ATWS CONTROL BY BORON INJECTION SYSTEM IN CAREM-25

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ABSTRACT

This work is oriented to analyze and develop knowledge for engineering support, about the response of an integral-type reactor in case of anticipated transients without SCRAM. Particularly, it was studied a diverse extinction system, called "Second Shutdown System" (SSS), consisting of an elevated tank, connected to a pressurizer that, by gravity force, injects a neutronic poison solution into the primary system coolant, interrupting the nuclear reaction.
A model of the SSS was developed using the well known code RELAP5, in order to represent the phenomena involved during the discharge stage. A multiple failure event is analyzed, a Station Black Out, including failure of SCRAM.
In this scenario, it is shown core power dependence with the coolant reactivity feedback due to changes in the secondary mass flow, before the shutdown is effective. On the one hand, a decrease of the secondary mass flow, induce a negative reactivity feedback in the reactor core, decreasing the core power generation. On the other hand, this produces an increment of the primary pressure and temperatures, leading the coolant reactor to a high energy state.
In this simulation, an important variable, due to physical characteristics of the studied nuclear reactor, is the growth of primary system coolant mixture volume. A margin is desirable in order the level to be far from steam intakes for safety systems. Different hypothesis are studied for the actuation of the system: an early actuation of the RHRS shows a maximum mixture volume increases of about 10.5 %; comparatively, if the RHRS is triggered by an alternative parameter (ie: high pressure), this increase would have been of 12.8 %. Thus, the adopted strategy shows an improvement in the level increment margin.
For this simulation, the SSS complies, in scenarios of Postulated Multiple Failure Events, with the shutdown of the reactor as diverse and redundant system of the neutron-absorbing elements system.

1. Introduction

1.1. About Multiple Failure Events and Defence in Depth

The concept of "Defence-in-Depth" [1] has been introduced in the field of nuclear safety in the early 1970s. This concept was gradually refined to constitute an increasingly effective approach combining both prevention of a wide range of postulated incidents and accidents and mitigation of their consequences. Incidents and accidents were postulated on the basis of single initiating events selected according to the order of magnitude of their frequency, estimated from general industrial experience.
The definitions of the different levels of DiD were set as to reflect an escalation from normal operation to accident so that if one level fails, a higher level comes into force. This does not mean that the situations considered in one level are systematically resulting from a failure of systems/features associated to the previous level of defence. The different levels of DiD were set as to cover the different situations that need to be considered in the design and operation of the plant. The approach was intended to provide robust means to ensure the fulfillment of each of the fundamental safety functions [2] of:

- Control Reactivity;
- Removal of heat from the reactor and from the fuel store;
- Confinement of radioactive material, shielding against radiation, as well as limitation of accident radioactive releases.

For new reactor designs, there is a clear expectation to address in the original design what was often "beyond design" for the previous generation of reactors, such as multiple failure events and core melt accidents, called Design Extension Conditions in [3].
The question has been discussed by RHWG [1] whether for multiple failure events, a new level of defence should be defined, because safety systems which are needed to control postulated single initiating events are postulated to fail and thus another level of defence should take over. However, the single initiating events and multiple failure events are two complementary approaches that share the same objective: controlling accidents to prevent their escalation to core melt condition.
Hence, at this stage of the discussion, it has been proposed to treat the multiple failure events as part of the 3rd level of DiD, but a clear distinction between means and conditions (sub-levels 3.a and 3.b). Level 4 remains now exclusively for core melt scenarios, which are dramatically different from Multiple Failure Events without core melt, which are now included in level 3b. Levels 1, 2 and 5 remain the same as in previous discussions.
The refined structure of the level of DiD proposed by RHWGR is shown in Tab 1.

