat curve is used to represent the residual power history following the reactor trip.

3.4.4.2. Protection and mitigation actions

Safety Systems

- Reactor protection and safeguard signals

The reactor protection signals documented in Sub-chapter 14.1 are calculated as part of the transient input. In the calculation the reactor is tripped on high secondary pressure drop MAX1 (-2 bar/min) and the safety injection signal is activated on low pressuriser pressure MIN3 (115 bar).

- Safety injection system

The four RIS [SIS] trains are simulated. The maximum delivery curves are applied for medium and low head safety injection pumps. The accumulators are assumed to be at low initial pressure and minimum liquid volume.

- Main steam relief train and pressure control

In the case of main steam line break, the GCT [MSB] is no longer available following closure of the main steam isolation valves. The primary heat transport and secondary pressure are controlled by the main steam relief trains.

Other Systems

- Pressuriser heaters and spray

The pressuriser heaters and normal spray are taken into account. Total heating power is 2592 kW. Two suction lines withdraw water from the cold leg of loops 2 and 3; the spray rate is 23 kg/s (minimum value). The auxiliary spray mode from the RCV [CVCS] is not considered, pressuriser pressure control is credited.

- Chemical and Volume Control System

A dedicated pressuriser level control is modelled. The high pressure letdown line withdraws coolant from the cross-over leg of loop 1. In normal operating conditions at 155 bar, one charging pump injects 10 kg/s of demineralised water at approximately 270°C into the cold leg of loops 2 and 4.

Other Assumptions

- Steam generator blowdown line

To avoid the steam generator overfeeding in the case of SGTR, the dedicated blowdown line may be opened by the operator at high liquid level. The discharge flow capacity is approximately 2% of the nominal feedwater flow.

3.5. RESULTS

The sequence of events is summarised in Section 16.4.3 - Table 2, abbreviations used in the graphs are explained in Section 16.4.3 - Table 3 and transient analytical results are presented in Section 16.4.3 - Figures 1 to 18.

Immediately after break occurrence, the main steam line pressure in the affected loop drops quickly which initiates a reactor trip on “high pressure drop MAX1” signal. In response to the reactor trip, turbine trip, control rod drop and main steam line isolation are activated automatically.

About 7 seconds after accident initiation, the four main steam isolation valves close completely, the leak discharge is terminated and the secondary pressure increases up to the VDA [MSRT] setpoint pressure. At around 59 seconds the VDA [MSRT] control valves in the unaffected main steam lines open to remove the heat from the core (around 131 seconds in the affected main steam line).

In the primary system, the loss of primary coolant and the fall in core power cause the pressure and liquid level to decrease in the pressuriser which initiates the start-up of the second RCV [CVCS] pump. At high pressure levels the injection capacity of two charging pumps cannot compensate for the tube rupture leakage, the liquid level continues to fall and the letdown line is isolated.

At about 1035 seconds the liquid level falls below 12% MR and the heaters are switched off. After this time the primary and secondary system pressures and temperatures stabilise at hot shutdown conditions.

At 1800 seconds after accident initiation, the operator is required to bring the plant to the final state by following the SGTR strategy. According to the system pressure and coolant temperature conditions at the end of the automatic phase, the operator performs the following:

- Activate the permissive signal P12: the safety injection system (RIS [SIS]) signal is switched over from the pressuriser pressure low (MIN3) signal to the saturated pressure margin (DPSAT) signal.

- Line-up the charging pump suction line to the IRWST, switch-on two RBS [EBS] pumps.

- Start-up the RCP [RCS] cooldown at a rate of -50°C/hour by reducing the VDA [MSRT] setpoint of the unaffected steam generators.

On the primary side the coolant temperatures begin to decrease in line with the secondary side cooldown gradient. During the RCP [RCS] cooldown, the evaporation of saturated water in the pressuriser (and later on in the vessel head after draining of the pressuriser) affects the pressure gradient: the saturation margin increases.

At 3300 seconds the saturation safety margin exceeds 50°C. The operator actuates the normal spray to keep the margin within the range defined in (P, T) diagram, which causes the fast decrease of RCP [RCS] pressure at this time.

In the affected SG, the liquid level increases continuously. Following the ‘level > 85% NR’ signal, at around 4790 seconds, the operator opens the blowdown line to reduce the inventory. Related to this signal the RCV [CVCS] charging line will be isolated at the end of the cooldown phase (the pressure setpoint of the unaffected SG is approximately 2 bar).

At around 5830 seconds the primary pressure falls below 101 bar and is lower than the affected steam generator pressure + 6 bar, the operator stops the depressurisation by switching off the normal spray. Following the EOP instructions, the operator manages the RCV [CVCS] flow rate in order to balance the RCP [RCS] cooldown and the two in service RBS [EBS] pumps.

At 10,960 seconds the core outlet temperature is less than 180°C, the primary pressure is at approximately 90 bar, system depressurisation is initiated by the operator in order to reach the RIS [SIS] in RHR mode operating conditions by performing the following:

- Stop the extra boration system (RBS [EBS]) injection.
- Start the depressurisation of the affected steam generator by stepwise lowering of the VDA [MSRT] setpoint pressure until the pressuriser pressure reaches 27 bar.
- Adjust the primary pressure within a range of 6 bar above the affected SG pressure using the normal spray.

At 11,060 seconds the affected SG level reaches the nominal value. The operator closes the blowdown line. The activity releases in steam and water phase are halted.

At around 13,000 seconds the unaffected SG setpoint pressure reaches 2.5 bar, the RCP [RCS] cooldown is ended, the charging line is isolated and the SGTR leakage is terminated. The RCP [RCS] pressure stabilises at 28 bar and the coolant temperature at 160°C, the RIS [SIS] in RHR mode operating conditions are reached. The operator may switch-over to the long term cooldown mode with the RIS [SIS] in RHR mode to bring the plant to the final state.

The primary releases to atmosphere are approximately 15.45 tonnes as steam and 39 kg as liquid.

3.6. CONCLUSION

The present analysis demonstrates that, in the case of main steam line break with SGTR in the affected SG, the plant can be managed to the final state without violating the acceptance criteria.

With the automatic countermeasures and dedicated SGTR strategy applied, the operator can bring the plant to a final state and the releases of primary coolant to atmosphere are limited.

The core subcriticality is ensured due to a large amount of borated water from the chemical and volume control system (RCV [CVCS]), extra boration system (RBS [EBS]) and safety injection systems (RIS [SIS]).

SECTION 16.4.3 – TABLE 1

Plant initial and boundary conditions

PARAMETER	VALUE
REACTOR COOLANT SYSTEM	
Nominal core power (MWth)	4590
Initial reactor power (% of Nominal Power)	102
Reactor coolant system mean temperature (°C)	312.4
T/H Reactor loop flow (kg/s)	22,110
PRESSURISER SYSTEM	
Initial pressuriser pressure (bar)	155
Pressuriser water volume / level (% MR)	56
Pressuriser safety valve setpoints (bar)	175 / 178 / 181
STEAM GENERATORS (WITH FOULING AND PLUGGING)	
Initial steam pressure (bar)	74.9
Initial Steam Generator level (% NR)	49
FEEDWATER SYSTEM	
Main feedwater flow (kg/s)	651
Initial main feedwater temperature (°C)	230
MAIN STEAM RELIEF TRAIN	
Steam flow rate per train (t/h) at 100 bar abs (55% nominal steam flow)	1291
EXTRA BORATION SYSTEM	
RBS [EBS] flow per loop (kg/s)	1.4

SECTION 16.4.3 - TABLE 2

Sequence of events

TIME (s)	EVENT
	AUTOMATIC PHASE
0	Accident initiation: MSLB + SGTR (one tube)
~1	Reactor trip actuated on high pressure drop > MIN1 in affected SG3.
~1-1.5	Responses related to RT: → Start of main steam line isolation → Start of rod drop → Start of feedwater full load valves close → Turbine trip
~1.5	Turbine valve completely closed.
~7	Main steam line isolation valves completely closed.
~16	Main feed water full load valve close.
57-59	VDA [MSRT] control valves of unaffected SG1, SG2, SG4 open.
131	VDA [MSRT] control valve of affected SG3 opens.
1035	Pressuriser liquid level < 12% MR High pressure letdown line isolated, cut off heaters.
	OPERATOR ACTIONS: BRINGING THE PLANT TO FINAL STATE
1800	Start of manual actions to bring the plant to final state following SGTR strategy: - Isolation the affected SG3 in steam and water. - Activate permissive P12 and start up 2 RBS pumps. - Increase VDA [MSRT] setpoint of affected SG3 to 99 bar.
1860	- Start of RCP [RCS] cooldown at rate of -50°C/h by reducing the VDA [MSRT] setpoint of the unaffected SG 1,2 and 4.
3300	Hot leg saturation margin > 50°C, operator activates the normal spray to adjust the saturation margin between within the range defined in the (P, T) diagram.
4789	Liquid level > MAX2, operator open the blowdown line to reduce the water inventory in the affected SG3. - Charging line will be isolated at end of cooldown phase
5827	Pressuriser pressure is ~95bar and is less than the affected SG3 pressure + 6bar, the operator stop to adjust the primary pressure (normal spray is stopped) and ongoing the RCS cooldown.

TIME (s)	EVENT
10,961	Core outlet temperature sink below 180°C, pressuriser pressure ~80 bar, the operator performs the RCP [RCS] depressurisation to meet the RIS [SIS] in RHR mode operating conditions: - reduce the VDA [MSRT] setpoint of the affected steam generator until the RCP [RCS] pressure reaches ~30 bar (stepwise down to 20 bar in the calculation) - activate the normal spray to adjust the primary pressure gradient within the range of affected SG3 pressure + 6 bar
11,011	Pressuriser pressure < 70 bar, saturation margin > 10°C: operator closes the accumulators.
11,060	Liquid level in the affected SG3 < nominal level setpoint, operator closes the blowdown line. Activity releases in steam and water phase are definitely cancelled.
~13,000	Pressuriser pressure reaches 28 bar and RCP [RCS] coolant temperature is ~160°C: the RIS [SIS] in RHR mode operating conditions are met. - end of cooldown: unaffected SG pressure is ~2.5 bar, charging line isolated, SGTR leakage is cancelled
>13,000	The operator isolates the charging line and switches over to the long term cooling mode with the RIS [SIS] in RHR mode (not simulated in the calculation). The final state is reached
15,000	End of analysis

SECTION 16.4.3 - TABLE 3

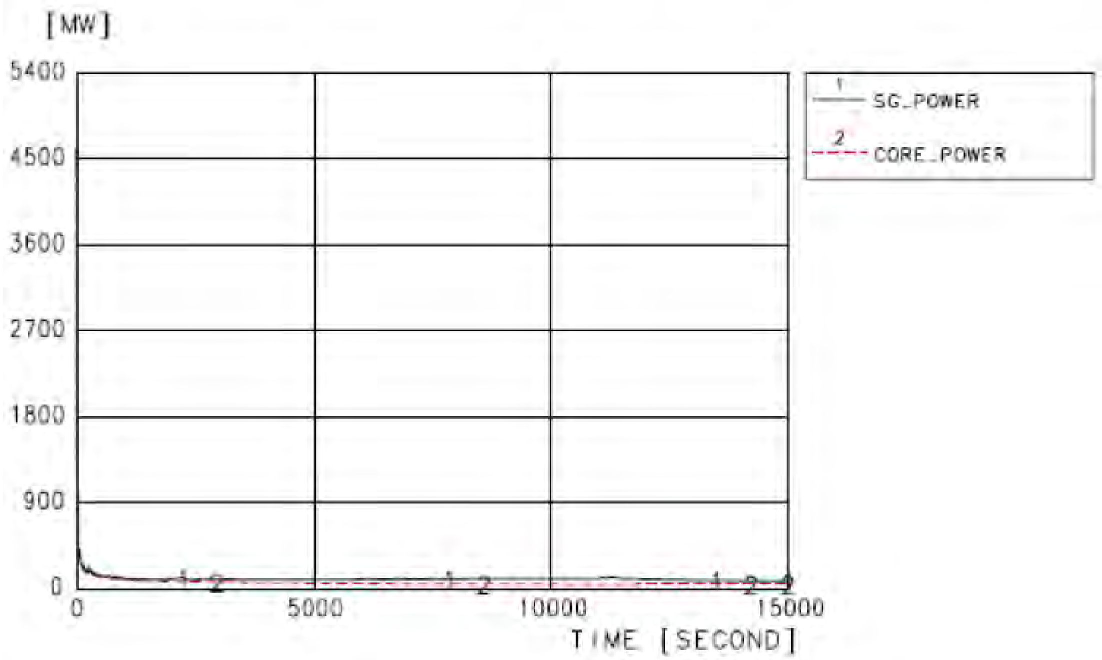
Explanation of graphic abbreviations

Figure No.	Titles and Description of Abbreviations
1	**Core Heat Release and Steam Generator Output** SG_POWER: total heat transfer rate in steam generator CORE_POWER: reactor core power
2	**Primary System Pressures** PRESSURISER : pressuriser steam dome COLDLEG1: cold leg loop 1 at vessel inlet COLDLEG2 : cold leg loop 2 at vessel inlet
3	**Coolant Temperatures in Hot Legs** HOTLEG1: hot leg loop 1 at vessel outlet HOTLEG2 : hot leg loop 2 at vessel outlet HOTLEG3: hot leg loop 3 at vessel outlet HOTLEGR : hot leg loop 4 at vessel outlet
4	**Coolant Temperatures in Cold Legs** COLDLEG1: cold leg loop 1 at vessel inlet COLDLEG2 : cold leg loop 2 at vessel inlet COLDLEG3: cold leg loop 3 at vessel inlet COLDLEGR : cold leg loop 4 at vessel inlet TCHARGING : charging injection temperature
5	**Coolant Temperatures in Reactor Pressure Vessel** PRESSU: liquid temperature in the pressuriser UPHEAD: liquid temperature in the vessel head UPLENUM: liquid temperature in the upper plenum
6	**Reactor Coolant System Liquid Levels** UPLENUM: liquid level in the upper plenum
7	**Cold Leg Loop Mass Flows** COLDLEG1: cold leg loop 1 at vessel inlet COLDLEG2 : cold leg loop 2 at vessel inlet COLDLEG3: cold leg loop 3 at vessel inlet COLDLEGR : cold leg loop 4 at vessel inlet
8	**Pressuriser Spray Rates** SUCK2: spray suction line loop 2 SUCK3: spray suction line loop 3 SPRAY2: spray charging line loop 2 SPRAY3: spray charging line loop 3
9	**Letdown and Charging Flow rates** LETDOWN: high pressure letdown line CHARGING: charging injection line

Figure No.	Titles and Description of Abbreviations
10	**Primary and Secondary Pressures** PZR: Pressuriser CAVITE1: Steam pressure of the unaffected SG loop 1 CAVITE3: Steam pressure of the affected SG loop 3 BARILET: Main steam header pressure
11	**Secondary threshold pressures** SET_SG1: VDA [MSRT] pressure setpoint unaffected loop 1 SET_SG2: VDA [MSRT] pressure setpoint unaffected loop 2 SET_SG3: VDA [MSRT] pressure setpoint affected loop 3 SET_SGR: VDA [MSRT] pressure setpoint unaffected loop 4 SET_MSB: GCT [MSB] pressure setpoint BARILET: MS header pressure (Break)
12	**Main Steam Relief Train Flow Rates** MSRT_SG1: Total mass flow rate VDA [MSRT] loop 1 MSRT_SG2: Total mass flow rate VDA [MSRT] loop 2 MSRT_SG3: Total mass flow rate VDA [MSRT] loop 3 MSRT_SGR: Total mass flow rate VDA [MSRT] loop 4
13	**Steam Generator Narrow Range Level** SG1: Narrow range level of unaffected SG loop 1 SG2: Narrow range level of unaffected SG loop 2 SG3: Narrow range level of affected SG loop 3 SGR: Narrow range level of unaffected SG loop 4
14	**Steam Generator Tube Rupture Mass Flow Rates** Total F: Leak discharge rates from the upstream side Total C: Leak discharge rates from the downstream side Total: Total leak discharge rate (Total F+Total C)
15	**Hot Leg Saturation Safety Margin** DT_SAT1: Saturation margin hot leg loop 1 DT_SAT2: Saturation margin hot leg loop 2 DT_SAT3: Saturation margin hot leg loop 3 DT_SATR: Saturation margin hot leg loop 4
16	**Blowdown Line Flow Rates** TOTAL: mass flow rate in the blowdown line SG3
17	**Primary Release as Steam** STEAM RELEASE: Primary release as steam
18	**Primary Release as Liquid** LIQUID RELEASE: Primary release as liquid