<table>
<tr><th colspan="2">Levels of Defence in Depth</th><th>Objective</th><th>Essential means</th><th>Radiological consequences</th><th>Associated plant condition categories</th></tr>
<tr><td rowspan="2">Level 3</td><td>3-a</td><td rowspan="2">Control of accident to limit radiological releases and prevent escalation to core melt conditions</td><td>Reactor protection system, safety systems, accident procedures</td><td rowspan="2">No off-site radiological impact or only minor radiological impact</td><td>Postulated single initiating events</td></tr>
<tr><td>3-b</td><td>Additional safety features, accident procedures</td><td>Postulated multiple failure events</td></tr>
<tr><td colspan="2">Level 4</td><td>Control of accidents with core melt to limit off-site releases</td><td>Complementary safety features to mitigate core melt, management of accidents with core melt (severe accident)</td><td>Off-site radiological impact may imply limited protective measures in area and time</td><td>Postulated core melt accidents (short and long term)</td></tr>
</table>

Tab 1: Levels of Defence in Depth, proposed by RHWG.

1.2. About CAREM-25 reactor

CAREM [4] is an Argentine project to achieve the development, design, and construction of an innovative simple and small Nuclear Power Plant (NPP), which is developed by CNEA (National Atomic Energy Commission). This nuclear plant has an indirect cycle reactor with some

distinctive and characteristic features that greatly simplify the design, and contributes to a higher safety level. Some of the high level design characteristics of the plant are: integrated primary cooling system, self-pressurised primary system and safety systems relying on passive features.
The CAREM nuclear power plant design is based on a light water integrated reactor. The whole high energy primary system, core, steam generators, primary coolant and steam dome, is contained inside a single pressure vessel (Fig 1).

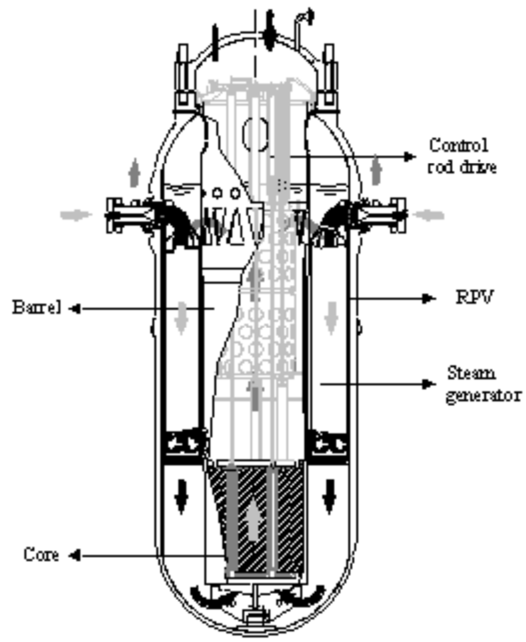
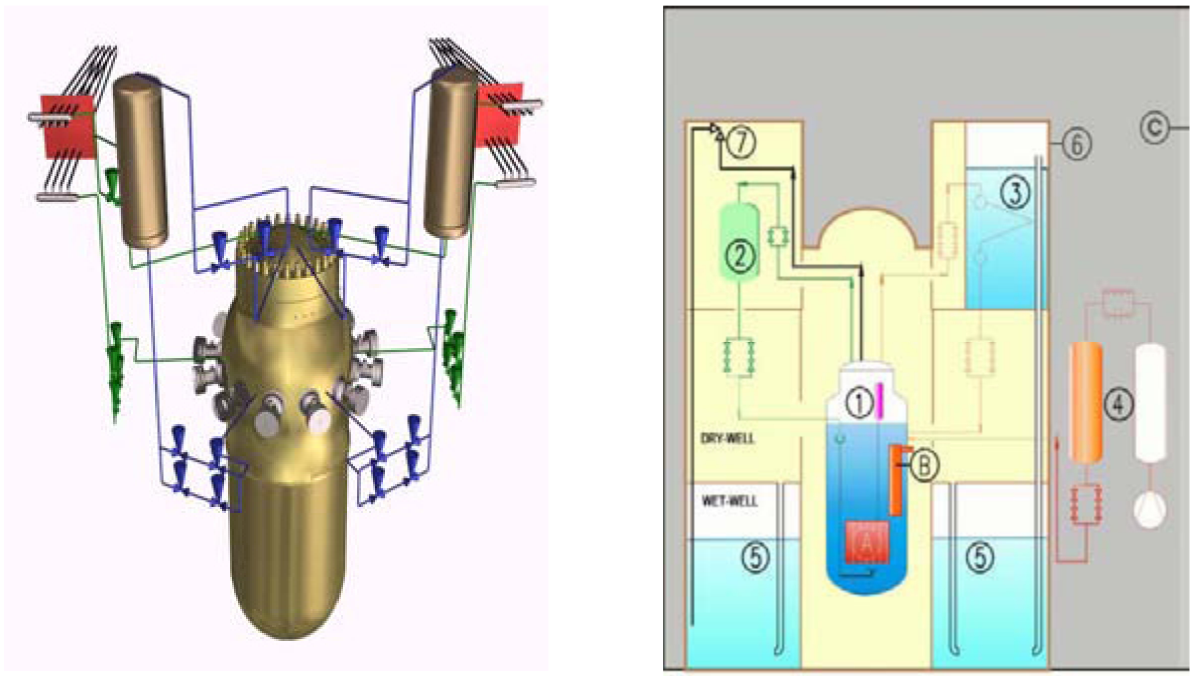