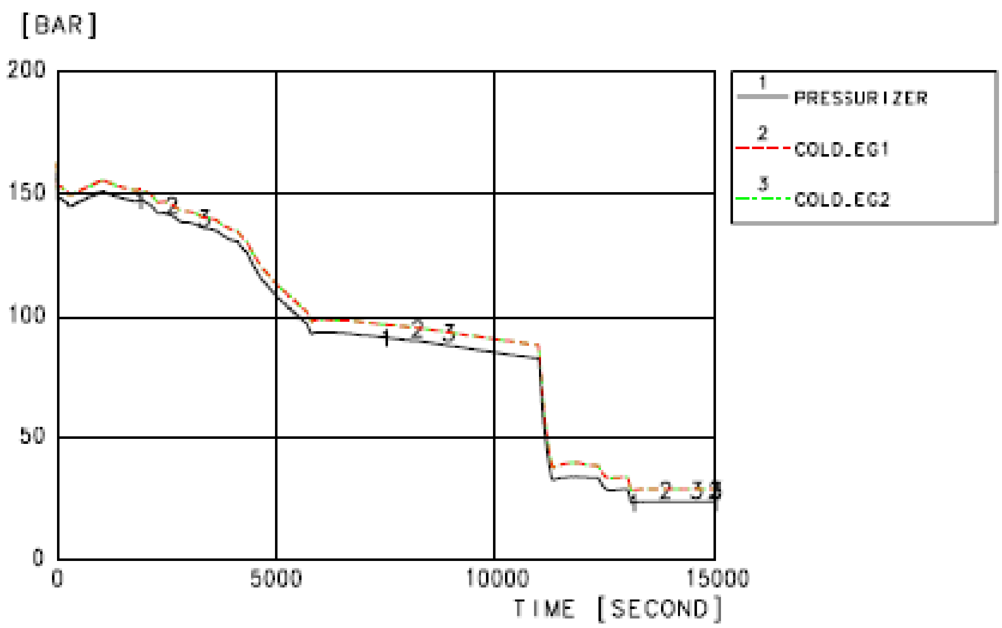
SECTION 16.4.3 - FIGURE 1

Core heat release and steam generator output



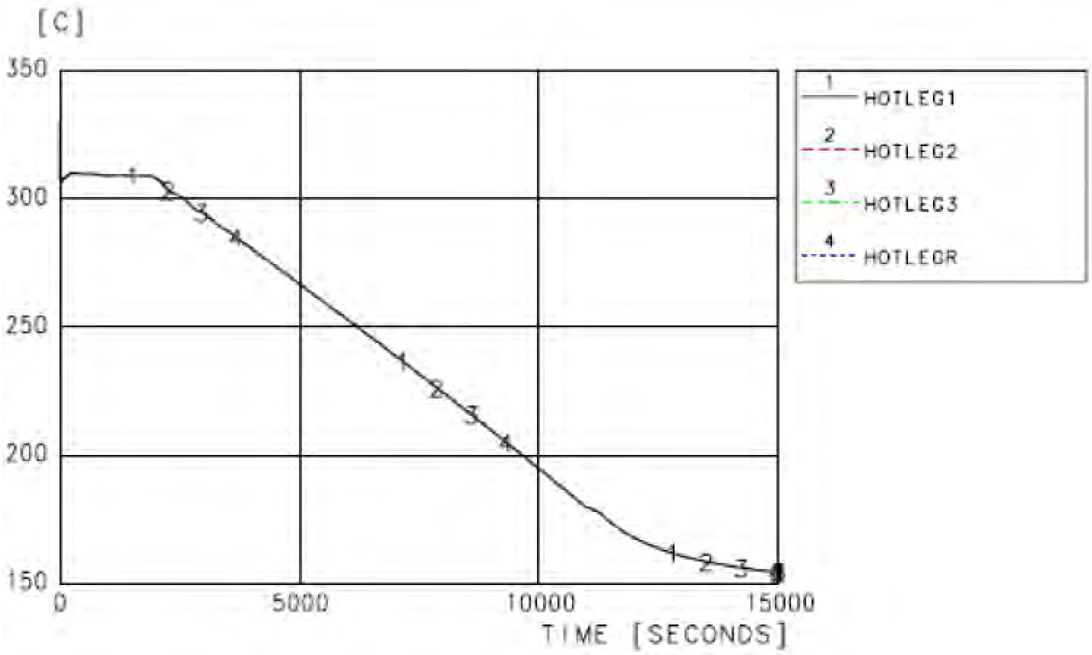
SECTION 16.4.3 - FIGURE 2

Primary system pressures



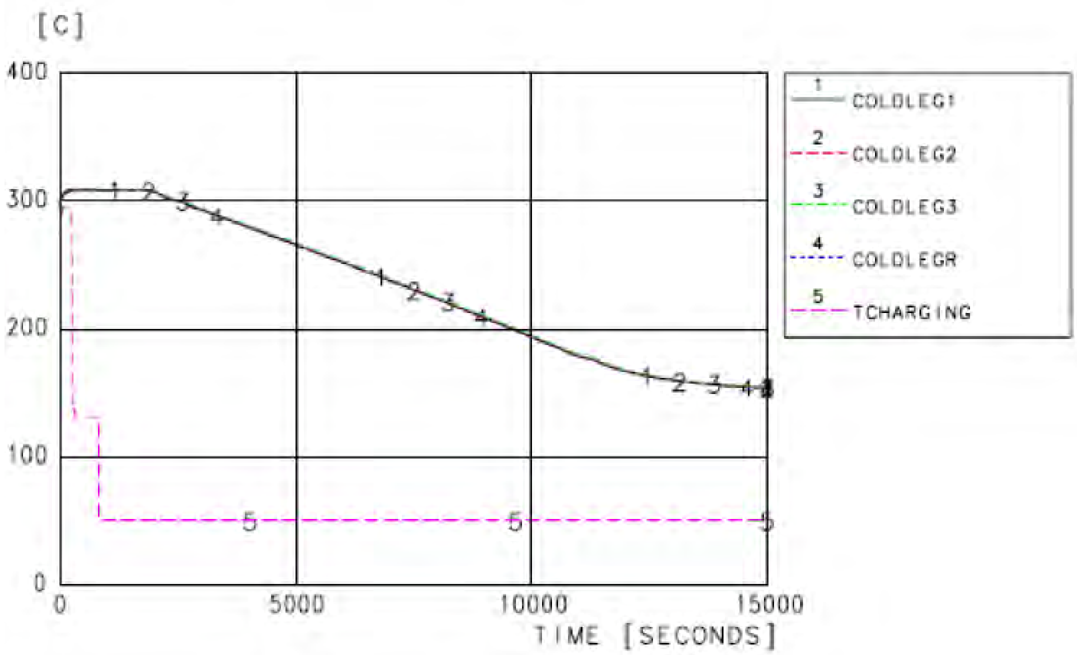
SECTION 16.4.3 - FIGURE 3

Coolant temperature in hot legs



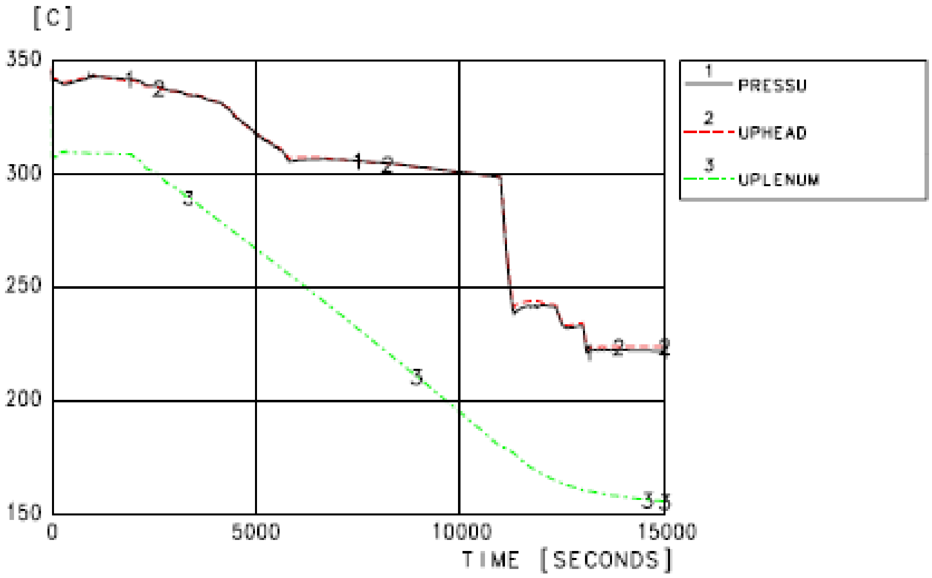
SECTION 16.4.3 - FIGURE 4

Coolant temperature in cold legs



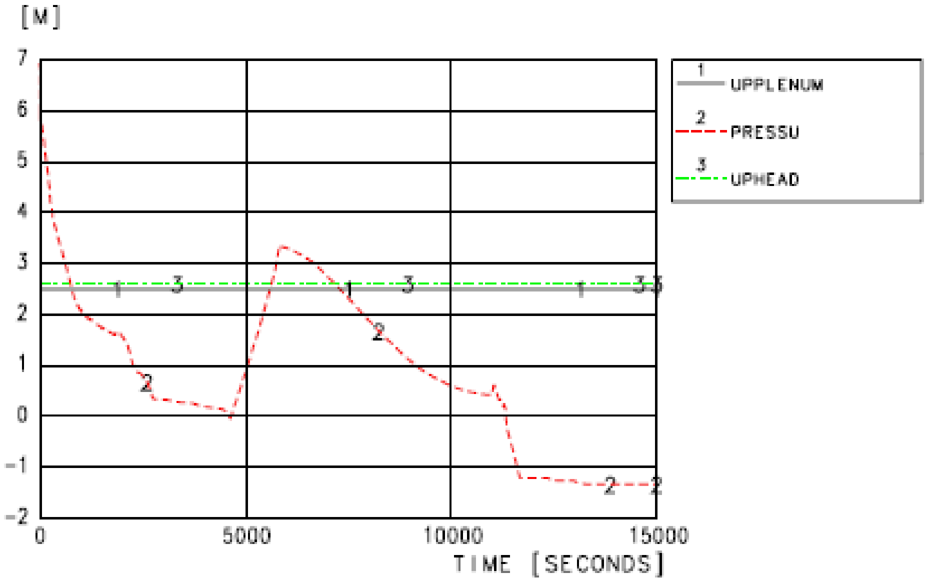
SECTION 16.4.3 - FIGURE 5

Coolant temperatures in reactor pressure vessel



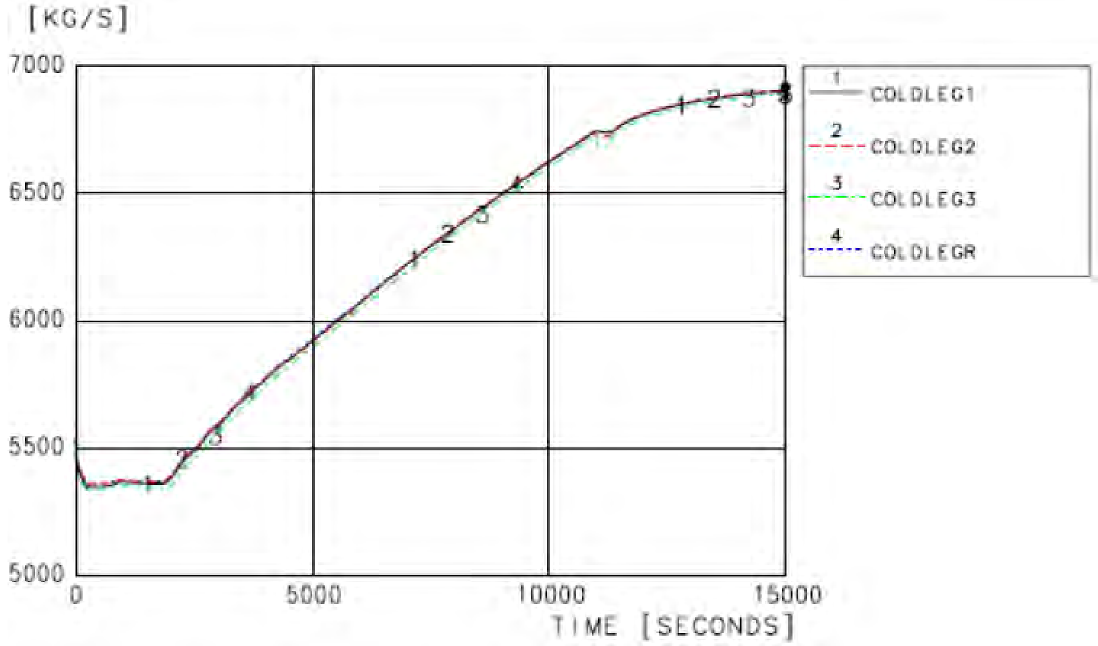
SECTION 16.4.3 - FIGURE 6

Reactor coolant system liquid levels



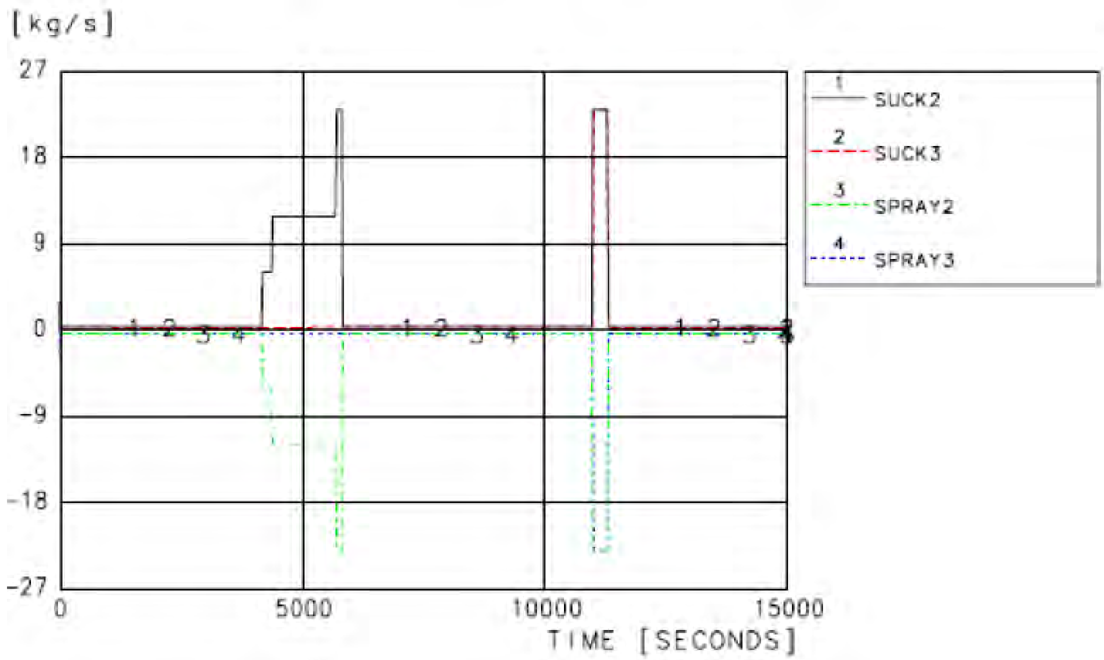
SECTION 16.4.3 - FIGURE 7

Cold leg loop mass flows



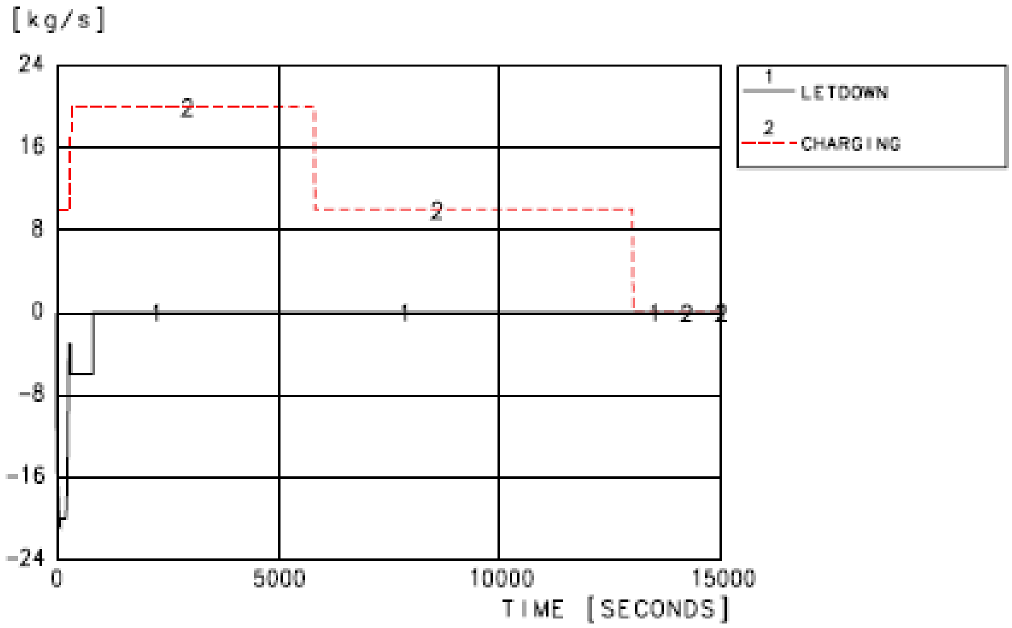
SECTION 16.4.3 - FIGURE 8

Pressuriser spray rates



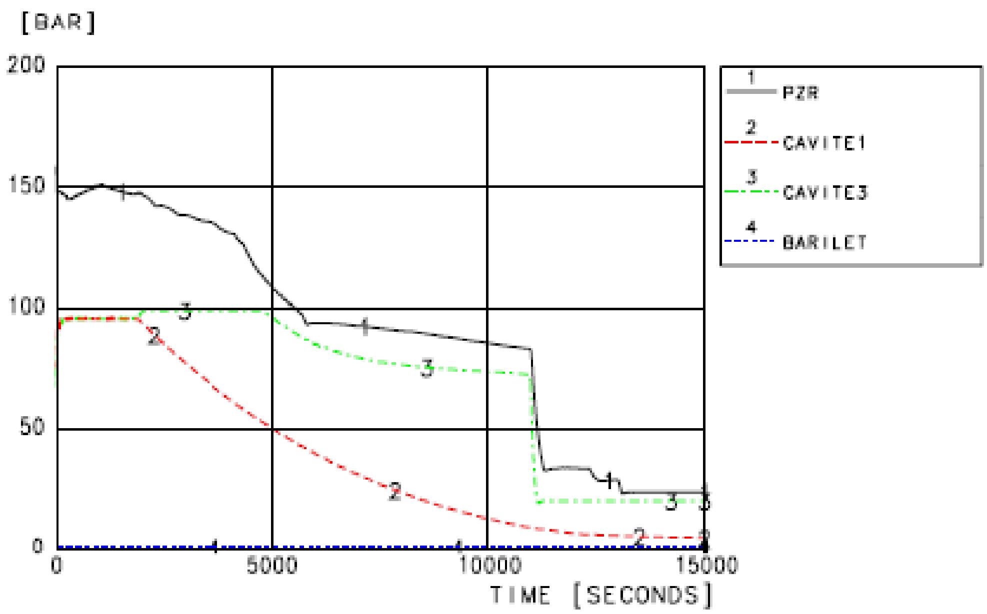
SECTION 16.4.3 - FIGURE 9

Letdown and charging flow rates



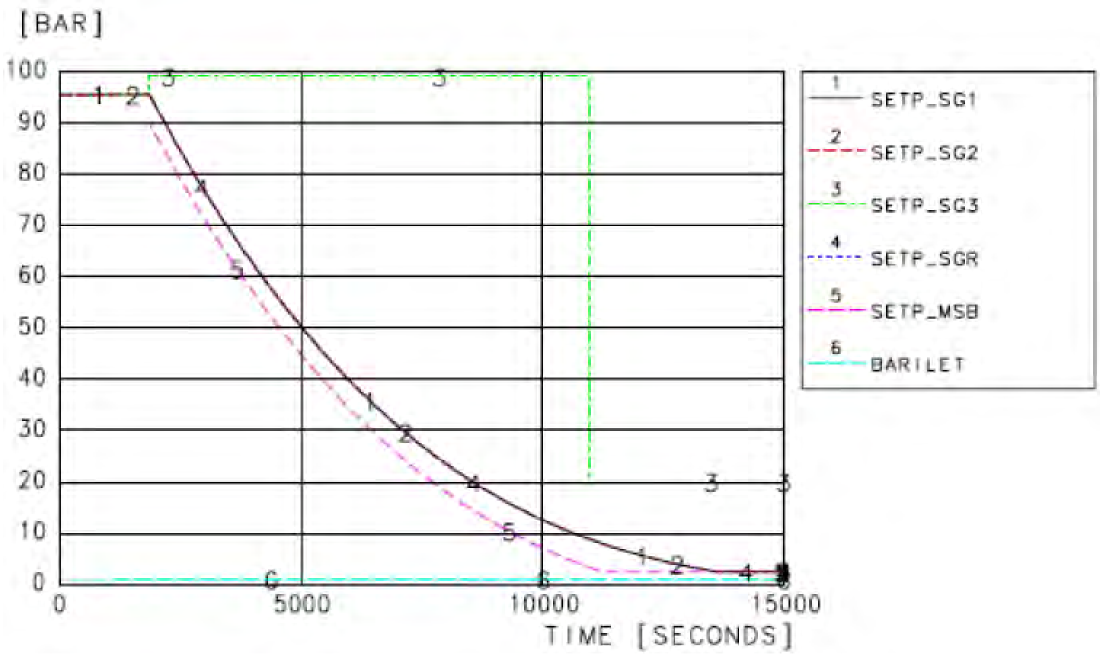
SECTION 16.4.3 - FIGURE 10

Primary and secondary pressures



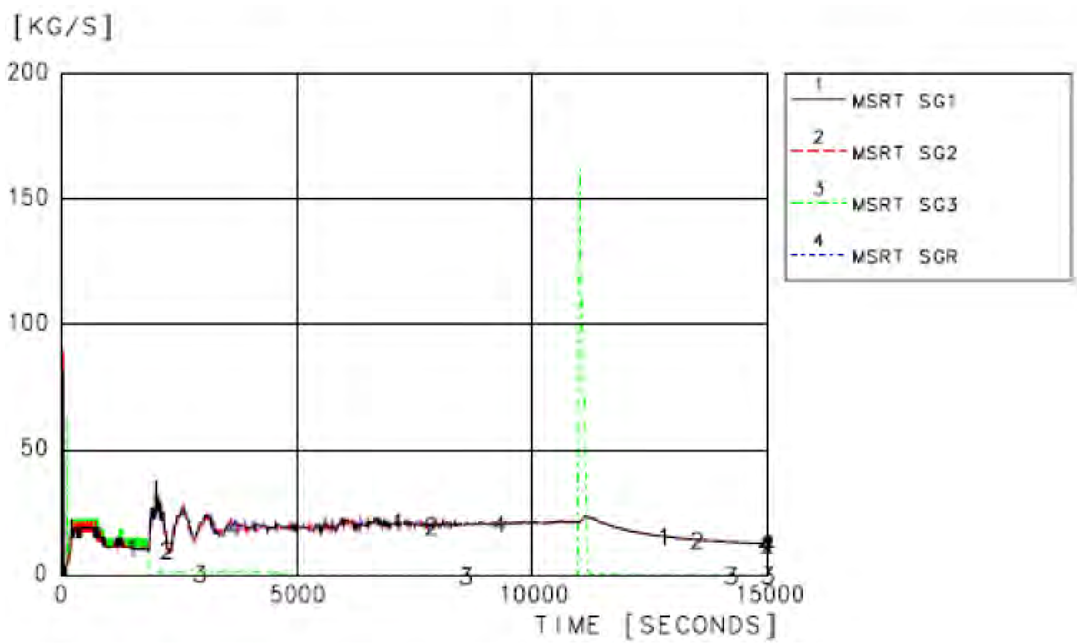
SECTION 16.4.3 - FIGURE 11

Secondary threshold pressures



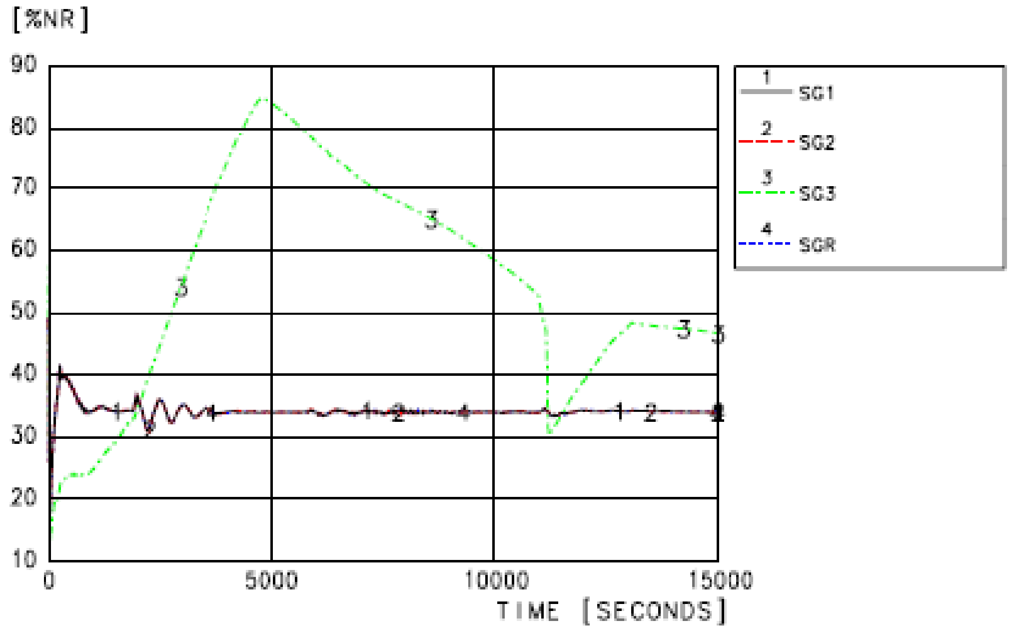
SECTION 16.4.3 - FIGURE 12

Main steam relief train flow rates



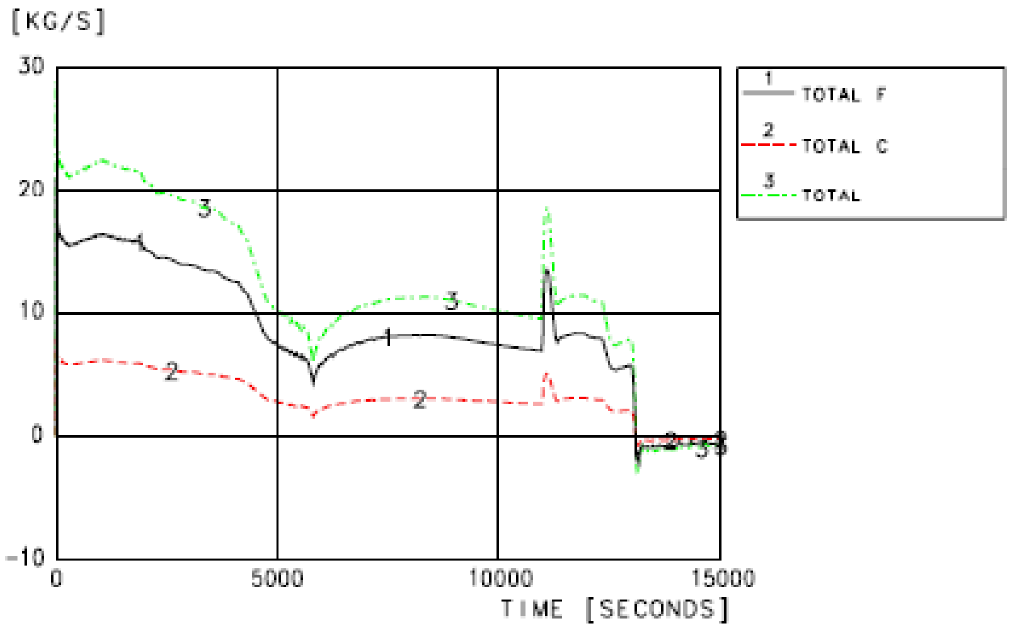
SECTION 16.4.3 - FIGURE 13

Steam generator narrow range level



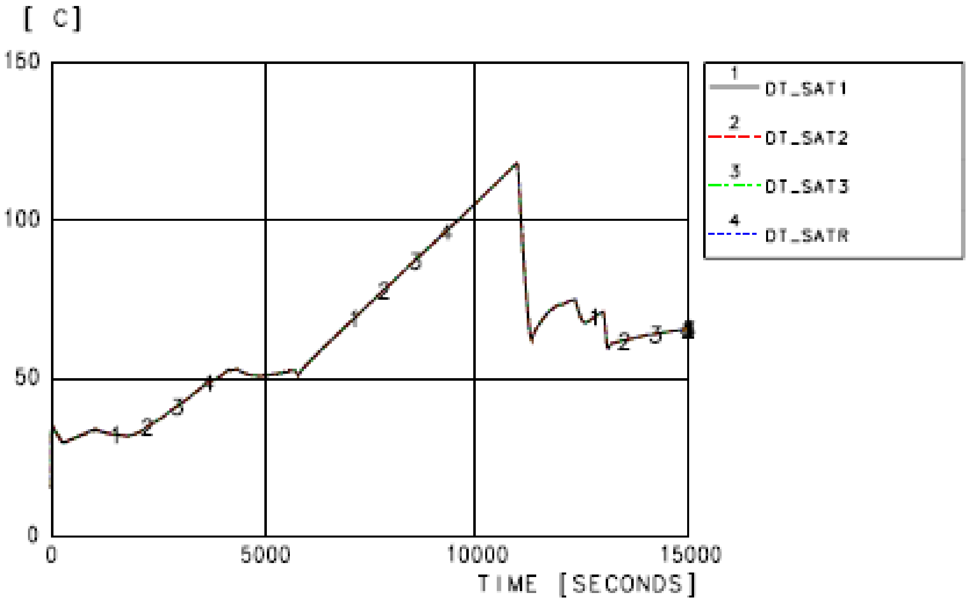
SECTION 16.4.3 - FIGURE 14

Steam generator tube rupture leak flow rates



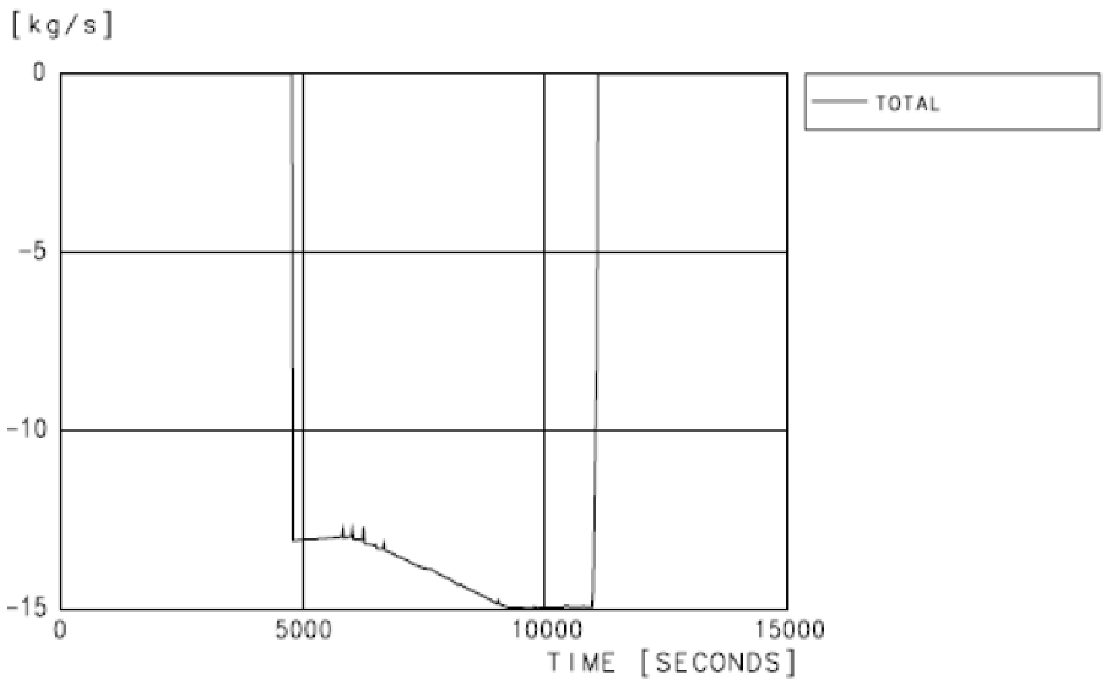
SECTION 16.4.3 - FIGURE 15

Saturation temperature safety margin



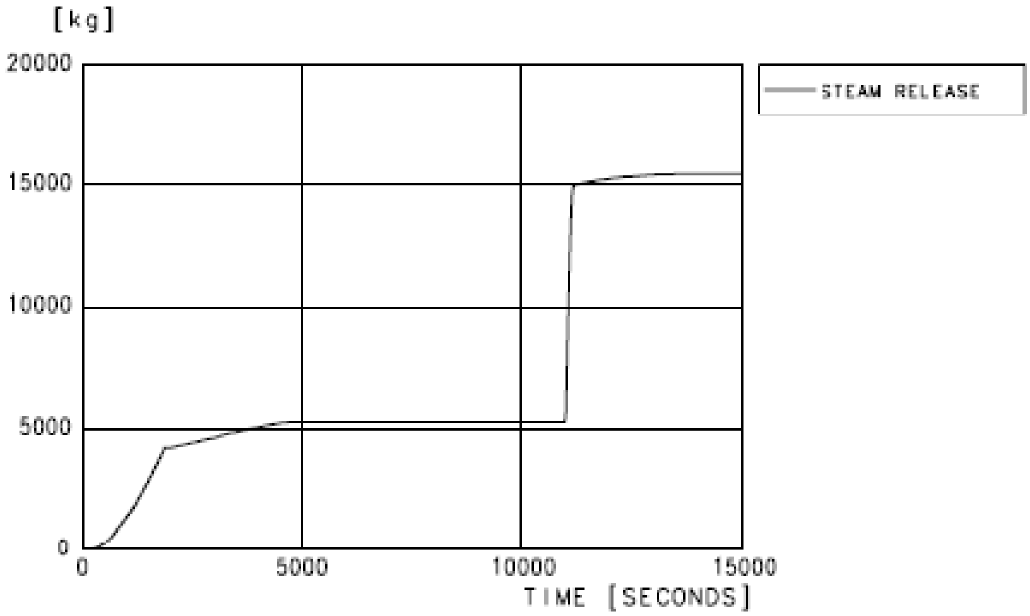
SECTION 16.4.3 - FIGURE 16

Blowdown line flow rates



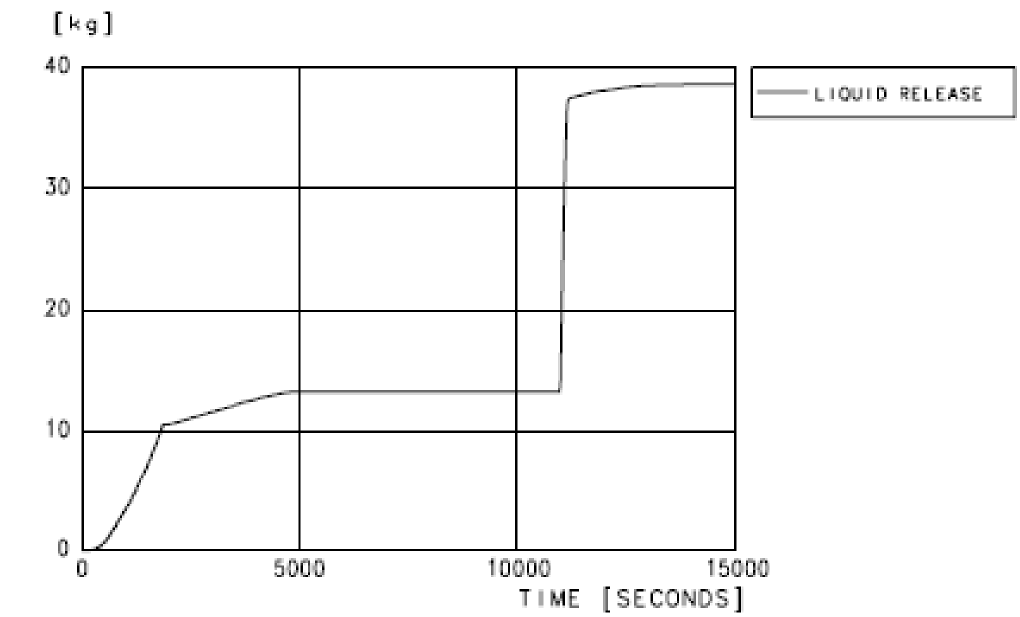
SECTION 16.4.3 - FIGURE 17

Primary release as steam



SECTION 16.4.3 - FIGURE 18

Primary release as liquid



4. STEAM GENERATOR TUBE RUPTURE (1 TUBE) WITH MAIN STEAM RELIEF TRAIN STUCK OPEN

4.1. INTRODUCTION

4.1.1. Accident description

The steam generator tube rupture (SGTR) leads to a loss of primary coolant, which is transferred to the steam generator affected by the rupture. The break induces a decrease in the primary pressure and contamination of the secondary side due to the tube leakage flow rate. The main consequence is the possible discharge of activity to the atmosphere via the main steam relief trains VDA [MSRT] (primarily the affected steam generator VDA [MSRT]).

Primary coolant may be contaminated by corrosion and fission products but the activity is limited by the maximum activity level allowed by the technical specifications.

A stuck open main steam relief train leads to increased heat removal from the core by the secondary side. This causes an overcooling event, principally leading, because of the negative moderator reactivity feedback, to a reactor power rise. It might, therefore, lead to fuel and cladding failure.

A stuck open main steam relief train combined with a steam generator tube rupture in the affected steam generator results in a non-isolatable flow path between the reactor coolant system and the atmosphere, and leads to an uncontrolled discharge of activity to the atmosphere until complete depressurisation of the reactor coolant system to atmospheric conditions occurs, in the worst case scenario.

A steam generator tube rupture (1 tube) combined with a stuck open main steam relief train in the affected steam generator in State A is considered as a specific study in this section.

4.1.2. Identification of causes

The scenario examined is the result of a tube rupture, as the initiating event, with two additional failures in reactor protection actions. In addition to the initiating event, the failures of both the following in the affected SG are assumed:

- the main steam relief control valve to close on demand
- the automatic and redundant closure of the main steam relief isolation valve.

In addition, the main steam bypass (GCT [MSB]) is postulated to be unavailable so that the main steam relief trains (VDA [MSRT]) are required to open.

4.1.3. Precautions limiting accident occurrence

The probability of a steam generator tube rupture is very low due to the following design features:

- The steam generator tube material is highly ductile.
- The blowdown system is located at the bottom of the SG tube bundle and is designed to prevent solid deposits on the tube plate.

- The secondary water is chemically conditioned, thereby protecting the SG tubes from corrosion phenomena.

- The steam generator support plates maintaining the tube bundle are designed (type of material, geometry of borings) to prevent denting of the tubes and, in the case of a double-ended guillotine tube rupture, to prevent the whipping phenomenon which could cause rupture of neighbouring tubes.

- The control of the activity in the secondary side steam generator water and steam allows for continuous monitoring of compliance with the established limits.

- On the mechanical side, the steam generators are designed to prevent projectiles, from the main feedwater, violently hitting one or several tubes.

The main steam relief train isolation valves are pilot-operated. The arrangement of pilot valves (two redundant control lines with two pilot valves in series on each line) reduces the risk of a failure to close.

4.2. TYPICAL SEQUENCE OF EVENTS

The following section describes the typical sequence of events during a double-ended guillotine rupture of one tube in one steam generator, which allows unimpeded blowdown via both ends of the severed tube. The tube rupture is located at bottom of the cold side tube leg. This location is chosen to maximise the primary coolant discharge into the secondary side.

As an additional failure, the main steam bypass is assumed to be unavailable and the main steam relief train of the steam generator affected by the tube rupture remains fully open after the opening (on demand) of the relief train valves.

In principle, the steam generator tube rupture is identified by monitoring of the high activity signal in the affected main steam line. The plant continues to operate at full power. In the affected SG, the increase in the liquid level due to the leak flow rate is compensated by the main feedwater control valve. The decrease in the pressuriser liquid level is compensated by the Chemical and Volume Control System RCV [CVCS]. At 1800 seconds after the occurrence of the accident, the operator performs a manual reactor trip which initiates the following:

- Reactor and turbine trip.

- Main feedwater full load line isolation.

As the turbine valve closes, the secondary pressure increases rapidly up to the VDA [MSRT] setpoint pressure and the VDA [MSRT] are actuated. The affected SG VDA [MSRT] valves remain stuck in the open position.

On the primary side, the pressure rapidly decreases as a consequence of the overcooling induced by the uncontrolled steam release and the coolant leak flow rate. The safety injection system is actuated on low pressuriser pressure (< MIN3) inducing the following actions:

- Partial cooldown.

- Start-up of the medium and low head safety injection pumps (RIS [SIS]).

- Opening of the accumulator isolation valves.

- Isolation of the reactor coolant pressure boundary.

All the main steam isolation valves are closed on detection of the steam generator pressure drop signal (MAX1) interrupting the uncontrolled depressurisation in the unaffected SGs. The pressure in the unaffected SGs is then controlled by the corresponding main steam relief trains, with the pressure following the pressure setpoint controlled by the partial cooldown.

If the pressure drop signal MAX2 is actuated in any SG, the dedicated main feedwater low load line is isolated and the feedwater supply is further restored by the dedicated emergency feedwater (ASG [EFWS]) supply upon reaching the reactor protection signal “SG level <MIN2”.

Due to the stuck open relief train, the pressure in the affected steam generator continues to decrease, passing the pressure limits MIN1, MIN2 and MIN3. Passing the first two limits repeats the reactor protection actions as already performed by the detection of pressure drop signals MAX1 and MAX2. Passing the third limit initiates the isolation of the appropriate VDA [MSRT], which has no effect on the secondary pressure transient (due to the assumed failure of the valves in the fully open position).

Thirty minutes after the first significant alarm, in addition to the reactor trip initiation, the operator requests the following in order to manage the plant to reach the final state:

- Reactor coolant system boration

  Boration is performed via the RCV [CVCS] or via the extra boration system (RBS [EBS]) if the RCV [CVCS] is not available. Since the RCV [CVCS] pumps can inject water up to the range of the RCP [RCS] design pressure, the charging injection will be isolated to avoid over-pressure in the primary and secondary systems (by the pressuriser or steam generator level high signal).

  The safety injection pumps (MHSI, LHSI) and accumulators also deliver borated water into the RCP [RCS] at low pressure levels.

- Reactor coolant system cooldown

  Together with the boration actions, a fast cooldown, consisting of a stepwise full-opening of the main steam relief train (VDA [MSRT]) of the unaffected steam generators, is performed.

- Reactor coolant system depressurisation

  For the depressurisation of the RCP [RCS], the accumulators must be isolated, the medium head safety injection pumps stopped and the RCV [CVCS] reactivated, once the dedicated criteria are fulfilled.

  Depressurisation is performed by the operator. The first phase involves activation of the normal pressuriser spray keeping a certain margin to saturation in the hot legs to avoid a restart of the medium head safety injection pumps. Once the RCP [RCS] pressure reaches the pressure limit for reactor coolant pump operation (> 25 bar), this first depressurisation phase is finished.

  When the coolant temperatures are sufficiently reduced by fast cooldown, the subsequent RCP [RCS] long term cooldown, via residual heat removal system (RHRS), is actuated.

Afterwards the safety injection signal limit is switched over to loop level and the main coolant pumps are successively stopped. Then the second depressurisation phase is started by the operator reducing the RCP [RCS] pressure via the pressuriser spray until leak termination is reached. Leak termination is obtained when the RCP [RCS] reaches atmospheric pressure.

To determine the radiological releases the final state is defined in the present analysis as the point at which the activity release is terminated.

4.3. ACCEPTANCE CRITERIA

The acceptance criteria for this study are the radiological limits given in Sub-chapter 14.6. For the present analysis the relevant criteria also have to be met:

- It must be demonstrated that the plant can be brought to a final state (as defined in section 4.2 above).
- Radiological releases must be below those of PCC-4 events.

4.4. METHOD OF ANALYSIS

The analysis has been performed using the CATHARE V 2.5 computer code described in Appendix 14A.

This code is an advanced, two-fluid, thermal-hydraulic computer code designed for use in realistic studies of accident thermal-hydraulics in pressurised water reactors. It provides a detailed representation of the primary and secondary systems.

4.4.1. Principle of the selected method

The reactor pressure vessel and the four main coolant loops with steam generators are simulated separately. The pressuriser is connected to the hot leg loop 3 the main steam line break and the steam generator tube rupture are located in loop 3.

The reactor core is modelled by one average channel and a conservative decay heat curve is applied to model the decay power history after reactor trip. On the secondary side, the main steam lines are equipped with the control and isolation valves. The main steam header, turbine and bypass station are also represented.

Plant auxiliary systems such as the feedwater systems, chemical and volume control system, pressuriser system, RIS [SIS], etc, are not geometrically simulated but their functional characteristics are known for the range of conditions modelled.

4.4.2. Choice of single failure and preventive maintenance

All plant systems not affected by the initiating event are assumed to be operating normally. The preventive maintenance and single failure assumptions are not taken into account for this specific study.

4.4.3. Initial state

For this specific study the nominal values of plant data are generally used as documented in Section 16.4.4 - Table 1, which shows the relevant initial and boundary conditions applied in the analysis.

4.4.4. Specific assumptions

In the case of an SGTR event, the automatic and operator actions are aimed at tripping the reactor, removing the residual heat, isolating the affected steam generator (both the feedwater supply and the steam line), limiting the contaminated water mass released to the atmosphere and bringing the plant to the final state.

4.4.4.1. Neutronic data

The initial state is at 102% nominal power. A conservative decay heat curve is used to represent the residual power history following the reactor trip.

4.4.4.2. Protection and mitigation actions

Safety Systems

- Reactor protection and safeguard signals

The reactor protection signals documented in Sub-chapter 14.1 are calculated as part of the transient input. In the calculation the reactor is assumed to be tripped manually and the safety injection signal is activated on low pressuriser pressure < MIN3 (115 bar).

- Safety injection system

The four RIS [SIS] trains are simulated. The maximum delivery curves are applied for medium and low head safety injection pumps. The accumulators are assumed to be at low initial pressure and minimum liquid volume.

- Main steam relief train and pressure control

The main steam relief trains are available (one per SG). After the main steam isolation valves close, the primary heat transport and secondary pressure are controlled by the main steam relief trains (VDA [MSRT]).

- Pressuriser safety valves

Three safety valves (PSV) with different opening pressure setpoints are available to limit the pressure increase in the reactor coolant system.

Other Systems

- Main steam bypass

The main steam bypass is assumed to be unavailable.

- Pressuriser heaters and spray

The pressuriser heaters and normal spray are taken into account. Total heating power is 2592 kW. Two suction lines withdraw water from the cold leg of loops 2 and 3; the spray rate is 23 kg/s (minimum value). The auxiliary spray mode from the RCV [CVCS] is not considered, pressuriser pressure control is credited.

- Chemical and Volume Control System

A dedicated pressuriser level control is modelled. The high pressure letdown line withdraws coolant from the cross-over leg of loop 1. In normal operating conditions at 155 bar, one charging pump injects 10 kg/s of demineralised water at approximately 270°C into the cold leg of loops 2 and 4.

4.5. RESULTS

The sequence of events is summarised in Section 16.4.4 - Table 2, abbreviations used in the graphs are explained in Section 16.4.4 - Table 3 and transient analytical results are presented in Section 16.4.4 - Figures 1 to 20.

Immediately after the steam generator tube rupture (one tube) occurs, the loss of primary coolant through the ruptured tube causes the pressure and level to decrease in the pressuriser. The increased feed to the affected steam generator is compensated for by the main feedwater control valve.

During the first 1800 seconds after initiation of the accident no reactor trip is activated. The only automatic actions are the start-up of the second charging pump to adjust the decrease in the pressuriser level and the increase in heater power for pressure control. When the pressuriser level sinks below the 12%MR limit, the charging line is isolated and the pressuriser heaters are switched off leading to a faster pressure gradient in the RCP [RCS].

Normally the operator would actuate a load reduction after the activity alarms (N16). But due to the conservative nature of the analysis, this load reduction is assumed not to be made during the first half hour.

At 1800 seconds after occurrence of the SGTR event the pressuriser is almost fully drained. At this time, it is assumed that the first operator action is to perform the manual reactor trip, which is more onerous than the load reduction requested by EOPs due to SG activity. In response, control rod (RCCA) drop, turbine trip and main feedwater full load line isolation are actuated by the F1 systems.

Due to the closure of the turbine valve, the main steam pressure increases up to the VDA [MSRT] actuation pressure of 95.5 bar (the GCT [MSB] is additionally assumed to be unavailable). When the VDAs [MSRT] are opened, the main steam relief control valve in loop 3 is then assumed to be stuck open leading to a rapid pressure decrease in the affected steam generator.

At about 1873 seconds the RCP [RCS] pressure decreases below 115 bar (MIN1), the safety injection signal is activated and therewith the start-up of partial cooldown via all VDAs [MSRT] (from 95.5 bar down to 60 bar). With respect to core subcriticality, the operator starts up the two RBS [EBS] pumps. At around 2071 seconds the pressure drop in the affected SG loop 3 exceeds the threshold MAX1, which leads to the isolation of all the main steam lines.

The operator then enters the “transition to cold shutdown in case of SGTR” to bring the plant to the final state starting with phase 1 of the EOP, “MHSI stop”.

At around 3064 seconds the partial cooldown is finished and the operator activates key P12 and therewith he can reset the SI-signal and restart the chemical and volume control system (RCV [CVCS]).

Subsequently, the operator performs the fast cooldown with the unaffected SGs down to atmospheric conditions and then switches off the three MHSI pumps (loops 1, 2 and 4). Only the MHSI pump in loop 3 remains in operation.

The fast depressurisation with the unaffected VDAs [MSRT] induces the isolation of low load feedwater injection at around 3120 seconds. As the RCP [RCS] pressure reaches 70 bar, at around 3235 seconds, the accumulators are manually closed.

At around 4103 seconds the core outlet temperature is below 180°C. Phase 1 of the EOP ends with the stopping of the last MHSI pump in operation.

The operator starts phase 4, “Transition to cold shutdown: depressurisation”, with manual termination of the RBS [EBS] pumps and start-up of normal spray to reduce the RCP [RCS] pressure. At around 4680 seconds the RCP [RCS] pressure is below 30 bar, normal spray is stopped and the operating conditions of the residual heat removal system are reached.

The operator will enter phase 5,”cooldown with RRA [RHRS] and depressurisation”, to connect the RRA [RHRS].

The RCP [RCS] pressure is stabilised above 25 bar to keep the reactor coolant pumps operational. Only the LHSI pump in loop 2 is kept in safety injection mode and the other three are switched over to residual heat removal operating mode. With the RRA [RHRS] in service the long term RCP [RCS] cooldown continues. In addition, the letdown line is re-activated to stop the increase of the pressuriser liquid level and to reach pressure equilibrium between the RCP [RCS] and the affected SG after switching off the reactor coolant pumps.

At around 11,649 seconds the core outlet temperature is below 90° C. The reactor coolant pumps are switched off, the permissive P15 is manually activated and the LHSI pump in loop 2 is stopped. As a result, the RCP [RCS] pressure decreases to atmospheric pressure so that the SG tube rupture leak flow is terminated.

The release to atmosphere is also terminated. At the end of the analysis, the primary releases to the atmosphere are approximately 54 tonnes in steam phase and 135 kg in liquid phase (residual steam moisture).

4.6. CONCLUSION

All the acceptance criteria concerning subcriticality, decay heat removal and activity release are fulfilled until isolation of the SGTR and further to RRA [RHRS] connection conditions.

With the automatic countermeasures and dedicated SGTR strategy applied, the operator can bring the plant to a final state.

The core subcriticality is ensured due to the amount of borated water from the chemical and volume control system (RCV [CVCS]), extra boration system (RBS [EBS]) and safety injection systems (RIS [SIS]).

SECTION 16.4.4 – TABLE 1

Plant initial and boundary conditions

PARAMETERS	VALUE
REACTOR COOLANT SYSTEM	
Nominal core power (MWth)	4590
Initial reactor power (% of Nominal Power)	102
Reactor coolant system mean temperature (°C)	312.4
T/H Reactor loop flow (kg/s)	22110
PRESSURISER SYSTEM	
Initial pressuriser pressure (bar)	155
Pressuriser water volume / level (% MR)	56
Pressuriser safety valve setpoints (bar)	175 / 178 / 181
STEAM GENERATORS (WITH FOULING AND PLUGGING)	
Initial steam pressure (bar)	74.9
Initial Steam Generator level (% NR)	49
FEEDWATER SYSTEM	
Main feedwater flow (kg/s)	651
Initial main feedwater temperature (°C)	230
MAIN STEAM RELIEF TRAIN	
Steam flow rate per train (t/h) at 100 bar abs (55% nominal steam flow)	1291
EXTRA BORATION SYSTEM	
RBS [EBS] flow per loop (kg/s)	1.4

SECTION 16.4.4 - TABLE 2

Sequence of events

TIME (s)	EVENT
>0.0	2A SGTR opening (1 tube)
1259.4	Pressuriser level < 49% MR, letdown flow reduces to minimum, second charging pump starts up Pressuriser level < 12% MR, heaters are cut off
1800	OPERATOR ACTIONS
1800.4	The operator performs a manual reactor trip: → Start of rod drop → Turbine trip
1801.6	→ Start of feedwater full load valves close → Turbine valve closes
1810.4	VDA [MSRT] in all loops active
1812.5	VDA [MSRT] control valves open due to SG pressure > 95.5 bar → VDA [MSRT] control valve loop 3 stuck opens at 100% position
1816.4	Main feedwater full load isolation valves are closed - Steam generator water level is controlled via the low load valve
1872.8	RIS [SIS] is actuated on low pressuriser pressure (<115 bar) → Start up the RIS [SIS] pumps (MHSI, LHSI) → Primary system isolation → Start of partial cooldown via VDA [MSRT] at rate of -100°C/h (1873.4s)
1874.4	Manual start of two RBS [EBS] pumps
1887.9	MHSI and LHSI pumps started
1897.8	RBS [EBS] pumps inject borated water into the RCP [RCS]
2070.8	SG pressure drop in loop 3 >MAX1: reactor trip RT → main steam isolation valves close in all loops (2075.8s)
3063.9	End of partial cooldown, the operator performs the following: - Activation of key P12 - Restart the RCV [CVCS] - Cut off 3 MHSI pumps, only the MHSI in loop 3 is operated - Start of fast cooldown to ambience pressure with the 3 unaffected SGs (1, 2 and 4) - Isolation of feedwater in affected SG
3120.2	Main feedwater low load line isolation in SG1, 2 and 4 due to steam generator pressure drop > MAX2
3235.3	RCP [RCS] pressure ~70bar accumulators are manually closed

TIME (s)	EVENT
4103.3	Core outlet temperature < 180°C, the operator performs the RCP depressurisation (EOP phase-4 “transition to cold shutdown: depressurisation”: - Cut off the last MHSI (loop3) - Stop the RBS [EBS] pumps - start the normal spray
4653	RCP [RCS] pressure ~30bar, the operating conditions of the RIS/RRA [SIS/RHRS] are reached, operator prepares the long term RCP [RCS] cooldown via RRA [RHRS]: - activates the permissive P14 - switches over the 3 LHSI loop 1, 3 and 4 to the RHR mode, the LHSI loop 2 remains in RIS mode - starts the RCP cooldown via RHRS
4679.7	RCP [RCS] pressure < 29 bar, RIS/RRA [SIS/RHRS] operating conditions reached
4691.2-4693.5	SG liquid level < MIN2, start up the ASG [EFWS] pumps in loop 1, 2 and 4.
4706.2-4708.6	Start of ASG [EFWS] injection in SG loop 1, 2 and 4.
9272	Manual restart of the RCV [CVCS] high pressure letdown line
11648.4	Core outlet temperature < 90°C The operator switches off all reactor coolant pumps, the loop circulation flows drop
11848.9	Operator activates the permissive P15 and switches off the LHSI in RIS [SIS] mode (loop 2).
> 15000	SGTR leak flow is nearly terminated, the final state is reached - primary system is depressurised
16000	End of analysis

SECTION 16.4.4 - TABLE 3

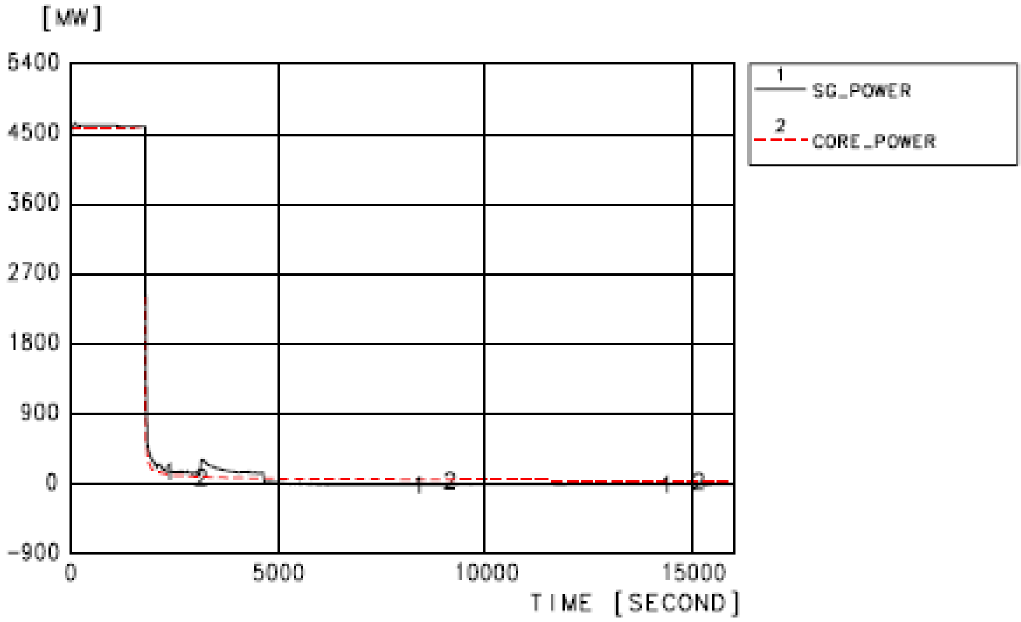
Explanation of graphic abbreviations

Figure No.	Titles and Description of Abbreviations
1	**Core Heat Release and Steam Generator Output** SG_POWER Total heat transfer rate in steam generator CORE-POWER: Reactor core power
2	**Primary System Pressures** PZR: pressuriser steam dome CL1,CL2,CL3,CLR: Cold Leg at vessel inlet loop 1/2/3/4
3	**Coolant Temperatures in Hot Legs** HL1,HL2,HL3,HLR: Hot Leg at vessel outlet loop 1/2/3/4
4	**Coolant Temperatures in Cold Legs** CL1,CL2,CL3,CLR: Cold Leg at vessel inlet loop 1/2/3/4
5	**Cladding temperature** INLET: Cladding temperature at core inlet, MIDDLE: Cladding temperature at core middle, PEAK33: Cladding temperature at the core peak
6	**Reactor Coolant System Liquid Levels** UPL: upper plenum PZR: pressuriser PZRCTRL: pressuriser measurement SURGE_LINE: surge line
7	**Cold Leg Loop Mass Flows** LOOP1, 2, 3 R Cold Leg at vessel inlet loop 1/2/3/4
8	**Pressuriser Spray and Heater Power** HEATER POWER Heated power of the heaters SPRAY2 POWER Energy of spray line loop2 SPRAY3 POWER Energy of spray line loop3
9	**Letdown and Charging Flow Rates** LETDOWN: High pressure letdown line CHARGING: Charging injection line
10	**Primary and Secondary Pressures** PZR: Pressuriser steam dome CL1, CLR: Cold Leg at vessel inlet loop 1and 4 SG1, 2, 3, R SG steam dome loop 1, 2, 3 and 4
11	**Main Steam Bypass Flow Rates** LIQUID Liquid mass flow rate STEAM Steam mass flow rate TOTAL Total mass flow rate

Figure No.	Titles and Description of Abbreviations
12	**Main Steam Relief Train Flow Rates** MSRT SG1 VDA flow rate SG loop 1 MSRT SG2 VDA flow rate SG loop 2 MSRT SG3 VDA flow rate SG loop 3 MSRT SGR VDA flow rate SG loop R
13	**Steam Generator Narrow Range Level** SG1, SG2, SG3, SGR : narrow range level of SG loop 1/2/3/4
14	**Steam Generator Wide Range Level** SG1, SG2, SG3, SGR : Wide range level of SG loop 1/2/3/4
15	**Saturation Temperature Safety Margin** DT_SAT1: Saturation margin hot leg loop 1 DT_SAT2: Saturation margin hot leg loop 2 DT_SAT3: Saturation margin hot leg loop 3 DT_SATR: Saturation margin hot leg loop 4
16	**Saturation Pressure Safety Margin** DT_SAT1: Saturation margin hot leg loop 1 DT_SAT2: Saturation margin hot leg loop 2 DT_SAT3: Saturation margin hot leg loop 3 DT_SATR: Saturation margin hot leg loop 4
17	**Steam Generator Tube Rupture Leak Flow Rates** TOTAL F: Leak discharge rate from the downstream side TOTAL C: Leak discharge rate from the upstream side TOTAL: Total leak discharge rate
18	**Blowdown Line Flow Rates** SG BLOWDOWN 3: Mass flow rate in the blowdown line SG3 TRANSFER R: Mass flow rate in the transfer line SG4 TRANSFER 3 Mass flow rate in the transfer line SG3
19	**Primary Release as Steam** STEAM RELEASE: Primary release as steam STEAM REL. GMBH (control model, not used)
20	**Primary Release as Liquid** LIQUID RELEASE: Primary release as steam LIQUID REL. GMBH (control model, not used)

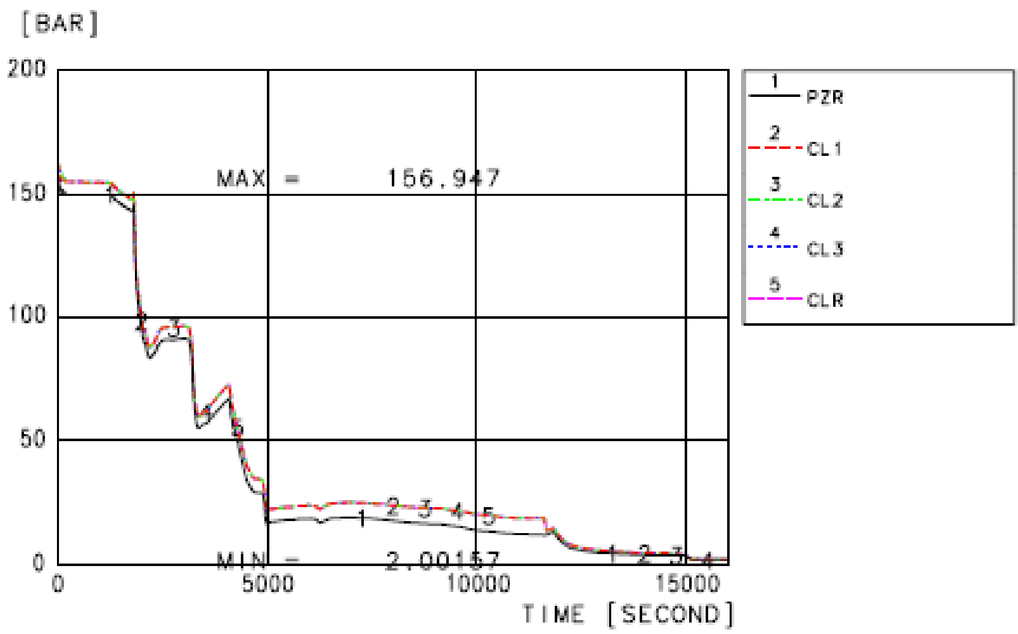
SECTION 16.4.4 - FIGURE 1

Core heat release and steam generator output



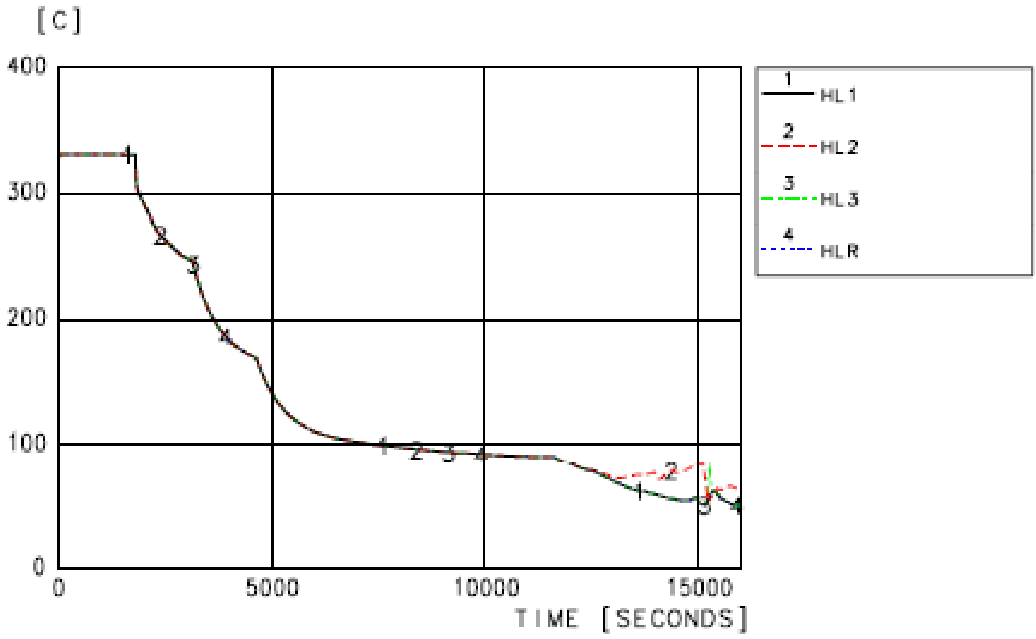
SECTION 16.4.4 - FIGURE 2

Primary system pressures



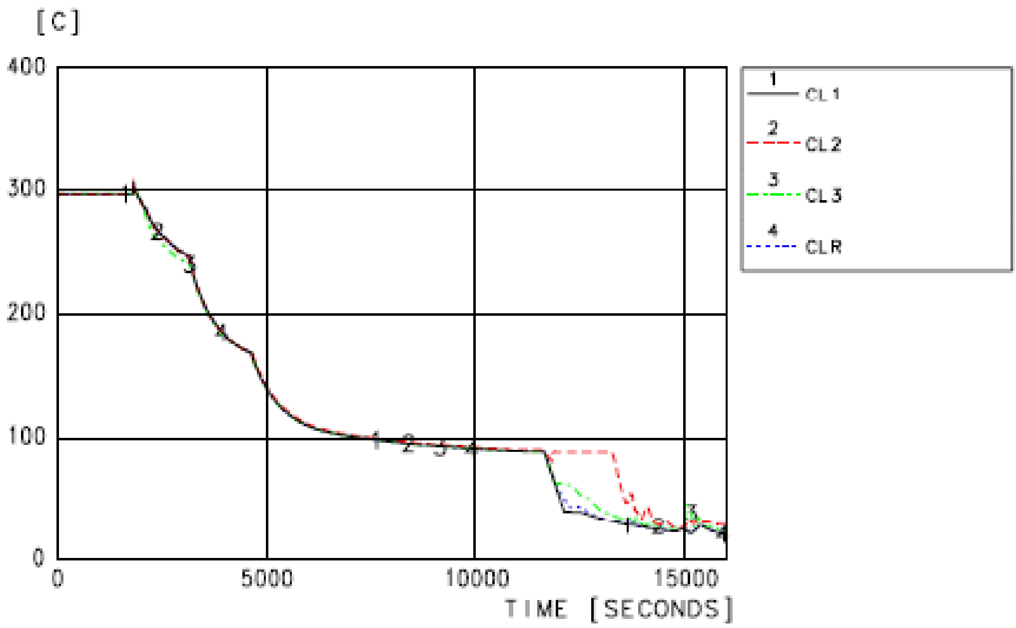
SECTION 16.4.4 - FIGURE 3

Coolant temperature in hot legs



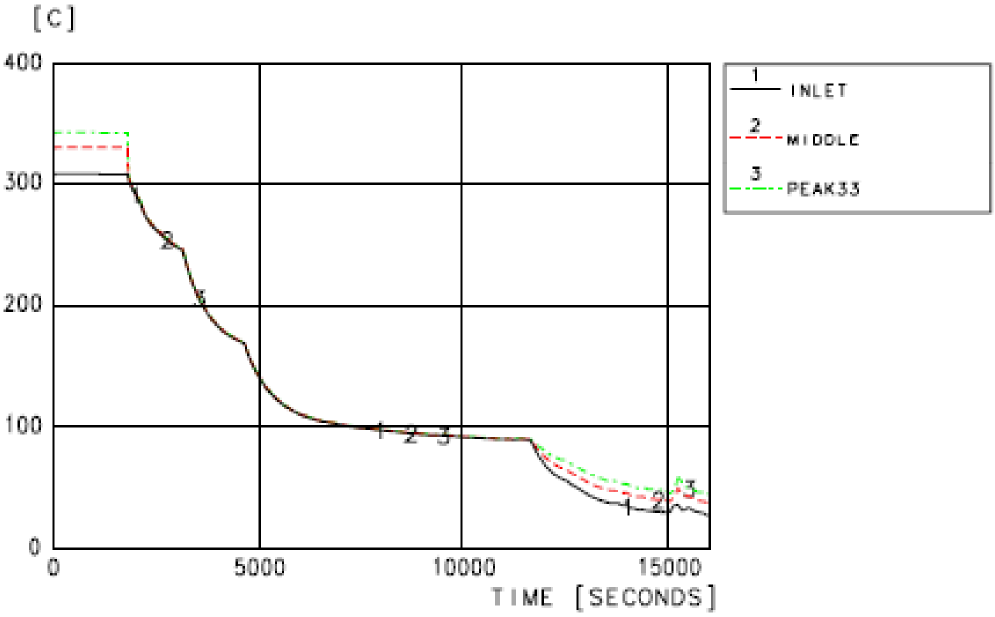
SECTION 16.4.4 - FIGURE 4

Coolant temperature in cold legs



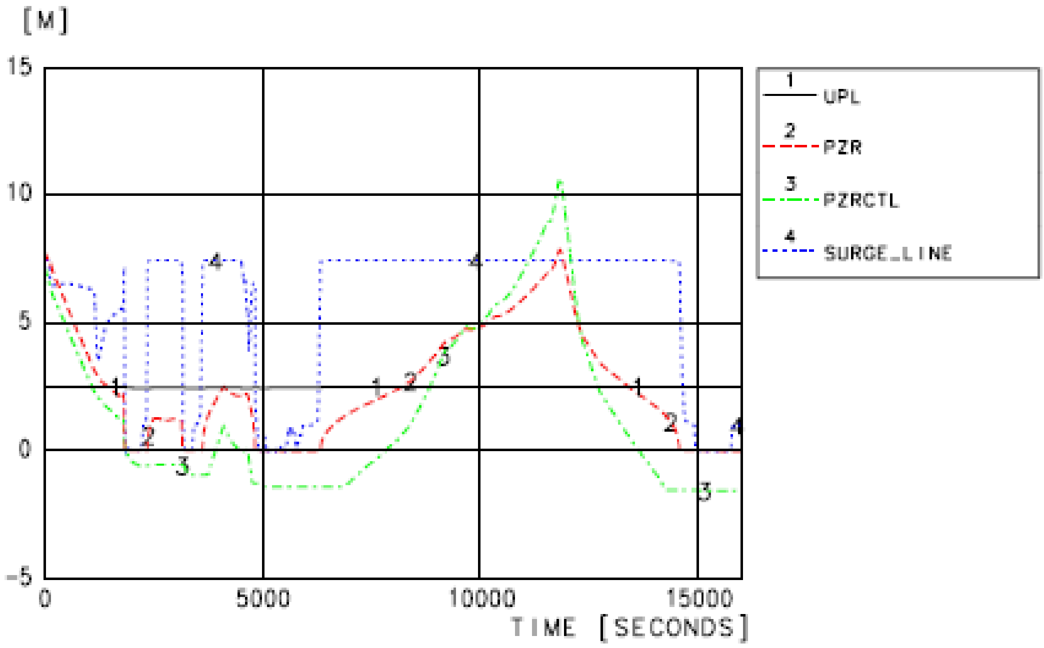
SECTION 16.4.4 - FIGURE 5

Cladding temperature



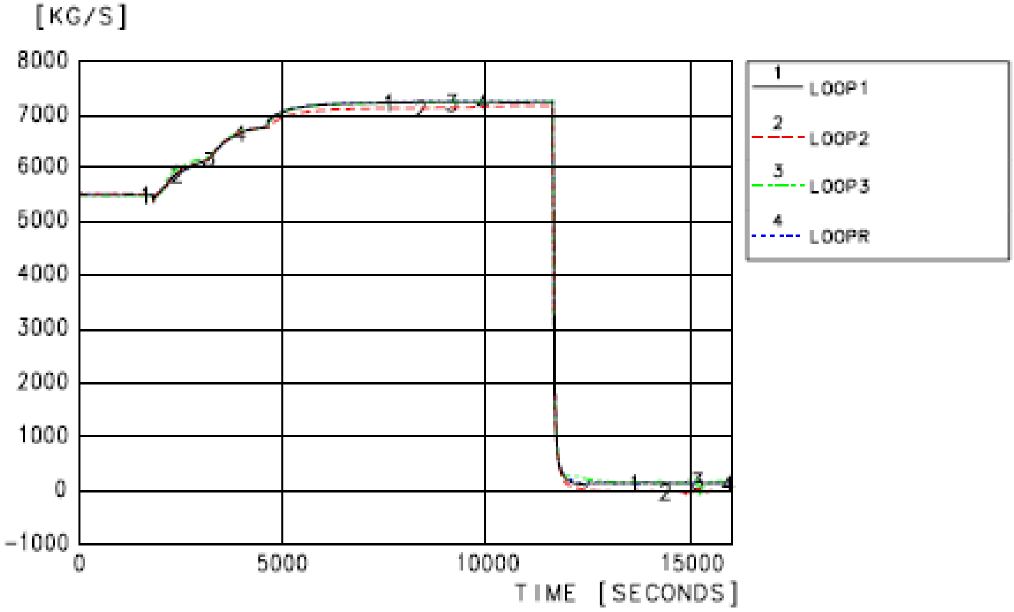
SECTION 16.4.4 - FIGURE 6

Reactor coolant system liquid levels



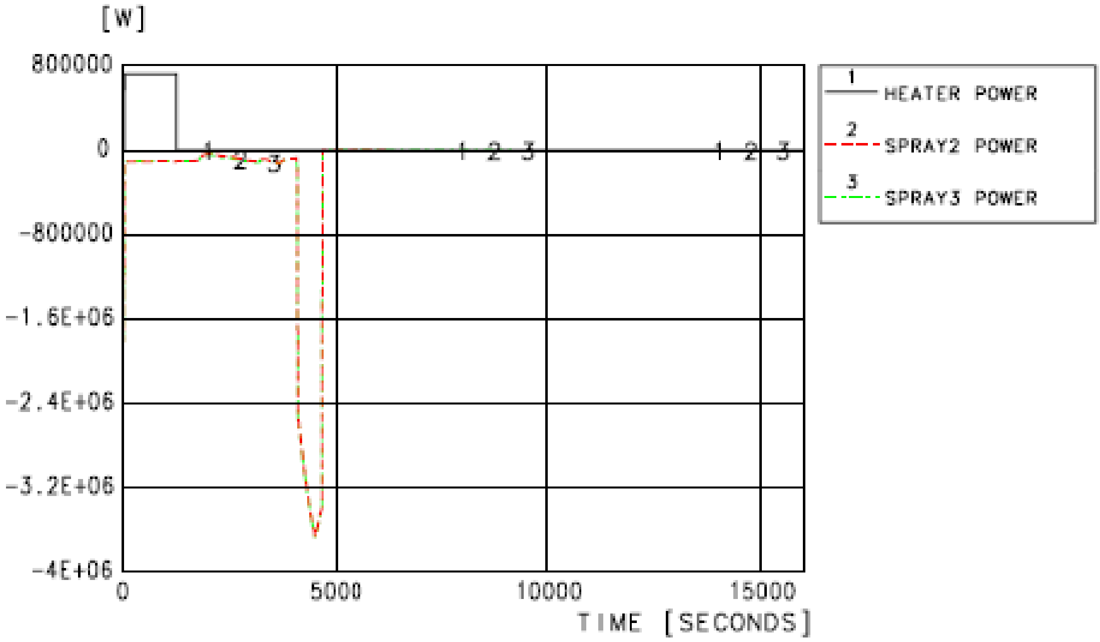
SECTION 16.4.4 - FIGURE 7

Cold leg loop mass flows



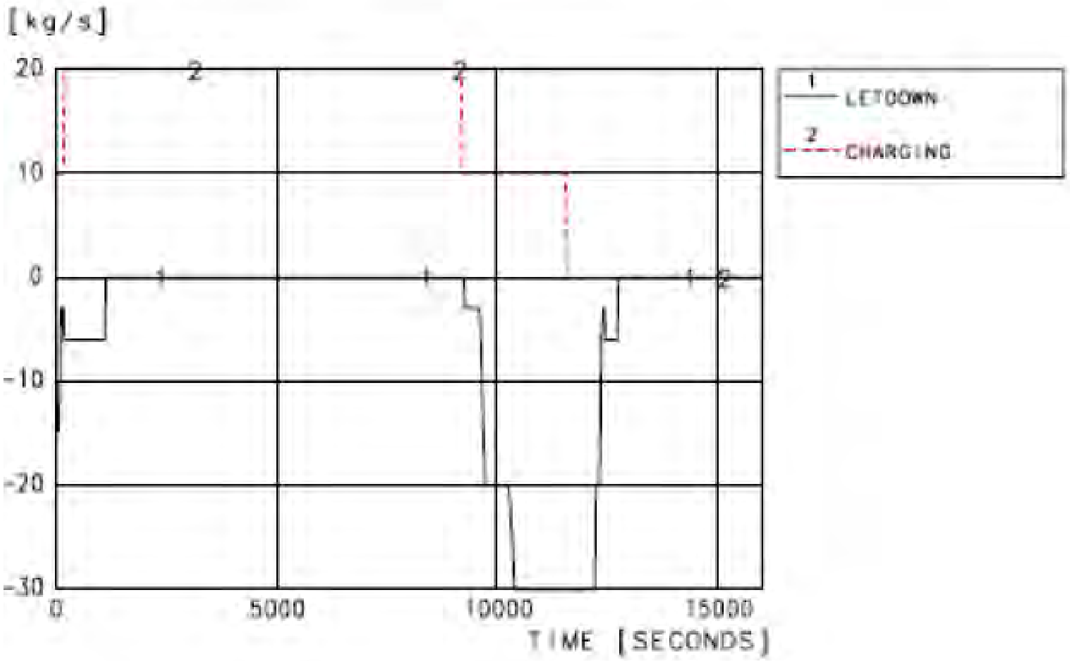
SECTION 16.4.4 - FIGURE 8

Pressuriser spray and heater power



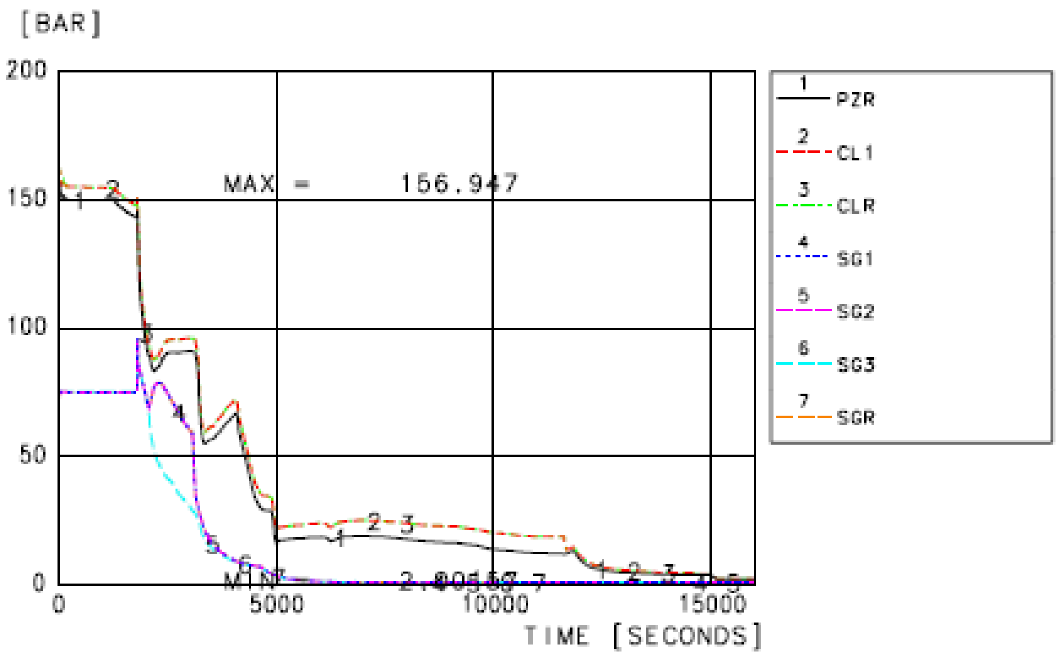
SECTION 16.4.4 - FIGURE 9

Letdown and charging flow rates



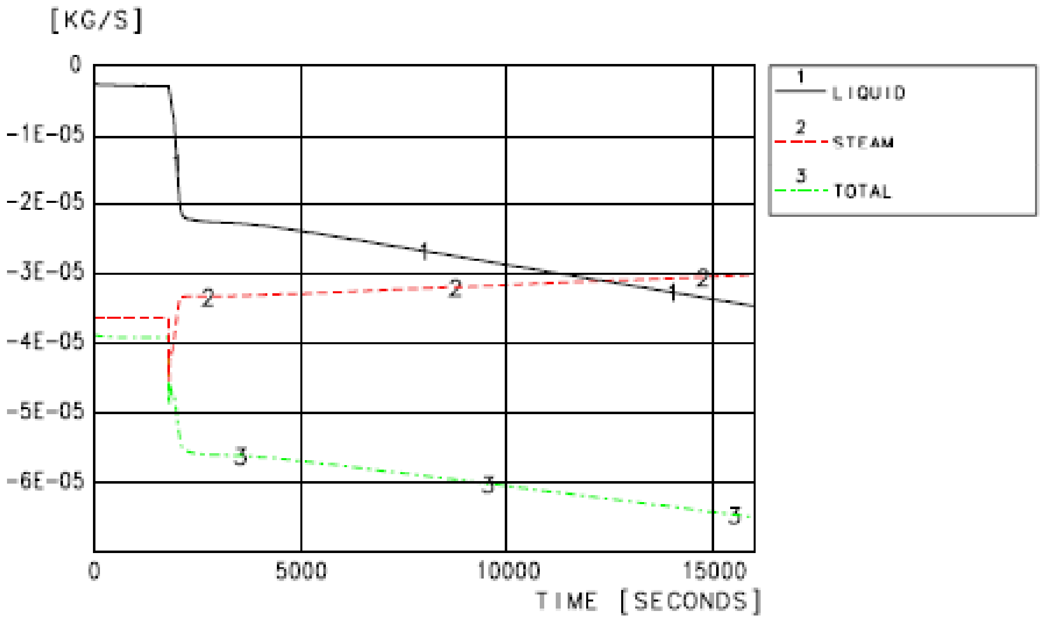
SECTION 16.4.4 - FIGURE 10

Primary and secondary pressures



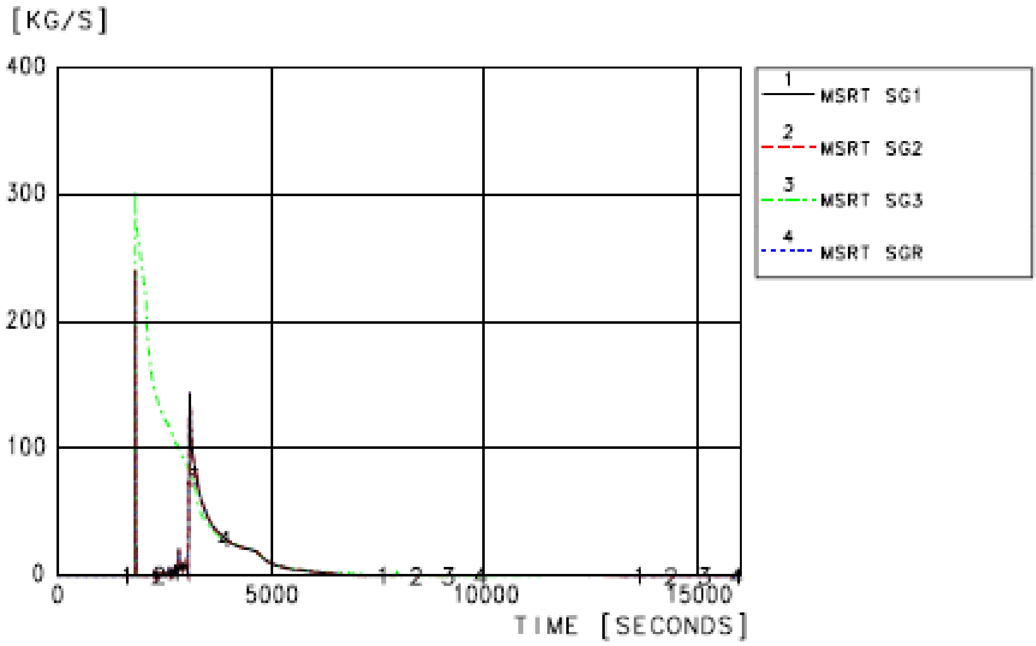
SECTION 16.4.4 - FIGURE 11

Main steam bypass flow rates



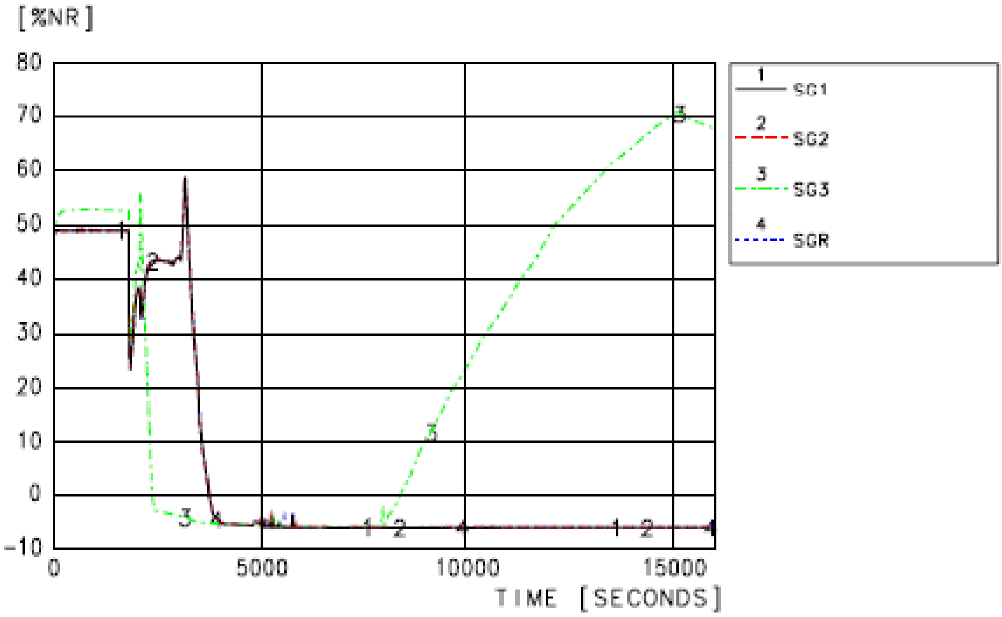
SECTION 16.4.4 - FIGURE 12

Main steam relief train mass flow



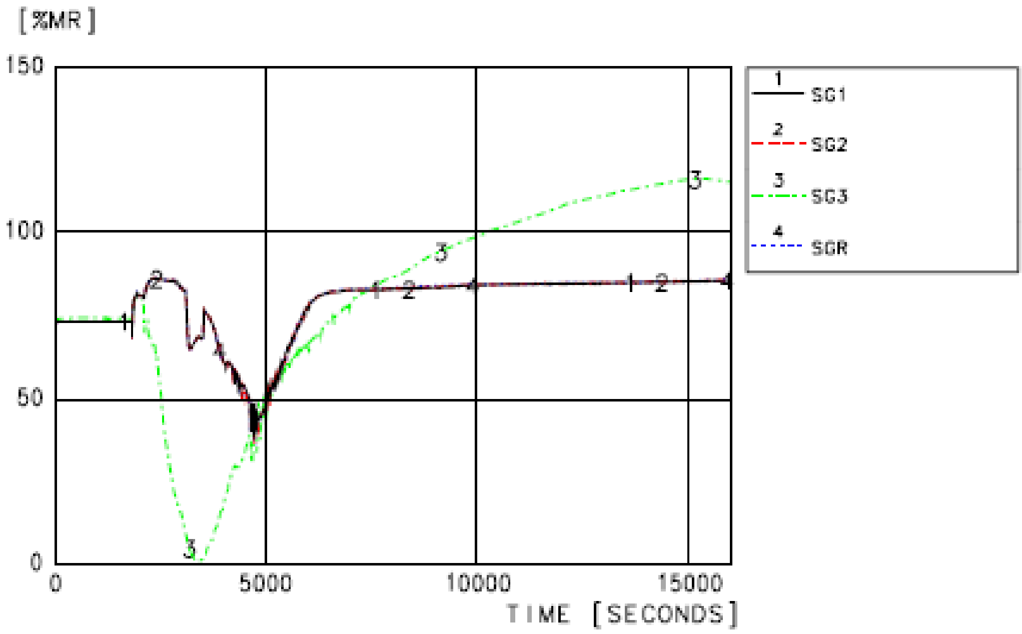
SECTION 16.4.4 - FIGURE 13

Steam generator narrow range level



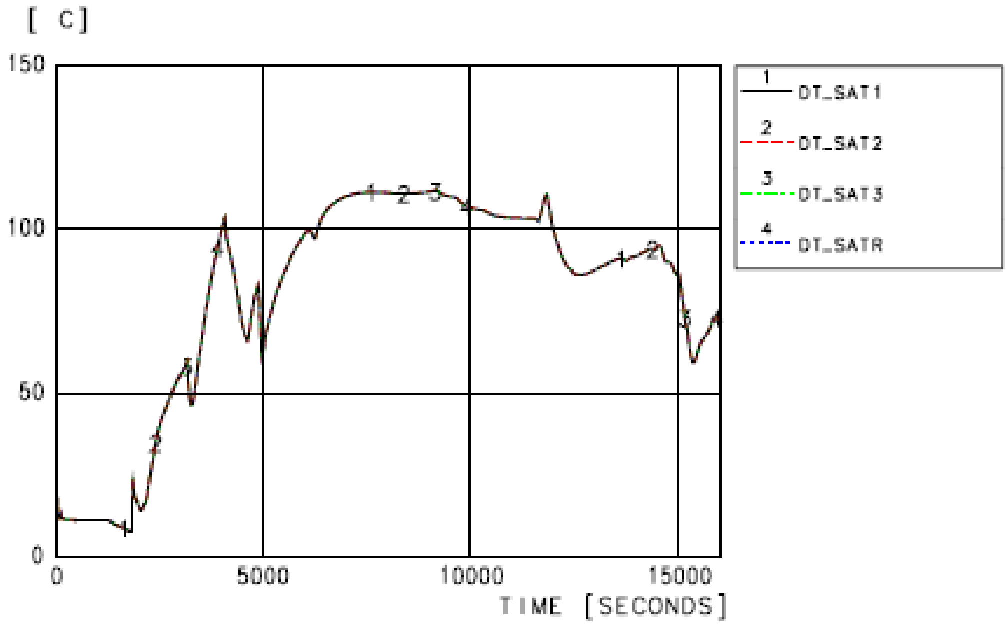
SECTION 16.4.4 - FIGURE 14

Steam generator wide range level



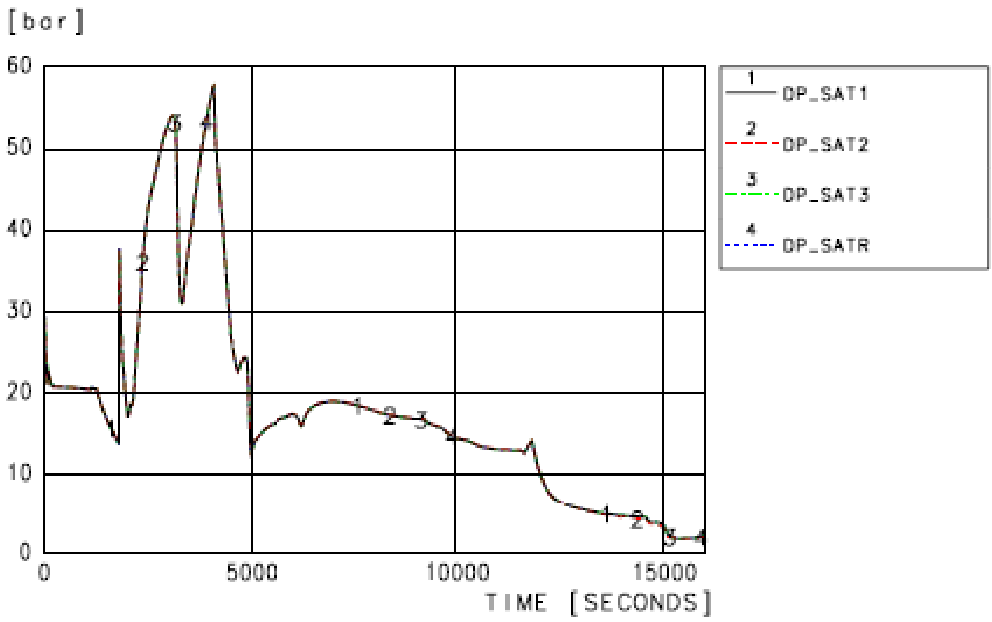
SECTION 16.4.4 - FIGURE 15

Saturation temperature safety margin



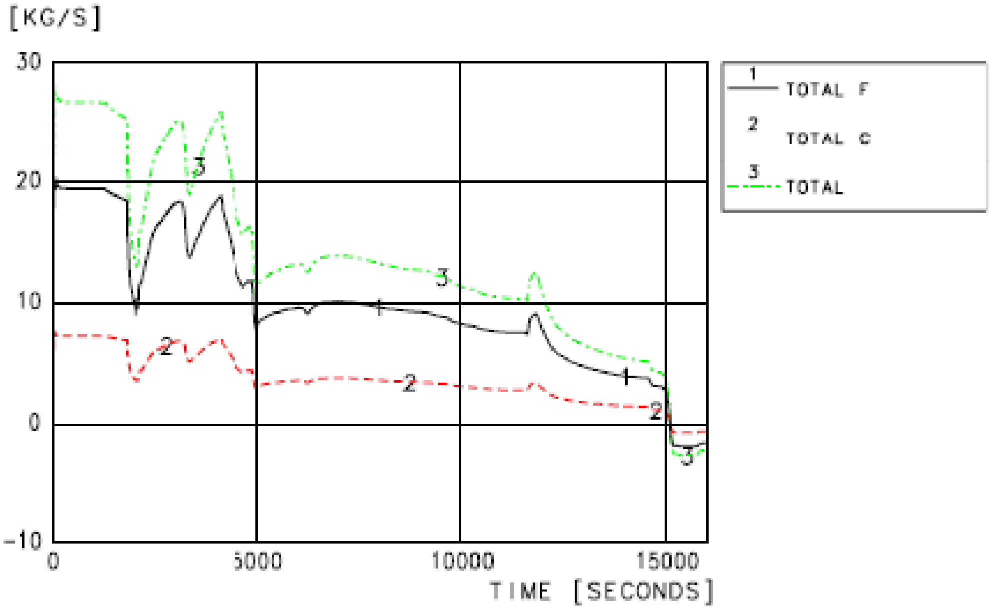
SECTION 16.4.4 - FIGURE 16

Saturation pressure safety margin



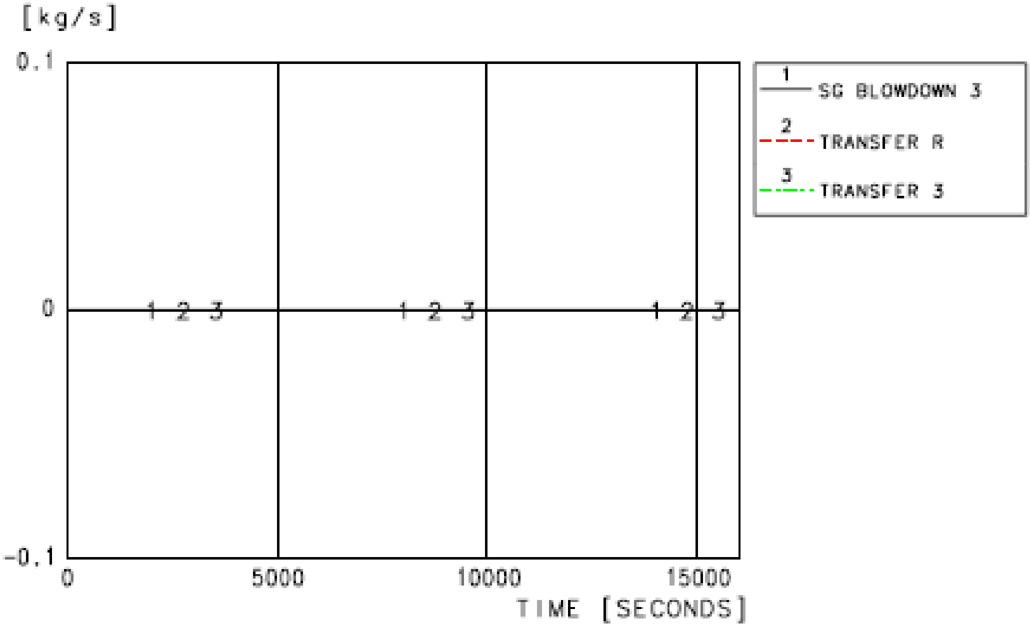
SECTION 16.4.4 - FIGURE 17

Steam generator tube rupture leak flow rates



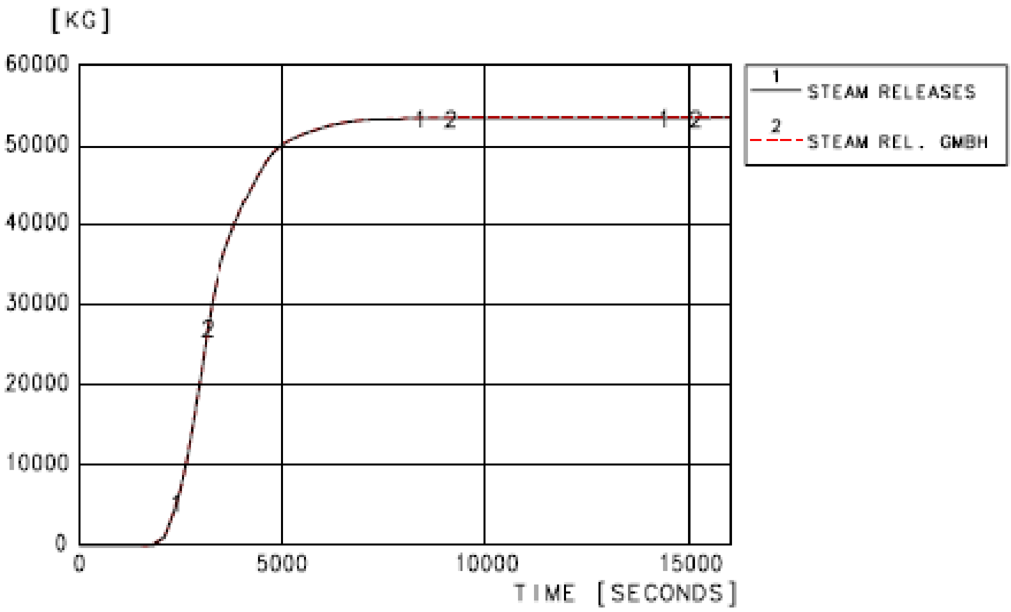
SECTION 16.4.4 - FIGURE 18

Blowdown line flow rates



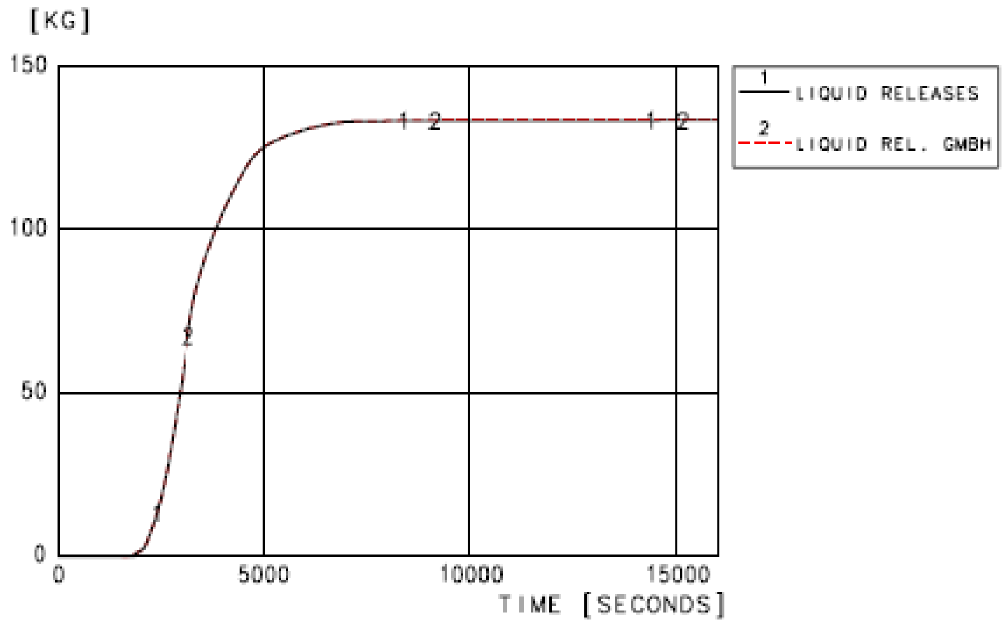
SECTION 16.4.4 - FIGURE 19

Primary release as steam



SECTION 16.4.4 - FIGURE 20

Primary release as liquid



5. MULTIPLE STEAM GENERATOR TUBE RUPTURE (10 TUBES IN ONE STEAM GENERATOR AT POWER)

5.1. DESCRIPTION

A steam generator tube rupture (SGTR) leads to a loss of primary coolant which is transferred to the steam generator affected by the rupture. The break causes a decrease in the primary circuit pressure and a contamination of the secondary side due to the tube leakage flow. The main consequence is a possible discharge of activity to the atmosphere via the main steam relief trains (primarily the affected steam generator). Primary coolant may be more or less contaminated by corrosion and fission products but its activity is nevertheless limited by the maximum activity level allowed by the technical specifications.

5.1.1. Identification of causes

The scenario examined is the complete severance of 10 tubes in one steam generator.

5.1.2. Precautions limiting accident occurrence

The probability of a steam generator tube rupture is very low due to the following design features:

- The steam generator tube material is highly ductile.
- The blowdown system is located at the bottom of the SG tube bundle and is designed to prevent solid deposits on the tube plate.
- The secondary water is chemically conditioned, thereby protecting the SG tubes from corrosion phenomena.
- Steam generator support plates supporting the tube bundle are designed (type of material, geometry of borings) to prevent denting in the tubes and, in the case of a double-ended guillotine tube rupture, to prevent pipe whip which could cause the rupture of neighbouring tubes.
- The control of the activity in the secondary side steam generator water and steam allows for continuous monitoring of compliance with established limits.
- On the mechanical side, the steam generators are designed to prevent projectiles, coming from the main feedwater system to hit directly one or several tubes.

Furthermore, should a double-ended guillotine rupture of a steam generator tube occur, pipe whip and rupture of any neighbouring tubes are prevented by the presence of 9 steam generator tube support plates. This makes the simultaneous rupture of two tubes very unlikely. However, for the purpose of demonstrating the capability of the plant, a completely theoretical scenario of the simultaneous rupture of 10 tubes is assessed.

Multiple steam generator tube rupture during power operation is considered as a specific study.

5.2. TYPICAL SEQUENCE OF EVENTS

A multiple steam generator tube rupture accident is characterised by a fast increase in the liquid level in the affected steam generator caused by the high leak flow. If no countermeasures are taken, the defective steam generator will be overfed with contaminated coolant from the primary side. In principle, the SGTR mitigation concept consists of two lines of automatic counter measures.

The first line initiates the following:

- Reactor and turbine trip. The reactor is tripped either on the “pressuriser pressure < MIN2” or on “steam generator liquid level > MAX1” signal.
- Pressuriser spray to lower the reactor coolant system pressure.
- Start-up the second RCV [CVCS] charging pump and isolation of the high pressure letdown line to make up the Reactor Coolant System (RCP [RCS]).

The second line follows a “SG level > MAX2” signal. The associated setpoint is chosen to prevent any SG overfilling following SGTR events. The following countermeasures are actuated:

- Partial cooldown of the RCP [RCS] by reducing the control valve pressure setpoint of the main steam bypass system (GCT [MSB]) or of the main steam relief trains (VDA [MSRT]) if the GCT [MSB] is not available.
- Isolation of the affected steam line by closing the main steam isolation valve at the end of the partial cooldown, increase of the VDA [MSRT] setpoint of the affected steam generator to 99 bar and isolation of the letdown line. The feedwater supply is shut off earlier in the transient via liquid level high signals (MAX0, MAX1).

The leak is terminated once the RCP [RCS] pressure and the affected SG pressure equalise. Any remaining leak flow can be compensated by the chemical and volume control system (RCV [CVCS]) and safety injection system (RIS [SIS]).

The liquid levels of the unaffected steam generators are maintained by the low load control valves.

The decay heat removal is provided by the unaffected SG VDA [MSRT].

At 30 minutes after the SGTR, the operator manoeuvres the plant to a final safe state using the SGTR strategy (identified by automatic diagnosis). The final state is defined in the present analysis as RCP [RCS] long term cooldown provided by the residual heat removal system (RRA [RHRS]) and the radioactive releases are terminated.

Different phases are contained in the SGTR emergency operating procedure (EOP) dependent on the actual plant conditions. Principally the following is undertaken by the operator to reach the RRA [RHRS] operating condition:

- Steam and water isolation of the affected steam generator

To limit the contaminated steam/water discharge, the operator has to confirm the feedwater supply has been stopped and steam line has been isolated before performing the RCP [RCS] cooldown. The VDA [MSRT] setpoint of the affected steam generator is therefore increased to 99[1] bar.

- Reactor coolant system boration

Boration is performed using the RCV [CVCS] or the extra borating system (RBS [EBS]) if the RCV [CVCS] is not available. The RCV [CVCS] pumps can inject water at pressures up to the RCP [RCS] design pressure. Therefore, the charging flow will be isolated to avoid any over-pressure in the primary and secondary systems (following a pressuriser or steam generator level high signal).

The Medium and Low Head Safety Injection pumps (MHSI, LHSI) and accumulators also deliver borated water into the RCP [RCS] at low pressure levels.

- Reactor coolant system cooldown

The reactor coolant system cooldown is performed using the unaffected steam generators by reducing the GCT [MSB] or VDA [MSRT] pressure setpoint at a cooldown rate dependent on the available rate of boron injection.

- Reactor coolant system depressurisation

At end of the RCP [RCS] manual cooldown phase, if the primary system pressure is still too high, the operator depressurises the RCP [RCS] using the pressuriser normal or auxiliary spray. If these are unavailable, the operator can open the VDA [MSRT] of the affected SG to perform a depressurisation of both the affected SG and the RCP [RCS].

5.3. ACCEPTANCE CRITERIA

The safety criteria for this event are the radiological limits defined for the plant. For the present analysis the relevant criteria to be met are:

- Bringing the plant to a final state: it shall be shown that the plant can be brought to a final state.

- Radiological releases are below those defined for PCC-4 events.

5.4. ANALYSIS FROM INITIATION OF EVENT TO FINAL STATE

5.4.1. Method of analysis

The analysis has been performed using the CATHARE V2.5 computer code described in Appendix 14A.

This code is an advanced, two-fluid, thermal hydraulic computer code designed for realistic studies of the accident thermal hydraulics in pressurised water reactors. It provides a detailed representation of the primary and secondary systems.

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[1] All pressures given in the present analysis are absolute.

The reactor pressure vessel and four main coolant loops with steam generators are simulated individually. The pressuriser is assumed to be connected to the hot leg of loop 3. The steam generator tube rupture is assumed to be located in the SG of loop 3.

The reactor core is modelled by one average channel and a conservative decay heat curve is applied to represent the decay power history after reactor trip. On the secondary side, the main steam lines are equipped with the control and isolation valves. The main steam header, turbine and bypass station are also represented.

Plant auxiliary systems such as the feedwater systems, chemical and volume control system, pressuriser system, emergency core cooling system, are not directly simulated but their functional characteristics are appropriately modelled.

5.4.2. Definition of studied case

The case studied here corresponds to the simultaneous double-ended guillotine rupture of 10 tubes (20A) in one steam generator. This allows full discharge from both ends of the severed tubes. The tube rupture is assumed to be located at the bottom of the cold side tube leg. This location is chosen to maximise the primary coolant discharge into the secondary side as a minimum break enthalpy leads to higher critical flows.

5.4.3. Choice of single failure and preventive maintenance

All plant systems not affected by the initiating of event are in their normal operating conditions. Preventive maintenance or single failure assumptions are not considered for this specific study.

5.4.4. Initial state

For the specific study the nominal values of plant data are generally used as shown in Section 16.4.5 - Table 1 which lists the relevant initial and boundary conditions applied in the analysis.

5.4.5. Specific assumptions

5.4.5.1. Neutronic data

The initial state of the plant is at a power of 4590 MWth. A conservative decay heat curve is used to represent the residual power history following the reactor trip.

5.4.5.2. Protection and mitigation actions

Following a SGTR event, automatic and operator actions trip the reactor, remove the residual heat and isolate the steam and water flow paths on the affected steam generator. These actions limit the contaminated water mass released to the atmosphere and bring the plant to the final safe state.

5.4.5.2.1. Safety Systems

- Reactor protection and safeguard signals

The reactor protection and safeguard signals listed in section 1 of Sub-chapter 15.0 are provided in the transient input. In the calculation the reactor is tripped following a low RCP [RCS] pressure MIN2 (135bar) signal and the safety injection signal is activated on a very low pressuriser pressure MIN3 (115bar) signal.

- Safety injection system

The four RIS [SIS] trains are modelled. The maximum delivery curves are applied for the medium and low head safety injection pumps. The accumulators are assumed to be at a low initial pressure and minimum liquid volume.

- Main steam relief train and pressure control

The main steam relief trains are available (one per SG). After the main steam isolation valves close, the secondary pressure is controlled by the VDA [MSRT].

- Pressuriser safety valves

Three pressuriser safety valves (PSV) with different opening pressure setpoints are available to limit any pressure increase in the reactor coolant system.

5.4.5.2.2. Non Safety Systems

- Main steam bypass

The main steam bypass is available until closure of the main steam isolation valve.

- Pressuriser heaters and spray

The pressuriser heaters and normal spray are taken into account. Total heater power is 2592 KW. Two suction lines take water from the cold legs of loops 2 and 3, the spray rate is 23 kg/s (minimum value). The auxiliary spraying mode from the RCV [CVCS] is not considered, pressuriser pressure control is credited.

- Chemical and Volume Control System

A dedicated pressuriser level control is modelled. The high pressure letdown line takes coolant from the cross-over leg on loop 1. In normal operation one charging pump injects 10 kg/s of demineralised water at about 270°C into the cold legs of loops 2 and 4.

5.4.5.2.3. Other assumptions

- Steam generator blowdown line

To prevent the overfeeding of the steam generator following a SGTR the dedicated blowdown line may be opened by the operator following indications of a high SG level. The discharge flow capacity is about 2% of the nominal feedwater flow.

5.4.5.3. Results

The sequence of events is summarised in Section 16.4.5 - Table 2 and Section 16.4.5 - Table 3, abbreviations used in the graphs are defined in Section 16.4.5 - Table 4 and Section 16.4.5 - Table 5 and transient analytical results are presented in Section 16.4.5 - Figure 1 to Figure 20.

During the automatic control phase (i.e. the first 1800 seconds after the SGTR occurs) the reactor trip is actuated following a low pressuriser pressure signal MIN2 (135 bar). The reactor trip signal initiates a checkback signal giving turbine trip and causing closure of the high load main feedwater line.

As the pressuriser level continues to fall, the second charging pump is actuated, subsequently the letdown line is isolated and the heaters are switched off.

When the pressuriser pressure decreases below 115 bar (MIN3), the safety injection signal is actuated. RIS [SIS] is actuated with the start-up of the MHSI and LHSI pumps. The safety injection signal also actuates a partial cooldown by means of the GCT [MSB] from 90 bar down to 55 bar.

When the affected SG water level reaches 69%NR the dedicated low load main feedwater line is closed. After the partial cooldown is finished and when the affected SG water level reaches the MAX2 (85%NR) threshold, the main steam isolation valve of the affected SG is closed, the RCV [CVCS] charging lines are isolated and the VDA [MSRT] setpoint of the affected SG is increased to 99 bar.

At 1800 seconds after start of the event, the operator is instructed to perform the emergency operating procedure (EOP) “transition to cold shutdown in case of SGTR“ to bring the plant to a final safe state.

In phase 1 ”MHSI stop”, the operator activates the P12 key to reset the safety injection signal, restarts the RCV [CVCS], starts up the two RBS [EBS] pumps and performs the RCP [RCS] cooldown at rate of -50°C/h.

In addition, three MHSI pumps (loop 1, 2 and 4) are shut down, the MHSI pump of the affected loop remaining in operation. As the affected SG level is high the operator opens the blowdown line to reduce the mass inventory.

After reaching a hot leg temperature below 180°C the last MHSI pump (loop 3) is switched off. The operator starts phase 4, “transition to cold shutdown: depressurisation”. The normal spray is actuated to reduce the primary pressure and to avoid backflow via the SGTR leak as far as possible. The secondary pressure of the affected SG is reduced using the GCT [MSB] whilst maintaining a 6 bar margin to the primary pressure. The RBS [EBS] pumps are stopped manually. When the RCP [RCS] pressure reaches 70 bar all accumulators are isolated from the RCP [RCS].

As a consequence of the fast pressure decrease (SG pressure drop > MAX1) the main steam lines are isolated. Consequently, the GCT [MSB] is no longer available and steam removal from the SG is continued via the main steam relief trains.

On the primary side the residual heat removal system operating conditions are reached when the RCP [RCS] pressure drops below 30 bar. The operator activates the permissive P14, stops the normal spray and switches over the LHSI pumps in safety injection mode to the residual heat removal mode and then performs the long term RCP [RCS] cooldown using the RRA [RHRS].

At the end of the analysis the SG tube rupture leak flow is almost terminated, activity release is stopped and the final state is reached. Primary releases to atmosphere are around 800 kg as steam and 2 kg as liquid (humidity of the steam).

5.4.6. Applicability of results to UK EPR

The present study utilises a plant thermal power of 4590 MWth. All other parameters are assumed at their nominal value. This thermal power is the main difference to the reference design configuration of the UK EPR [Ref-1]. It leads to a slightly higher SG feedwater flow rate and a slightly different RCP [RCS] average temperature.

Considering the criteria to be met (primarily the radiological consequences), the SGTR transient assessed in the current study encompasses a similar study for the UK EPR.

5.5. CONCLUSION

The present analysis demonstrates that, in the case of a multiple SG tube rupture (10 tubes) event, the plant can be taken to the final state without violating the acceptance criteria.

With the automatic counter measures and dedicated SGTR strategy applied, the operator can bring the plant to a final state, whilst limiting the releases of primary coolant to atmosphere.

The core sub-criticality is maintained due to the amount of borated water injected from the chemical and volume control system (RCV [CVCS]), extra borating system (RBS [EBS]) and safety injection systems (RIS [SIS]).