Fig 1. Reactor pressure vessel

For low power modules (which is the case of the prototype, CAREM-25, studied in this work), the flow rate in the reactor primary systems is achieved by natural circulation. Fig 1 shows a diagram of the natural circulation of the coolant in the primary system. Water enters the core from the lower plenum. After it's heated the coolant exits the core and flows up through the riser to the upper dome. In the upper part, water leaves the riser through lateral windows to the external region. Then it flows down through modular steam generators, decreasing its enthalpy. Finally, the coolant exits the steam generators and flows down through the down-comer to the lower plenum, closing the circuit. The driving forces obtained by the differences in the density along the circuit are balanced by the friction and form losses, producing the adequate flow rate in the core in order to have the sufficient thermal margin to critical phenomena. Reactor coolant natural circulation is produced by the location of the steam generators above the core.
Nuclear safety [5] has been incorporated in CAREM 25 since the beginning of the design. The defence-in-depth concept has specially been considered. Many intrinsic characteristics contribute to the avoidance or mitigation of eventual accidents.
CAREM 25 safety systems (Fig 2) are based on passive features and must guarantee no need of active actions to mitigate the accidents during a long period. They are duplicated to fulfill the redundancy criteria. The shutdown system should be diversified to fulfill regulatory requirements.
The First Shutdown System (FSS) is designed to shut down the core when an abnormality or a deviation from normal situations occurs, and to maintain the core sub-critical during all shutdown states. This function is achieved by dropping a total of 25 neutron-absorbing elements into the core by the action of gravity.
The Second Shutdown System (SSS) is a gravity-driven injection device of borated water at high pressure. It actuates automatically when the Reactor Protection System (RPS) detects the failure of the FSS. The system consists of two tanks located in the upper part of the

containment. Each of them is connected to the reactor vessel by two piping lines: one from the steam dome to the upper part of the tank, and the other from a position below the reactor water level to the lower part of the tank. When the system is triggered, the valves open automatically and the borated water drains into the primary system by gravity. The discharge of a single tank produces the complete shutdown of the reactor.



1: First Shutdown System
2: Second Shutdown System
3: Residual Heat Removal System
4: Emergency Injection System
5: Pressure suppression pool
6: Containment
7: Safety valves
A: Core B: Steam Generators C: Reactor Building

Fig 2. Safety systems

The Residual Heat Removal System (RHRS) has been designed to reduce the pressure on the primary system and to remove the decay heat in case of loss of heat sink. It is a simple and reliable system that operates condensing steam from the primary system in emergency condensers. The emergency condensers are heat exchangers consisting of an arrangement of parallel horizontal U tubes between two common headers. The top header is connected to the reactor vessel steam dome, while the lower header is connected to the reactor vessel at a position below the reactor water level. The condensers are located in a pool filled with cold water inside the containment building.