SECTION 16.4.5 – TABLE 1

Plant initial and boundary conditions

PARAMETERS	VALUE
REACTOR COOLANT SYSTEM	
Nominal core power (MWth)	4590
Reactor coolant system mean temperature (°C)	312.4
T/H Reactor loop flow (kg/s)	22110
PRESSURISER SYSTEM	
Initial pressuriser pressure (bar)	155
Pressuriser water volume / level (%MR)	56
Pressuriser safety valve setpoints (bar)	175 / 178 / 181
STEAM GENERATORS (WITH FOULING AND PLUGGING)	
Initial steam pressure (bar)	74.9
Initial Steam Generator level (%NR)	49
FEEDWATER SYSTEM	
Main feedwater flow (kg/s)	651
Initial main feedwater temperature (°C)	230
MAIN STEAM RELIEF TRAIN	
Steam flow rate per train (t/h) at 100 bar abs (55% nominal steam flow)	1291
EXTRA BORATION SYSTEM	
RBS [EBS] flow per loop (kg/s)	1.4

SECTION 16.4.5 - TABLE 2

Sequence of events – Automatic phase

Time (s)	**Event**
AUTOMATIC PHASE	
>0.0	20A SGTR opening (10 tubes)
58	Reactor trip actuated on low pressuriser pressure <MIN2
59	Responses related to Reactor Trip: → Start of rod drop → Start of feedwater full load valves close (25s) → Turbine trip
~74	Start-up of second charging pump (pressuriser level <49%MR)
~75	Letdown line isolation (pressuriser level <15%MR) Heaters are cut off (pressuriser level <12%MR)
114	Safety Injection signal actuated on low low pressuriser pressure <MIN3 → Start of partial cooldown with GCT [MSB] → Start-up of MHSI and LHSI pumps (129s) → RCP [RCS] isolation
505	SG Level >MAX0, main feed water low load line closed on affected SG3
1258	Partial cooldown via GCT [MSB] finishes, SG pressure is ~55bar
1304	Affected SG level > MAX2 → main steam isolation valve loop 3 closes (full closed 1310s) → VDA [MSRT] pressure setpoint increased to 99bar → RCV [CVCS] charging injection isolated
1309.	Pressuriser liquid level < 12%MR High pressure letdown line isolated, heaters cut off Controlled state is reached

SECTION 16.4.5 - TABLE 3

Sequence of events – operator actions

Time (s)	Operator Actions
1800	EOP phase 1 “transition to cold shutdown in case of SGTR” : - Activation of P12 key, start-up two RBS [EBS] pumps (injection 1824 s) - Switch-off the three MHSI loops 1, 2 and 4 - MHSI loop 3 remains in service - restart the RCV [CVCS] - manual opening of SG blowdown line loop 3 - Start of RCP [RCS] cooldown via GCT [MSB] at rate of -50°C/h
8630.3	- Manual stop the RBS [EBS] pumps - Manual stop of the last MHSI pump (loop 3) due to hot leg temperature below 180°C - Start of “transition to cold shutdown in case of SGTR” EOP phase 4: “transition to cold shutdown: depressurisation”
9023.3	RCP [RCS] pressure < 70bar: all accumulators are isolated
9630.4	Start manual spray to reduce reactor coolant system pressure
9630.7	VDA [MSRT] active in loop 3 to reduce the SG pressure during the manual spray
9717.7	RRA [RHRS] connection conditions reached - Start of “transition to cold shutdown in case of SGTR” EOP phase 5: “cooldown with RRA [RHRS] and depressurisation” to connect the RRA [RHRS] - manual actuation of key P14 - start-up RIS [SIS] for Residual Heat Removal mode in loops 1, 2 and 4
9718.2	Stop manual spray
15000	End of analysis: SGTR leak flow nearly terminated Secondary pressure of unaffected steam generators at approximately 2 bar Secondary pressure of affected steam generator at approximately 25 bar Primary pressure at below 30 bar Primary coolant temperature below 180°C RRA [RHRS] in operation since approximately 10000 s

SECTION 16.4.5 - TABLE 4

Explanation of graphic abbreviations

Section 16.4.5 - Figure No.	Titles and Description of Abbreviations
1	**Core Heat Release and Steam Generator Output** SG_POWER Total heat transfer rate in steam generator CORE-POWER Reactor core power
2	**Primary System Pressures** PZR pressuriser steam dome CL1,CL2,CL3,CLR Cold Leg at vessel inlet loop 1/2/3/4
3	**Coolant Temperatures in Hot Legs** HL1,HL2,HL3,HLR Hot Leg at vessel outlet loop 1/2/3/4
4	**Coolant Temperatures in Cold Legs** CL1,CL2,CL3,CLR Cold Leg at vessel inlet loop 1/2/3/4
5	**Cladding temperature** INLET Cladding temperature at core inlet, MIDDLE Cladding temperature at core middle, PEAK33 Cladding temperature at the core peak
6	**Reactor Coolant System Liquid Levels** UPL upper plenum PZR pressuriser PZRCTRL pressuriser measurement SURGE_LINE surge line
7	**Cold Leg Loop Mass Flows** LOOP1, 2, 3 R Cold Leg at vessel inlet loop 1/2/3/4
8	**MHSI Pump Injection Rates** MHSI1,2, 3, R MHSI mass flow rate loop 1, 2, 3 and 4
9	**Extra Borating System Injection Rates** Q_EBS_1, 2, 3, R RBS [EBS] mass flow rate loop 1, 2, 3 and 4
10	**Primary and Secondary Pressures** PZR Pressuriser steam dome CL1, CLR Cold Leg at vessel inlet loop 1and 4 SG1, 2, 3, R SG steam dome loop 1, 2, 3 and 4
11	**Main Steam Bypass Flow Rates** LIQUID Liquid mass flow rate STEAM Steam mass flow rate TOTAL Total mass flow rate

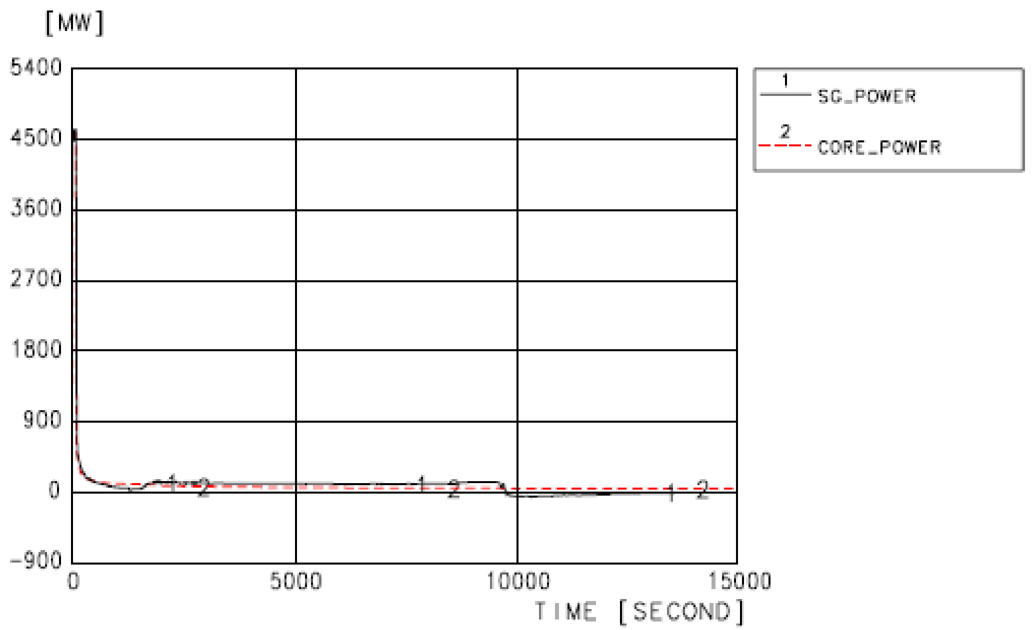
SECTION 16.4.5 - TABLE 4

Explanation of graphic abbreviations (Continued)

12	**Main Steam Relief Train Flow Rates** MSRT SG1 VDA [MSRT] flow rate SG loop 1 MSRT SG2 VDA [MSRT] flow rate SG loop 2 MSRT SG3 VDA [MSRT] flow rate SG loop 3 MSRT SGR VDA [MSRT] flow rate SG loop R
13	**Steam Generator Narrow Range Level** SG1, SG2, SG3, SGR narrow range level of SG loop 1/2/3/4
14	**Steam Generator Wide Range Level** SG1, SG2, SG3, SGR Wide range level of SG loop 1/2/3/4
15	**Saturation Temperature Safety Margin** DT_SAT1 Saturation margin hot leg loop 1 DT_SAT2 Saturation margin hot leg loop 2 DT_SAT3 Saturation margin hot leg loop 3 DT_SATR Saturation margin hot leg loop 4
16	**Saturation Pressure Safety Margin** DT_SAT1 Saturation margin hot leg loop 1 DT_SAT2 Saturation margin hot leg loop 2 DT_SAT3 Saturation margin hot leg loop 3 DT_SATR Saturation margin hot leg loop 4
17	**Steam Generator Tube Rupture Leak Flow Rates** TOTAL F Leak discharge rate from the downstream side TOTAL C Leak discharge rate from the upstream side TOTAL Total leak discharge rate
18	**Blowdown Line Flow Rates** SG BLOWDOWN 3 Mass flow rate SG3 blowdown line TRANSFER R Mass flow rate SG4 transfer line TRANSFER 3 Mass flow rate SG3 transfer line
19	**Primary Release as Steam** STEAM RELEASE Primary release as steam STEAM REL. GMBH (control model, not used)
20	**Primary Release as Liquid** LIQUID RELEASE Primary release as steam LIQUID REL. GMBH (control model, not used)

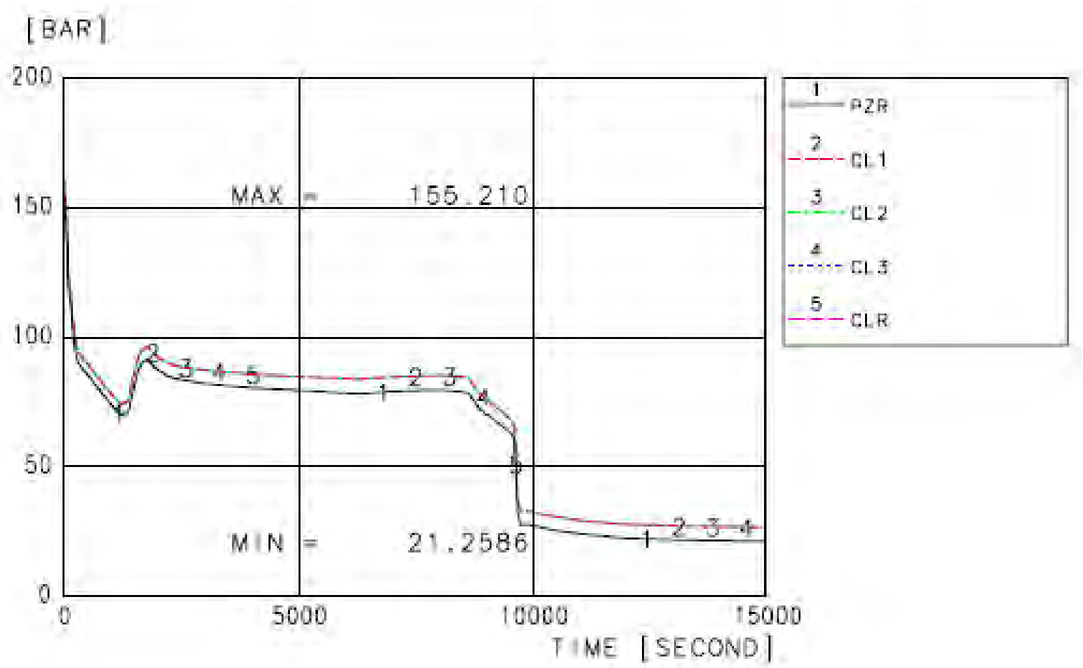
SECTION 16.4.5 - FIGURE 1

Core thermal power and power removed by the steam generators



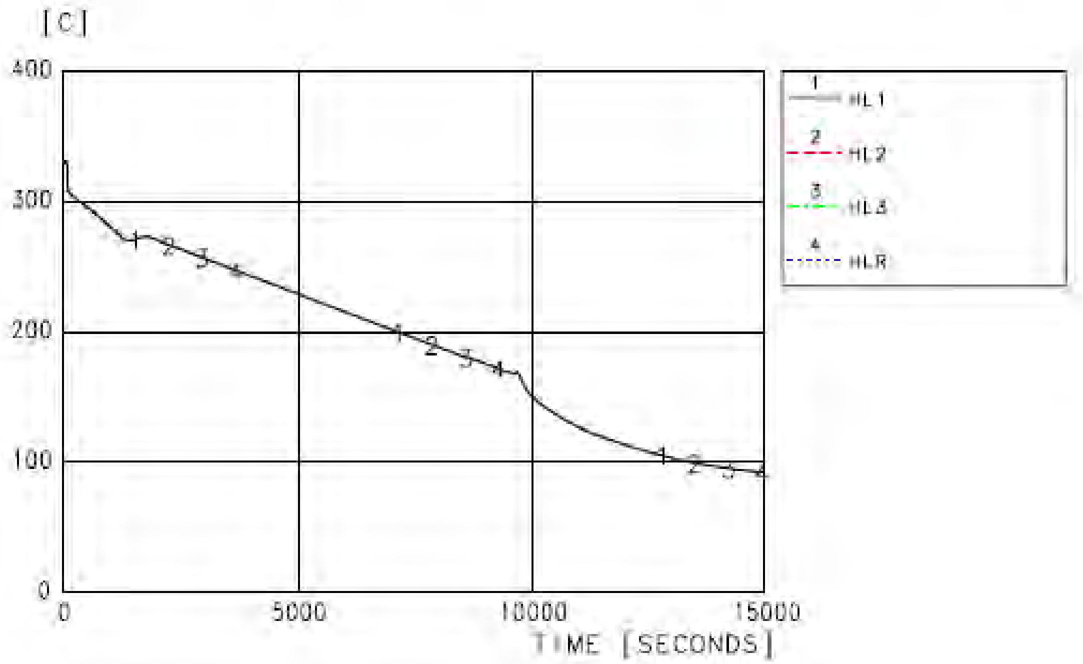
SECTION 16.4.5 - FIGURE 2

Reactor coolant system pressures in pressuriser and cold legs



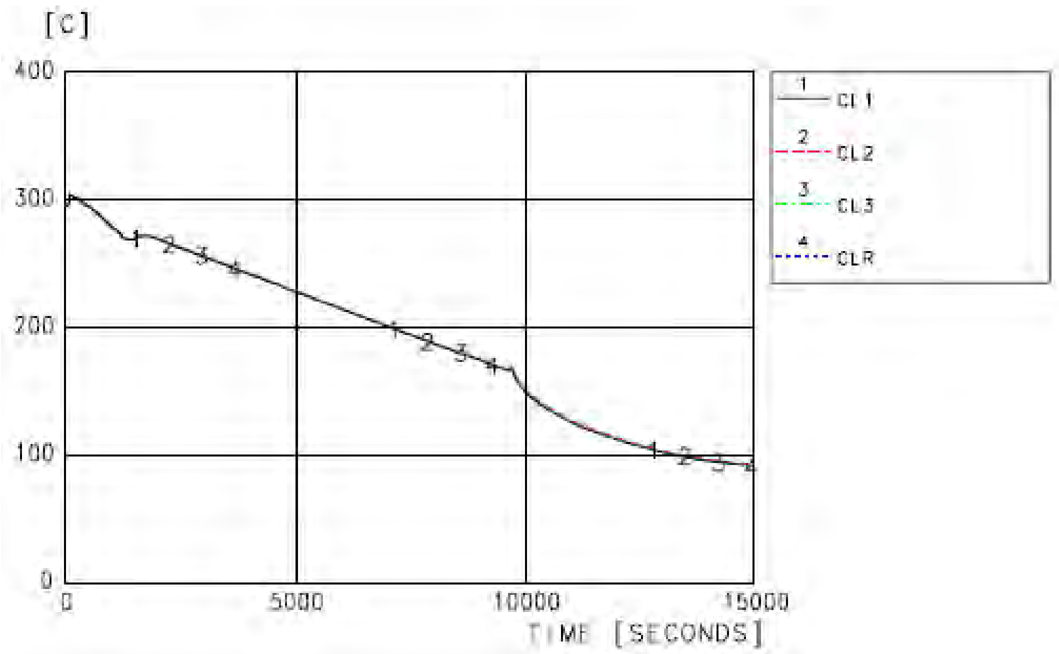
SECTION 16.4.5 - FIGURE 3

Reactor Coolant System Temperatures in hot legs



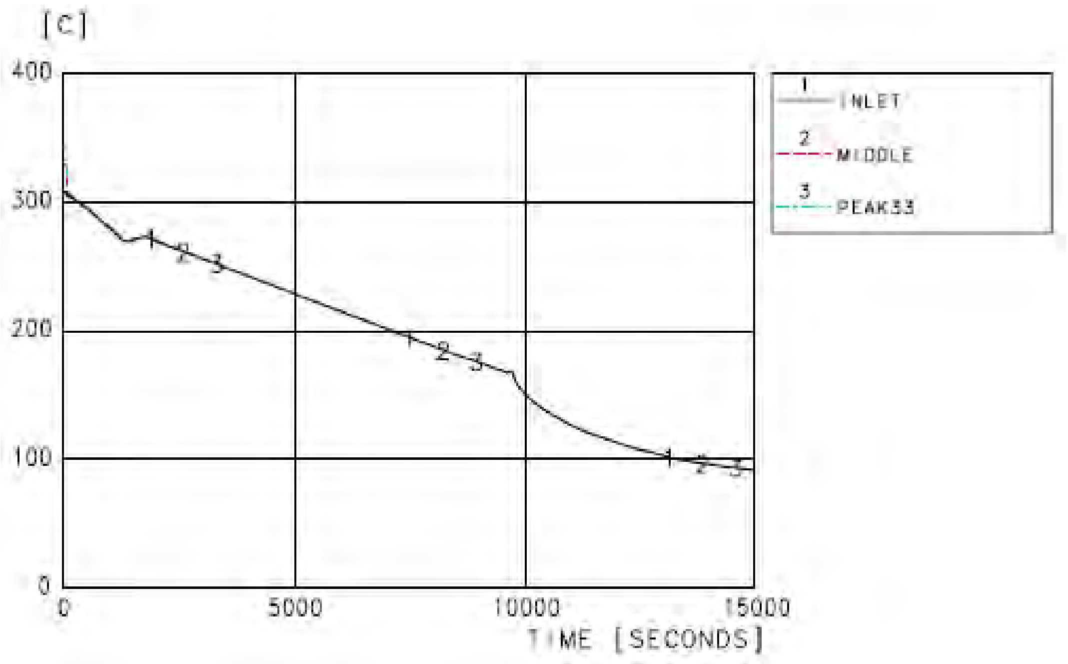
SECTION 16.4.5 - FIGURE 4

Reactor coolant system temperatures in cold legs



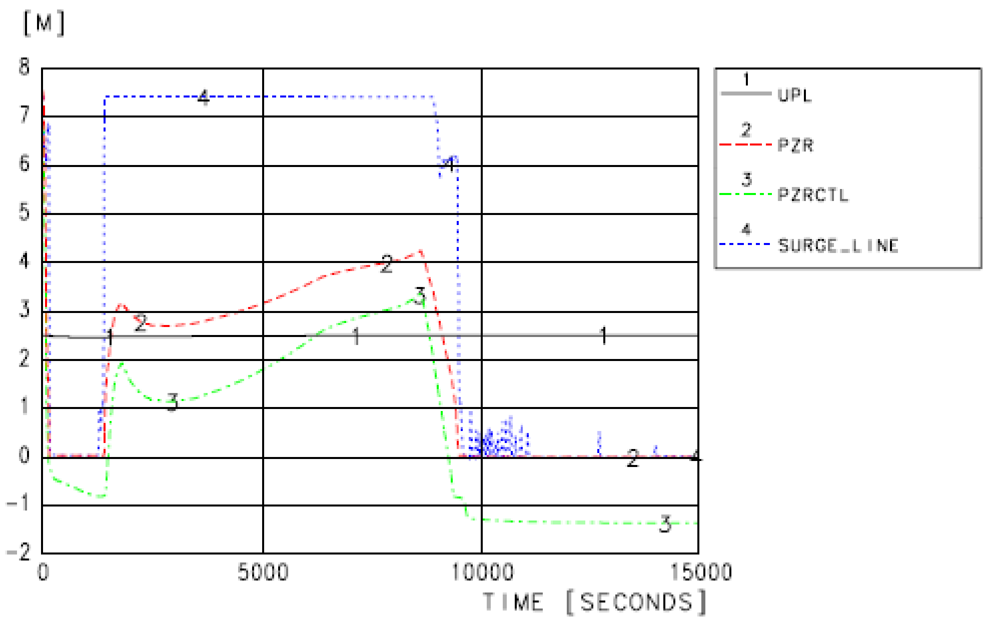
SECTION 16.4.5 - FIGURE 5

Fuel cladding temperature



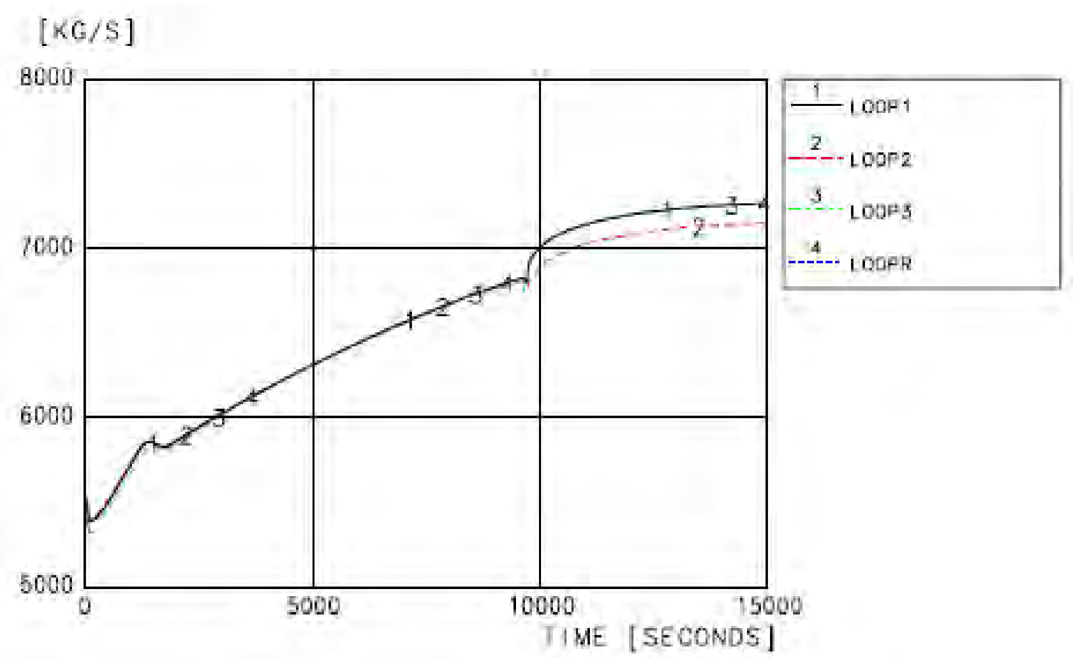
SECTION 16.4.5 - FIGURE 6

Reactor coolant system liquid levels



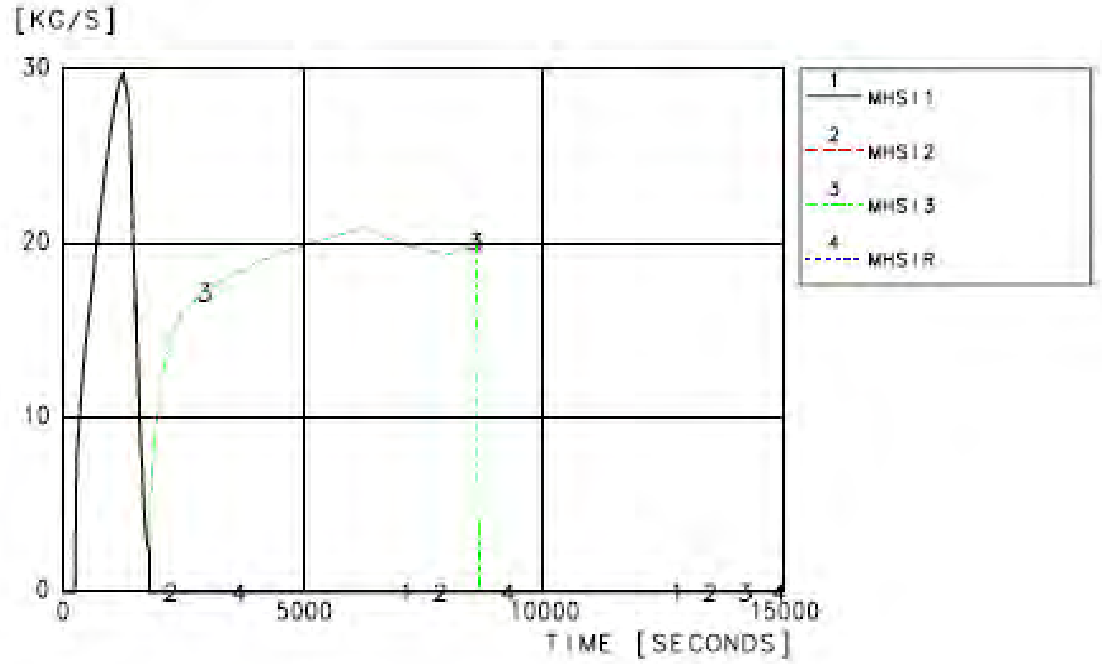
SECTION 16.4.5 - FIGURE 7

Reactor coolant system mass flow rates in cold legs



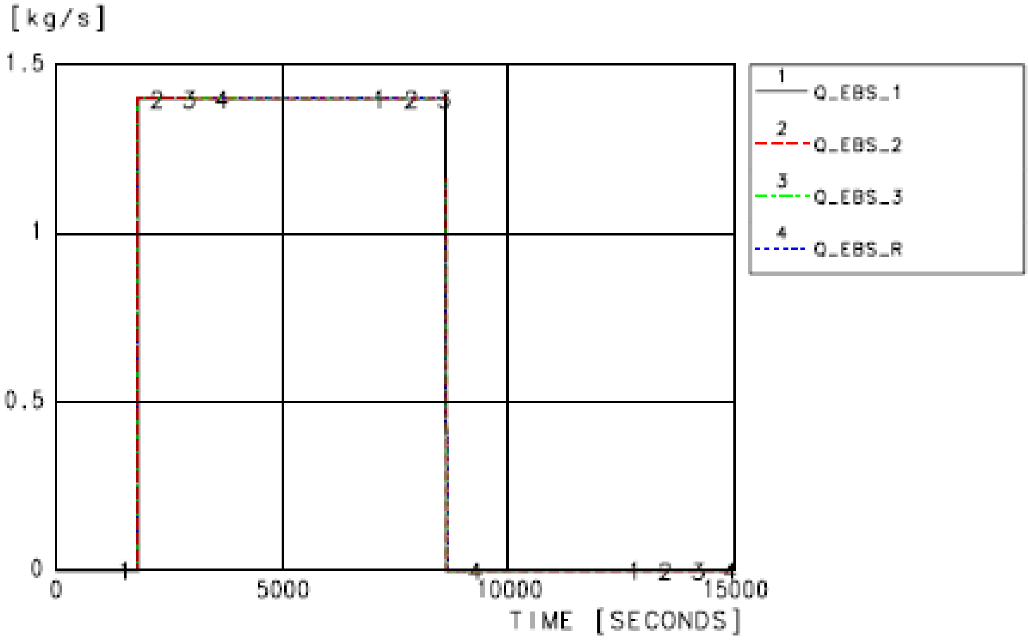
SECTION 16.4.5 - FIGURE 8

Medium Head Safety Injection mass flow rates



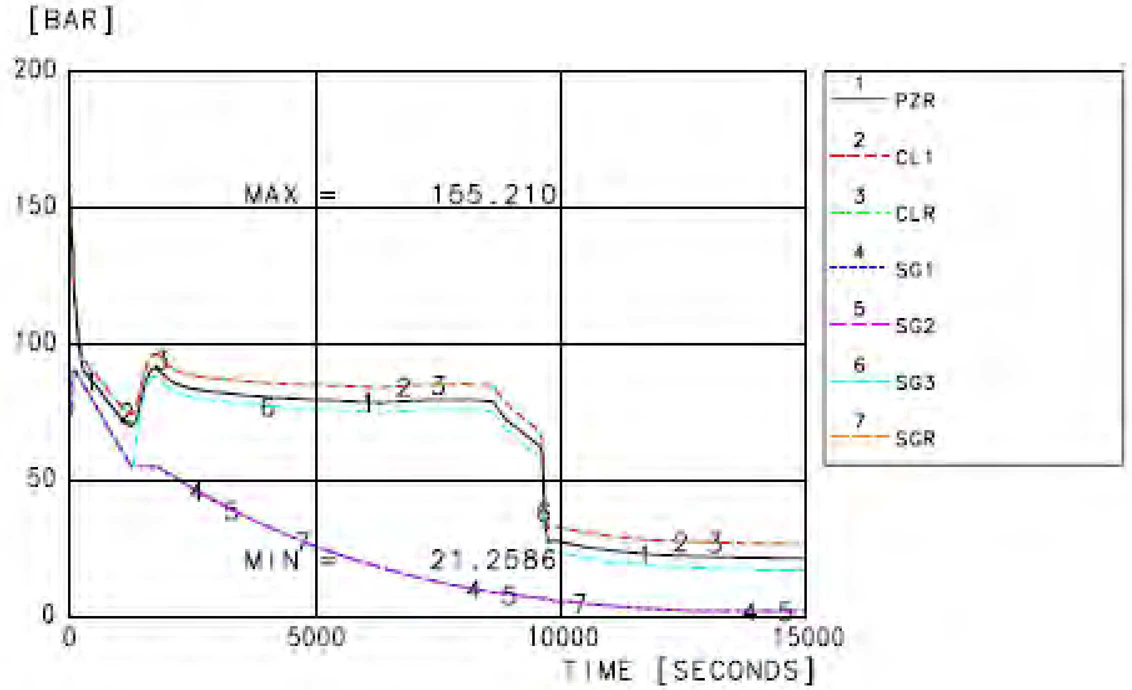
SECTION 16.4.5 - FIGURE 9

Extra borating system injection mass flow rates



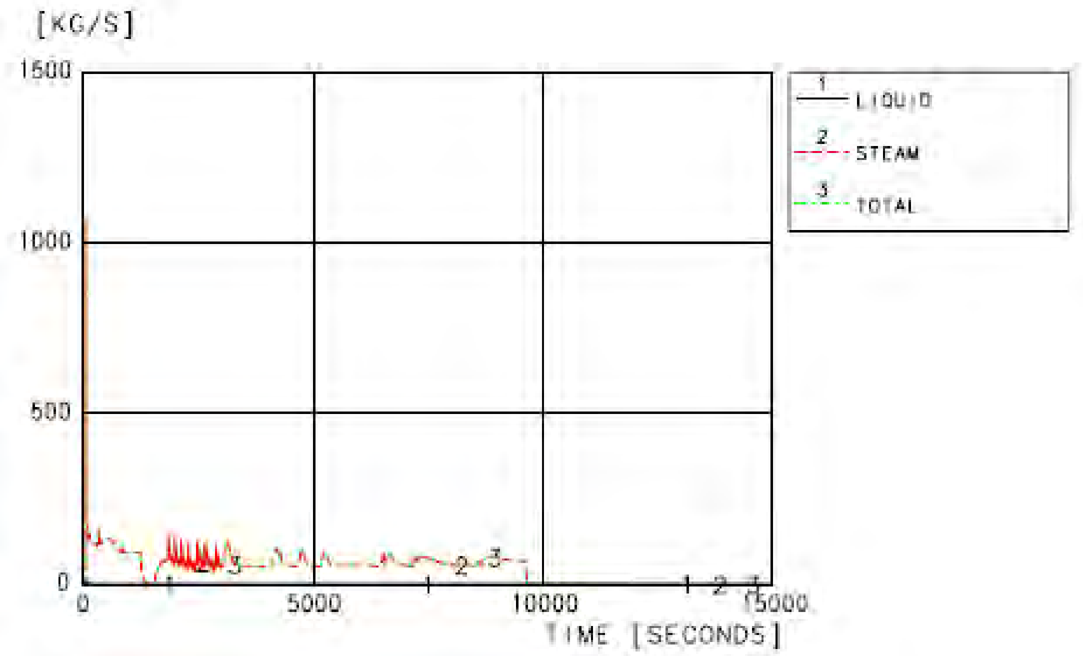
SECTION 16.4.5 - FIGURE 10

Reactor coolant system and steam generators pressures



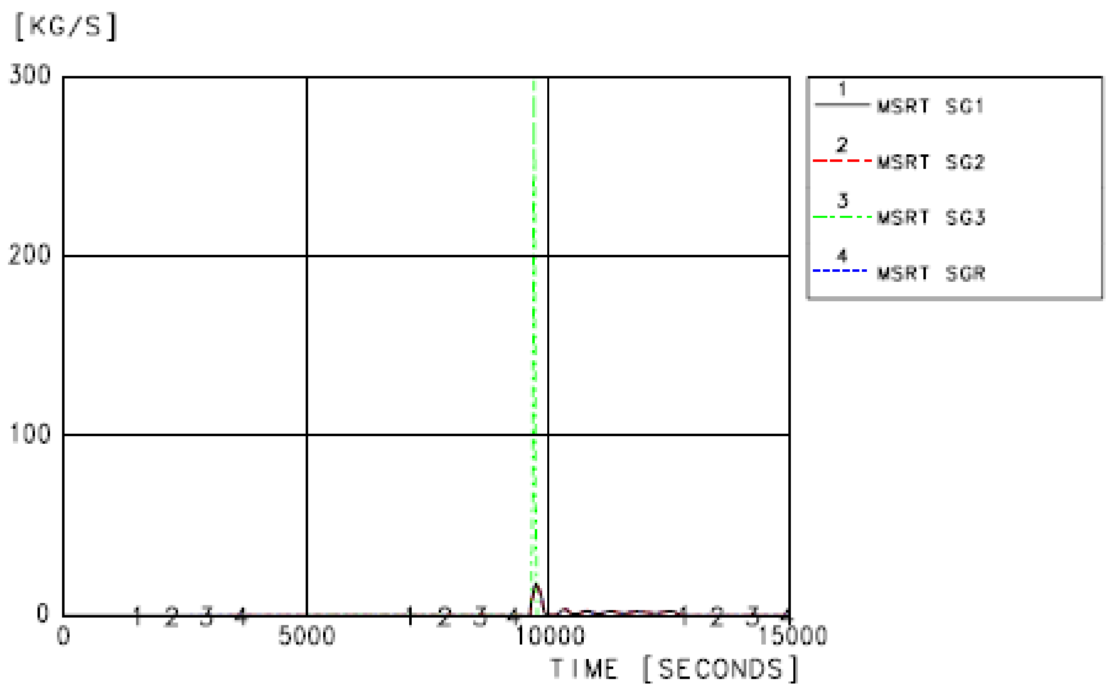
SECTION 16.4.5 - FIGURE 11

Main steam bypass mass flow rates



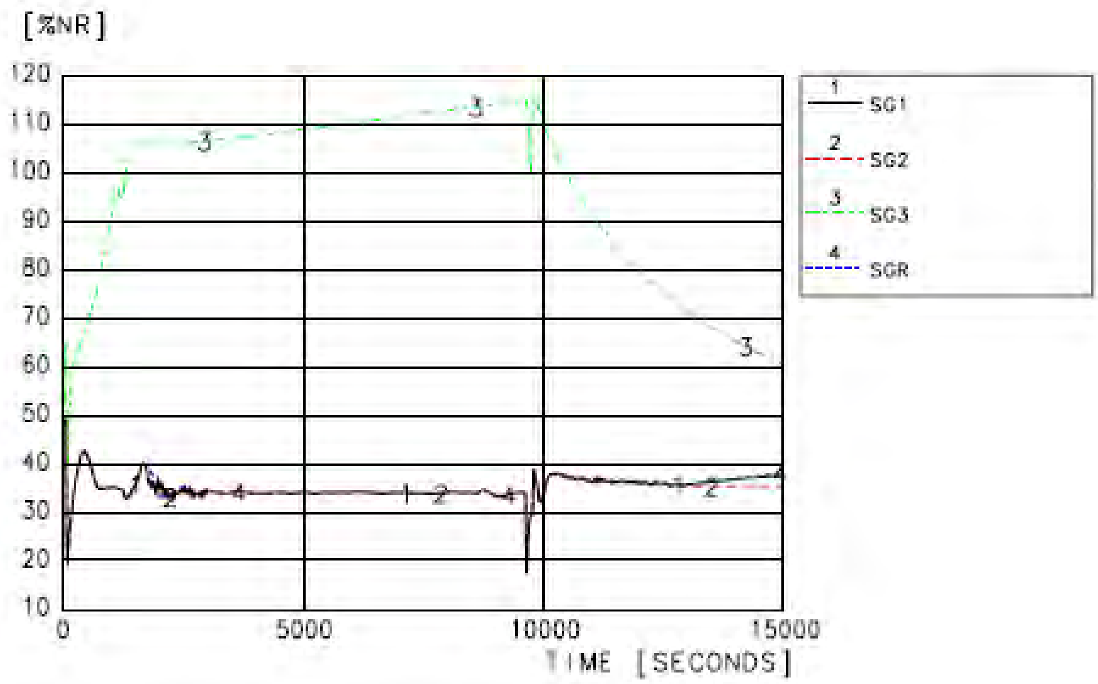
SECTION 16.4.5 - FIGURE 12

Main steam relief train mass flow rates



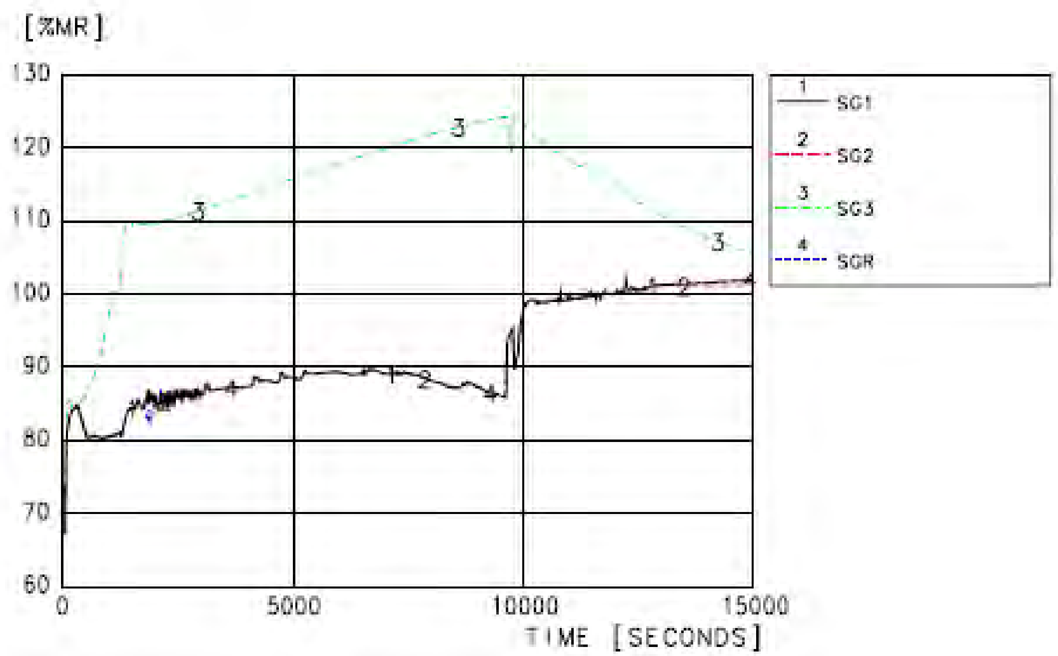
SECTION 16.4.5 - FIGURE 13

Steam generator narrow range level



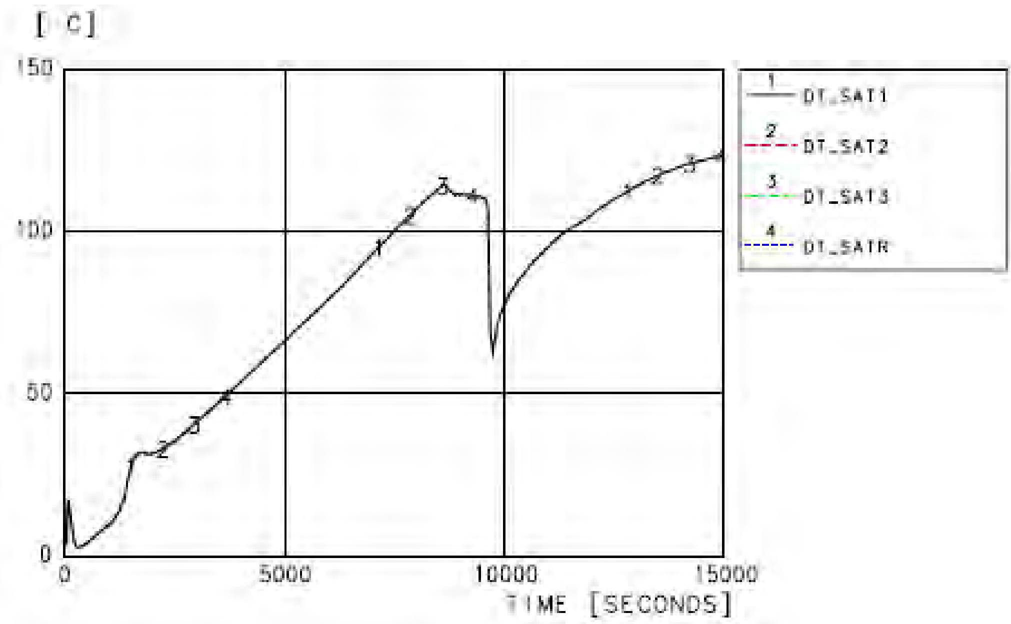
SECTION 16.4.5 - FIGURE 14

Steam generator wide range level



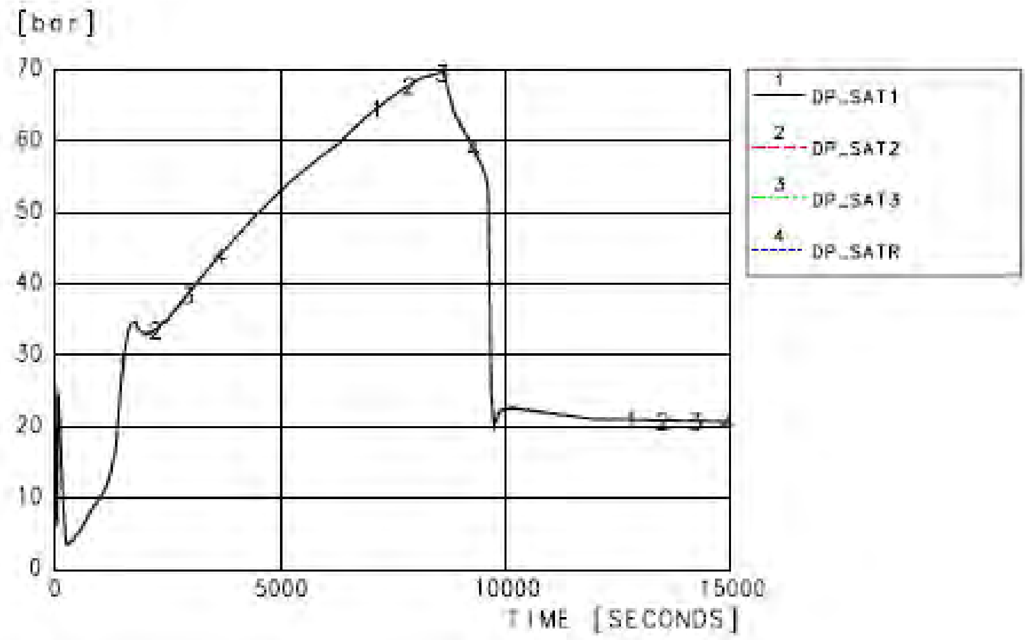
SECTION 16.4.5 - FIGURE 15

Saturation temperature margin



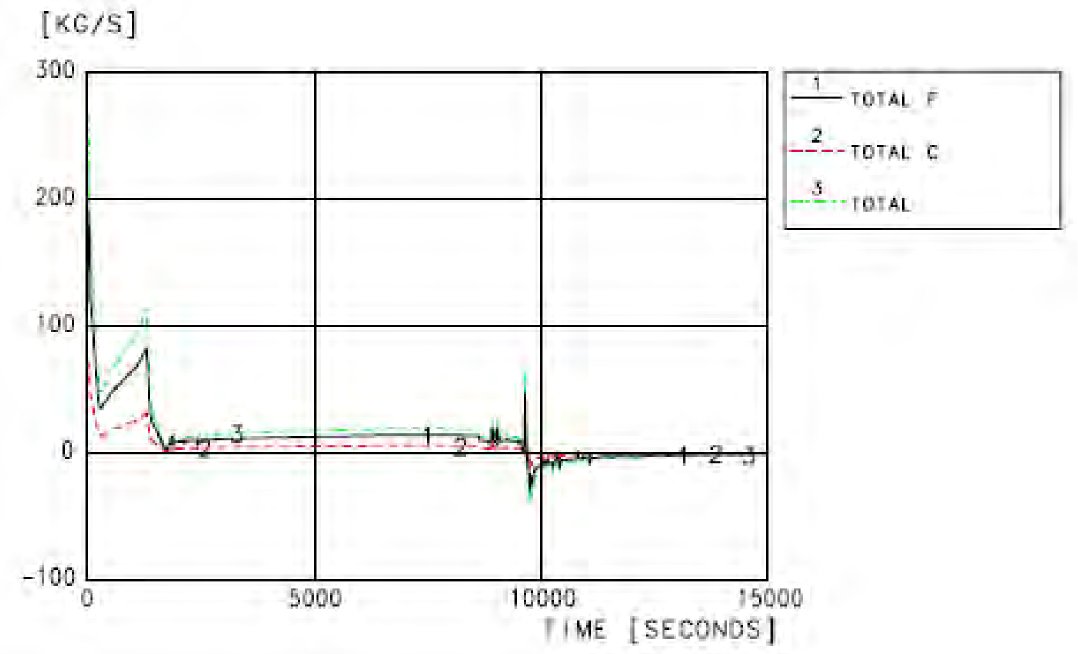
SECTION 16.4.5 - FIGURE 16

Saturation pressure margin



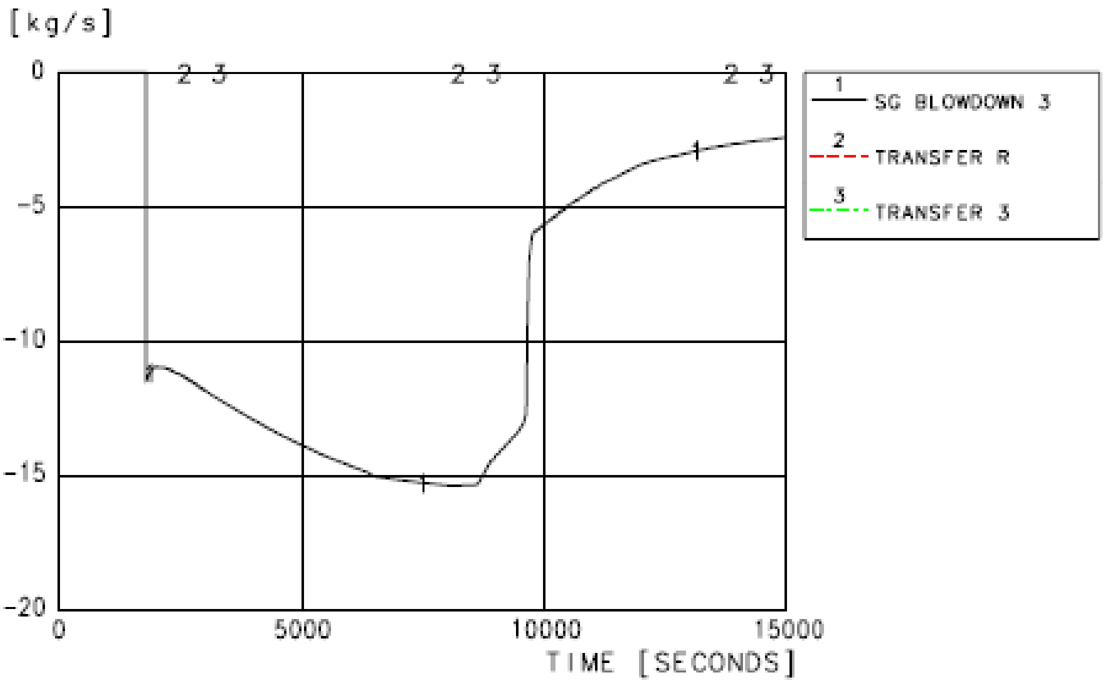
SECTION 16.4.5 - FIGURE 17

Steam generator tube rupture leak mass flow rates



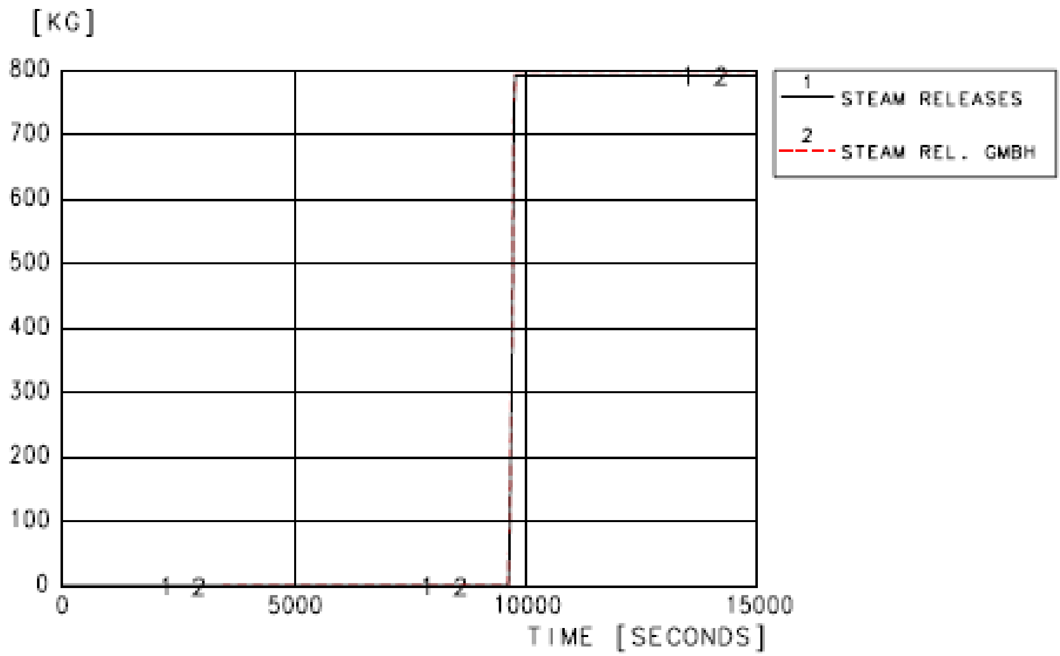
SECTION 16.4.5 - FIGURE 18

Blowdown line mass flow rates



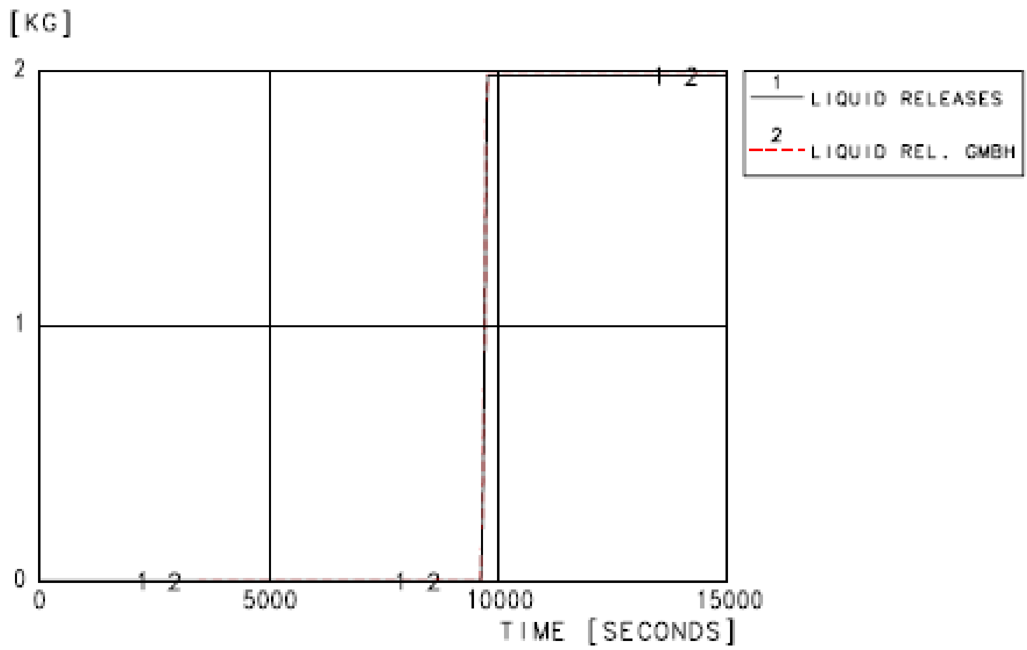
SECTION 16.4.5 - FIGURE 19

Primary release as steam



SECTION 16.4.5 - FIGURE 20

Primary release as liquid (humidity contained in steam)



6. SPURIOUS ACTUATION OF REACTOR PROTECTION SYSTEM

6.1. INTRODUCTION

The reactor Protection System (RPR [PS]) is the primary protection system claimed in PCC-2, PCC-3, PCC-4 and certain RRC-A events: it contributes to the basic safety functions of reactivity control, residual heat removal and limitation of radioactivity release. The RPR [PS] performs automatic reactor shutdown and actuates the safeguard systems, e.g. Emergency Feedwater System (ASG [EFWS]), Safety Injection System (RIS [SIS]).

Failure of the RPR [PS] may lead to spurious actuation of a safeguard system.

In considering possible spurious actuation of a safeguard system by the RPR [PS], the RPR [PS] is considered unavailable to provide accident mitigation, as an aggravating factor, since the event is initiated by an RPR [PS] failure. The Safety Automation System (SAS) functions are considered available to provide protection as a diversified means of fault mitigation.

The analysis of RPR [PS] functions presented in PCSR Sub-chapter 7.3, shows that some of the functions could not initiate an accident sequence if they were to be actuated by a spurious signal, and would therefore be unable to challenge the normal operating condition of the plant. Other RPR [PS] functions are identified whose spurious actuation could lead to an accident condition. Both classes of function are discussed below.

6.2. RPR [PS] FUNCTIONS WHOSE SPURIOUS ACTUATION DOES NOT LEAD TO AN ACCIDENT

The following RPR [PS] functions are identified as functions which cannot initiate an accident sequence if triggered spuriously.

ASG [EFWS] isolation

Under normal operating conditions, the emergency feedwater system (ASG [EFWS]) is not actuated. Therefore, spurious isolation of this safeguard system cannot initiate an accident sequence.

VDA [MSRT] isolation

Under normal operating conditions, the main steam relief trains (VDA [MSRT]) are not required to open. Therefore, spurious isolation of this safeguard system cannot initiate an accident sequence.

VDA [MSRT] setpoint increase

Under normal operating conditions, the VDA [MSRT] are not required to open. Therefore, a spurious increase of the opening setpoint of this safeguard system cannot initiate an accident sequence.

6.3. IDENTIFICATION OF EVENTS POTENTIALLY INITIATED BY SPURIOUS ACTUATION OF RPR [PS] FUNCTIONS

The RPR [PS} functions which could potentially initiate an accident are described below. In some cases, no actions would be required to mitigate the event. In some other cases, spurious actuation would lead to consequences covered by anticipated transient without scram (ATWS) events with loss of the RPR [PS], for which diversity pf protection is demonstrated in PCSR Sub-chapter 16.5. In the remaining cases diverse functions in the SAS are available to mitigate the accident sequence. Details are provided below.

6.4. EVENTS INVOLVING DECREASE IN HEAT REMOVAL

Reactor Trip (RT)

Spurious reactor trip is covered by the ATWS events with loss of the RPR [PS] presented in PCSR Sub-chapter 16.5, where diversity is demonstrated. The most representative case is the ATWS event (with loss of the RPR [PS]) with loss of main feedwater (ARE [MFWS]).

Turbine Trip

Spurious turbine trip is covered by the Loss of Off-site Power ATWS event (with loss of the RPR [PS]) presented in PCSR Sub-chapter 16.5, where diversity is demonstrated.

In the Loss Of Off-site Power (LOOP) ATWS event (with loss of the RPR [PS]), the turbine is expected to be tripped at the same time as the initiating event. Delays in safeguard actions are maximised to take account of the loss of off-site power, allowing for the fact that emergency diesel generators (EDGs) will be actuated first.

The RCP [RCS] pumps are tripped as a result of the initiating event, resulting in decreased primary/secondary heat transfer and consequently an increase in core temperature.

Therefore the LOOP ATWS event is assumed to bound turbine trip with loss of the RPR [PS].

VIV [MSIV] closure

Spurious closure of one main steam isolation valve (VIV [MSIV]) is an event resulting in a decrease in heat removal. Only three steam generators (SG) are available since the fourth SG is isolated by the spurious closure signal. The pressure in the affected SG reaches the opening setpoint of the main steam safety valve (MSSV). The unaffected SGs compensate for the reduced heat removal capacity, and the water inventory in these SGs decreases in the same way as in a loss of main feedwater (ARE [MFWS]) event. Reactor trip is triggered on the diverse low SG level trip (wide range (WR)).

Regarding SG overfeed protection, the diverse main feedwater (ARE [MFWS]) high load and low load isolation function (on SG level > MAX WR) prevents filling of the SGs.

The accident sequence is bounded by ATWS (with loss of the RPR [PS]) combined with loss of main feedwater (ARE [MFWS]) presented in PCSR Sub-chapter 16.5, which involves a more extreme loss of heat removal capability.

Spurious closure of the four VIV [MSIVs] combined with failure of the RPR [PS] will lead to pressure increases on both the primary and secondary systems, which will be controlled by the primary and secondary safety valves and by the diverse reactor trip on high hot leg pressure which occurs before the opening setpoint of the pressuriser safety valve (PSV) is reached. The safety margins with regard to the over-pressure criteria in the case of closure of all four VIV [MSIVs] are high, as can be seen by considering the extreme case of failure of all PSVs. With regard to core damage criteria, the consequences of spurious closure of the four VIV [MSIVs] are similar to those in a turbine trip event, since the GCT [MSB] is not credited in safety analysis. Therefore, 'Spurious actuation of the four VIV [MSIVs] with failure of the RPR [PS]' is covered by LOOP ATWS with loss of the RPR [PS] event presented in PCSR Sub-chapter 16.5, where diversity is demonstrated.

ARE [MFW] isolation

Spurious isolation of the main feedwater system (ARE [MFWS]) is an event resulting in a decrease in heat removal. The consequences are similar to those of the ATWS (with loss of the RPR [PS]) combined with loss of main feedwater (ARE [MWFS]) event presented in PCSR Sub-chapter 16.5 where diversity is demonstrated.

Trip of RCP [RCS] pumps

Spurious trip of RCP [RCS] pumps is covered by the LOOP ATWS with loss of the RPR [PS] event, presented in PCSR Sub-chapter 16.5, in which all the pumps are tripped as a result of the initiating event.

6.5. EVENTS INVOLVING INCREASE IN HEAT REMOVAL

LHSI/RHR train isolation

Spurious isolation of a LHSI/RHR train is analysed in states C3 and D when the RCP [RCS] coolant inventory is at its lowest level (¾ loop). In these states, three out of four LHSI/RHR trains are in operation (see PCSR Sub-chapter 14.0). As shown in PCSR Sub-chapter 14.3 section 17, two LHSI/RHR trains are sufficient to remove the residual heat from the core in these states: RCP [RCS] temperatures are maintained within a range which will allow continuous operation of the LHSI/RHR trains without additional counter-measures.

The RPR [PS] is not required to mitigate the accident. Therefore spurious isolation of a LHSI/RHR train is covered by analyses already presented in the PCSR.

RIS [SIS] actuation

Spurious RIS [SIS] actuation will directly trigger reactor trip, Partial Cooldown initiation, RIS [SIS] pumps actuation, and reactor coolant pressure boundary isolation. This event is considered to be an 'increase in heat removal' accident since the Partial Cooldown causes a secondary side cooldown with a rate of -250°C/h. Since the failure of the RPR [PS] is assumed, the Partial Cooldown can only be performed by the RCSL by means of the Main Steam By-pass. The isolation of the reactor coolant pressure boundary does not affect this event.

Reactor trip is triggered by the initiating event. The rod cluster control assembly (RCCA) drop keeps the core sub-critical during the entire transient. There is no risk of return to criticality since the cooling performed by the SGs terminates when the secondary pressure reaches 54.5 bar.

RIS [SIS] injection is initiated when the RCP [RCS] pressure falls below 85 bar. The primary system pressure stabilises at the MHSI shut-off injection pressure when the secondary system pressure stabilises at 54.5 bar at the end of the Partial Cooldown.

The controlled state is reached without the requirement to actuate any SAS functions.

Partial Cooldown

Spurious Partial Cooldown (PCD) actuation initiates an ‘increase in heat removal’ event. The PCD is performed at a rate of -250°C/h by the Main Steam By-pass as failure of the RPR [PS] is assumed. The PCD terminates when the secondary system pressure reaches 54.5 bar.

This event sequence is similar to that in excessive increase in steam flow with ATWS (with loss of the RPR [PS]) presented in PCSR Sub-chapter 16.5. The only difference in the sequences is that reactor trip is triggered earlier on the diverse SAS signal “high neutron flux (power range) > 120%FP”.

Following the reactor trip, RCCA drop ensures that the core remains sub-critical throughout the transient. There is no risk of return to criticality since the cooldown by the steam generators terminates when the secondary pressure reaches 54.5 bar. The RCP [RCS] pressure cannot fall below the diverse threshold “hot leg pressure (WR) < MIN3” at which the RIS [SIS] is actuated. RIS [SIS] injection is initiated when the RCP [RCS] pressure falls below 85 bar. The primary system pressure then stabilises at the MHSI pump shut-off injection pressure when the secondary system pressure is stabilised at 54.5 bar at the end of the PCD. The controlled state is reached utilising SAS diverse functions.

VDA [MSRT] opening

Spurious opening of a main steam relief train (VDA [MSRT]) initiates an increase in heat removal by the secondary system leading to a power increase. As a result of the VDA [MSRT] opening, the SG level drops until the diverse RT threshold on low Wide Range SG level is reached.

The sequence of events is similar to that in ATWS (with loss of the RPR [PS]) combined with excessive increase in steam flow presented in PCSR Sub-chapter 16.5. However an earlier RT signal is triggered since the VDA [MSRT] opening results in a more rapid decrease in level in the affected SG. The maximum core power is thus lower in this case. After the reactor trip occurs, the residual heat is removed by the open VDA [MSRT] or by the MSSVs with a lower pressure within the SGs. At the end of the grace period, the operator is assumed to isolates the affected SG locally.

Therefore this event is considered to be covered by the event: ATWS with loss of the RPR [PS] with an excessive increase in steam flow.

ASG [EFWS] actuation

Spurious actuation of the Emergency Feed-water System (ASG [EFWS]) in one or all SGs increases the feedwater flow rate. The bounding case is actuation of all four ASG [EFWS] trains since the RCP [RCS] cooldown rate is then most onerous. In the most onerous case (maximum ASG [EFWS] flow rate of 200 m^3/h (1bar) at a minimum temperature of 10°C - see PCSR Sub-chapter 14.1) the heat removal increases by 13%. Therefore, the reactor power may also increase by about 13% and stabilise at this power level. A controlled state is reached, and the operator is able to bring the plant to the safe shutdown state using post-accident safe state functions.

It is possible that the extra feedwater flow rate caused by the spurious ASG [EFWS] actuation could overfill the Steam Generators. This is prevented by SG overfeed protection triggered by the diversified MFW high load and low load isolation function (on “SG level > MAX WR”) which would prevent further filling of the SGs and terminate RCP [RCS] over-cooling.

With the ARE [MFWS] isolated, the ASG [EFWS] flow rate is insufficient to remove core residual heat, resulting in a sharp decrease in the SG water inventory. The event sequence is covered by the ATWS (with loss of the RPR [PS]) combined with loss of main feedwater (ARE [MFWS]) event presented in PCSR Sub-chapter 16.5.

6.6. DECREASE IN PRIMARY SYSTEM COOLANT INVENTORY

RCV [CVCS] isolation

Spurious isolation of the charging line of the chemical and volume control system (RCV [CVCS]) would lead to a decrease in RCP [RCS] coolant inventory. If the letdown flow is sufficiently high, the RCP [RCS] pressure may drop to the threshold “hot leg pressure (WR) < MIN2” which actuates a diversified reactor trip from the SAS: the pressure may then drop further to the threshold “hot leg pressure (WR) < MIN3” triggering the following diverse signals: RIS [SIS] actuation, partial cooldown (PCD) and Reactor Coolant Pressure Boundary (RCPB) isolation.

RCPB isolation closes the RCV [CVCS] letdown line cancelling the accident initiator. The SAS safeguard actions are thus sufficient to mitigate this event.

Since discharge of RCP [RCS] coolant is via the RCV [CVCS], there are no radiological releases into the containment.

In this transient, the RCP [RCS] pressure drop will not result in a challenge to the minimum DNBR. This protection aspect is bounded by the case of ATWS (with total loss of the RPR [PS]) combined with spurious spray actuation presented in PCSR Sub-chapter 16.5.

PSV opening

Spurious opening of a pressuriser safety valve (PSV) could initiate a LOCA event. Spurious opening of one PSV is covered by the small break LOCA (20 cm^2) since the PSV leak is located at the top of the pressuriser. A 20 cm^2 break at the bottom of the cold leg is considered more onerous than spurious PSV opening since lower enthalpy fluid is discharged. As steam is not released directly the depressurisation of the RCP [RCS] is slower, delaying the RIS [SIS] injection and resulting in a faster reduction in primary coolant inventory which in turn presents a greater challenge to core cooling.

Therefore, spurious PSV opening with failure of the RPR [PS] is bounded by the case of ATWS with small break LOCA (with failure of the RPR [PS]) presented in PCSR Sub-chapter 16.5.

Containment isolation

Spurious containment isolation triggers several actions. However, the only action which challenges the safety of the plant is the isolation of the letdown line of the RCV [CVCS], with the charging line remaining open. In this case, the RCP [RCS] coolant inventory, and the coolant level in the pressuriser would increase. This over-filling via the RCV [CVCS] could lead to an RCP [RCS] overpressure event.

Even with the additional failure of the RPR [PS], the RCP [RCS] is still adequately protected against over-pressure: reactor trip would be triggered on the diversified signal "High Wide Range hot leg pressure". The PSVs would open at their dedicated threshold (175 / 178 / 181 bar) releasing the excess primary coolant. There is no potential for radiological release outside containment since spurious containment isolation is the initiating event. The operator is able to isolate the charging line of the RCV [CVCS] with a dedicated SAS function to terminate the transient. The controlled state is thus reached by means of diverse SAS functions.

6.7. CONCLUSION

The above analysis confirms that events initiated by a spurious actuation of RPR [PS] functions combined with the total loss of the RPR [PS], can be mitigated in all cases by use of SAS diverse safety functions.

7. SAFETY CASE FOR HETEROGENEOUS BORON DILUTION FAULT

7.1. INTRODUCTION

In the UK EPR, as in other PWRs, reactivity of the fuel is compensated by addition of soluble boron to the primary coolant, which acts as a neutron absorber. Over a fuel cycle, excess reactivity progressively decreases with the increase in fuel burn-up, and the boron concentration in the primary circuit is reduced accordingly, reaching zero at the end of the cycle. Soluble boron is also used to compensate for the surplus reactivity generated when the plant is moved to cold shutdown conditions, to ensure an adequate margin of subcriticality.

Due to these fundamental PWR design characteristics, it is possible to generate a significant reactivity transient by inadvertently inserting deborated water into the core. Such a reactivity transient could, in principle, lead to a power excursion, potentially resulting in fuel damage and a radioactivity release to the environment.

In the safety analysis of hypothetical boron dilution in the UK EPR, different types of event are considered, namely:

- Homogeneous boron dilution events, where the boron concentration is approximately uniform throughout the RCP [RCS]. Homogeneous boron dilution transients normally result in moderate reactivity transients.

- Heterogeneous boron dilution events, where a water slug containing water at low or zero Boron Concentration (BC) is formed within the RCP [RCS] when the Main Coolant Pumps (MCPs) are temporarily stopped. Restart of the MCPs results in transport of the low BC slug into the core causing a reactivity excursion.

Heterogeneous boron dilution events are subdivided further into external and intrinsic dilution events. In external events, water of low or zero Boron Concentration (BC) is accidentally injected into a main coolant loop from an externally connected system. In intrinsic dilution events, low BC water forms in the RCP [RCS] by condensation or evaporation processes within the RCP [RCS] (e.g. by condensation of steam in SBLOCA events).

The safety analysis of external heterogeneous dilution events in the UK EPR ([Ref-1] to [Ref-5]) is performed in three steps as follows:

- Thermal-hydraulic and neutronic calculations are performed to define the maximum volume of a pure water slug (critical slug volume) that could be injected into the reactor core without threatening the subcriticality margin.

- The critical slug volume is used to define prevention and mitigation measures which will avoid such a slug being injected in credible dilution scenarios.

- Probabilistic analysis is performed to confirm that the design of prevention and mitigation measures implemented in the design will ensure that the risk of injection of a slug above the critical size is negligible (i.e. ensuring that reactivity excursions due to external heterogeneous dilution events are "practically eliminated").

For the UK EPR, a safety case has been produced for external heterogeneous boron dilution faults that does not rely on probabilistic analysis to “practically eliminate” reactivity excursions due to external heterogeneous dilution events. Instead the postulated faults are grouped systematically according to their frequency and unmitigated consequences. The protection measures against each fault group are identified and shown to be adequate to meet UK requirements for diversity, redundancy and safety classification. Validation evidence has also been provided for the CFD modelling of slug transport in the RCP [RCS], to show that adequate margins have been allowed for in the calculation of the critical slug size to take into account modelling uncertainties.

This section presents the safety case for external heterogeneous boron dilution faults in the UK EPR. The safety case is organised as follows:

a) The maximum acceptable volume of an unborated water slug that could be injected into the RCP [RCS] without risk of a reactivity excursion is first derived, by considering a range of injection scenarios in the most adverse plant operating state. Validation evidence for the CFD methods used to derive the critical slug size is presented, and it is shown that the derived value of 2 m^3 contains large safety margins (see Section 16.4.7 – Appendix 1 at the end of this section).

b) For each system connected to the RCP [RCS], credible scenarios are identified which, if unmitigated, could result in injection of a slug of low BC water (above the critical size) into the RCP [RCS]. An initiating event frequency is derived for each credible scenario, taking into account preventive measures such as administrative controls, automatic control system actions and alarms.

c) The initiating events are grouped into a small number of Bounding Initiating Events (BIE), depending on their frequency and unmitigated consequences. For each BIE, protection functions are identified which either prevent the slug entering the RCP [RCS], or prevent the slug entering the core on restart of the MCPs.

d) Improvement options are proposed to address shortfalls against UK requirements for diversity, redundancy and safety classification of the identified protection functions. An assessment is presented which confirms that the proposed modifications reduce the risk due to external heterogeneous dilution events to ALARP.

7.2. MAXIMUM ACCEPTABLE SLUG SIZE

7.2.1. Assessment of maximum acceptable slug size

A large number of analyses were carried out using the CFD model STAR-CD [Ref-1] to determine the transient boron concentration at the reactor core inlet following restart of an MCP in a main coolant loop containing a hypothetical unborated water slug. The purpose of the analysis is to model mixing of a slug of pure water with the borated water in the loop pipework, downcomer and lower plenum as it is transported to the core. Results are used to check if the boron concentration at the core inlet could fall to a level where there was a possibility of core criticality, and hence to determine a “critical slug size” (defined as the volume of largest pure water slug which would not present a significant risk of core re-criticality). Calculations are performed assuming the most conservative transient flow conditions, namely initial start-up of an MCP in the loop containing the unborated water slug.

Calculations consider a 2 m^3 slug at three possible locations in a MCP inlet U-leg (adjacent to the MCP inlet, centre of U-leg, adjacent to the SG outlet). A 4 m^3 slug in the U-leg is also considered as a sensitivity study. MCP start-up transients are modelled corresponding to three pump start-up times between 15 and 40 seconds. The flow boundary conditions at the inlet to the cold leg downstream of the operational MCP, and the reverse flow boundary conditions at the inlets to the cold legs of the other loops, are obtained from calculations using the MANTA [Ref-2] system code.

The initial conditions in the RPV and main coolant loops at start-up of the MCP are assumed to be 24.5 bar and 55°C. In practice, the initial boron concentration on start-up of the reactor will depend on the fuel management that is implemented. For the STAR-CD calculations, the fuel management with the highest critical boron concentration is chosen as the bounding condition (initial boron concentration = 2293 ppm). In this core state the critical boron concentration (i.e. the minimum concentration required to prevent criticality is 1300 ppm (=1200 ppm + 100 ppm uncertainty allowance).

The STAR-CD model contains approximately 1.9 million cells and represents the RPV, four cold legs, downcomer, lower plenum, core support plate, flow distribution device and half the height of the total core. The modelling and input assumptions are based on those used to simulate tests in the JULIETTE mock-up, which were used to validate the STAR-CD model for this application (see section 7.2.2 below).

The success criterion was that the minimum boron concentration anywhere across the core inlet during the transient should be above the critical concentration. It should be noted that this criterion is conservative as if the critical boron concentration is only reached locally, this does not necessarily imply re-criticality of the core or a risk of core damage.

Results of the calculations showed that, for the cases with a 2 m^3 water slug, the minimum local boron concentration at the core inlet was 1602 ppm. For the sensitivity calculation with the 4 m^3 slug, the minimum local boron concentration at the core inlet was 1474 ppm. These results show that, with a slug size below a critical value of 2 m^3, the risk of re-criticality should be avoided with a substantial margin of safety. Even with a doubling of the slug size to 4 m^3, a significant margin to criticality is achieved. Hence, there is confidence that accidental introduction of a water slug below 2 m^3 during shutdown would not lead to a significant risk of a criticality accident.

The methods used to derive the critical slug size presented above show that the derived value of 2 m^3 contains large safety margins. A quantification of this safety margin is given in Section 16.4.7 – Appendix 1, at the end of this section.

7.2.2. Validation of CFD Modelling

Validation studies of STAR-CD [Ref-1] and its associated numerical model demonstrate the ability of the model to simulate boron dilution transients, and in particular external dilution events. The validation tests are performed in the JULIETTE mock-up, which is representative of the EPR geometry for the lower internals at 1/5 scale. The validation study focuses on two external dilution tests which use different slug sizes, plug kinetics and test conditions. It is concluded that the CFD modelling of external dilution transients in the EPR reactor pressure vessel is likely to be conservative (in terms of minimum boron concentration at core inlet), based on the JULIETTE mock-up tests results.

The JULIETTE mock-up design and instrumentation should allow an accurate representation of dilution transients in the EPR reactor and a precise simulation of the slug behaviour and mixing inside the reactor vessel.

7.3. ANALYSIS OF POTENTIAL BORON DILUTION FAULTS AND PROTECTION MEASURES

The current section reviews potential heterogeneous external dilution scenarios that might arise due to injection of low BC water by systems connected to the RCP [RCS] when the MCPs are stopped. The faults are grouped into a small number of Bounding Initiating Events (BIEs) for which initiating frequencies are derived using conservative assumptions for system failure frequencies and human performance. These BIEs are events which, if unmitigated, could result in injection of an unborated water slug with a volume greater than 2 m^3 into the core on restart of MCPs.

The mitigation measures available to protect against slug injection into the core are reviewed for each of the BIEs. Improvement options are proposed to address shortfalls against UK requirements for diversity, redundancy and safety classification of protection features.

7.3.1. Review of Dilution Scenarios

For scenarios in which malfunctions occur in external systems connected to the RCP [RCS], which could lead to possible injection of an unacceptable slug of unborated water into the reactor core on start-up of MCPs, the main connected systems of concern are:

- Chemical and Volume Control System (RCV [CVCS]) and connected systems (Reactor Boron and Water Make-up System (REA [RBWMS]), Coolant Treatment, Purification and Gasification System (TEP [CSTS]) etc)
- Safety Injection System (RIS [SIS]) (including the LHSI, MHSI and Accumulator Systems)
- Extra Borating System (RBS [EBS])
- Nuclear Sampling System (REN [NSS])
- Nuclear Vent Drain System (RPE [NVDS])
- Component Cooling Water System (RRI [CCWS])
- Steam Generators (SG)

Section 16.4.7 - Figure 1 shows the connections of these systems to the RCP [RCS].

The external boron dilution scenarios can be divided into 3 groups.

a) Dilution scenarios in which the MCPs are intentionally shutdown (reactor States C3, D, E) (44 scenarios)

b) Dilution scenarios in which the MCPs are shutdown due to a Loss of Offsite Power (LOOP) (45 scenarios)

c) Dilution scenarios in which the MCPs are shutdown due to a Total Loss of Heat Sink (TLOCC) (45 scenarios)

Some scenarios are discounted on the basis that they are physically impossible, or that the scenario is enveloped by another scenario with a much higher frequency of occurrence.

Section 16.4.7 - Table 1 presents a schedule which summarises the boron dilution scenarios showing the postulated initial failures and the preventive measures available to prevent the initial failure from escalating into a dilution event that would challenge the protection systems.

The schedule in Section 16.4.7 - Table 1 also contains estimates of the frequency of the initial malfunction and the unreliability of the preventive measures. The Initiating Event (IE) frequency for a dilution event which challenges the protection systems is defined as the product of the initial frequency of the malfunction and the unreliability of all the available independent preventive measures.

The proposed analysis is not intended to replace or be a substitute for the existing PSA studies on external boron dilution scenarios. In particular, conservative values are proposed for probabilities of failures in order to give illustrative figures for the risk of external dilution faults, suitable for a deterministic assessment of protection requirements[1]. This conservative assessment is an artefact aimed at:

- screening the different Potential Initiating Events,
- grouping all the Potential Initiating Events,
- giving a simple quantitative assessment of the of the fault frequency in order to assess the potential need for a design improvement.

Note that, due to the conservative and simplistic derivation of the BIE frequencies, it cannot be concluded that infrequent BIEs with frequencies close to the Frequent-Infrequent boundary (10^{-3}/yr) will necessarily require two diverse protection systems to comply with UK EPR requirements in Sub-chapter 3.2, unless the BIE frequency can be confirmed by PSA analysis.

The values given for event frequency and unreliability are based on simple conservative assumptions as follows:

- The probability of failure of a pre-accident human action controlled by a procedure is taken as 0.03 fpd which is the value used in the Human Reliability Model applied for the UK EPR PSA, discounting any reduction due to possible operator error recovery (which is intrinsically extremely conservative).
- The probability of failure of a control system or an alarm is assumed to be dominated by the failure probability of the digital I&C. A failure probability of 0.1 fpd is conservatively used.
- The frequency of a significant failure of a heat exchanger tube or pump seal is taken as 0.01/yr. The failure frequency during a shutdown state when the reactor is depressurised is obtained by multiplying the annual failure rate by the fraction of time the reactor is in a depressurised shutdown state (0.03).
- The frequency of a LOOP coinciding with REA [RBWMS] injecting low BC water into the RCV [CVCS] is assumed to be 10^{-3}/yr.
- The frequency of a TLOCC coinciding with REA [RBWMS] injecting low BC water into the RCV [CVCS] is assumed to be $1E^{-5}$/yr.

[1] Note that the reliabilities used are particularly conservative regarding human factor and PSA claims.

7.3.2. Review of Protection Functions

The following protection systems are provided in the UK EPR design to prevent a slug of borated water from entering the core on start-up of an MCP.

7.3.2.1. Protection against Boron Dilution

In many cases, following a system malfunction, multiple preventive measures would be available to help prevent unborated water being transferred to the RCP [RCS]. Such measures include administrative controls (such as routine sampling of the BC in tanks or pipework), automatic control system actions to prevent the delivery of unborated water by a support system, and alarms to alert the operators to the equipment malfunction. Only if all preventive measures fail (i.e. the initial system malfunction is undetected) can the scenario lead, (if unmitigated by the action of protection functions), to an unborated water slug entering the reactor core.

7.3.2.1.1. Boron meters downstream of Charging Pumps due to faults in RCV [CVCS]

If water at an abnormally low BC is delivered to the RCV [CVCS] charging pump suction lines due to a system malfunction, when the MCPs are stopped, the low BC would be detected by four Class 1 boron meters in the charging pump discharge line. On detection of the low BC, the suction of the charging pumps would be automatically realigned to the IRWST, and the normal suction line and the letdown line automatically isolated, by a Class 1 protection system. Therefore automatic Class 1 protection is available to prevent water at low BC from entering the RCP [RCS] via the normal make-up routes.

7.3.2.1.2. MCP start-up procedure

If the Class 1 protection system in section 7.3.2.1.1 failed and a low BC water slug was to enter the RCP [RCS] from the RCV [CVCS] via the normal make-up routes, the unborated water would not necessarily reach the reactor core. Additional protection is provided by the procedure followed to start up the MCPs. The procedure is as follows:

a) Following filling of the RCP [RCS], the RCV [CVCS] is operated for an extended period to remove or reduce the size of any slugs of low BC water that might be located in the Loop 1 cross-over leg or the Loop 1 SG, via the letdown line connected to the Loop 1 cross-over leg. Water removed from Loop 1 will flow though RCV [CVCS] charging line and diluted slug would be detected by boron meter. It will be otherwise re-injected into all four loops via the make-up lines and the MCP seal injection lines, where it will be mixed with a much larger volume of borated water.

b) MCP 1 is then started up. This establishes a reverse flow in Loops 2, 3 and 4 that is about ten times smaller than the flow in Loop 1. The reverse flow will mix with the forward flow in the upper plenum. In addition to the path to the core being much longer in the reverse flow loops, any unborated water slug in these loops would mix with a much larger volume of borated water before reaching the core than would be the case if there was forward flow, and mixing would be enhanced by the vessel internals. These physical phenomena will ensure that any slug located in Loops 2, 3 and 4 will be sufficiently well mixed to avoid a reactivity excursion.

c) MCPs 2, 3 and 4 are started up.

7.3.3. Fault and Protection Schedule

Using the schedule of external boron dilution scenarios in Section 16.4.7 - Table 1 and the protection functions identified in section 7.3.2, it is possible to combine similar scenarios for which the available protection measures are similar into a small number of BIEs. The frequency of the BIEs is the combined frequency of the scenarios which they envelop.

This bounding process results in the five BIEs shown in Section 16.4.7 - Table 2. The protection functions available against each BIE are indicated:

7.3.4. Review of Prevention/Protection Functions and Proposed Improvements

7.3.4.1. Review of current protection functions

Inspection of Section 16.4.7 - Table 2 indicates that the protection functions available against external heterogeneous dilution events have some shortfalls with respect to UK EPR requirements.

In particular, it is notable that the only protection function available against Bounding Initiating Events 2 to 5 is the MCP start-up procedure described in section 7.3.2.1.2, which is designed to prevent an unborated water slug from entering the core. UK EPR safety requirements in Sub-chapter 3.2 request that at least one Class 1 protection function is available to mitigate against all credible initiating events. It is not clear that the manual start-up procedure implemented by administrative controls can be claimed as equivalent to a Class 1 protection function or that it can be performed with the required level of reliability.

Therefore a study was performed to investigate if it would be practicable to implement a Class 1 automatic protection function instead of the manual procedure currently used. The proposed modification is described below.

7.3.4.2. Proposed modification of prevention and protection functions

7.3.4.2.1. Design improvements

Some practicable procedural measures, which could be implemented to reduce the risk of a low BC slug forming due to a tube rupture in a heat exchanger cooled by the RRI [CCWS], are identified. The following design changes will be implemented to reinforce the EPR design regarding preventive isolation of heat exchangers in shutdown states.

- Isolation of the RCV [CVCS] letdown line HP cooler: In addition to the existing local manual heat exchanger isolation, the operator will also close the containment isolation valves from the main control room. New alarms would alert the operator to non-closure of the valves.

- Isolation of the RIS [SIS] / RRA [LHSI/RHRS] pump mechanical seal heat exchanger: The current isolation will be performed automatically when the RRA [RHRS] pump is stopped in order to rely less on manual operator action. An alarm would alert the operator to non-closure of the valve.

- Isolation of the RCP [RCS] pump thermal barrier: In addition to the existing automatic isolation, the operator will also close the containment isolation valves from the main control room.

7.3.4.2.2. Procedure improvements

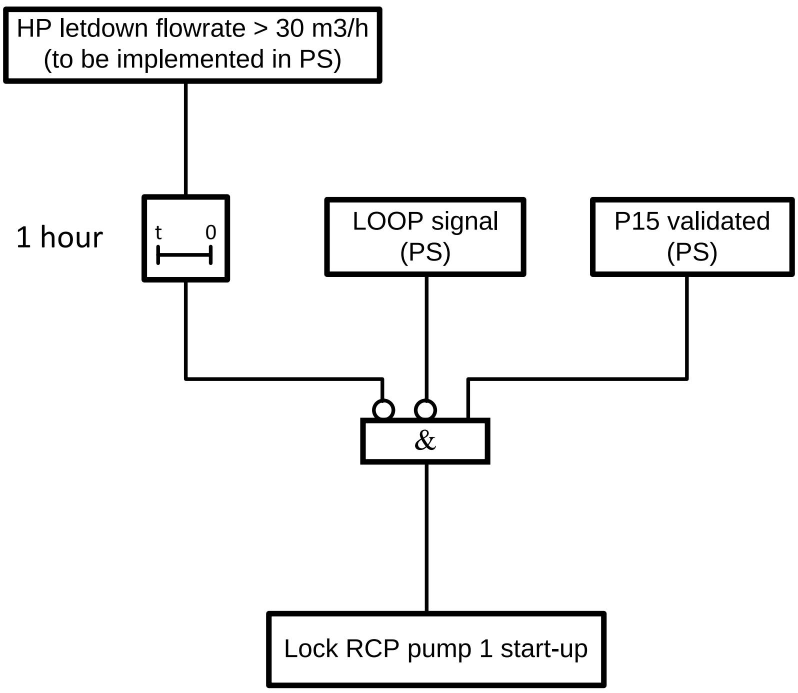
The proposed modification is to create an interlock which will prevent the operator from starting MCP n°1 if the RCV [CVCS] letdown has not run sufficiently to clear unborated water slugs for Loop 1. A further interlock is proposed to prevent restart of MCPs 2, 3 and 4 until MCP 1 has run for a sufficient time to clear any unborated water slugs from the other three loops.

7.3.4.2.2.1. Interlocks for MCP pump n°1 start-up

One hour running of the letdown system with 36 m^3/h flow rate (RCV [CVCS] normal letdown flow rate) would allow displacement of a volume of primary coolant equivalent to the volume of one half of the {CCI Removed} from Loop 1. If a slug is located in the cross-over leg or cold leg, which is the most likely location, it would be eliminated or substantially reduced in size. This automatic procedure is considered the most effective Class 1 means of supplementing the existing RCV [CVCS] Class 1 protection. The procedure significantly reduces the risk of scenarios involving injection of large low BC slugs.

Increasing the length of the delay period beyond one hour would increase the effectiveness of the slug removal. However, after one hour, the gain is limited, and further extensions cannot be demonstrated to give further meaningful improvement in the removal of an unborated water slug from Loop 1. Extending the delay would also begin to impact the overall start-up schedule and hence plant availability, without producing a demonstrable benefit.

The figure below summarises the interlocks for MCP n°1 start-up.



Interlocks on MCP pump n°1 Start-up

Other features of the proposed protection system as are follows:

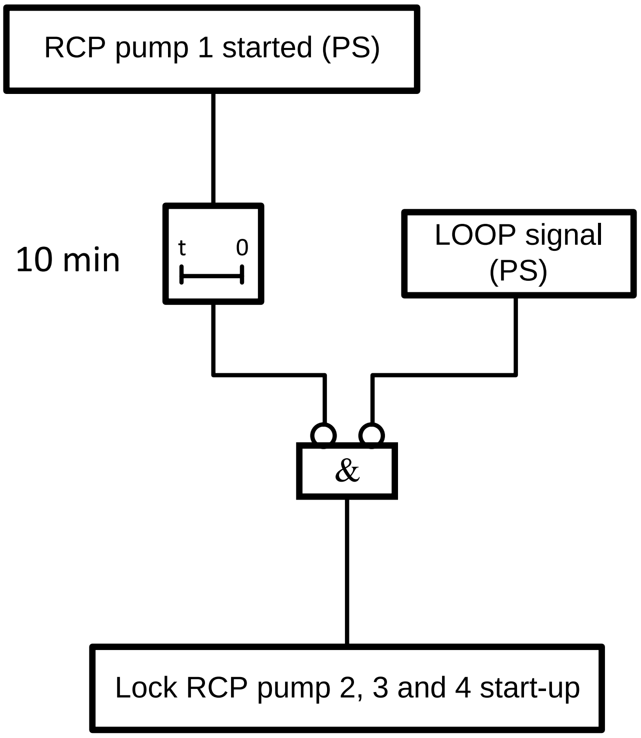
- Anti-dilution signal interlock: an interlock would be implemented so that if a dilution is detected by RCV [CVCS] boron meters, the MCPs could not be started up until the dilution causes are cleared,
- P15 permissive: if any MCP pump had already run, there is no risk of a heterogeneous dilution event due the start-up of MCP 1. MCP 1 can hence be restarted,
- If MCP 1 was stopped because of a Loss Of Offsite Power, the operator shall not be prevented from restarting it.

7.3.4.2.2.2. Interlocks for MCP 2, 3, and 4 start-up

Once MCP 1 has been started, reverse flow is quickly established in Loops 2, 3, and 4 (nominal flow rate reached in less than one minute). Any unborated water slug in these loops will be rapidly mixed with borated water in the RCP [RCS].

A delay of 2 minutes between MCP 1 start-up and start-up of the other MCPs is imposed as this should be sufficient to clear any slugs of unborated water without impacting plant availability.

The figure below summarises the interlocks for MCP 2, 3 and 4 start-up.



Interlocks on MCP pump n°2, 3, and 4 Start-up

Other features of the protection system as are follows:

- Anti-dilution signal interlock: if a dilution is detected by RCV [CVCS] boron meters, the MCPs cannot be started up until the dilution causes are cleared,

- If the MCP n°1 was stopped because of a Loss Of Offsite Power, the operator shall not be prevented from restarting it.

7.4. ALARP REVIEW

It is seen from Section 16.4.7 - Table 2 that implementation of a Class 1 interlock to prevent start up of MCPs until unborated water slugs have been cleared from the loops ensures that at least one Class 1 protection function is available against all credible external heterogeneous boron dilution faults and two Class 1 protection functions are available against ‘frequent’ boron dilution events (f > 10^{-3}/yr).

Due to the reliability of the Class 1 protection functions, fault sequences resulting in potential reactivity excursion are adequately protected.

The total frequency of uncontrolled boron dilution events due to all scenarios is negligible, even considering the highly pessimistic assumptions used to derive the frequencies of the Bounding Initiating Events in Section 16.4.7 - Table 3. Hence the risk due to uncontrolled external heterogeneous boron dilution is considered insignificant and to be reduced as low as reasonably practicable.

It is noted that the two Class 1 protection functions are not fully independent as they are each implemented within the Protection System (RPR [PS]) within the TXS digital platform. Hence both protection functions would be unavailable in the event of failure of the RPR [PS]. However, in practice this is not of concern since MCP start-up would not be carried out if the RPR [PS] was unavailable.

As UK EPR safety principles are fully satisfied in respect of external heterogeneous boron dilution faults, and the risk to nuclear safety from such events is reduced to an insignificant level, no further measures to achieve ALARP are considered necessary.

7.5. CONCLUSIONS

EPR prevention and mitigation measures against external heterogeneous boron dilution have been analysed:

- The initiating events potentially leading to external boron dilution faults have been analysed and bounding initiating events have been identified;

- The current protection measures against these faults and proposed design improvements are considered to achieve an ALARP position in respect of EPR boron dilution faults.