The Emergency Injection System (EIS) prevents core exposure in case of LOCA. In the event of such accident, the primary system is depressurised with the help of the emergency condensers to less than 15 bar, with the water level over the top of the core. At 15 bar a low pressure water injection system comes into operation.

2. Analysis of postulated multiple failure events

As we seen in section 1.1, a Postulated Multiple Failure Event (PMFE) is located at the sub-level 3-b of Defence in Depth, this means that, in addition to the Postulated Single Initiating Event (PSIE), it is assumed the failure of an important safety system. In this paper, We are interested in events that are composed by the failure of the First Shutdown System (FSS) as the additional failure in the PSIE, be this failure given by the non-detection of the PSIE by the First Reactor

Protection System (FPRS), or by the failure of the closure of the control rod mechanisms feedwater valve, or by a control rod mechanical failure. In all these cases it is assumed that the reactor extinction is given by the Second Shutdown System (SSS).
The annual occurrence frequency of these events is less than 10^{-4} 1/year, so the acceptation criterion limits, for safety important variables, should be less restrictive than those applied to sub-level 3-a. However, these limits will be taken as the reactor performance parameters.

2.1. Station Black Out with FSS failure.

2.1.1. Simulation hypothesis.

For Station Black Out simulation with FSS failure, and success of SSS with RHRS conditioned to SSS actuation, and discharge of both SSS tanks, the following hypotheses are proposed:

- Initial power: 100 %.
- Failure of diesel generators and systems that can contribute to the primary system cooling.
- Loss of feedwater supply to steam generators in 2 seconds. Thus, there is not heat removal from secondary system.
- Abrupt closing of MSIV at 2 s
- Failure of FSS.
- No credit to the "loss of feedwater" SCRAM signal.
- FSS demanded by 2nd valid SCRAM signal.
- SSS demanded 3 s after the FSS demand, if the neutron flux keeps higher than 20%.
- Passive RHRS is demanded with SSS signal.
- Energy removed for RHRS at nominal pressure 2.4 % of nominal power.
- FSS
  - It is assumed a mechanical failure of control rod, so the neutron-absorbing elements cannot be inserted into the reactor core.
- SSS
  - Discharge success of both SSS tanks is assumed.

2.1.2. Station Black Out with FSS failure, success of SSS and RHRS

Phenomenologically, the PMFE can be described in three phases:

<u>Phase 1 (0 s – 55.1 s)</u>:

This event starts with a total loss of electrical energy, onsite and offsite. Because of that, the pumps cannot keep the steam generators mass flow, reducing the flow from 100% to 0 % in 2 s. On the other hand, due to the loss of power mass flow to the FSS control rod mechanisms is also interrupted, so it is expected the imminent reactor SCRAM. By simulation hypothesis, the neutron-absorbing control rods had a mechanical failure, so the SCRAM by the FSS is not possible.
During the first second, the "low voltaje" and "low feedwater flow" are detected and discarded by simulation hypothesis.
The pressure of the secondary system increases, due to MSIV closure, causing "secondary system's high pressure" at t = 3 s. By simulation hypothesis, the reactor shutdown is required with a 1 s delay by the FRPS.
Due to the rod drop failure, SSS is demanded because the neutron flux keeps higher than 20% 3 s after FSS demand. At this moment the pressure equalization valves are opened.

Decrease of power removal from primary system (Fig 3), increases the cold leg temperature (Fig 4), initially at steam generators exits – primary side – and then through the downcomer. This causes a coolant expansion increasing the primary system's mixture volume (Fig 5), thus compressing the vapor zone and then increasing the primary system pressure (Fig 6) and the saturation temperature, so the primary system remains subcooled.
The increase in mixture volume is in fact due to three contributors: the imbalance between the core power generation and the removed from the primary system, the void generation by action of the RHRS, and finally, by the incoming volume of borated solution from the SSS tanks.
On the other hand, the buoyancy force is reduced by the increment of the downcomer temperature leading to a decrease of the primary system natural circulation mass flow (Fig 7).
A slight increase in core power is observed during the first seconds. This is due to self-pressurization dynamic: due to the power removal loss from the primary system (Fig 3), the primary pressure increases (Fig 6), causing a bubble collapse in the hot leg, particularly in the core region, inducing a positive reactivity feedback by moderator density (Fig 8). Nevertheless, the reduction of the natural circulation mass flow causes a longer residence time of primary system coolant in the nucleus region, increasing the coolant average temperature (Fig 4), which compensates the self-pressurization effect keeping the maximum power below 105 %.
Later, core power generation decreases.
At t = 12 s the SSS discharge valve opens, due to piping distribution the borated solution injection to the primary system coolant will starts at t = 21 s.
As a result of the decrease of core power, the clad and fuel temperatures decreases and the critical heat flux margins will increase, without risking the fuel integrity.
Due to the power imbalance in the primary system the mixture volume (Fig 5) and pressure will continue to grow during this phase.
At t = 17 s RHRS is triggered by simulation hypothesis (5 s later than the SSS discharge valves been opened). The RHRS action is relevant during this transient. On the one hand, it removes power from the primary system, in particular removing steam from the RPV dome, limiting the primary system pressure rise within a maximum of 12.8 MPa at 27 s. On the other hand, the RHRS trigger increase the hot leg void fraction generation increasing the buoyancy force resulting in a primary mass flow of approximately 300 kg/s (Fig 7).
At t = 30 s, the hot thermal front reaches to core inlet, decreasing the moderator density and increasing the negative reactivity feedback accordingly. Therefore, at this point, the core power generation has a decrease only due to negative reactivity feedback.
The end of this phase occurs at t = 55 s, when the boron concentration (Fig 9) introduces a negative reactivity (Fig 8) which is capable to shutdown the reactor core. In this moment, the reactor power is equal to 47 MW.

<u>Phase 2 (55 s – 170 s):</u>

This phase begins with the arrival of the boron front to the core region. The core power generation will decrease throughout this phase to decay power.
The primary system pressure starts to decrease due to the RHRS action and the saturation temperature accordingly. This causes a void fraction increase in the hot leg. However, because of the loss of steam generators heat removal, the temperatures of the primary system (particularly in the cold leg regions) are increased. Therefore, the buoyancy force in the primary system is reduced, achieving a minimum 175 kg/s.
Due to the action of the RHRS the primary coolant reaches saturation, void fraction increases in the hot leg and the mass flow increases to 275 kg/s at the end of this phase.
This phase ends when the core power generation equals to decay power and the primary system achieves saturation.

Phase 3 (170 s – 5000 s):

After both legs reach saturation temperature, it begins a saturated depressurization phase; the RHRS removes the decay power and cools the primary system.

Nevertheless, the primary system level continues to increase due to the mass injected by the SSS to the primary system, until t = 670 s when the boron solution injection ends. The maximum mixture volume reached is close to 10.5 %. Thus, the primary system mixture volume begins to decrease due to RHRS power removal that causes a density coolant increase.

A correct shutdown of the reactor core and a correct extraction of heat caused by the decay power are observed until the simulation end.