Prevention measures against boron dilution faults play a significant role in reducing the risk of such faults. It is intended to review prevention measures again in the Nuclear Site Licensing phase of the UK EPR, to confirm that no further reduction in the risk from dilution events is reasonably practicable.

The design improvements proposed in the current safety case will be implemented through a dedicated Change Management Form.

SECTION 16.4.7 – TABLE 1

Summary and Analysis of External Heterogeneous Boron Dilution Scenarios

Scenario	Scenario Description (Causal event frequency/yr)	Causal Event Freq (/yr)	Prevention Function P1 (Unreliability fpd)	Prevention Function P2 (Unreliability fpd)	Prevention Function P3 (Unreliability fpd)	Prevention Function P4	Init Event Freq (/r.y.) (FMEA)
	FAULT SEQUENCES WHERE MCPs ARE SHUTDOWN IN NORMAL OPERATIONAL STATE (REACTOR STATES C, D, E)						
S1	Injection of unborated water by RCV [CVCS] due to operator setting incorrect BC set-point in REA [RBWMS]. (3E-2 for operator error)	<<3.00E-02	Control system (PAS) automatically adjust BC to normal level (1E-1)	RCSL Isolates Demin Water supply to REA [RBWMS] on detection of low BC in injection flow (Class 2) (1E-2)			<<3.00E-05
S2	Injection of unborated water by RCV [CVCS] due control system malfunction (PAS) adjusting BC to below normal value for refuelling (1E-1 for PAS failure)	<<1.00E-01	Alarm in control room on low BC delivered by REA [RBWMS] (1E-1)	RCSL Isolates Demin Water supply to REA [RBWMS] on detection of low BC in injection flow (Class 2) (1E-2)			<<1.00E-04
S3A	Injection of unborated water by RCV [CVCS] due to control system malfunction resulting in BC REA [RBWMS] Buffer Tank (6110) being too low (1E-1 for PAS control system error)	<<1.00E-01	Sampling of buffer tank contents before transfer to storage Tanks 1110/2110. Sampling of contents of storage tanks before injection (1e-2)	Alarm in control room on low BC delivered to buffer tank detected by 1oo2 boron meters (1E-2)			<<1.0E-05
S3B/C	Injection of unborated water by RCV [CVCS] due to filling error resulting in too low BC in Buffer Tank or Mixing Tank (3E-2 for operator error)	<<3.00E-02	Sampling of buffer tank contents before transfer to storage Tanks 1110/2110. Sampling of contents of storage tanks before injection (3e-2)				<<9.00E-04
S4	Injection of unborated water into RCV [CVCS] due to tube rupture in REA [RBWMS] injection recirculation cooler while MCPs shutdown (3E-4 for tube leakage with reactor in S/D state)	<<3.00E-04					<<3.00E-04
S5	Injection of unborated water by RCV [CVCS] due to tube rupture in RCV [CVCS] HP cooler while MCPs shutdown (3E-4 for tube leakage in reactor S/D state)	<<3.00E-04	HP coolers isolated by manual closure of valves 1231/1241/1238/1248 when MCPs are stopped. (1E-2)				<<3.00E-06
S6	Injection of unborated water by RCV [CVCS] due to failure of RCV [CVCS] charging pump seal resulting in injection of Demin water from SED (3E-4 for failure of seal in S/D state)	<<3.00E-04	Alarm in control room on high seal water flow (1E-1)	Alarm in control room on low seal injection pressure (1E-1)			<<3.00E-06
S7	Accidental injection of unborated water from REN [NSS]	No connection of REN [NSS] to significant water source					
S8	Injection of unborated water by RCV [CVCS] due to tube rupture in REN [NSS] heat exchanger	REN [NSS] aligned to RIS system under cold shutdown conditions. Leakage would be into REN [NSS]					

Scenario	Scenario Description (Causal event frequency/yr)	Causal Event Freq (/yr)	Prevention Function P1 (Unreliability fpd)	Prevention Function P2 (Unreliability fpd)	Prevention Function P3 (Unreliability fpd)	Prevention Function P4	Init Event Freq (/r.y.) (FMEA)
S9	Injection of unborated water by RCV [CVCS] due to failure to isolate resin tank from Demin water supply after resin replacement (3E-2 for initial operator error to isolate tank)	<<3.00E-02	Operator would observe filling of the faulty demineraliser (3E-2)				<<9.00E-04
S10	Injection of unborated water by RCV [CVCS] due operator ignoring instruction to avoid reconnecting demineraliser after resin replacement while MCP pumps stopped (1E-2)	<<1.00E-02	Demineraliser output sampling to confirm BC suitable for reconnection (1E-2)				<<1.00E-04
S11	Injection of unborated water by RCV [CVCS] due to temperature reduction in letdown flow causing reduction in BC at outlet of demineraliser (1E-2)	<<1.00E-02	Demineraliser automatically bypassed on detection of inlet low temperature (PAS control) (1E-1)				<<1.00E-03
S12	Injection of unborated water by RCV [CVCS] due to tube rupture in degasifier condenser during operation of degasifier (100h/yr) (1E-4 for tube rupture during short operating period)	<<1.00E-04	Leak detection by water level in degasifier column. Degasifier automatically isolated (PAS control) (1E-1)				<<1.00E-05
S13	Injection of unborated water by RCV [CVCS] due to operator reconnecting degasifier without previously draining (3E-2 for operator error)	<<3.00E-02					<<3.00E-02
S14	Injection of unborated water by RCV [CVCS] due to leakage of Demin water from TEG {GWPS] into gas compressor seals	Not possible to inject unborated water into RCV [CVCS]					
S15	Injection of unborated water by RCV [CVCS] due to flushing of chemical additive tanks and pipework with Demin water	Injected water volumes too small to cause significant dilution of RCP [RCS]					
S17	Injection of unborated water by RCV [CVCS] due to Demin water being left in RCV [CVCS] systems after maintenance, due to failure of administrative controls	<<3.00E-02					<<3.00E-02
S18	Injection of unborated water from RHR system due to tube leakage in RHR heat exchanger when RHR/LHSI pump not operating (1E-2 for tube leakage frequency)	<<1.00E-02	Isolation of LHSI Heat exchanger (Valve 1420) when LHSI pumps stopped prevents formation of unborated water plug in RHR pipework (3E-2)	Start up procedure involves RHR pipework being initially flushed with IRWST water by operation of RHR pumps before connection to RCP [RCS] (3E-2)	RHR heat exchanger water box sampling is performed to ensure BC normal carried out before connection (3E-2)	Radioactivity monitor on RRI [CCWS] would detect RHR leakage on start-up of RHR pump in recirc mode before connection to RCP [RCS] (1E-1)	<<3.00E-08
S19	Injection of unborated water from RHR system due to tube rupture in LHSI pump seal cooling heat exchanger when LHSI pump is not operating (1E-2 for tube leakage frequency)	<<1.00E-02	Isolation of LHSI Heat exchanger (Valve 1237) when LHSI pumps stopped prevents formation of unborated water plug in RHR pipework (3E-2)	Start up procedure involves RHR pipework being initially flushed with IRWST water by operation of RHR pumps before connection to RCP [RCS] (3E-2)	RHR heat exchanger water box sampling is performed to ensure BC normal carried out before connection (3e-2)		<<3.00E-07

Scenario	Scenario Description (Causal event frequency/yr)	Causal Event Freq (/yr)	Prevention Function P1 (Unreliability fpd)	Prevention Function P2 (Unreliability fpd)	Prevention Function P3 (Unreliability fpd)	Prevention Function P4	Init Event Freq (/r.y.) (FMEA)
S20	Leakage of primary coolant into RHR system during normal operation creates low BC slug in RHR which is injected on connection of RHR system. (1E0 for operation of RHR in S/D state).	<<1.00E+00	2 Class 1 isolation valves prevent leakage of primary coolant into LHSI pipework (1E-4)	MCR Alarm (PAS) on detection of high P, T in pipework between isolation valves. (1E-1)	Start up procedure involves RHR pipework being initially flushed with IRWST water by operation of RHR pumps before connection to RCP [RCS] (1E-2)	RHR heat exchanger water box sampling is performed to ensure BC normal carried out before connection (1e-2)	<<1.00E-09
S21	Leakage of Demin water from REN [NSS] into RHR system	Risk excluded. REN [NSS] not connected to significant Demin water sources					
S22	Tube leakage in REN [NSS] heat exchanger results in backflow of unborated water into depressurised RHR	Risk excluded due to the presence of multiple check valves and normal closed isolation valves, preventing backflow.					
S24	Injection of unborated water from RHR system due to unborated water being left in RHR circuit following maintenance due to failure of administrative procedures (1E0 for operation of RHR in S/D state)	<<1.00E+00	Administrative procedures to avoid unborated water being left in pipework (3E-2)	Start up procedure involves RHR pipework being initially flushed with IRWST water by operation of RHR pumps before connection to RCP [RCS] (3E-2)	RHR heat exchanger water box sampling is performed to ensure BC normal carried out before connection (3E-2)		<<2.70E-05
S25	Leakage of primary coolant into MHSI pipework system during normal operation creates low BC slug which is injected on accidental/spurious operation of MHSI with MCPs stopped. (1E-4 for operation of MHSI pumps in S/D state).	<<1.00E-04	2 Class 1 isolation valves prevent leakage of primary coolant into MHSI pipework (1E-4)	MCR Alarm (PAS) on detection of high P, T in pipework between isolation valves. (1E-1)			<<1.00E-09
S27	Injection of unborated water from MHSI system due to unborated water being left in circuit following maintenance due to failure of administrative procedures (1E-4 for spurious or intentional operation of MHSI in S/D state)	<<1.00E-04	Administrative procedures to avoid unborated water being left in pipework (3e-2)				<<3.00E-06
S28	Injection of unborated water from Accumulator system due to unborated water being diverted into Accumulator following maintenance operations on MHSI pipework or incorrect BC in IRWST (1E-1 for test of Accumulator in S/D state once every 10 years)	<<1.00E-01	Checks to avoid unborated water being left in pipework following maintenance on MHSI + Checks on IRWST BC (9E-4)	Boron sampling of accumulators before injection into RCP [RCS] (1E-2)			<<1.00E-06
S29	Leakage of primary coolant into accumulator during normal operation creates low BC slug in accumulator which is injected on intentional/spurious operation of accumulator with MCPs stopped. (1E-1 for test of Accumulator in S/D state once every 10 years)	<<1.00E-01	2 Class 1 isolation valves prevent leakage of primary coolant into MHSI pipework (1E-4)	MCR Alarm (PAS) on detection of high P, T in pipework between isolation valves. (1E-1)	Significant leakage into accumulator would cause alarm on high accumulator pressure (1e-2)	Boron sampling of accumulators before injection into RCP [RCS] (3E-2)	<<3.00E-10
S30	Injection of unborated water following maintenance operations followed by accidental or intentional operation of accumulator (1E-1 for test of Accumulator in S/D state once every 10 years)	<<1.00E-01	Administrative procedures to avoid unborated water being left in pipework (3E-2)	Boron sampling of accumulators before injection into RCP [RCS] (3E-2)			<<9.00E-05

Scenario	Scenario Description (Causal event frequency/yr)	Causal Event Freq (/yr)	Prevention Function P1 (Unreliability fpd)	Prevention Function P2 (Unreliability fpd)	Prevention Function P3 (Unreliability fpd)	Prevention Function P4	Init Event Freq (/r.y.) (FMEA)
S31	Injection of unborated water by RBS [EBS] due to water with too low BC being transferred from REA [RBWMS] Mixing Tank (1110) to RBS [EBS] storage tanks (1140/4140). (3e-2 for filling error in REA [RBWMS] mixing tank)	<<3.00E-02	Sampling of mixing tank contents before transfer to storage Tanks. Sampling of contents of storage tanks before injection (9E-4)				<<3.00E-05
S32	Leakage of primary coolant into RBS [EBS] pipework during normal operation creates low BC slug in RBS [EBS] pipework which is injected intentional/spurious operation of the RBS [EBS] with MCPs stopped. (1E-2 for operation of EBS in S/D state)	Consequences insignificant as accident analysis of transients involving RBS [EBS] assume RBS [EBS] pipework primed with unborated water					
S33	Leakage of unborated cooling water into RBS [EBS] pumps	N/A as RBS [EBS] pumps are air-cooled					
S34	Injection of unborated water into RBS [EBS] pipework in maintenance operations followed by accidental or intentional operation of RBS [EBS]	Consequences insignificant as accident analysis of transients involving RBS [EBS] assume RBS [EBS] pipework primed with unborated water					
S35	Unborated water introduced into RCP [RCS] in 10 yearly hydrotest.	Scenario excluded as RCP [RCS] is filled with borated water from the IRWST and VCT during hydrotest					
S36	Borated water injected into loops by backflow from REN [NSS] sampling lines	Scenario excluded as no fixed connections exist from the demineralised water system to the sampling pipes.					
S37	Leakage of RRI [CCWS] cooling water into a REN [NSS] heat exchanger when RCP [RCS] is depressurised causes unborated water to enter loops via REN [NSS] sampling lines (2E-4 for failure of REN [NSS] heat exchanger tubes in S/D state)	<<2.00E-04	RCP [RCS] sampling points isolated by 3 series motorised valves in S/D condition when RCP [RCS] pressure is below RRI [CCWS] pressure. (1E-2 for failure of operators to perform isolation)	REN [NSS] boron meter would indicate presence of low BC water (1E-1)			<<2.00E-07
S38	Unborated water accidentally transferred from PRT to Pressuriser due to over-filling of PRT	Scenario excluded as Pressuriser isolated from PRT during filling of the PRT. Also make-up to PRT would be automatically isolated on high PRT level.					
S39	Leakage of unborated water from RRI [CCWS] into thermal barrier of an MCP during shutdown. (2E-4 for failure of thermal barrier in S/D state)	<<2.00E-04	Thermal barrier automatically isolated when MCPs are shutdown when RCP [RCS] pressure falls below set point level (above RRI [CCWS] pressure) (1E-1)	Thermal barrier automatically isolated on detection of high flow (1E-1)			<<2.00E-06
S40	Leakage of unborated water in RCP [RCS] due to SG tube failure in a secondary system hydrostatic pressure test	Scenario excluded as hydrostatic pressure test conducted with MCL loops drained and SG waterboxes open and under visual observation by operators. Undetected leakage into open loops could not occur.					
S41	Leakage of unborated water from secondary side due to faulty plugging of failed SG tube (1E-2)	<<1.00E-02	Leak would be detected as SG refilling following SG tube repair conducted with MCL loops drained and SG waterboxes open and under visual observation by operators. (3E-2)				<<3.00E-04

Scenario	Scenario Description (Causal event frequency/yr)	Causal Event Freq (/yr)	Prevention Function P1 (Unreliability fpd)	Prevention Function P2 (Unreliability fpd)	Prevention Function P3 (Unreliability fpd)	Prevention Function P4	Init Event Freq (/r.y.) (FMEA)
S42	Accidental dilution of water in reactor pool when reactor is open	Scenario excluded. Not conceivable that an undetected leakage could occur from a water system in the RB that was large enough to cause significant dilution of the water in the pool.					
S43	Unborated water introduced into RCP [RCS] by refilling of instrument impulse lines	Scenario excluded as procedures require filling of impulse lines with borated water. Even if procedure fails, volume of water in impulse lines would be too small to cause significant dilution.					
S44	Accidental addition of unborated water into RCP [RCS] during maintenance actions such as reactor pool cleaning when reactor is open. (2E-1 for frequency of maintenance with unborated water once every 5 years)	<<2.00E-01	Administrative procedures to avoid unborated water being left in pipework (3E-2)	Manual sampling of BC in RCP [RCS] carried out after refilling (3E-2)			<<1.80E-04
		FAULT SEQUENCES WHERE MCPs ARE SHUTDOWN DUE TO LOOP (REACTOR STATES A, B, C)					
S100	LOOP while REA [RBWMS] is injecting low BC water into RCV [CVCS] causes unborated water slug to form in RCV [CVCS] pipework which is injected into RCP [RCS] pipework on start-up of RCV [CVCS] charging pumps (1E-3 for LOOP event coinciding with REA [RBWMS] injecting low BC water into RCV [CVCS])	<<1.00E-03					<<1.00E-02
S101	Injection of unborated water by RCV [CVCS] due to operator setting incorrect BC set-point in REA [RBWMS] coincident with LOOP	Discounted as included in scenario S100					
S102	Injection of unborated water by RCV [CVCS] due control system malfunction (PAS) adjusting BC to low value coincident with LOOP.	Discounted as included in scenario S100					
S103	Injection of unborated water by RCV [CVCS] due to control system malfunction resulting in BC in REA [RBWMS] Buffer Tank or Mixing Tank being too low, coincident with LOOP.	Discounted as included in scenario S100					
S104	LOOP coinciding with injection of unborated water into RCV [CVCS] due to tube rupture in REA [RBWMS] injection recirculation cooler while MCPs shutdown	Discounted as protection similar to S3 and sequence frequency much lower					
S105	LOOP coinciding with injection of unborated water by RCV [CVCS] due to tube rupture in RCV [CVCS] HP cooler while MCPs shutdown	Scenario excluded as RCP [RCS] pressure exceeds RRI [CCWS] pressure in normal operation					
S106	LOOP coinciding with injection of unborated water by RCV [CVCS] due to failure of RCV [CVCS] charging pump seal resulting in injection of Demin water from SED (2E-4)	Discounted as protection similar to S4 and sequence frequency much lower					
S107	Accidental injection of unborated water from REN [NSS]	Discounted as no connection of REN [NSS] to significant water source					
S108	Injection of unborated water by RCV [CVCS] due to tube rupture in REN [NSS] heat exchanger	Scenario excluded as RCP [RCS] pressure exceeds RRI [CCWS] pressure in normal operation					

Scenario	Scenario Description (Causal event frequency/yr)	Causal Event Freq (/yr)	Prevention Function P1 (Unreliability fpd)	Prevention Function P2 (Unreliability fpd)	Prevention Function P3 (Unreliability fpd)	Prevention Function P4	Init Event Freq (/r.y.) (FMEA)
S109	Injection of unborated water by RCV [CVCS] due to failure to isolate resin tank from Demin water supply after resin replacement coincident with LOOP	Scenario excluded as connection of demineraliser not permitted during LOOP					
S110	Injection of unborated water by RCV [CVCS] due operator ignoring instruction to avoid reconnecting demineraliser after resin replacement while MCP pumps stopped (1E-2)	Scenario excluded as connection of demineraliser not permitted during LOOP					
S111	Injection of unborated water by RCV [CVCS] due to temperature reduction in letdown flow causing reduction in BC at outlet of demineraliser coinciding with LOOP (1E-4 for initiating event coinciding with LOOP)	Discounted as protection similar to S11 and sequence frequency much lower					
S112	Injection of unborated water by RCV [CVCS] due to tube rupture in degasifier condenser during operation of degasifier (100h/yr) coincident with LOOP	Scenario excluded as RCP [RCS] pressure exceeds RRI [CCWS] pressure in normal operation					
S113	Injection of unborated water by RCV [CVCS] due to operator reconnecting degasifier without previously draining coincident with LOOP	Scenario excluded as reconnection of degasifier not permitted after LOOP					
S114	Injection of unborated water by RCV [CVCS] due to leakage of Demin water from TEG {GWPS] into gas compressor seals coincident with LOOP	Discounted as protection similar to S14 and sequence frequency much lower					
S115	Injection of unborated water by RCV [CVCS] due to flushing of chemical additive tanks and pipework with Demin water coincident with LOOP	Discounted as protection similar to S15 and sequence frequency much lower					
S116	RCV [CVCS] deliberately or unintentionally realigned to inject water from IRWST which has incorrectly low BC coincident with LOOP	Discounted as protection similar to S16 and sequence frequency much lower					
S117	Injection of unborated water by RCV [CVCS] due to Demin water being left in RCV [CVCS] systems after maintenance, due to failure of administrative controls, followed by LOOP	Scenario excluded as MCPs have operated since maintenance					
S118	Injection of unborated water from RHR system due to tube leakage in RHR heat exchanger when RHR/LHSI pump not operating, coinciding with LOOP	Discounted as protection similar to S18 and sequence frequency much lower					
S119	Injection of unborated water from RHR system due to tube rupture in LHSI pump seal cooling heat exchanger when LHSI pump is not operating coinciding with LOOP	Discounted as protection similar to S18 and sequence frequency much lower					
S120	Leakage of primary coolant into RHR system during normal operation creates low BC slug in RHR which is injected on connection of RHR system coinciding with LOOP.	Discounted as protection similar to S18 and sequence frequency much lower					
S121	Leakage of Demin water from REN [NSS] into RHR system coinciding with LOOP	Discounted as protection similar to S21 and sequence frequency much lower					

Scenario	Scenario Description (Causal event frequency/yr)	Causal Event Freq (/yr)	Prevention Function P1 (Unreliability fpd)	Prevention Function P2 (Unreliability fpd)	Prevention Function P3 (Unreliability fpd)	Prevention Function P4	Init Event Freq (/r.y.) (FMEA)
S122	Tube leakage in REN [NSS] heat exchanger results in backflow of unborated water into depressurised RHR coinciding with LOOP	Discounted as protection similar to S22 and sequence frequency much lower					
S123	Injection of unborated water from RHR system due to water in IRWST having incorrectly low BC coinciding with LOOP	Discounted as protection similar to S23 and sequence frequency much lower					
S124	Injection of unborated water from RHR system due to unborated water being left in RHR circuit following maintenance due to failure of administrative procedures, coinciding with LOOP	Scenario excluded as MCPs have operated since maintenance					
S125	Leakage of primary coolant into MHSI pipework system during normal operation creates low BC slug which is injected on accidental/spurious operation of MHSI with MCPs stopped coinciding with LOOP	Discounted as protection similar to S25 and sequence frequency much lower					
S126	Injection of unborated water from MHSI system due to water in IRWST having incorrectly low BC coinciding with LOOP	Discounted as protection similar to S26 and sequence frequency much lower					
S127	Injection of unborated water from MHSI system due to unborated water being left in circuit following maintenance due to failure of administrative procedures coinciding with LOOP	Scenario excluded as MCPs have operated since maintenance					
S128	Injection of unborated water from Accumulator system due to unborated water being diverted into Accumulator following maintenance operations on MHSI pipework or incorrect BC in IRWST coinciding with LOOP	Discounted as protection similar to S28 and sequence frequency much lower					
S129	Leakage of primary coolant into accumulator during normal operation creates low BC slug in accumulator which is injected on intentional/spurious operation of accumulator with MCPs stopped due to LOOP	Discounted as protection similar to S29 and sequence frequency much lower					
S130	Injection of unborated water following maintenance operations followed by accidental or intentional operation of accumulator following LOOP	Scenario excluded as MCPs have operated since maintenance					
S131	Injection of unborated water by RBS [EBS] due to water with too low BC being transferred from REA [RBWMS] Mixing Tank (1110) to RBS [EBS] storage tanks (1140/4140) coinciding with LOOP	Discounted as protection similar to S29 and sequence frequency much lower					
S132	Leakage of primary coolant into RBS [EBS] pipework during normal operation creates low BC slug in RBS [EBS] pipework which is injected intentional/spurious operation of the RBS [EBS] with MCPs stopped. (1E-2 for operation of EBS in S/D state)	Discounted as protection similar to S29 and sequence frequency much lower					
S133	Leakage of unborated cooling water into RBS [EBS] pumps	Discounted as protection similar to S29 and sequence frequency much lower					

Scenario	Scenario Description (Causal event frequency/yr)	Causal Event Freq (/yr)	Prevention Function P1 (Unreliability fpd)	Prevention Function P2 (Unreliability fpd)	Prevention Function P3 (Unreliability fpd)	Prevention Function P4	Init Event Freq (/r.y.) (FMEA)
S134	Injection of unborated water into RBS [EBS] pipework in maintenance operations followed by accidental or intentional operation of RBS [EBS]	Scenario excluded as MCPs have operated since maintenance					
S135	Unborated water introduced into RCP [RCS] in 10 yearly hydrotest coincident with LOOP	Scenario excluded as hydrotest would not be carried out under LOOP conditions					
S136	Borated water injected into loops by backflow from REN [NSS] sampling lines coincident with LOOP	Scenario excluded as no fixed connections exist from the demineralised water system to the sampling pipes.					
S137	Leakage of RRI [CCWS] cooling water into a REN [NSS] heat exchanger when RCP [RCS] is depressurised causes unborated water to enter loops via REN [NSS] sampling lines coincident with LOOP	Scenario excluded as RCP [RCS] pressure above RRI [CCWS] pressure					
S138	Unborated water accidentally transferred from PRT to Pressuriser due to over-filling of PRT coincident with LOOP	Scenario excluded as RCP [RCS] is pressurised					
S139	Leakage of unborated water from RRI [CCWS] into thermal barrier of coincident with LOOP	Scenario excluded as RCP [RCS] pressure above RRI [CCWS] pressure					
S140	Leakage of unborated water in RCP [RCS] due to SG tube failure in a secondary system hydrostatic pressure test coincident with LOOP	Risk evaluated in SGTR safety case					
S141	Leakage of unborated water from secondary side due to faulty plugging of failed SG tube coincident with LOOP	Risk evaluated in SGTR safety case					
S142	Accidental dilution of water in reactor pool when reactor is open coincident with LOOP	Covered by S42					
S143	Unborated water introduced into RCP [RCS] by refilling of instrument impulse lines coincident with LOOP	Discounted as protection similar to S43 and sequence frequency much lower					
S144	Accidental addition of unborated water into RCP [RCS] during maintenance actions such as reactor pool cleaning when reactor is open, coincident with LOOP	Scenario excluded as MCPs have operated since maintenance					
		FAULT SEQUENCES WHERE MCPs ARE SHUTDOWN DUE TO TLOCC (REACTOR STATES A, B, C)					
S200	TLOCC occurs while REA [RBWMS] injects low BC water into RCV [CVCS]. Low BC slug injected into main coolant loops (1E-5 for TLOCC event while REA [RBWMS] injecting low BC water)	<<1.00E-05					<<1.00E-05
S201	Injection of unborated water by RCV [CVCS] due to operator setting incorrect BC set-point in REA [RBWMS] coincident with TLOCC	Discounted as included in scenario S200					
S202	Injection of unborated water by RCV [CVCS] due control system malfunction (PAS) adjusting BC to low value coincident with TLOCC.	Discounted as included in scenario S200					

Scenario	Scenario Description (Causal event frequency/yr)	Causal Event Freq (/yr)	Prevention Function P1 (Unreliability fpd)	Prevention Function P2 (Unreliability fpd)	Prevention Function P3 (Unreliability fpd)	Prevention Function P4	Init Event Freq (/r.y.) (FMEA)
S203	Injection of unborated water by RCV [CVCS] due to control system malfunction resulting in BC in REA [RBWMS] Buffer Tank or Mixing Tank being too low, coincident with TLOCC.	Discounted as included in scenario S200					
S204	TLOCC coinciding with injection of unborated water into RCV [CVCS] due to tube rupture in REA [RBWMS] injection recirculation cooler while MCPs shutdown	Discounted as protection similar to S3 and sequence frequency much lower					
S205	TLOCC coinciding with injection of unborated water by RCV [CVCS] due to tube rupture in RCV [CVCS] HP cooler while MCPs shutdown	Scenario excluded as RCP [RCS] pressure exceeds RRI [CCWS] pressure in normal operation					
S206	TLOCC coinciding with injection of unborated water by RCV [CVCS] due to failure of RCV [CVCS] charging pump seal resulting in injection of Demin water from	Discounted as protection similar to S4 and sequence frequency much lower					
S207	Accidental injection of unborated water from REN [NSS] coinciding with TLOCC	Discounted as no connection of REN [NSS] to significant water source					
S208	Injection of unborated water by RCV [CVCS] due to tube rupture in REN [NSS] heat exchanger coinciding with TLOCC	Scenario excluded as RCP [RCS] pressure exceeds RRI [CCWS] pressure in normal operation					
S209	Injection of unborated water by RCV [CVCS] due to failure to isolate resin tank from Demin water supply after resin replacement coincident with TLOCC	Scenario excluded as connection of demineraliser not permitted during LOOP					
S210	Injection of unborated water by RCV [CVCS] due operator ignoring instruction to avoid reconnecting demineraliser after resin replacement while MCP pumps stopped coincident with TLOCC	Scenario excluded as connection of demineraliser not permitted during LOOP					
S211	Injection of unborated water by RCV [CVCS] due to temperature reduction in letdown flow causing reduction in BC at outlet of demineraliser coinciding with TLOCC (1E-4 for initiating event coinciding with TLOCC)	Discounted as protection similar to S11 and sequence frequency much lower					
S212	Injection of unborated water by RCV [CVCS] due to tube rupture in degasifier condenser during operation of degasifier (100h/yr) coincident with TLOCC	Scenario excluded as RCP [RCS] pressure exceeds RRI [CCWS] pressure in normal operation					
S213	Injection of unborated water by RCV [CVCS] due to operator reconnecting degasifier without previously draining coincident with TLOCC	Scenario excluded as reconnection of degasifier not permitted after LOOP					
S214	Injection of unborated water by RCV [CVCS] due to leakage of Demin water from TEG {GWPS] into gas compressor seals coincident with TLOCC	Discounted as protection similar to S14 and sequence frequency much lower					
S215	Injection of unborated water by RCV [CVCS] due to flushing of chemical additive tanks and pipework with Demin water coincident with TLOCC	Discounted as protection similar to S15 and sequence frequency much lower					

Scenario	Scenario Description (Causal event frequency/yr)	Causal Event Freq (/yr)	Prevention Function P1 (Unreliability fpd)	Prevention Function P2 (Unreliability fpd)	Prevention Function P3 (Unreliability fpd)	Prevention Function P4	Init Event Freq (/r.y.) (FMEA)
S216	RCV [CVCS] deliberately or unintentionally realigned to inject water from IRWST which has incorrectly low BC coincident with TLOCC	Discounted as protection similar to S16 and sequence frequency much lower					
S217	Injection of unborated water by RCV [CVCS] due to Demin water being left in RCV [CVCS] systems after maintenance, due to failure of administrative controls, followed by TLOCC	Scenario excluded as MCPs have operated since maintenance					
S218	Injection of unborated water from RHR system due to tube leakage in RHR heat exchanger when RHR/LHSI pump not operating, coinciding with TLOCC	Discounted as protection similar to S18 and sequence frequency much lower					
S219	Injection of unborated water from RHR system due to tube rupture in LHSI pump seal cooling heat exchanger when LHSI pump is not operating coinciding with TLOCC	Discounted as protection similar to S18 and sequence frequency much lower					
S220	Leakage of primary coolant into RHR system during normal operation creates low BC slug in RHR which is injected on connection of RHR system coinciding with TLOCC.	Discounted as protection similar to S18 and sequence frequency much lower					
S221	Leakage of Demin water from REN [NSS] into RHR system coinciding with TLOCC	Discounted as protection similar to S21 and sequence frequency much lower					
S222	Tube leakage in REN [NSS] heat exchanger results in backflow of unborated water into depressurised RHR coinciding with TLOCC	Discounted as protection similar to S22 and sequence frequency much lower					
S223	Injection of unborated water from RHR system due to water in IRWST having incorrectly low BC coinciding with TLOCC	Discounted as protection similar to S23 and sequence frequency much lower					
S224	Injection of unborated water from RHR system due to unborated water being left in RHR circuit following maintenance due to failure of administrative procedures, coinciding with TLOCC	Scenario excluded as MCPs have operated since maintenance					
S225	Leakage of primary coolant into MHSI pipework system during normal operation creates low BC slug which is injected on accidental/spurious operation of MHSI coinciding with TLOCC	Discounted as protection similar to S25 and sequence frequency much lower					
S226	Injection of unborated water from MHSI system due to water in IRWST having incorrectly low BC coinciding with TLOCC	Discounted as protection similar to S26 and sequence frequency much lower					
S227	Injection of unborated water from MHSI system due to unborated water being left in circuit following maintenance due to failure of administrative procedures coinciding with TLOCC	Scenario excluded as MCPs have operated since maintenance					
S228	Injection of unborated water from Accumulator system due to unborated water being diverted into Accumulator following maintenance operations on MHSI pipework or incorrect BC in IRWST coinciding with TLOCC	Discounted as protection similar to S28 and sequence frequency much lower					

Scenario	Scenario Description (Causal event frequency/yr)	Causal Event Freq (/yr)	Prevention Function P1 (Unreliability fpd)	Prevention Function P2 (Unreliability fpd)	Prevention Function P3 (Unreliability fpd)	Prevention Function P4	Init Event Freq (/r.y.) (FMEA)
S229	Leakage of primary coolant into accumulator during normal operation creates low BC slug in accumulator which is injected on intentional/spurious operation of accumulator with MCPs stopped due to TLOCC	Discounted as protection similar to S29 and sequence frequency much lower					
S230	Injection of unborated water following maintenance operations followed by accidental or intentional operation of accumulator following TLOCC	Scenario excluded as MCPs have operated since maintenance					
S231	Injection of unborated water by RBS [EBS] due to water with too low BC being transferred from REA [RBWMS] Mixing Tank (1110) to RBS [EBS] storage tanks (1140/4140) coinciding with TLOCC	Discounted as protection similar to S29 and sequence frequency much lower					
S232	Leakage of primary coolant into RBS [EBS] pipework during normal operation creates low BC slug in RBS [EBS] pipework which is injected intentional/spurious operation of the RBS [EBS] with MCPs stopped.	Discounted as protection similar to S29 and sequence frequency much lower					
S233	Leakage of unborated cooling water into RBS [EBS] pumps	Discounted as protection similar to S29 and sequence frequency much lower					
S234	Injection of unborated water into RBS [EBS] pipework in maintenance operations followed by accidental or intentional operation of RBS [EBS]	Scenario excluded as MCPs have operated since maintenance					
S235	Unborated water introduced into RCP [RCS] in 10 yearly hydrotest coincident with TLOCC	Scenario excluded as hydrotest would not be carried out under LOOP conditions					
S236	Borated water injected into TLOCCs by backflow from REN [NSS] sampling lines coincident with TLOCC	Scenario excluded as no fixed connections exist from the demineralised water system to the sampling pipes.					
S237	Leakage of RRI [CCWS] cooling water into a REN [NSS] heat exchanger when RCP [RCS] is depressurised causes unborated water to enter TLOCCs via REN [NSS] sampling lines coincident with TLOCC	Scenario excluded as RCP [RCS] pressure above RRI [CCWS] pressure					
S238	Unborated water accidentally transferred from PRT to Pressuriser due to over-filling of PRT coincident with TLOCC	Scenario excluded as RCP [RCS] is pressurised					
S239	Leakage of unborated water from RRI [CCWS] into thermal barrier of coincident with TLOCC	Scenario excluded as RCP [RCS] pressure above RRI [CCWS] pressure					
S240	Leakage of unborated water in RCP [RCS] due to SG tube failure in a secondary system hydrostatic pressure test coincident with TLOCC	Risk evaluated in SGTR safety case					
S241	Leakage of unborated water from secondary side due to faulty plugging of failed SG tube coincident with TLOCC	Risk evaluated in SGTR safety case					
S242	Accidental dilution of water in reactor pool when reactor is open coincident with TLOCC	Covered by S42					
S243	Unborated water introduced into RCP [RCS] by refilling of instrument impulse lines coincident with TLOCC	Discounted as protection similar to S43 and sequence frequency much lower					

Scenario	Scenario Description (Causal event frequency/yr)	Causal Event Freq (/yr)	Prevention Function P1 (Unreliability fpd)	Prevention Function P2 (Unreliability fpd)	Prevention Function P3 (Unreliability fpd)	Prevention Function P4	Init Event Freq (/r.y.) (FMEA)
S244	Accidental addition of unborated water into RCP [RCS] during maintenance actions such as reactor pool cleaning when reactor is open, coincident with TLOCC	Scenario excluded as MCPs have operated since maintenance					

SECTION 16.4.7 – TABLE 2

Fault and Protection Schedule for External Heterogeneous Boron Dilution Faults

BIE No	Bounding Fault	Scenarios Enveloped	BIE illustrative bounding frequency/r.y	Protection/Mitigation M1	Protection/Mitigation M2
BIE1	Malfunction of RCV [CVCS] results in injection of low BC water into reactor loops with MCPs stopped due to normal operating or fault condition. (Injection via MCP seals or by normal makeup lines).This includes the failure of MCP due to LOOP or TLOC while REA [RBWMS] injecting low BC water.	S1-S4, S6-S17, S101-104,S106-117, S100,-S200, S201-204, S206-217	<<7.5E-02	Boron meters on RCV [CVCS] automatically realign RCV [CVCS] suction to IRWST on detection of low BC in injection (Class1 PS Action)	Operating procedures for MCP start up prevents MCP start-up until unborated water slugs have been removed from loops
BIE2	Failure of tubes in RCV [CVCS] Letdown Heat Exchanger with MCPs stopped normally or due to normal operating or fault condition. Injection of low BC water into Main Coolant Loop #1 via letdown line.	S5, S105, S205	<<3.0E-06	Operating procedures for MCP start up prevents MCP start-up until unborated water slugs have been removed from loops	
BIE3	Inadvertent injection of low BC water in MHSI/LHSI/RHR/Accumulator/RBS [EBS]/REN [NSS]/RRI [CCWS] pipework and subsequent injection into Main Coolant Loops with MCPs stopped due to normal operating or fault condition	S18-39, S118-139, S218-239	<<1.0E-04	Operating procedures for MCP start up prevents MCP start-up until unborated water slugs have been removed from loops	
BIE4	Leakage of secondary coolant into Main Coolant Loop following SG tube repair of SG hydrotest	S40-41, S140-S141, S240-241	<<1.0E-05	Operating procedures for MCP start up prevents MCP start-up until unborated water slugs have been removed from loops	
BIE5	Inadvertent introduction of low BC water into loop during RCP [RCS] refilling following maintenance outage	S42-44, S142-144, S242-244	<<5.0E-04	Operating procedures for MCP start up prevents MCP start-up until unborated water slugs have been removed from loops	

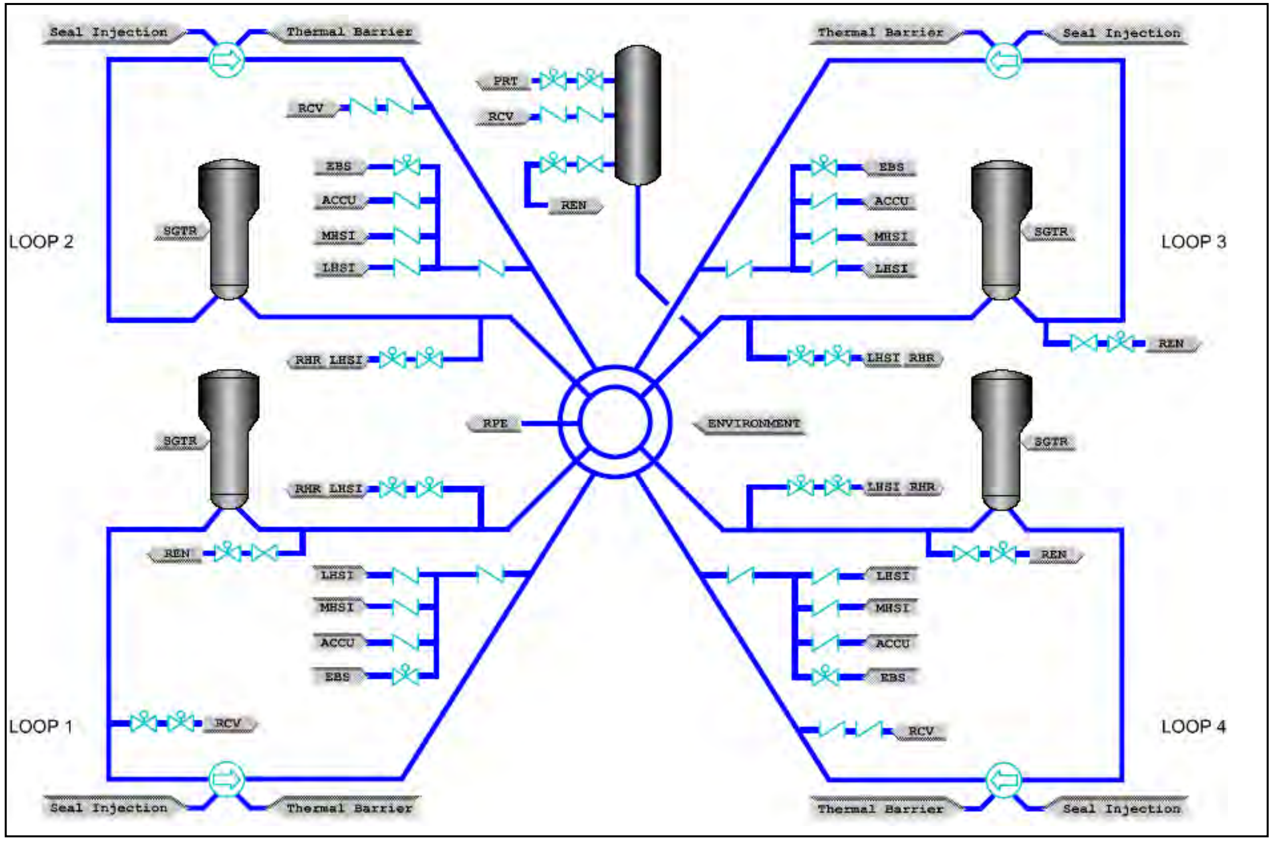
SECTION 16.4.7 – TABLE 3

Final Fault and Protection Schedule for External Heterogeneous Boron Dilution Faults

BF No	Bounding Fault	Scenarios Enveloped	BIE Illustrative bounding frequency /r.y	Protection/Mitigation M1	Protection/Mitigation M2
BF1	Malfunction of RCV [CVCS] Results in Injection of low BC water into reactor loops with MCPs stopped due to normal operating or fault condition. (Injection via MCP seals or by normal makeup lines). This includes the failure of MCP due to LOOP or TLOC while REA [RBWMS] injecting low BC water.	S1-S4, S6-S17, S101-104,S106-117, S201-204, S206-217	<<7.5E-02	Boron meters on RCV [CVCS] automatically realign RCV [CVCS] suction to IRWST on detection of low BC in injection (Class1 PS Action) (standard Class 1 reliability)	New Class 1 interlocks prevent MCP start-up until unborated water slugs have been removed from loops (standard Class 1 reliability)
BF2	Failure of tubes in RCV [CVCS] Letdown Heat Exchanger with MCPs stopped normally or due to normal operating or fault condition. Injection of low BC water into Main Coolant Loop #1 via letdown line.	S5, S105, S205	<<3.0E-06	New Class 1 interlocks prevent MCP start-up until unborated water slugs have been removed from loops (standard Class 1 reliability)	
BF3	Inadvertent injection of low BC water in MHSI/LHSI/RHR/Accumulator/RBS [EBS]/REN [NSS]/RRI [CCWS] pipework and subsequent injection into Main Coolant Loops with MCPs stopped due to normal operating or fault condition	S18-39, S118-139, S218-239	<<1.0E-04	New Class 1 interlocks prevent MCP start-up until unborated water slugs have been removed from loops (standard Class 1 reliability)	
BF4	Leakage of secondary coolant into Main Coolant Loop following SG tube repair of SG hydrotest	S40-41, S140-S141, S240-241	<<1.0E-05	New Class 1 interlocks prevent MCP start-up until unborated water slugs have been removed from loops (standard Class 1 reliability)	
BF5	Inadvertent introduction of low BC water into loop during RCP [RCS] refilling following maintenance outage	S42-44, S142-144, S242-244	<<5.0E-04	New Class 1 interlocks prevent MCP start-up until unborated water slugs have been removed from loops (standard Class 1 reliability)	

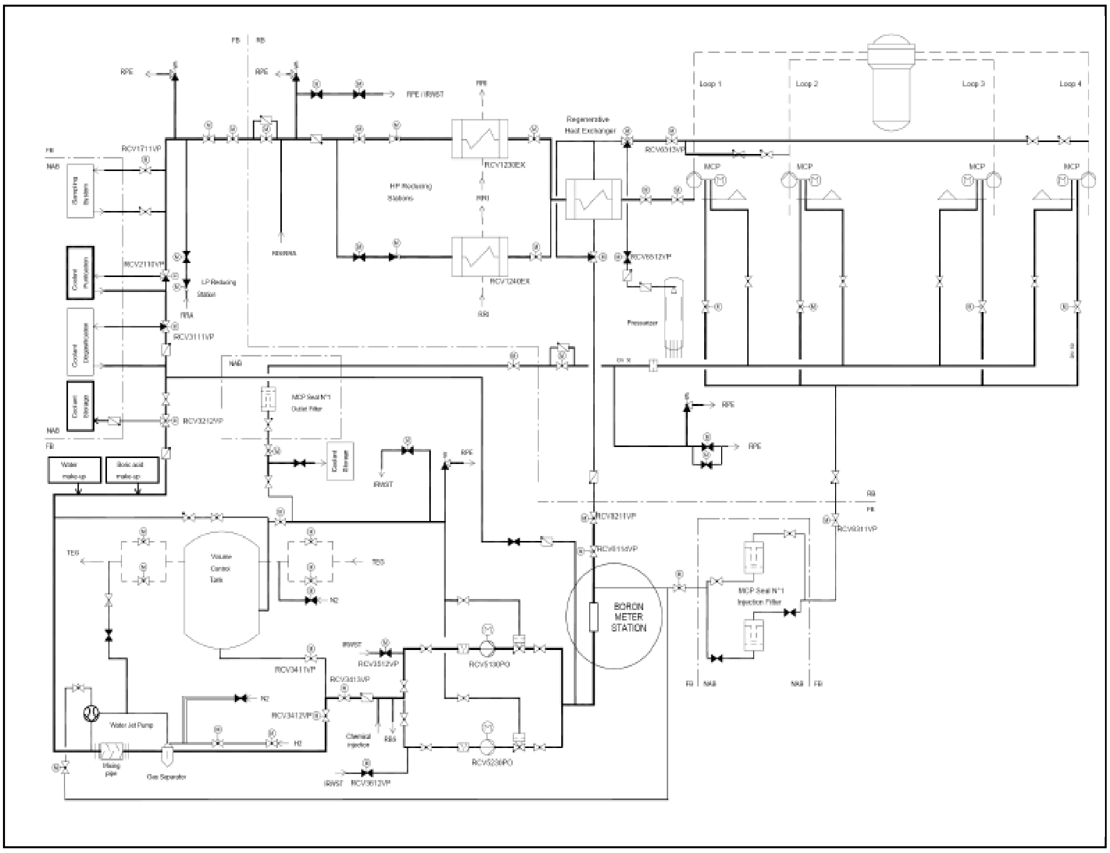
SECTION 16.4.7 – FIGURE 1

External systems connected to RCP [RCS]



SECTION 16.4.7 – FIGURE 2

UK EPR RCV [CVCS] System



SECTION 16.4.7 – APPENDIX 1

ILLUSTRATION OF SAFETY MARGINS ASSOCIATED WITH THE CRITICAL SLUG SIZE DETERMINATION

The methodology used to assess the critical plug size is based on a CFD analysis which derives the local minimum boron concentration at the core inlet, in order to ensure that this minimum boron concentration remains above the critical concentration. This criterion is highly conservative as, if the critical boron concentration is reached locally, this does not necessarily imply re-criticality of the core or a risk of core damage. The purpose of this appendix is to demonstrate the large safety margins associated with this methodology.