Figures:

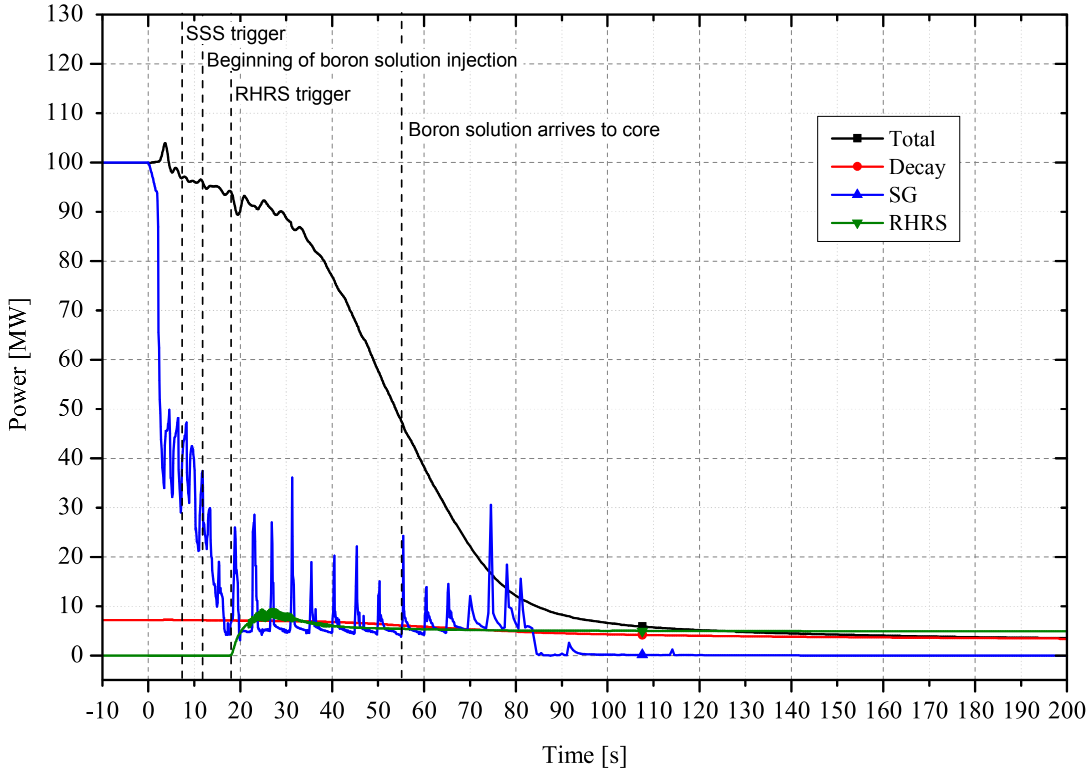


Fig 3: Power.