A sensitivity study with CFD analyses was performed to investigate the heterogeneous boron dilution for different volumes of the injected unborated water slug. This sensitivity study (range from 4 m^3 to 9 m^3 investigated) has concluded that, for an injected unborated water slug with a volume of 6 m^3, the minimum boron concentration at the core inlet is just slightly above the critical boron concentration (see Figure A1). The average boron concentration at the core inlet remains above the critical boron concentration for a slug size up to 9 m^3. In the figures the dashed horizontal line marks the critical boron concentration.

In conclusion, for water slug volumes greater than 6 m^3, the local minimum boron concentration is below the critical boron concentration and the non-return to criticality for these water slugs cannot be guaranteed by the decoupling approach.

A1. Coupling of core-physics and computational fluid dynamics

In order to obtain a quantitative estimate of the margins associated with the use of the critical boron concentration, additional sensitivity studies have been performed.

Their purpose is to use the minimum boron concentration given from STAR-CD and to perform a criticality calculation using the actual map of the boron distribution. The STAR-CD maps of minimum boron concentration at the core inlet are applied uniformly to all axial elevations of the 3D core model. This conservatism assumes no further mixing of the fluid as it passes through the core region. A static calculation of core criticality (using the SMART code) is then performed assuming the 3D distribution of boron concentrations.

The case of interest conservatively applies the minimum local boron concentration within the assembly to the entire assembly, at the worst transient time (when boron concentration in minimum).

Although the STAR-CD physical validation focuses on minimal boron concentration at core inlet (local value), as indicated above, the sensitivity performed here is used to provide a quantitative assessment of the margins associated to the use of the decoupled critical boron concentration.

a) Assumptions

The initial value of core BC for the homogeneous dilution scenario is taken as 2293 ppm. The temperature and pressure for all cases are constant at 55ºC and 24.5 bar (0.9867 g/cm^3).

The conditions of the calculation are the following:

- Zero power,
- Cycle: A4,

- Burnup: BLX (150 MWd/t),
- Zero Xenon,
- All rods in,
- 55°C and 24.5 bar.

The initial core boron concentration assumed in the STAR-CD calculation (2293 ppm) corresponds to the refuelling boron concentration required for the cycle with the highest critical boron concentration and includes an additional 180 ppm for overall uncertainties. Considering a pessimistic boron efficiency, this corresponds to 1385 pcm.

To be consistent with the pessimistic boron concentration used as an initial condition in the STAR-CD model, a corresponding penalty on the calculated reactivity is applied prior to comparison to the limit value of 1.0.

The reactivity (ρ = 1-1/keff) predicted by the SMART 3D static calculation is therefore pessimised by 1385 pcm.

b) Realistic fuel management scheme

When using the STAR-CD maps of minimum boron concentration as input data to the SMART 3D code, using the methodology described above, the maximum K-effective never exceeds 0.99 for all the configurations analysed.

A 9 m^3 slug of unborated water does not lead to any re-criticality.

c) Pessimised fuel management schemes

In order to provide some additional conservatism, the assembly-wise boron concentrations have been collapsed into six values of boron concentrations (called plateaus). The boron concentration applicable to each assembly is then decreased to the nearest plateau.

This leads to further pessimism of the core average boron concentration, although it has already been minimised as far as possible.

Such modelling is then used to locate the lowest boron concentration around the most reactive assemblies. The minimum boron concentration corresponding to the plateau values is placed on the assembly providing the peak power and then the BCs are distributed in concentric circles from low concentration to high concentration The below shows six regions, and all assemblies in the same coloured region are at a constant boron concentration (the plateau value) with the minimum boron concentration centred within the rings (indicated in orange):

This logically gives the most pessimistic results in terms of recriticality.

With such a bounding approach (ignoring the mixing at core inlet, assuming a very bounding distribution of boron concentrations and underestimating the mean core boron concentration), calculations performed still demonstrate that a 7 m^3 slug of unborated water does not lead to recriticality.

Non borated water slug volume (m^3)	Minimum BC (ppm)	K-Effective
Initial Value	2293	0.9146
4	1466	0.9268
5	1422	0.9543
6	1382	0.9582
7	1162	0.9726
8	861	0.9802
9	689	0.9896

TABLE A1: Results of 3D core calculations for realistic configurations

Non-borated water slug volume (m3)	Minimum BC (ppm)	Impact of Plateau on Average BC (ppm)	Configuration Concentric zones
			K-Effective
Initial Value	2293	0	0.9146
4	1450	-89	0.9562
7	1150	-185	0.9948

TABLE A2: Results of 3D core calculations for pessimised concentric configurations

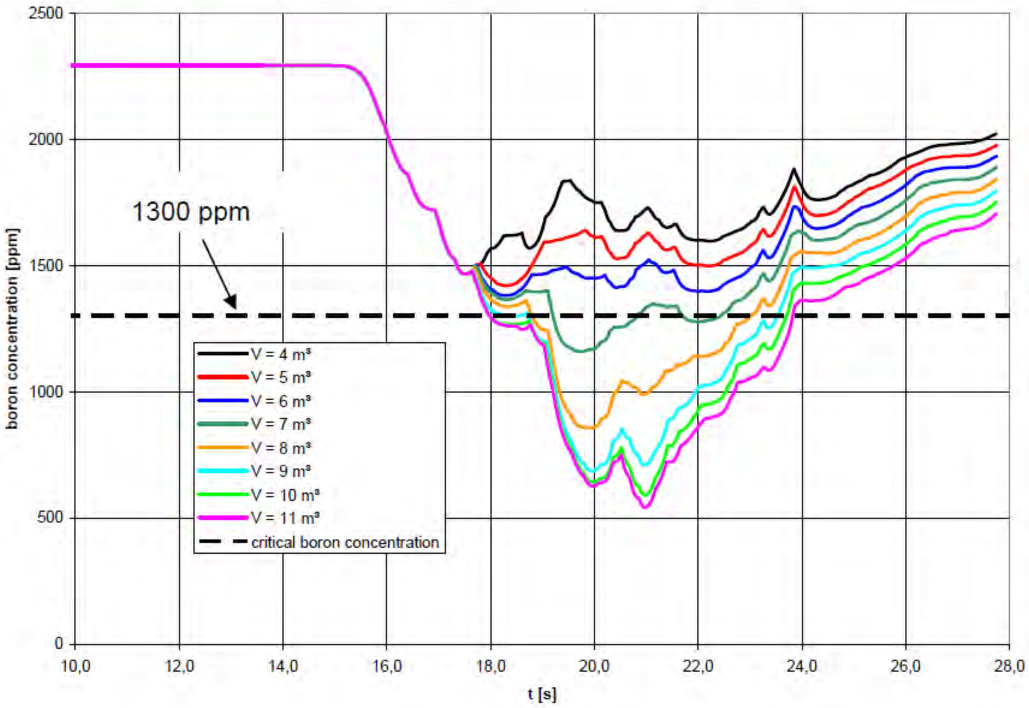


FIGURE A1*: Local minimum boron concentration at core inlet*

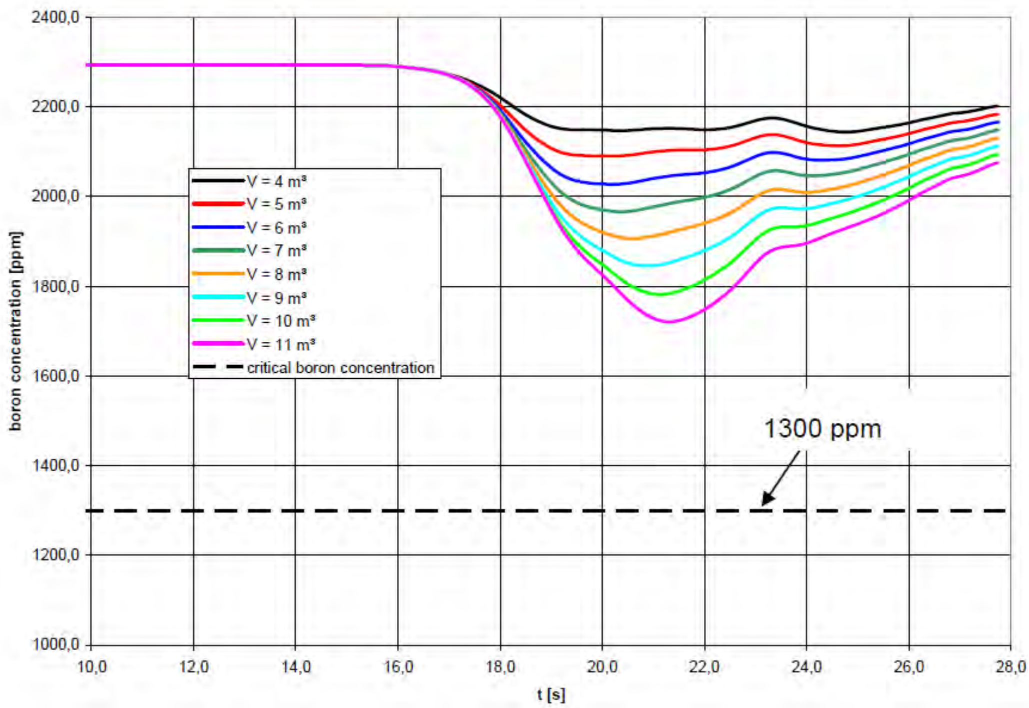


FIGURE A2: Average boron concentration at core inlet

8. SPENT FUEL POOL DRAINAGE FAULTS

8.1. INTRODUCTION

Sub-chapter 9.1 of the PCSR provides a safety assessment of the Spent Fuel Pool and the fuel handling systems used to transport fuel assemblies between the reactor and Spent Fuel Pool, and from the Spent Fuel Pool to the spent fuel cask. The safety assessment analyses postulated faults associated with failures of these fuel handling and storage systems. However, there are a number of potential non-isolable failures of penetrations/openings in the Spent Fuel Pool and connected compartments that are not considered in Sub-chapter 9.1 as they were previously regarded as outside the design basis for the UK EPR due their assumed low frequency of occurrence. Such events could in principle lead to drainage of the Spent Fuel Pool and overheating of fuel assemblies undergoing handling.

A safety case has been produced [Ref-1] that addresses the additional potential failures and identifies changes to the UK EPR design to reduce the risk due to such events to ALARP. The design changes are:

- Removal of the personnel access doors in the Reactor Cavity, Reactor Building Transfer Compartment and Fuel Building Transfer Compartment, to eliminate the risk of failure of doors leading to drainage of pools during fuel handling operations;

- Provision of secondary containment of the Fuel Transfer Tube to prevent significant loss of pool inventory due to a postulated gross failure of the Fuel Transfer Tube;

- Provision of temporary standpipes and cover plates to cover PTR [FPCS/FPPS] drain lines in the Reactor Pool, the Reactor Building Internals Compartment, the Reactor Building and Fuel Building Transfer Compartments, and the Cask Loading Pit. The modification reduces the risk of pool draining due to non-isolable failures of PTR [FPCS/FPPS] pipework in fuel handling;

- Changes to the operating procedure for transferring fuel assemblies from the Spent Fuel Pool to the Cask to ensure that the fixed gate between the Spent Fuel Pool and the Cask Loading Pit is closed whenever the Penetration Upper Cover is open. This modification prevents draining of the Spent Fuel Pool following failure of the bellows connecting to the cask penetration, facilitating recovery of Spent Fuel Pool cooling and recovery of a fuel assembly being handled in the Cask Loading Pit;

- Increase in the safety class of the Safety Feature Group providing the Spent Fuel Pool make-up function (part of the JAC/JPI system) from class 2 to class 1.

This PCSR section summarises the analysis of the additional failures presented in the safety case [Ref-1] and demonstrates that, with the proposed design modifications, the risk due to the postulated failures has been reduced to ALARP.

8.2. SCOPE OF ASSESSMENT

8.2.1. Plant and Processes Considered

The plant areas considered in this PCSR section are the Spent Fuel Pool and the other pools used to transfer irradiated fuel assemblies (Reactor Pool, Reactor Internals Compartment, Fuel Transfer Compartments in the Reactor Building and Fuel Building and the Cask Loading Pit) and the fuel handling systems (PMC). The fuel handling system comprises the Refuelling Machine, the Fuel Transfer Facility, Fuel Transfer Tube and Spent Fuel Mast Bridge.

8.2.2. Failures Considered

The following failures are considered in this assessment:

- Non-isolable breaks in the PTR [FPCS/FPPS] pipework;
- Gross failure of Fuel Transfer Tube;
- Gross failure of a Technical Opening in the Reactor Pool;
- Drainage of the Cask Loading Pit due to gross failure of bellows;

The assessment considers the effects of these faults on the coolability of fuel assemblies being stored in the Spent Fuel Pool or being handled, assuming the failure occurs in the most unfavourable plant configuration. It addresses the ability of operators to provide make-up water to the pools to keep the fuel assemblies covered, and to move a fuel assembly that was being handled at the time of the fault to a coolable location. It also considers the internal flooding consequences due to water draining from the pools into other parts of the Reactor Building or Fuel Building.

8.2.3. Plant States and Configurations

The reactor states that are considered are given in Section 16.4.8 - Table 1, taken from Sub-chapter 14.0 of the PCSR.

Within the different plant states defined in Section 16.4.8 - Table 1, four plant configurations are identified corresponding to the different filled compartments and the potential drainage locations {CCI Removed} b [Ref-1]). The plant configurations are identified as follows:

1. Configuration 1: Normal Operation/Shutdown – No Fuel Handling {CCI Removed} b

In this configuration, the reactor is in operational or shutdown States A to D: the fuel assemblies are either in the Reactor Pressure Vessel or in the underwater fuel storage rack in the Spent Fuel Pool. The Spent Fuel Pool and Fuel Building Transfer Compartment are filled with water. The Fuel Transfer Tube is isolated and empty of water and the Reactor Cavity, Internals Compartment and Reactor Building Transfer Compartment are empty of water.

2. Configuration 2: Fuel Loading and Unloading {CCI Removed} b

In this configuration, the reactor is in State E, undergoing loading/unloading. The penstocks are removed between the Reactor Pool and the Internals Compartment, the Internals Compartment and the Reactor Building Transfer Compartment, and the Fuel Building Transfer Compartment and the Spent Fuel Pool. The fixed gate between the Fuel Building Transfer Compartment and the Spent Fuel Pool is open. The Fuel Transfer Tube is un-isolated for fuel assembly transfers. All compartments are filled with water except for the Cask Loading Pit, which is empty. The door between the Spent Fuel Pool and the Cask Loading Pit is closed and the penstock is installed between the Spent Fuel Pool and the Cask Loading Pit.

A fuel assembly may be being handled by the Refuelling Machine in the Reactor Building {CCI Removed} b, or by the Fuel Transfer Facility {CCI Removed} b, or by the Spent Fuel Mast Bridge in the Fuel Building {CCI Removed} b.

3. Configuration 3: Fuel fully Unloaded into the Spent Fuel Pool {CCI Removed} b

This configuration corresponds to plant State F: the core is fully unloaded into the Spent Fuel Pool. {CCI Removed} b.

4. Configuration 4: Fuel being Loaded into Fuel Cask {CCI Removed} b

The reactor is in an operational state (plant State A), but the penstock between the Spent Fuel Pool and the Cask Loading Pit is removed and the door is open between these compartments. The Spent Fuel Pool and the Cask Loading Pit are filled with water. All other compartments are empty. The door between the Spent Fuel Pool and the Fuel Building Transfer Compartment is closed and the penstock is installed between the Spent Fuel Pool and the Fuel Building Transfer Compartment.

8.3. FAULT ANALYSIS

8.3.1. Non-isolable Pipe Breaks in the Fuel Pool Cooling System (PTR [FPCS])

Top Entry Pipes

Postulated breaks in PTR [FPCS/FPPS] pipework between the pool wall and the first isolation valve cannot be isolated by the operator in the Main Control Room. Taking into account the pipework design and construction quality, the inspection regime, and the limited extent of the pipework which is vulnerable, such failures are considered to be infrequent events which lie in the frequency range of PCC-4 events: they were previously excluded from the UK EPR design basis by applying the break preclusion principle.

For the non-isolable pipework sections connected to the top of the fuel pool, the worst case is a break of a suction line of the third PTR [FPCS/FPPS] train which could theoretically lead to drainage of the pool to +16.2 m {CCI Removed} [b]. With regard to cooling of the fuel in the Spent Fuel Pool, the worst case plant state would be State F, as this corresponds to the maximum heat load in the Spent Fuel Pool and the most rapid drop in pool level. The event sequence in State F would be similar to the corresponding case for isolable breaks, which are considered in PCSR Sub-chapter 14.4. However, recovery of the final safe state would now require repair of the pipe break so that the Spent Fuel Pool could be refilled using the JAC/JPI [NIFPS] make-up system to allow a non-faulted PTR [FPCS/FPPS] train to be restarted. Up to the time of the repair, a controlled state would be maintained by supplying make-up to the Spent Fuel Pool using the JAC/JPI [NIFPS] system to compensate for mass losses due to boiling.

Strategies for repairing non-isolable breaks are being developed. If a feasible method of repair cannot be found, further design modifications will be implemented.

Floor Entry Pipes

A non-isolable break of a floor entry pipe could lead to complete drainage of the affected compartment to floor level, and partial or complete drainage of connected compartment. Such a failure during fuel handling operations (configurations 2 and 4 in section 8.2.3) could result in a stranded fuel assembly that could not be cooled. Taking into account the pipework design and construction quality, the inspection regime, the limited extent of the vulnerable pipework, and the short time at risk when fuel handling is taking place, the probability a failure occurring in a situation which could lead to fuel damage is considered to be very low (below the frequency range of PCC-4 events).

To enable the risk of overheating of a stranded fuel assembly due to failure of a floor entry pipe to be discounted, the 2008 Reference Design will be modified to provide moveable standpipes to cover the floor drains in the Reactor Pool and Internals Compartment, and removable cover plates to cover the floor drains in the Transfer Compartments and Cask Loading Pit, when fuel transfer operations are taking place.

The moveable standpipes will ensure that the level in the Reactor Pool, Internals Compartment, Transfer Compartments and Spent Fuel Pool will only drop to +16.2 m on failure of a PTR [FPCS/FPPS] purification line in the Reactor Pool or Internals Compartment. The standpipes will be permanently located inside the compartments and will be lowered into position manually when the Reactor Pool and the Internals Compartment are filled with water and fuel assemblies are being handled. Purification of these compartments is not affected as the purification lines are not closed, suction being from the top of the standpipe.

Provision of these standpipes means that, if there were to be a pipe break, sufficient water would be maintained over the spent fuel in the Spent Fuel Pool and a handled assembly could be transferred into a safe cooling position in the reactor or Spent Fuel Pool.

The removable covers for the drain lines in the Reactor Building Transfer Compartment, Fuel Building Transfer Compartment and Cask Loading Pit will be actuated by pull rods and will be closed before fuel handling commences. Thus, a drain line break would not result in significant water leakage from any of the water filled compartments.

There are no bottom penetrations in the Spent Fuel Pool and therefore there is no potential for the Spent Fuel Pool to drain when it is isolated from the other compartments.

8.3.2. Gross Failure of Fuel Transfer Tube

Gross failure of the Fuel Transfer tube during fuel handling operations (configuration 2 in section 8.2.3) could result in a stranded fuel assembly that could not be cooled. Additionally, the internal flooding consequences could be unacceptable if the failure occurred inside the Reactor Building Annulus [Ref-1]. Taking into account the design and construction quality, the inspection regime, the limited tube length, and the short time at risk when fuel handling is taking place, the probability a large failure of the Fuel Transfer tube occurring in a situation where fuel uncovering could occur is considered to be very low (below the frequency range of PCC-4 events). Such failures were not previously considered in the design basis for the UK EPR because of their assumed low frequency of occurrence.

To enable the risk of overheating of a stranded fuel assembly or unacceptable flooding consequences due to failure of the Fuel Transfer Tube to be discounted, the 2008 Reference Design will be modified to ensure that such failures will only lead to a small loss in inventory from the pools. The modification will be achieved by designing the rooms enclosing the Fuel Transfer Tube to be watertight to a pressure corresponding to the maximum water level in the pools. The sealing arrangements will be seismically qualified at SC1.

Due to the modification, the fall in water level due to failure of the Fuel Transfer Tube will be insignificant: therefore, fuel cooling will not be affected and any handled fuel assembly can be placed in a safe position, with no limitations on how quickly this has to be achieved.

8.3.3. Gross Failure of a Personnel Access Door

{CCI Removed}

[b] Gross failure of one of these doors when the compartments are flooded was previously excluded from the design basis for the UK EPR as the doors are substantial positive-locking bulkhead doors equipped with multiple seals, similar in design to submarine bulkhead doors. The locking mechanism ensures that the door is tightly held against the bulkhead, with the water pressure providing further positive closure. Therefore, no mechanism can be envisaged that would result in gross failure. If failure of a door is nonetheless postulated in plant configurations 2 and 3 (see section 8.2.3), the result could be a stranded fuel assembly that could not be cooled.

To enable the risk of overheating of a stranded fuel assembly due to failure of a personnel access door to be discounted, the 2008 Reference Design will be modified to remove the doors and replace them with alternative means of access to the affected compartments. Implementation of this modification will eliminate the risk due to door failure.

8.3.4. Gross Failure of a Technical Opening

The UK EPR design includes Technical Openings in the walls of the reactor cavity, which provide essential Heating, Venting and Air Conditioning (HVAC), electrical and aeroball access to the reactor cavity in normal operation {CCI Removed} [b]. Before the reactor cavity is flooded, the openings are sealed using watertight doors.

The doors to the Technical Openings are positive-locking bulkhead doors, which are similar in design to submarine bulkhead doors. The locking mechanism ensures that the door is tightly held against the bulkhead, with the water pressure providing further positive closure. The doors have double static seals and their watertightness is monitored continuously by the liner leak detection system. If a seal leakage was to occur, the fault would be detected when the pool was initially filled, before the start of fuel handing. Although it is possible that degradation of the door seals could occur during fuel handling operations, this would be rapidly detected by the leak monitoring system: no mechanism can be identified which could result in a sudden gross failure of the door, given the low applied loads and the large margins inherent in the structural design. Maintenance of the sealing of doors will be carried out annually and includes a gas test to check sealing tightness (test equipment can be connected to the space between two seals).

Taking into account the robust design, the leakage monitoring system, the absence of an identified degradation mechanism that could result in gross failure, and the limited time at risk when fuel handling operations are taking place, the probability of a gross failure of a door in a situation which could lead to fuel uncovering is considered to be very low (below the frequency range of PCC-4 events).

The consequences of failure of a door to a Technical Opening in configuration 2 (State E) are described in section 8.3.4.1 below.

8.3.4.1. Consequences of Failure

The bounding case for failure of a Technical Opening is failure of one of the four HVAC openings at the +9.9 m level {CCI Removed} {CCI Removed} [b]. Gross failure would result in the pool level in the Spent Fuel Pool, Reactor Building Transfer Compartment and Fuel Building Transfer Compartment falling to +10.9 m: the level in the Reactor Pool would fall to +9.9 m. The final pool levels would stabilise {CCI Removed} [b].

8.3.4.2. Establishment of Controlled State

{CCI Removed}

[b]

Fuel handling operations in the Reactor Building and Fuel Building are automated so, if the machine handling a fuel assembly at the time of the fault was allowed to continue operating, the fuel assembly would be automatically moved to a location where it was covered by water (either in the Reactor Pressure Vessel, the Fuel Transfer Facility or the Spent Fuel Pool). The maximum time taken to move the assembly to the safe underwater position would be about 3 minutes. Therefore, if fuel handling operations were taking place in the Reactor Building or Fuel Building at the time of the fault, the only operator action required would be to allow the machine to carry out its pre-programmed movement to take the fuel assembly to its final safe location.

As the fuel movement is rapid, the fuel assembly would not become uncovered, and the additional radiation exposure of the operators of the Refuelling Machine or Spent Fuel Mast Bridge would be small. If the fuel assembly was in the Fuel Transfer Facility at the time of the event, no operator action would be needed for many hours as the fuel assembly would remain covered by water. Numerical simulations show that a high power fuel assembly in the Fuel Transfer Tube (the most unfavourable location for a fuel assembly in the Fuel Transfer Facility) would not overheat provided it was water covered.

The fuel handling systems (Refuelling Machine, Fuel Transfer Facility and Spent Fuel Mast Bridge) are operational systems, which are not currently safety classified. As these systems would be in use prior to the initiating event and should not be affected by it, there is high confidence in their continued availability: thus, the probability of a random failure occurring in combination with the initiating event is judged insignificant. Therefore, it is not considered necessary to make improvements to the design of these systems. The classification of safety and operational systems of the UK EPR is being re-assessed against the current UK EPR classification principles, and this assessment will take into account the role of the fuel handling systems in mitigating pool drainage faults.

A controlled state would be reached with all fuel assemblies in either the Reactor Pressure Vessel, the Spent Fuel Pool or the Fuel Transfer Tube:

- The fuel assemblies in the Reactor Pressure Vessel would be cooled by the Low Head Safety Injection [LHSI] system operating in Residual Heat Removal (RHR) mode. Adequate shielding would be provided by the water layer above the fuel in the Reactor Pressure Vessel and Reactor Pool to allow access to operators above the Reactor Pool;

- The fuel assemblies in the Spent Fuel Pool would be cooled by supplying make-up water to the Spent Fuel Pool via the class 1 JAC/JPI [NIFPS] make-up safety feature, to compensate for the level drop due to evaporation.

- If a fuel assembly was in the Fuel Transfer Facility, it would be rotated to the horizontal position to maximise radiation shielding. If necessary, make-up water would be supplied to the Transfer Compartments via the class 1 JAC/JPI [NIFPS] make-up safety feature supplying the Spent Fuel Pool, to compensate for the level drop due to evaporation.

8.3.4.3. Establishment of Safe Shutdown State

The radiation levels in the Spent Fuel Pool Hall with the water level in the Spent Fuel Pool at +10.9 m and the core fully unloaded to the Spent Fuel Pool, which is bounding for the present case, are calculated [Ref-1]. The radiation levels are above the limits at which personnel would be evacuated for the Spent Fuel Pool Hall: therefore, it is desirable to establish a long term safe state without the need for operators to access the Spent Fuel Pool Hall. One option would be for operators to replace the penstock between the Reactor Building Fuel Transfer Compartment and the Reactor Internals Compartment in the Reactor Building. A normal water level could then be established in the Reactor and Fuel Building Transfer compartments and the Spent Fuel Pool by transferring water from the In-Containment Refuelling Water Storage Tank [IRWST] or using the JAC/JPI [NIFPS] system. When normal radiation levels were restored in the Fuel Building, a fuel assembly in the Fuel Transfer Facility could be transferred into the Spent Fuel Pool, and a final safe state established by reinstating the PTR [FPCS/FPPS] system.

Finally, the Reactor Pressure Vessel would be re-closed, the Reactor Pool drained and repairs effected on the failed HVAC door.

Internal flooding consequences of the breach are discussed in section 8.3.6 of this sub-chapter and shown to be acceptable.

8.3.5. Leakage from the Cask Loading Pit due to Bellows Failure

Leakages from the Cask Loading Pit are only relevant to cask loading operations when fuel assemblies are being transferred from the Spent Fuel Pool into the cask for onward transfer to an external fuel store. {CCI Removed} [b] The Cask Loading Pit is filled with water by moving the penstock from between the Spent Fuel Pool and the Cask Loading to between the Spent Fuel Pool and the Fuel Building Transfer Compartment and transferring the water in the Fuel Building transfer Compartment to the Cask Loading Pit.

The bounding failure considered during cask loading is gross failure of the bellows between the bottom penetration in the Cask Loading Pit and the cask itself. Gross failure of the bellows has previously been excluded from the UK EPR design basis, as the bellows are comprised of two barriers, both of which would have to fail to cause a leakage. The bellows also have an internal and external protective shell to protect them against mechanical damage. Taking into account the robust design and the time at risk when a fuel cask is being loaded, gross failure of the bellows is considered to be an infrequent event within the frequency range of PCC-4 events. The consequences of bellows failure are analysed below.

8.3.5.1. Consequences of Failure

Following postulated gross failure of the penetration bellows, it is estimated that the leakage would result in complete drainage of the Cask Loading Pit and drainage of the Spent Fuel Pool to the +10.9 m level {CCI Removed} [b], if the door between the Cask Loading Pit and the Spent Fuel Pool was open. Drainage of the Spent Fuel Pool could result in high radiation fields at the Centralised Control Unit for the Spent Fuel Mast Bridge in the Fuel Pool Hall. As the operation of lowering a fuel assembly into the Cask is not automated, the operators may not have time to recover a handled assembly before having to evacuate the Fuel Pool Hall.

In view of the difficulty in returning the exposed fuel assembly to a coolable location following gross failure of the bellows, it is proposed to modify the operating method for transferring a fuel assembly to the cask, to prevent draining of the Spent Fuel Pool in the event of bellows failure. The modified procedure will ensure that the door between the Spent Fuel Pool and the Cask Loading Pit is closed whenever the Penetration Upper Cover is open. Hence, in the event of bellows failure, only the Cask Loading Pit will be drained. Although a fuel assembly inside the Cask Loading Pit could still be uncovered, analysis of radiation levels in the Spent Fuel Pool area show that access would be possible and the operators would be able to perform local operations to recover the stranded assembly.

8.3.5.2. Establishment of Controlled State

Taking into account air cooling, thermal equilibrium would be reached before fuel damage occurred [Ref-1], implying that there would be no time limit on recovery of the stranded assembly. Recovery of the assembly would be possible by closing the Penetration Upper Cover and refilling the Cask Loading Pit using the class 1 JAC/JPI [NIFPS] make-up system {CCI Removed} [b].
Alternatively, the operators could lower the fuel assembly into the cask by operating the Spent Fuel Mast Bridge from the Centralised Control Unit. The radiation fields in the Spent Fuel Pool area would be acceptable for such local actions.

8.3.5.3. Establishment of Safe Shutdown State

A final safe state would be established either by reflooding the Cask Loading Pit and returning the fuel assembly to the Spent Fuel Pool where it would be cooled by the PTR [FPCS/FPPS], or by placing the fuel assembly in the cask and securing the cask using normal procedures.

8.3.5.4. Review of other Cask Loading Pit Faults

A Failure Modes and Effects Analysis (FMEA) has been carried out to identify failure sequences which could lead to draining of the Cask Loading Pit [Ref-1]. The FMEA identified the failure modes for each of the components that could lead to draining of the Cask Loading Pit and hence partial draining of the Spent Fuel Pool. The reliabilities of the components that could fail were estimated using operating experience from the EDF fleet of PWRs [Ref-2]. These data were used to estimate the frequency of the postulated initiating events that could cause the Cask Loading Pit to drain [Ref-3]. Using the data, initiating events were identified for which a Design Basis Analysis (DBA) should be performed, those which should be included in the Probabilistic Safety Analysis (PSA), and those whose frequency was low enough not to require further analysis. Only one fault was identified that was frequent enough to require DBA (rupture of a hose connected to cask orifice A, B or C during fuel assembly transfer operations). The consequences of this fault are bounded by gross failure of the bellows: therefore, no further DBA of faults in the Cask Loading Pit is considered necessary.

8.3.6. Flooding Consequences of Pool Leakage Events

Following the design changes described above, the only non-isolable leakages that could cause internal flooding in the Reactor Building or Fuel Building are PTR [FPCS/FPPS] pipework failures, failure of the Fuel Transfer Tube, failure of a Technical Opening, or draining of the Cask Loading Pit.

Due to the modification to make the rooms enclosing the Fuel Transfer Tube watertight, failure of the Fuel Transfer Tube would lead to only a small amount of water entering the Reactor Building and the Fuel Building and the flooding consequences of such a failure are insignificant.

The consequences of flooding in the Reactor Building in the bounding case of failure of a personnel access door in the Reactor Cavity (note that this door is to be removed) have been assessed [Ref-1]. This failure bounds the case of failure of a Technical Opening, as it results in total drainage of the Reactor Cavity and Reactor Internals Compartment, and partial drainage of both Transfer Compartments and the Spent Fuel Pool, causing up to 2645 m^3 of water to be released into the Reactor Building. The water released would flow down into the IRWST. It is shown that the IRWST water level would remain acceptable and no safety significant equipment in the Reactor Building would be vulnerable to damage by flood water [Ref-1].

With regard to drainage events in the Fuel Building, the bounding case of failure of a personnel access door at the bottom of the Fuel Building Transfer Compartment (note that this door is to be removed) is considered [Ref-1]. This hypothetical event results in almost complete drainage of both transfer compartments and draining of the Reactor Pool, Internals Compartment and Spent Fuel Pool to the +10.9 m level, resulting in a volume of 2359 m^3 of water being released into the Fuel Building. The water released would flow down to the basement of the Fuel Building, flooding Fuel Building Division 4. One main train of the PTR [FPCS/FPPS] cooling system located in Fuel Building Division 4 could be lost, but cooling of the Spent Fuel Pool could still be performed by the remaining PTR [FPCS/FPPS] main train located in Division 1 of the Fuel Building or the third train of the PTR [FPCS/FPPS] which is located in Division 1 of the safeguard building. Hence, the consequences for fuel cooling would be acceptable.

Drainage of the Cask Loading Pit would result in a maximum volume of approximately 1145 m^3 being drained to the Fuel Building, if the door to the Spent Fuel Pond was open. With the modified cask loading procedure implemented in GDA, this volume would be reduced to 215 m^3. The only difference to failure of the door in the Transfer Compartment is that Division 1 of the Fuel Building would be flooded, potentially damaging the PTR [FPCS/FPPS] cooling train in the basement of Fuel Building Division 1. Cooling of the Spent Fuel Pool could be achieved by the PTR [FPCS/FPPS] train located in Division 4 of the Fuel Building or the third train of the PTR [FPCS/FPPS] located in Division 1 of the safeguard building.

It is concluded that the flooding consequences of non-isolable failures of PTR [FPCS/FPPS] pipework, or drainage of the Cask Loading Pit, would be acceptable in terms of their effects on essential safety functions in the Fuel Building.

8.3.7. Criticality risk

Following drainage of water from the Spent Fuel Pool, make-up would be provided if necessary using the JAC/JPI [NIFPS] system. The water in the Spent Fuel Pool is borated and therefore making up the pond with clean water will dilute the boron concentration in the Spent Fuel Pool.

Sub-chapter 9.1 of the PCSR provides the design specification for the Spent Fuel Pool. The design of the underwater fuel storage rack ensures that fuel assemblies are stored in an arrangement that ensures that criticality cannot occur even in the case of zero boron concentration in the pool water. Hence, criticality is not of concern in any of the bounding faults considered in this sub-chapter.

The minimum boron content specified by the technical specifications ensures a K_{eff} value below 0.98 in credible accident conditions including following dropping of an assembly into the Spent Fuel Pool and this also applies to the Cask Loading Pit.

8.4. CONCLUSION

The above analysis demonstrates that a safe shutdown state can be achieved with limited consequences in terms of radiation dose for non-isolable drainage events associated with the Spent Fuel Pool, subject to the implementation of modifications to the 2008 Reference Design, described in this PCSR section. As the consequences of these initiating events are insignificant, it is considered that the risk due to faults associated with leakages in the Spent Fuel Pool and connected pools has been reduced to a level that is as low as reasonably practicable.

SECTION 16.4.8 - TABLE 1

Plant States

State	**Description**
State A	Power states and hot and intermediate shutdown (P > 130 bar).
State B	Intermediate shutdown above 120°C (P < 130 bar).
State C	Intermediate and cold shutdown with cooling by LHSI/RHR. The RCP [RCS] closed or can be rapidly reclosed enabling SGs to be made available for heat removal. 3 sub-states are defined State C1 • RCP [RCS] pressure between 24.5 – 32 bar, temperature between 120°C -100°C • two SGs participating in heat removal • RIS/RRA [SIS/RHRS] operating via two LHSI/RHR trains, the other two trains are on stand-by State C2 • RCP [RCS] pressure between 24.5 – 32 bar, temperature between 100°C- 55°C • RIS/RRA [SIS/RHRS] operating via all 4 LHSI/RHR trains State C3 • RCP [RCS] pressure between 32 -1 bar,] temperature around 55°C • RCP [RCS] water inventory between pressuriser level at hot zero power conditions and low level operation (3/4 loop) • RIS/RRA [SIS/RHRS] operating via three LHSI/RHR trains, the other train is on standby.
State D	Cold shutdown with RCP [RCS] open so that the SGs cannot be used for decay heat removal. The RCP [RCS] level may be at ¾ loop level. 3oo4 LHSI/RHR trains are required to be in operation, fourth train on standby.
State E	Cold shutdown with the reactor cavity flooded for refuelling.
State F	Cold shutdown with the core fully unloaded. During this state works are performed on RCP [RCS] components.

{CCI Removed}

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b

9. ANALYSIS OF FAULTS IN ESSENTIAL SUPPORT SYSTEMS

9.1. INTRODUCTION

The list of initiating Events (IEs) considered within the design basis for the UK EPR is based on the design basis faults identified for FA3, which were based on the list used for earlier PWR designs supplemented by 30 years of operational feedback from the international PWR fleet (from analytical studies, operational experience and reviews by national safety authorities). As described in Chapter 14 of the PCSR, the IEs are grouped into Plant Condition Categories PCC-2 to PCC-4 depending on their frequency of occurrence. The following frequency ranges are defined:

- PCC-1: normal operating transients
- PCC-2: design basis transients (F > 10^{-2}/r.y)
- PCC-3: design basis incidents (10^{-2}/r.y > F > 10^{-4}/r.y)
- PCC-4: design basis accidents (10/$^{-4}$r.y > F > 10^{-6}/r.y)

Design basis analysis is performed to demonstrate that in all the PCC-2 to PCC-4 events, the plant can be brought to a controlled state and ultimately to the safe shutdown state using conservative analysis rules and assumptions presented in PCSR Sub-chapter 14. 0. PCCs considered in the design basis cover event frequencies down to 10^{-6}/yr.

The PCC analyses are supplemented by analyses of Design Extension Conditions, (referred to as RRC-A conditions) in which initiating events are combined with a failure of the safeguard systems providing a main line of protection. The RRC-A sequences, which are defined based on probabilistic considerations, are analysed in Sub-chapter 16.1 of the PCSR, using less conservative analysis rules and plant modelling assumptions than those applied to PCCs. Additional plant safety features are implemented to enable the UK EPR to reach a controlled state in RRC-A sequences: these safety features provide a diverse line of protection for frequent faults.

EDF and AREVA carried out a study to reconcile the IEs modelled in the design basis analyses of the UK EPR with the IEs modelled in the PSA [Ref-1]. This study concluded that the majority of PSA IEs are either bounded by PCC events or that the relevant PSA sequence is bounded by an RRC-A transient. However, it was found that some PSA initiating events were not bounded by PCC events, even though they were in the PCC frequency range. A commitment was made to perform further design basis analysis of these cases.

The design basis analysis previously performed for UK EPR generally excluded initiating events arising in support systems such as electrical or ventilation systems. In the UK it is normal practice to consider failures in support systems within the scope of design basis analysis. Therefore EDF/AREVA has agreed to extend the Design Basis Analysis (DBA) to address such events.

This PCSR section presents the initial results of the DBA of faults in essential support systems. Faults in the following essential support systems are addressed.

- Safety Cooling Chain systems
- HVAC systems
- Electrical Systems
- Nitrogen Gas Distribution Systems
- Instrument Air System

Faults initiated by failures in I&C systems have been excluded from the analysis as the I&C design in the UK EPR has been substantially modified during GDA and the allocation of safety functions to different platforms is not yet finalised. Therefore DBA of faults in I&C systems is deferred until the NSL phase of licensing.

The present PCSR section describes the DBA for new Design Basis Events (DBEs) due to faults in support systems. The DBEs considered involve failures in individual trains of a multiple train system, and Common Cause Failures (CCF) affecting multiple safety trains concurrently. It is found that to meet design requirements on safety systems used to mitigate the consequences of DBEs, modifications are required to the reference design of the UK EPR [Ref-2]. It is intended to implement these modifications in the UK EPR design if it is reasonably practicable to do so. This sub-chapter shows that, taking into account the proposed modifications, the safeguard systems will meet the deterministic requirements defined for the UK EPR, for the new DBEs for failures in support systems.

This PCSR section is organised as follows. Sub-section 9.2 describes the methods used to identify and analyse the new design basis faults, and the applicable success criteria. Sub-section 9.3 describes the identified plant modifications. The DBA for the new DBEs is described in sub-section 9.4 and the overall conclusions of the fault analysis presented in sub-section 9.5.

9.2. METHODOLOGY FOR DESIGN BASIS ANALYSIS

9.2.1. Identification of faults to be analysed

The steps in identifying the IEs in essential support systems for which new DBA was potentially required were performed as follows:

- a review of possible faults on essential support systems affecting single or multiple system trains was performed to identify events that could result in a reactor transient or a transient in the spent fuel pool that, if unmitigated, could lead to significant radiological consequences. Where appropriate, IEs with similar consequences were grouped together to define a bounding IE.

- An initiating event frequency was then determined for each bounding IE.

- If the potential consequences of the IE were not bounded by an existing PCC event, a safety functional capability analysis was undertaken to demonstrate that the remaining safeguard trains are capable of performing their intended function using assumptions and success criteria appropriate to the frequency of the IE (see section 9.2.2 below). In cases where the potential consequences are bounded, a check is made to ensure the increase in the frequency of the bounding PCC does not require the PCC study to be re-performed.

- New transient analysis was carried out if the fault transient could not be bounded by an existing analysis.
- Reasonably practicable plant modifications were proposed in cases where shortfalls against applicable deterministic criteria were identified.

9.2.2. Rules applied in Design Basis Analysis

The analysis rules and success criteria applied in new design basis analysis are dependent on the assessed frequency of the IE.

9.2.2.1. Initiating events with frequency > 10^{-5}/r.y.

The analyses of the new DBEs are performed using the rules and success criteria applicable to PCC studies (see PCSR Sub-chapter 14.0) which are summarised as follows.

- The single failure criterion (active or passive failure) is applied to equipment which needs to change state to fulfil its functions and has a beneficial effect on the transient.
- Coincident preventive maintenance is assumed at the time of occurrence of the IE.
- Coincident Loss of Offsite Power (LOOP) is assumed if it is conservative, at the time of turbine trip, for faults in the PCC-3 and PCC-4 frequency range. However for faults involving CCF of multiple trains of essential support systems, co-incident LOOP is not generally assumed unless it is a direct consequence of the initiating event.
- Standard deterministic acceptance criteria are applied related to key fuel, fuel clad, core, and primary circuit parameters.
- The safety analysis must demonstrate that a controlled state can be reached for each PCC event and the safe shutdown state for each group of similar PCC events.
- Safety features needed to reach the controlled state must be Class 1. Safety features required to reach the safe shutdown state must be at least Class 2.
- Safety criteria must be met with a high level of confidence using appropriate levels of conservatism in the transient analysis.
- Initial conditions must assume the most conservative operating conditions allowing for uncertainties etc.
- Minimum time limitations are placed on claims for operator actions.
- If the new DBE has a frequency ≥ 10^{-3}/r.y it should be demonstrated that there is an additional line of protection provided for each fundamental safety function, which is diverse to the main line of protection. The safety features providing the diverse line of protection function should be at least Class 2.

Slightly more relaxed assumptions and acceptance criteria are applied in the analysis for the spent fuel pool due to the slower transients and extended response times (see section 2.10 of PCSR Sub-chapter 14.0).

9.2.2.2. Common Cause Failure Initiating Events.

Common Cause Failure (CCF) events are events resulting in failures of multiple trains of an essential support system. These events generally have frequencies < 10^{-5}/r.y. The analysis rules and success criteria applicable to these DBEs were similar to those used for frequencies above 10^{-5}/r.y. with the following exceptions:

- With regard to assumptions of additional single failure, maintenance and LOOP the general rule applied was that the frequency of the sequences to be analysed using DBA should not be less than 10^{-7}/r.y. As the probability of an additional single failure or coincident LOOP can be taken as below 10^{-2}, these assumptions were not applied in the DBA. However the probability of Preventive Maintenance on a safety train may be above 10^{-2}. Preventive Maintenance was therefore included on a case by case basis depending on whether the resulting sequence had a frequency greater than 10^{-7}/r.y.

- The classification requirements on systems required to reach a controlled state and safe shutdown state were graduated according to the initiating event frequency and ALARP considerations. For IEs close to 10^{-5}/r.y. the safety features required to reach the controlled state are expected to be at Class 1. Class 3/NC is considered sufficient for safety features required to reach the controlled state for IEs with a frequency close to 10^{-7}/r.y.

9.3. REFERENCE DESIGN FOR LOSS OF SUPPORT SYSTEMS ANALYSES

Several design changes have been implemented in the UK EPR design following the initial design basis analyses on essential support systems. This section describes the design reference to be considered for the design basis analysis of the loss of essential support systems in section 9.4. The main modifications are described below and summarised in Section 16.4.9 - Table 1.

9.3.1. Cooling Chain

The main cooling chain is composed of the Component Cooling Water System (RRI [CCWS]) and Essential Service Water System (SEC [ESWS]). The modifications proposed include classification upgrades of RRI [CCWS] safety features (e.g. automatic isolation of the RRI [CCWS] train in case of leakage, automatic switchover to the stand-by train). The design of the RRI [CCWS] and SEC [ESWS] systems are described in PCSR Sub-chapter 9.2 (sections 1 and 2).

An additional electrical supply to the RRI [CCWS] common load isolation valves and RRI [CCWS] / SEC [ESWS] heat exchanger regulation valves will be provided by the LA system on trains 1 and 4 and by the LV system on trains 2 and 3 [Ref-1] in order to cope with potential losses of these electrical supplies.

A common header (see PCSR Sub-chapter 9.2, section 2) will be added on the RRI [CCWS] side of the thermal barrier to allow cooling of the four thermal barriers in the event of loss of one of the RRI/SEC [CCWS]/[ESWS] trains.

9.3.2. Ventilation Systems

The design of the main UK EPR ventilation systems is presented in Sub-chapter 9.4.

The main ventilation system cooling the I&C and electrical rooms is the DVL [SBVSE] system, which is cooled by chilled water from the DEL [SCWS]. The chilled water system is necessary to cool the DVL [SBVSE] system when the outside temperature is above a certain temperature limit. Modifications such as an upgrade of the main ventilation DVL [SBVSE] and DEL [SCWS] trains to Class 1 and creation of two new back up ventilation trains in case of loss of the main ventilation systems are to be implemented [Ref-1] in the UK EPR design. In addition, automatic Class 1 switchover to the back-up DVLnew trains is also to be implemented. Complete diversification (electrical supply, I&C supply and mechanical diversification) will be provided between the four main trains and the two back-up trains. In order to ensure mitigation against complete loss of the 690 V LJ system, the back-up DVL/DEL [SBVSE]/[SCWS] trains will be designed to cool two electrical and two I&C divisions in circumstances of a total loss of the LJ system.

The Main Control Room (MCR) is ventilated by the DCL [CRACS] system described in PCSR Sub-chapter 9.4 (section 8). The DCL [CRACS] system is to be modified to ensure that sufficient cooling is provided for 24 hours even in case of the loss of three main cooling trains. In addition provisions will be taken to ensure ventilation of the Main Control Room in case of Common Cause Failure (CCF) of the four main cooling trains.

Ventilation of the main pumping station is provided by the DVP system described in PCSR Sub-chapter 9.4 (section 12). The DVP ensures cooling of the rooms where SEC [ESWS] pumps are located. In order to reduce the frequency of a potential Total Loss Of Cooling Chain (TLOCC) event arising from Common Cause Failure (CCF) of the DVP ventilation function, I&C associated with the ventilation function start-up or I&C sensors associated with the temperature monitoring (to be defined depending on the grace period), will be duplicated on the Class 1 TXS platform. Additional mechanical diversification will be provided if required, following completion of probabilistic studies.

9.3.3. Electrical and I&C Systems

9.3.3.1. I&C systems

The design of the I&C systems is described in PCSR Chapter 7. In order to improve the management of some situations following common cause failure of the cooling chain (TLOCC), improvements in the reliability of the required safety features groups [Ref-1] has been implemented (upgrade of classification of the required signals, the manual actions and /or relevant instrumentation). Safety functions and corresponding SSC requirements will be upgraded to Class 1 and manual actions implemented on TXS platform in order to improve the reliability.

9.3.3.2. Electrical systems

The design of the electrical systems is presented in PCSR Chapter 8. Re-allocation of the electrical supply of several components (e.g. RRI [CCWS] valves, PTR [FPCS] pumps) has been implemented in order to improve plant response in case of total loss of electrical system at the same voltage level. In addition, consideration is being given to using cross connections (normally used for maintenance) to recover an electrical division or the electrical supplies to at least one of the RBS [EBS] pumps in situations when only one electrical division is available, or neither of the RBS [EBS] pumps are available [Ref -1].

PSA studies will be conducted in the Nuclear Site Licensing (NSL) phase to identify if a 2x2 diversification of the components within the 4 electrical divisions is required.

9.3.4. Other Support Systems

9.3.4.1. Steam Generator Blowdown System (APG [SGBS])

The design of the Steam Generator Blowdown System (APG [SGBS]) is presented in PCSR Sub-chapter 10.4 (Section 7). In some situations such as complete loss of electrical supplies from LV switchboards, where the full ASG [EFWS] flow rate is required to remove primary heat through the steam generators [Ref-1], isolation of the Steam Generator Blowdown System (APG [SGBS]) is required. The electrical supply of one of three isolation valves on each of the APG [SGBS] trains will be re-allocated to the LA system to ensure availability of the isolation function in all situations.

9.3.4.2. Fuel Pool Cooling System (PTR [FPCS])

The design of the Fuel pool cooling system (PTR [FPCS]) is presented in PCSR Sub-chapter 9.3 (section 2). In order to ensure that fuel pool cooling is always available in case of loss of electrical supply from the LJ system, electrical supply of two PTR [FPCS]) pumps is being re-allocated to one of the 400V electrical systems [Ref-1].

9.3.4.3. Chemical and Volume Control System (RCV [CVCS])

The design of the Chemical and Volume Control System (RCV [CVCS]) is described in Sub-chapter 9.3 (section 2). In order to cope with situations where electrical supply from LV switchboards is not available, modifications to ensure seal injection will be implemented. Re-allocation of electrical supply of some valves will ensure sufficient functionality of the RCV [CVCS] system to maintain seal injection function.

9.3.5. Conclusion

The design modifications presented in this section and summarised in Section 16.4.9 - Table 1 will be implemented if it is reasonably practicable to do so. The detailed design of these modifications will be performed in the NSL phase.

9.4. DESIGN BASIS ANALYSES FOR MAIN SUPPORT SYSTEMS

9.4.1. Faults in Cooling Chain (RRI/SEC [CCWS]/[ESWS])

9.4.1.1. Faults in individual trains and sub-systems

Faults in individual trains and sub-systems within the RRI [CCWS] and SEC [ESWS] systems are analysed [Ref-1]. Three bounding IEs are identified which are not bounded by current PCC events, and for which new DBA is therefore required. The bounding IEs and the new resulting DBA are as follows:

1) Mechanical failure of an RRI/SEC [CWCS/ESWS] train

The RRI [CCWS] cooling functions are performed by four cooling trains two of which are normally operating and two of which are on standby (see PCSR Sub-chapter 9.1). In the event of a mechanical failure of an operating train there is an automatic switchover to the standby train to maintain the RRI [CCWS] cooling functions. However, if the standby train is not available (due to preventive maintenance or additional single failures assumption) cooling of the essential systems connected to common headers associated with failed RRI [CCWS] train could be lost. This could result in a small LOCA due to Main Coolant Pump seal failure and also multiple failures in other safety systems. The Standstill Seal System DEA [SSSS] is not claimed to mitigate the event as the relevant safety features are designed to Class 2.

Plant modifications introduced in section 9.3 will ensure that in this DBE sequence a controlled state can be established using Class 1 safety features, and the safe shutdown state can be established with safety features at least at Class 2. Establishment of the controlled state is achieved by using one of the available LHSI trains in Division 1 or Division 4, (which are cooled by air-cooled trains within the DEL [SCWS] safety chilled water system) to inject water into the reactor to compensate for the leak [Ref-2].

This DBE is considered as a frequent event ($F>10^{-3}$/r.y.): it is therefore necessary to show that the controlled state and the safe shutdown state could be established by alternate diverse means in the event of failure of the main line of protection. An alternate cooling route is available by using the Class 2 DEA [SSSS] to prevent seal leakage. Reactor cooling could then be achieved via the SGs using the Class 1 ASG [EFWS] safety features (see PCSR Sub-chapter 16.5, section 2.5.11).

2) Leak or break of within an RRI [CCWS] train

A leak or break on an RRI [CCWS] train would lead to isolation of the associated common headers, and prevention of the switchover to the alternate RRI [CCWS] train. Hence cooling of components and of the associated common loads would be lost. This DBE is considered as a frequent event ($F>10^{-3}$/r.y.).

The consequences of the DBE and the protection measures available are the same as identified above for mechanical failure of an RRI/SEC [CWCS/ESWS] train.

3) Leak/Break of a common RRI [CCWS] circuit

A leak/ break on a common header leads to isolation of both common headers associated with an RRI [CCWS] circuit, and a lock on the switchover to the standby RRI [CCWS] train. Hence cooling of the associated common loads is lost. This DBE is considered as a frequent event ($F>10^{-3}$/r.y.).

The consequences of the DBE and the protection measures available are the same as identified above for mechanical failure of an RRI/SEC [CWCS/ESWS] train.

9.4.1.2. Faults involving multiple safety trains

CCF of the four RRI [CCWS] pumps or the four SEC [ESWS] pumps is identified as a DBE for which DBA is required [Ref-1]. The event is referred as a Total Loss of Cooling Chain (TLOCC) and was previously treated as an RRC-A event in the UK EPR safety analysis (Sub-chapter 16.1). A TLOCC event in at-power states gives rise to the tripping and seal failure on all four Reactor Coolant Pumps leading to a Small Break-Loss of Coolant Accident (SB-LOCA) and multiple failures in other safety systems. The Standstill Seal System (DEA) [SSSS] is not claimed in to mitigate the event as the relevant safety features are designed to Class 2. The frequency of the TLOCC event in at-power states is identified as around 10^{-5}/r.y. [Ref-1].

A DBA of TLOCC in at-power and shutdown states has been carried out [Ref-1]. As the IE frequency is close to 10^{-5}/r.y. the analysis rules and success criteria in section 9.2.2.2 were applied. As the sequence frequency with coincident LOOP was around 10^{-7}/r.y. coincident LOOP was not assumed.

The analysis shows that taking into account the plant modifications introduced in section 9.3, the controlled state and the safe shutdown state can be established using safety features which are appropriately classified, taking into account plant maintenance states. To achieve the required availability in I&C functions it is necessary to claim operation of the new back-up chilled water system DELnew which is Class 2. The use of Class 2 equipment is considered to be justified as the safety claim is only required for external air temperatures above 25°C. As the frequency of a TLOCC event coinciding with such high ambient temperature conditions is of order 10^{-7}/r.y., the use of Class 2 safety features for fault recovery is considered to be justified under ALARP principles.

9.4.2. Faults in HVAC Systems

The ventilation systems of Nuclear Island buildings play a direct role in supporting the main safety function "Containment of radioactivity", and indirectly supporting other main safety functions by maintaining ambient conditions within acceptable limits for correct operation of safety-related systems. The systems are described in Sub-chapter 9.4 of the PCSR.

A screening assessment has been carried out [Ref-1] to identify the impact of failures in individual HVAC systems on normal operation and plant safety systems: the assessment includes postulated failures within individual trains of HVAC systems as well as CCF events involving multiple trains.

Functional analysis has been performed to evaluate the impact of the postulated IEs and establish whether the consequential transient was already covered by existing PCC analyses. The analysis indentified a number of bounding IEs which are not enveloped by current PCC events, and for which new DBA is therefore required. The bounding IEs and the new DBA results are described below.

9.4.2.1. Faults in the DVL/DEL [SBVSE/SCWS] system

The DVL/DEL [SBVSE/SCWS] systems are used to cool the Electrical and I&C Rooms within the Safeguard Buildings. Therefore postulated failures in these systems can lead to electrical and I&C failures that impact multiple safety functions. The new DBEs identified for faults in these systems, and the design basis analysis of these events [Ref-1] are summarised below:

9.4.2.1.1. Faults in individual trains

The DBE identified was loss of a single DVL [SBVSE] or DEL [SCWS] train.

Failure of one DVL [SBVSE] or DEL [SCWS] train results in the rapid raising of the ambient temperature in electrical and I&C equipment rooms, potentially leading to a total loss of electrical supply and I&C operability in one safety division. This can cause possible reactor transients such as malfunction of the Main Feedwater System ARE [MFWS], Chemical Volume and Control System RCV [CVCS] and/or the drifting plant parameters, which potentially results in a reactor trip transient. Loss of a single DVL [SBVSE] or DEL [SCWS] train are identified as frequent events (F>10^{-3}/r.y.).

The DBA of this event [Ref-1] assumes implementation of the design enhancements in section 9.3 of this sub-chapter, and uses the analysis rules and success criteria in section 9.2.2.1. It is concluded that with the identified modifications the controlled reactor state can be established using Class 1 safety features, and the safe shutdown state can be established with safety features at least Class 2 [Ref-2]. Additionally a safe reactor state can be established by alternate diverse means in the event of failure of the main line of protection, satisfying the diversity requirement for frequent faults (see PCSR Sub-chapter 16.5, section 2.5.11).

The consequences for cooling of the Spent Fuel Pool are bounded by existing PCC events.

9.4.2.1.2. Faults in multiple safety trains

Three new DBEs were identified as follows:

1) <u>CCF on two DVL [SBVSE] trains (1&4 or 2&3) with external air temperatures below 25°C.</u>

The estimated frequency of this DBE is >10^{-5}/r.y. and the DBE is therefore analysed with the rules and success criteria presented in section 9.2.2.1.

The event leads to the start-up of the Class 1 DVLnew trains and hence cooling of the electrical and I&C rooms is maintained. The bounding sequence involves combination of the initiating event with a preventive maintenance on a DVL/DEL [SBVSE/SCWS] train and an additional single failure on the start-up of a Class 1 DVLnew train. The DBA analysis shows with the identified modifications that the controlled reactor state can be established using Class 1 safety features, and the safe shutdown state can be established with safety features at least Class 2. Relevant Spent Fuel Pool safety criteria are met.

2) <u>CCF on two DEL [SCWS] trains (1&4 or 2&3) with external air temperature below 25°C</u>

The estimated frequency of this DBE is >10^{-5}/r.y. and the DBE is therefore analysed with the rules and success criteria presented in section 9.2.2.1.

In this scenario DVL [SBVSE] continues to operate in case of failure of its dedicated DEL [SCWS] train. As the external temperature is below 25°C, cooling of the I&C and electrical rooms provided by the DVL [SBVSE] is sufficient. However, CCF of two DEL [SCWS] trains leads to failure of two LHSI pumps, two associated DCL [CRACS] trains and two RBS [EBS] pumps if the failed DEL [SCWS] trains are the trains 1 and 4. The analysis shows that with the identified modifications the initiating event does not lead to a transient in the reactor or Spent Fuel Pool and that ambient temperatures in the MCR remain acceptable for operability of electrical and I&C equipment.

3) The CCF on two DEL [SCWS] trains (1&4 or 2&3) with external air temperatures above 25°C

The estimated frequency of this DBE is <10^{-5}/r.y. and the DBE is therefore analysed with the rules and success criteria presented in section 9.2.2.2. Hence the IE is combined with preventive maintenance, but the single failure criterion is not applied for the reasons given in section 9.2.2.2.

The DBA analysis shows that, with the design modifications identified in section 9.3, a controlled reactor state (and the safe shutdown state) can be established with safety features that are appropriately classified at Class 1 (and respectively Class 2). Relevant Spent Fuel Pool safety criteria are met. Ambient temperatures in the MCR remain acceptable for operability of electrical and I&C equipment.

9.4.2.2. Faults in other HVAC systems

The screening analysis [Ref-1] considered the impact of failures in other HVAC systems on normal operation and plant safety systems. For the majority of the HVAC systems, either no reactor transient was induced by the failure, or the associated reactor transient was already covered by an existing PCC analysis. The following HVAC failures were identified where resulting transients might not be bounded by existing PCC analysis:

- Loss of the DCL [CRACS] MCR HVAC system, which could lead to the need to evacuate the MCR due to the increased temperatures.
- Loss of the Circulating Water Pumping Station Ventilation System (DVP), which could cause failure of all SEC [ESWS] pump leading to a TLOCC event.

Taking into account the design modifications identified in section 9.3, it is considered that relevant safety criteria will now be met following these DBEs. The study [Ref-1] is continuing and further modifications to the detailed plant design may be identified in the NSL phase of licensing.

9.4.3. Faults in Electrical Systems

The Nuclear Island electrical distribution system provides essential support to all the main safety functions of the UK EPR. The design of the systems is described in Sub-chapter 8.3 of the PCSR.

Functional analysis has been performed to evaluate the impact of postulated IEs in the electrical support system and establish whether consequential transient was already covered by existing PCC analyses [Ref-1]. The analysis included postulated failures of the electrical supplies from individual switchboards as well as CCF of supplies from multiple switchboards at the same voltage level. The analysis has identified a number of bounding IEs which are not enveloped by current PCC events, and for which new DBA is therefore required. The bounding IEs and the new DBA results are described below.

9.4.3.1. Loss of supplies from single switchboard

The consequences of failures of supplies from a single switchboard in the LH (10kV), LJ (690V), LV (400V) and LA (220V) electrical distribution systems have been analysed [Ref-1]. It is concluded that loss of supplies from an LV or LA switchboard would not lead to a reactor or fuel pool transient, but that loss of supplies from an LH switchboard or an LJ switchboard would lead to a reactor trip. Loss of supplies from an LH switchboard was identified as a bounding DBE both in terms of consequences with respect to the equipment affected, and to the frequency of occurrence. This new DBE is considered as a frequent event (F>10^{-3}/r.y.).

A DBA of the new IE which involves failure of the electrical supplies from an LH switchboard has been performed [Ref-1], assuming the implementation of the design enhancements in section 9.3 of this sub-chapter. The analysis applies the rules and success criteria given in section 9.2.2.1. The analysis [Ref-1] concludes that taking into account the proposed design modifications, the controlled state can be established using Class 1 safety features, and the safe shutdown state can be established with safety features at least at Class 2. Additionally, a safe shutdown state can be established by alternate diverse means in the event of failure of the main line of protection, satisfying the diversity requirement for frequent faults. The consequences for cooling of the Spent Fuel Pool are bounded by existing PCC events.

9.4.3.2. Loss of supplies from multiple switchboards

A preliminary assessment of the consequences of a CCF of supplies from switchboards belonging to different electrical systems, which have the same voltage level was performed [Ref-1]. Preliminary analyses of CCF events in all sub-systems (LH, LL, LO, LA, LV and LJ) showed that the consequences of the CCF events on LH, LO, LA and LL sub-systems are less onerous than the loss of LV and LJ cases.

Therefore, the two following DBEs were identified as requiring further analysis in GDA: further analysis of the other cases is to be deferred until the NSL stage of licensing:

- Loss of supplies from LV 400V AC Uninterruptible Power Supply (UPS) System across all four divisions
- Loss of supplies from LJ 690V AC Power Supply System across all four divisions.

The estimated frequencies of these DBEs were in the range 10^{-5} to 10^{-6}/r.y.

The DBEs were analysed using the rules and success criteria applied in section 9.2.2.2 but with the following additional assumptions:

- Analysis was performed only for reactor in state A (power operation).
- The loss of electrical supplies is considered to be recovered after a period of approximately 12 hours.
- The possibility of multiple CCFs between electrical system switchboards within different systems was discounted (i.e. between LV, LJ, LH, LO or LL).
- No consideration was made of impacts of loss of supplies from LV or LJ on the Conventional Island.
- External temperatures are assumed to be in the range assumed for fault studies in GDA (-15°C/+36°C) but not to vary by more than 10 to 15°C in the period considered.
- The analysis was required to demonstrate the ability to reach the controlled state through using heat removal via the steam generators within the first 12 hours, and to demonstrate that the safe shutdown state can be reached afterwards following recovery of the electrical switchboards.

Results of the DBA are presented below.

1) <u>Total loss of supplies from LV (400V UPS) switchboards</u>

The estimated frequency of this DBE is <10^{-6}/r.y. Based on the previous reference design of the UK EPR (i.e. prior to the modifications described in section 9.3 above), the DBE would potentially lead to failures in multiple safeguard systems including:

- Automatic isolation of the RRI [CCWS] common headers, leading to loss of seal injection and loss of thermal barrier cooling on the four main cooling pumps, resulting in pump trip and potential small break LOCA due to failure of pump seals.

- Automatic closure of all four VIV [MSIV] valves occurs, leading to a reactor trip on main steam pressure.

- Reduced flow rate capability on all four ASG [EFWS] trains.

It is noted that the I&C systems remain available following this event because they will continue to be supplied from the LA switchboards.

The DBA shows that with the design modifications identified in section 9.3, a controlled reactor state can be established for 12 hours and the safe shutdown state can be established subsequently after recovery of the failed electrical supplies. The safety features claimed for fault recovery are appropriately classified. Spent Fuel Pool safety criteria are met and ambient temperatures in the MCR remain acceptable for operability of electrical and I&C equipment.

2) <u>Total loss of supplies from LJ (690V) switchboards</u>

The estimated frequency of this DBE is 10^{-5} to 10^{-6}/r.y. Based on the previous GDA reference design of the UK EPR, the DBE would potentially lead to failures in multiple safeguard systems including:

- Loss of all four main DVL/DEL [SBVSE/SCWS] trains so that HVAC in all four divisions is lost.

- Loss of HVAC in the MCR (Main Control Room) and the RSS (Remote Shutdown Station) which are supplied by DEL [SCWS] trains 2 and 3. This would lead to a requirement for evacuation of the MCR within less than one hour, and potential failure of the associated I&C equipment.

- Loss of spent fuel pool cooling through the loss of four main PTR [FPCS] pumps.

- Loss of ASG [EFWS] pumps in divisions 1 and 4.

- Unavailability of VDA [MSRT] valves, as two neighbouring I&C divisions are not available.

The DBA shows that with the design modifications identified in section 9.3, the controlled reactor state can be established for 12 hours and the safe shutdown state can be established subsequently after recovery of the failed electrical supplies. The safety features claimed for fault recovery are appropriately classified. Spent Fuel Pool safety criteria are met and ambient temperatures in the MCR remain acceptable for operability of electrical and I&C equipment.

9.4.4. Faults in Nitrogen Gas Distribution Systems

The purpose of the Nitrogen Gas Distribution System (SGN) is to ensure the supply of nitrogen which is required for various circuits and auxiliaries contained within the Nuclear Island. The Nitrogen supplied by the SGN is used for the following activities (see PCSR Sub-chapter 9.5):

- Flushing and filling of tanks
- Mixing chemical reagents in the secondary water of the steam generators
- Actuation of the standstill seals of the Main Coolant Pumps
- Pressure support for tanks

An assessment of the safety impact of potential faults in the Nitrogen Gas Distribution System (SGN) has been carried out [Ref-1]. The analysis concludes that faults in this system would not lead to any event which would cause the reactor to deviate from its normal operating range. Hence faults in the Nitrogen Gas Distribution System (SGN) do not need to be considered in this PCSR sub-chapter.

9.4.5. Faults in Instrument Air Systems

The main safety role of the compressed air system in the UK EPR is production and distribution of compressed air that supplies pneumatic valves, pneumatic control valves and other pneumatic equipment within Nuclear and Conventional Islands (see PCSR Sub-chapter 9.5).

An assessment of the safety impact of potential faults in the compressed air distribution system has been carried out [Ref-1]. It was identified that faults in individual sub-systems would only cause a reactor transient at reactor power levels above 4 %, for which the maximum consequences were bounded by existing PCC analyses or by failures in the RRI [CCWS] system already identified as a DBE in the assessment of faults in the cooling chain (see section 9.4.1). Similarly the consequences of CCF faults in the compressed air system were found to be bounded by TLOCC events already identified in the assessment of cooling chain faults. It was concluded that no new DBA analyses were required due to potential faults in the compressed air system.

9.5. CONCLUSIONS

This PCSR sub-chapter has provided Design Basis Analysis of faults in the following essential systems which support the operations of safeguard systems in the UK EPR:

- Safety Cooling Chain systems
- HVAC systems
- Electrical Systems
- Nitrogen Gas Distribution Systems
- Instrument Air System

New postulated Design Basis Events (DBEs) have been identified in the analysis, including initiating events resulting from both single failures in system trains and Common Cause Failures (CCFs) affecting multiple trains.

The new DBEs have been subjected to a Design Basis Analysis using analysis rules and success criteria applicable to PCC-type events for faults with frequencies above 10^{-5}/r.y. For CCF event which generally have frequencies below 10^{-5}/r.y., analysis rules and criteria have been adapted to take account of the low frequencies of the corresponding fault sequences.

This analysis has considered both reactor transients and transients in the spent fuel pool.

A number of design modifications have been identified to resolve those shortfalls against deterministic requirements. It has been shown that, taking into account the design enhancements, deterministic criteria applicable to UK EPR design basis faults are met for the support system analysed.

The majority of design changes presented in this sub-chapter have been developed to a level which will allow detailed design to begin. However, in some areas further design development is required, including selection of the preferred design option. This work will be carried out in the NSL phase of licensing.

This DBA of essential support systems, and its extension to all plant states, will be developed further in the NSL phase of licensing.

SECTION 16.4.9 - TABLE 1

Summary of Design Modifications to cope with Faults on Essential Support Systems

Related System	Related Fault	Description of Modification
RRI [CCWS]	LOCC - Single Train/Subsystem	Upgrade to Class 1 of the Reactor Coolant Pump thermal barrier cooling function and design improvement to increase robustness against RRI [CCWS] losses.
RRI [CCWS]	LOCC - Single Train/Subsystem	Upgrade to Class 1 of the Automatic Isolation of the Component Cooling Water system RRI [CCWS] train in case of leakage
RRI [CCWS]	LOCC - Single Train/Subsystem	Upgrade to Class 1 of the Automatic switchover to the stand-by RRI/SEC [CCWS/ESWS]) in case of loss of the operating RRI/SEC [CCWS/ESWS] train.
Reactor Coolant Pump	LOCC- Single Train/Subsystem	Upgrade to Class 1 of the Reactor Coolant Pump trip in case of loss of cooling to its motor bearings or in case of simultaneous loss of seal injection flow and thermal barrier cooling.
ASG [EFWS]	LOCC - Single Train/Subsystem	Provision of capability for re-alignment of the ASG [EFWS] Pump discharge from the Main Control Room (MCR).
DVL [SBVSE]) /DEL [SCWS]	HVAC - Single Train & CCF	Modifications on the DVL [SBVSE]) and DEL [SCWS] regarding the ventilation and air-conditioning function of electrical and I&C rooms including requirements upon the functional diversity of electrical power supply of the DVL/DEL [SBVSE/SCWS] trains.
RBS [EBS]	LOCC, HVAC & Electrical - Single Train/Subsystem	Implementation of a back-up electrical supply for RBS [EBS] trains (and required support systems)
I&C systems	LOCC, HVAC & Electrical - Single Train/Subsystem	Implementation of a back-up electrical supply for the I&C cabinet to recover one division
RIS [SIS]	CCF TLOCC	Upgrading/reliability improvement of the Automatic transfer of the cooling for the Low Head Safety Injection Pumps (LHSI) in trains 1 and 4 from the RRI [CCWS] to the DEL [SCWS].
RIS [SIS]	CCF TLOCC	Upgrading/reliability improvement of Automatic initiation of LHSI pumps in RRA [RHR] mode (Reactor State Cb)
RIS [SIS]	CCF TLOCC	Upgrading/reliability improvement of Automatic start of the LHSI in Safety Injection reduced flow rate mode (Reactor State Cb)

SECTION 16.4.9 - TABLE 1 (CONT'D)

Summary of Design Modifications to cope with Faults on Essential Support Systems

Related System	Related Fault	Description of Modification
EVU [CHRS] SRU [UCWS]	CCF TLOCC	Upgrade/reliability improvement of the cooling of the In-Reactor Water Storage Tank [IRWST] by the Containment Heat Removal system EVU [CHRS], including pumps and motors for the EVU [CHRS], the intermediate EVU [CHRS] and the Ultimate Cooling Water System SRU [UCWS].
VDA [MSRT]	CCF TLOCC	Upgrading/reliability improvement of actuation signals for Initiation of manual cooldown via the Main Steam Relief Train VDA [MSRT]
RIS [SIS]	CCF TLOCC	Upgrading/reliability improvement of actuation signals for Manual Isolation of the accumulators
EVU [CHRS]	CCF TLOCC	Upgrading/reliability improvement of actuation signals for Manual Actuation of the EVU [CHRS]
RCP [RCS]	CCF TLOCC	Upgrading/reliability improvement of actuation signals for manual Feed & Bleed actuation. This also includes upgrading to Class 1 of the associated RCP [RCS] valves and their electrical supplies.
RCP [RCS]	CCF TLOCC	Upgrading/reliability improvement of actuation signals for manual Steam Generator (SG) level control
PTR [FPCS] / IRWST	CCF TLOCC	Upgrading/reliability improvement of actuation signals for manual establishment of recirculation from the reactor pool to the IRWST via the PTR [FPCS] valves
PTR [FPCS]	CCF Electrical	Introduction of functional diversity for the electrical power supply of main PTR [FPCS] cooling train pumps
RRI [CCWS]	CCF Electrical	Introduction of functional diversity for the electrical power supply to the RRI [CCWS] Switchover valves (common user isolation valves) and RRI/SEC [CCWS/UCWS] heat exchanger regulation valves
APG [SGBS]	CCF Electrical	Introduction of functional diversity for the electrical power supply among the three isolation valves on each train for the APG [SGBS]
DVP	CCF HVAC	Mitigation against and/or control of potential consequences of failure of the DVP system
RCV [CVCS]	CCF Electrical	RCV [CVCS] modification – Seal injection of the RCS [RCP] pumps

SECTION 16.4.9 - TABLE 1 (CONT'D)

Summary of Design Modifications to cope with Faults on Essential Support Systems

Related System	Related Fault	Description of Modification
DEL [SCWS]	CCF HVAC	Recovery of 1 DEL [SCWS] train to support the MCR air-conditioning
DCL [CRACS]	CCF HVAC	Solution to manage CCF on DCL [CRACS] systems – DCL [CRACS] diversification and/or reliability improvements - Introduction of functional diversity for the electrical power supply of DCL [CRACS] trains
PTR [FPCS]	CCF HVAC	Management of scenarios on CCF of DVL/DEL with no PTR [FPCS] trains available

SUB-CHAPTER 16.4 – REFERENCES

External references are identified within this sub-chapter by the text [Ref-1], [Ref-2], etc at the appropriate point within the sub-chapter. These references are listed here under the heading of the section or sub-section in which they are quoted.

1. DOUBLE ENDED BREAK OF THE MAIN COOLANT LINE (2A-LOCA)

1.4. METHOD OF ANALYSIS

1.4.2. Key Assumptions

1.4.2.1. Computer Code and Modelling

1.4.2.1.1. Computer Code

[Ref-1] Science V2 nuclear code package – qualification report. NFPSD DC 89 Revision A. AREVA. March 2004. (E)

[Ref-2] Summary note on qualification of the COPERNIC/TRANSURANUS software version 2.4.
TFJC-DC 1547 Revision E. FRAMATOME-ANP. October 2003. (E)

[Ref-3] The CATHARE Code. DTP/SMTH/LMDS/EM/2001-063. CEA. 2001. (E)

1.4.2.2. Physical Models

[Ref-1] CATHARE 2 V2.5_1: Description of the FUEL sub-module SSTH-LDAS-EM-2005-043. CEA. 2005. (E)

[Ref-2] M5™ Properties Data (BE). FF DC 00267 Revision D. FRAMATOME-ANP. January 2005. (E)

1.5. RESULTS

1.5.1. Assessment of the IN-OUT UO_2 18 months fuel management – peak power located at the top of the core

1.5.1.1. Determination of the fuel data and hot assemblies

1.5.1.1.1. Fuel data

[Ref-1] Summary note on qualification of the COPERNIC/TRANSURANUS software version 2.4.
TFJC-DC 1547 Revision E. FRAMATOME-ANP. October 2003. (E)

1.5.2. Assessment for IN-OUT UO2 22 months fuel management – peak power located at the top of the core

1.5.2.1. Determination of the fuel data and hot assemblies

1.5.2.1.1. Fuel data

[Ref-1] Summary note on qualification of the COPERNIC/TRANSURANUS software version 2.4.
TFJC-DC 1547 Revision E. FRAMATOME-ANP. October 2003. (E)

1.5.3. Assessment for IN-OUT UO_2 18 months fuel management – peak power located at mid-core

1.5.3.1. Determination of the fuel data and hot assemblies

1.5.3.1.1. Fuel data

[Ref-1] Summary note on qualification of the COPERNIC/TRANSURANUS software version 2.4.
TFJC-DC 1547 Revision E. FRAMATOME-ANP. October 2003. (E)

1.5.5. Assessment for IN-OUT 30% MOX 18 months fuel management – peak power located at the top of the core

1.5.5.1. Determination of the fuel data and hot assemblies

1.5.5.1.1. Fuel data

[Ref-1] Summary note on qualification of the COPERNIC/TRANSURANUS software version 2.4.
TFJC-DC 1547 Revision E. FRAMATOME-ANP. October 2003. (E)

1.5.6. Assessment for IN-OUT 30% MOX 18 months fuel management – peak power located at mid-core

1.5.6.1. Determination of the fuel data and hot assemblies

1.5.6.1.1. Fuel data

[Ref-1] Summary note on qualification of the COPERNIC/TRANSURANUS software version 2.4.
TFJC-DC 1547 Revision E. FRAMATOME-ANP. October 2003. (E)

2. DOUBLE ENDED BREAK OF THE MAIN STEAM LINE OUTSIDE THE CONTAINMENT

2.2. MAIN STEAM LINES OUTSIDE THE REACTOR BUILDING

2.2.3. Description of studied cases

2.2.3.5. DNBR Criteria

[Ref-1] Doroschuk, V.E. and Lantsman, F.P. Effect of Pressure and Mass Flow Rate on Burnout Heat Fluxes in a Water and Steam-Water Mixture Flow in Tubes. International Journal of Heat and Mass Transfer V7 pp. 187-190. Pergamon Press. 1964. (E)

5. MULTIPLE STEAM GENERATOR TUBE RUPTURE (10 TUBES IN ONE STEAM GENERATOR AT POWER)

5.4. ANALYSIS FROM INITIATION OF EVENT TO FINAL STATE

5.4.6. Applicability of results to UK EPR

[Ref-1] UK EPR GDA Project - Reference Design Configuration. UKEPR-I-002. EDF/AREVA. (E)

7. SAFETY CASE FOR HETEROGENEOUS BORON DILUTION FAULT

7.1. INTRODUCTION

[Ref-1] UK EPR - Design Improvements for Heterogeneous Boron Dilution Fault. PEPR-F DC 97 Revision A. AREVA. June 2012. (E)

[Ref-2] EPR™ - Heterogeneous Boron Dilution Resulting from an Improper Reactor Coolant Pump Start-up. PEPR-G DC 100032 Revision A. AREVA. November 2011. (E)

[Ref-3] Heterogeneous Boron Dilution Fault Schedule. PEPR-F DC 67 Revision A. AREVA. July 2011. (E)

[Ref-4] UK EPR – Safety case for the inherent boron dilution following LOCA. PEPR-F DC 24 Revision A. AREVA. July 2011. (E)

[Ref-5] UK EPR – Countermeasures against Heterogeneous Dilution Initiators. PEPR-F DC 70 Revision A. AREVA. July 2011. (E)

7.2. MAXIMUM ACCEPTABLE SLUG SIZE

7.2.1. Assessment of maximum acceptable slug size

[Ref-1] STAR-CD – Qualification on dilution phenomena. NFPSD DC 119. AREVA. October 2004. (E)

[Ref-2] MANTA – Code Synthetic Qualification Assessment. NFPSD DC 85 Revision D. AREVA. September 2008. (E)

7.2.2. Validation of CFD Modelling

[Ref-1] CFD simulation of the Juliette Tests – STAR-CD Physical Validation for external and inherent dilution. PEPD-F DC 13 Revision B. AREVA. November 2011. (E)

8. SPENT FUEL POOL FAULTS

8.1. INTRODUCTION

[Ref-1] UK EPR GDA PROJECT: Safety Case for Spent Fuel Pool Cooling and Pool Drainage Faults. PTS DC 10 Revision C, AREVA. November 2012. (E)

8.2. SCOPE OF ASSESSMENT

8.2.3. Plant States and Configurations

[Ref-1] UK EPR GDA PROJECT: Safety Case for Spent Fuel Pool Cooling and Pool Drainage Faults. PTS DC 10 Revision C, AREVA. November 2012. (E)

8.3. FAULT ANALYSIS

8.3.2. Gross Failure of Fuel Transfer Tube

[Ref-1] Safety Assessment of Consequences Due to Gross Failure of Penetrations in the Fuel Pools. PEPS-G/2012/en/1033 Revision A. AREVA. September 2012. (E)

8.3.3. Failure of Technical Openings

8.3.3.2. Establishment of Controlled State

[Ref-1] Safety Assessment of Consequences Due to Gross Failure of Penetrations in the Fuel Pools. PEPS-G/2012/en/1033 Revision A. AREVA. September 2012. (E)

8.3.4. Leakage from the Cask Loading Pit

8.3.4.2. Establishment of Controlled State

[Ref-1] UK-EPR™ - Calculation of Discharge Outflow Rates for Different Scenarios. PEPR-G-12-0074 Revision B. AREVA. September 2012. (E)

8.3.4.3 Establishment of Safe Shutdown State

[Ref-1] Safety Assessment of Consequences Due to Gross Failure of Penetrations in the Fuel Pools. PEPS-G/2012/en/1033 Revision A. AREVA. September 2012. (E)

8.3.5. Leakage from the Cask Loading Pit due to Bellows Failure

8.3.5.2. Establishment of Controlled State

[Ref-1] FS03 – Impact on the Probabilistic Safety Assessment of the Initiating Events Identified for the Cask Loading Process. ECESN120833 Revision A. EDF. October 2012. (E)

8.3.5.4. Review of other Cask Loading Pit Faults

[Ref-1] FS03 – Failure Modes and Effect Analysis for the Spent Fuel Pool Isolation Gates, the Spent Fuel Transfer Facility and Burn Up Device.
ECESN110231 Revision A. EDF. May 2012. (E)

[Ref-2] Estimated reliability parameters of PTR [FPCS], PMC [FHS] and DMK (Spent Fuel Handling) systems equipment used for Fuel Building pool leak prevention and fuel handling. D4550.34-11/4134 Revision 0. EDF. April 2012. (E)

[Ref-3] FS-03 – Quantification of the new initiating events identified for the spent fuel cask transfer facility and spent fuel cask loading process.
ECESN120308 Revision A. EDF. May 2012. (E)

8.3.6. Flooding Consequences of Pool Leakage Events

[Ref-1] Safety Assessment of Consequences Due to Gross Failure of Penetrations in the Fuel Pools. PEPS-G/2012/en/1033 Revision A. AREVA. September 2012. (E)

9. ANALYSIS OF FAULTS IN ESSENTIAL SUPPORT SYSTEMS

9.1. INTRODUCTION

[Ref-1] UK EPR – Consistency between PSA list and PCC list. NEPR-F DC 584 Revision A. AREVA. July 2010. (E)

[Ref-2] UK EPR GDA Project - Reference Design Configuration. UKEPR-I-002. EDF/AREVA. (E)

9.3. REFERENCE DESIGN FOR LOSS OF SUPPORT SYSTEMS ANALYSES

9.3.1. Cooling Chain

[Ref-1] Faults in Essential Support System – ALARP Assessments, Proposed Design Changes and Justification of Resultant Design.
ECESN121088 Revision A. EDF. November 2012. (E)

9.3.2. Ventilation Systems

[Ref-1] GDA – DEL / DVL – Conceptual Design Note. ECECS121567 Revision A. EDF. November 2012. (E)

9.3.3. Electrical and I&C Systems

9.3.3.1. I&C systems

[Ref-1] Safety frame for common cause failure events on the cooling chain, and analysis of classification upgrade of TLOCC mitigation means
PEPSD-F/12.328. AREVA. July 2012. (E)

9.3.3.2. Electrical Systems

[Ref-1] Faults in Essential Support System – ALARP Assessments, Proposed Design Changes and Justification of Resultant Design. .
ECESN121088 Revision A. EDF. November 2012. (E)

9.3.4. Other Support Systems

9.3.4.1. Steam Generator Blowdown System (APG [SGBS])

[Ref-1] Faults in Essential Support System – ALARP Assessments, Proposed Design Changes and Justification of Resultant Design.
ECESN121088 Revision A. EDF. November 2012. (E)

9.3.4.2. Fuel Pool Cooling System (PTR [FPCS])

[Ref-1] Faults in Essential Support System – ALARP Assessments, Proposed Design Changes and Justification of Resultant Design.
ECESN121088 Revision A. EDF. November 2012. (E)

9.4. DESIGN BASIS ANALYSES FOR MAIN SUPPORT SYSTEMS

9.4.1. Faults in Cooling Chain (RRI/SEC [CCWS]/[ESWS])

9.4.1.1. Faults in individual trains and sub-systems

[Ref-1] Design Basis Analysis of single faults on essential support systems
ECESN120355 Revision A. EDF. June 2012. (E)

[Ref-2] Loss of Support Systems – Design Basis Analyses. PEPR-F DC 103 Revision B. AREVA. October 2012. (E)

9.4.1.2. Faults involving multiple safety trains

[Ref-1] Faults in Essential Support System – ALARP Assessments, Proposed Design Changes and Justification of Resultant Design.
ECESN121088 Revision A. EDF. November 2012. (E)

9.4.2. Faults in HVAC Systems

[Ref-1] Screening of the HVAC systems to establish the impact of their loss on normal operation systems and on safety systems. ECESN120253 Revision A. EDF. October 2012. (E)

9.4.2.1 Faults in the DVL/DEL [SBVSE/SCWS] system

[Ref-1] Faults in Essential Support System – ALARP Assessments, Proposed Design Changes and Justification of Resultant Design.
ECESN121088 Revision A. EDF. November 2012. (E)

9.4.2.1.1. Faults in individual trains

[Ref-1] Faults in Essential Support System – ALARP Assessments, Proposed Design Changes and Justification of Resultant Design.
ECESN121088 Revision A. EDF. November 2012. (E)

[Ref-2] Loss of Support Systems – Design Basis Analyses. PEPR-F DC 103 Revision B. AREVA. October 2012. (E)

9.4.2.2 Faults in other HVAC systems

[Ref-1] Screening of the HVAC systems to establish the impact of their loss on normal operation systems and on safety systems. ECESN120253 Revision A. EDF. October 2012. (E)

9.4.3. Faults in Electrical Systems

[Ref-1] Faults in Essential Support System – ALARP Assessments, Proposed Design Changes and Justification of Resultant Design.
ECESN121088 Revision A. EDF. November 2012. (E)

9.4.3.1. Loss of supplies from single switchboard

[Ref-1] Faults in Essential Support System – ALARP Assessments, Proposed Design Changes and Justification of Resultant Design.
ECESN121088 Revision A. EDF. November 2012. (E)

9.4.3.2. Loss of supplies from multiple switchboards

[Ref-1] Faults in Essential Support System – ALARP Assessments, Proposed Design Changes and Justification of Resultant Design.
ECESN121088 Revision A. EDF. November 2012. (E)

9.4.4. Faults in Nitrogen Gas Distribution Systems

[Ref-1] Faults in Essential Support System – ALARP Assessments, Proposed Design Changes and Justification of Resultant Design.
ECESN121088 Revision A. EDF. November 2012. (E)

9.4.5. Faults in Instrument Air Systems

[Ref-1] Faults in Essential Support System – ALARP Assessments, Proposed Design Changes and Justification of Resultant Design.
ECESN121088 Revision A. EDF. November 2012. (E)