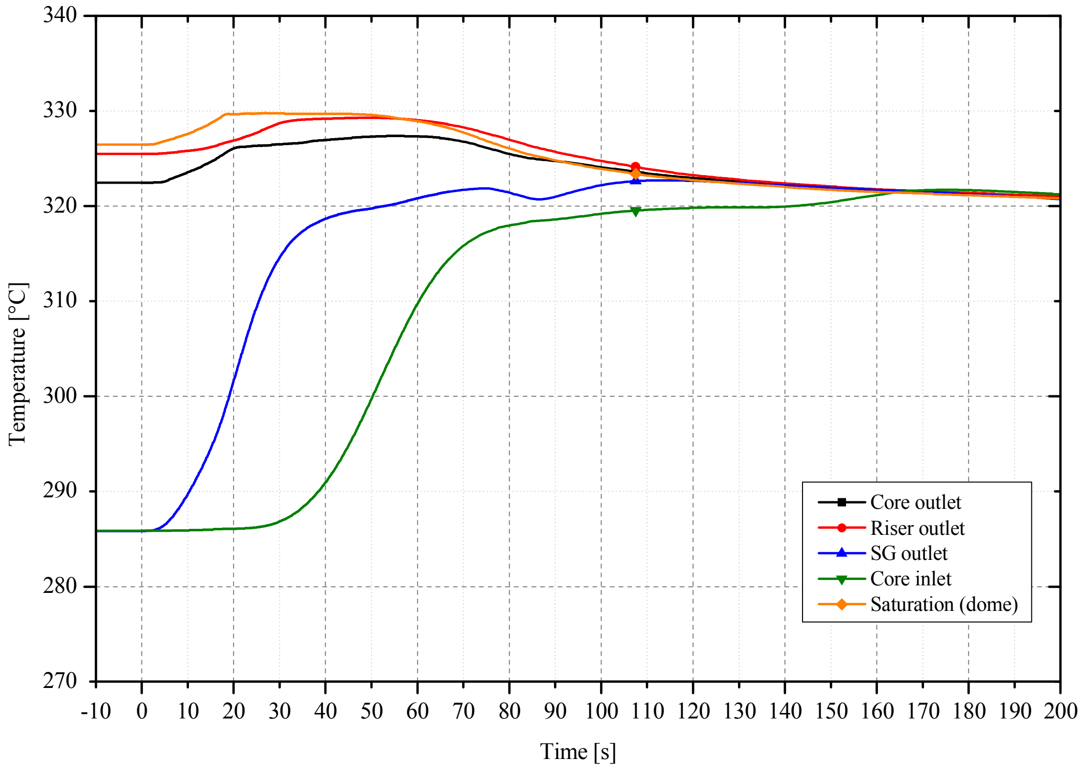


Fig 4: Primary system temperature at different locations, short therm.

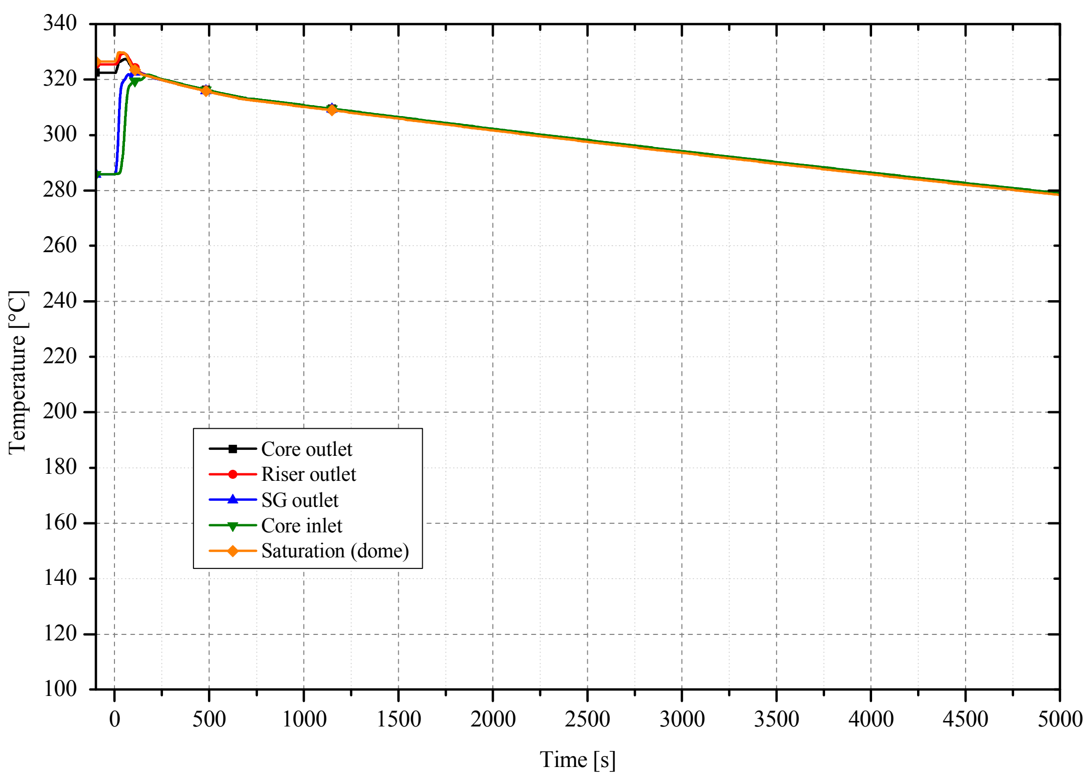


Fig 4_a: Primary system temperature at different locations, long therm.

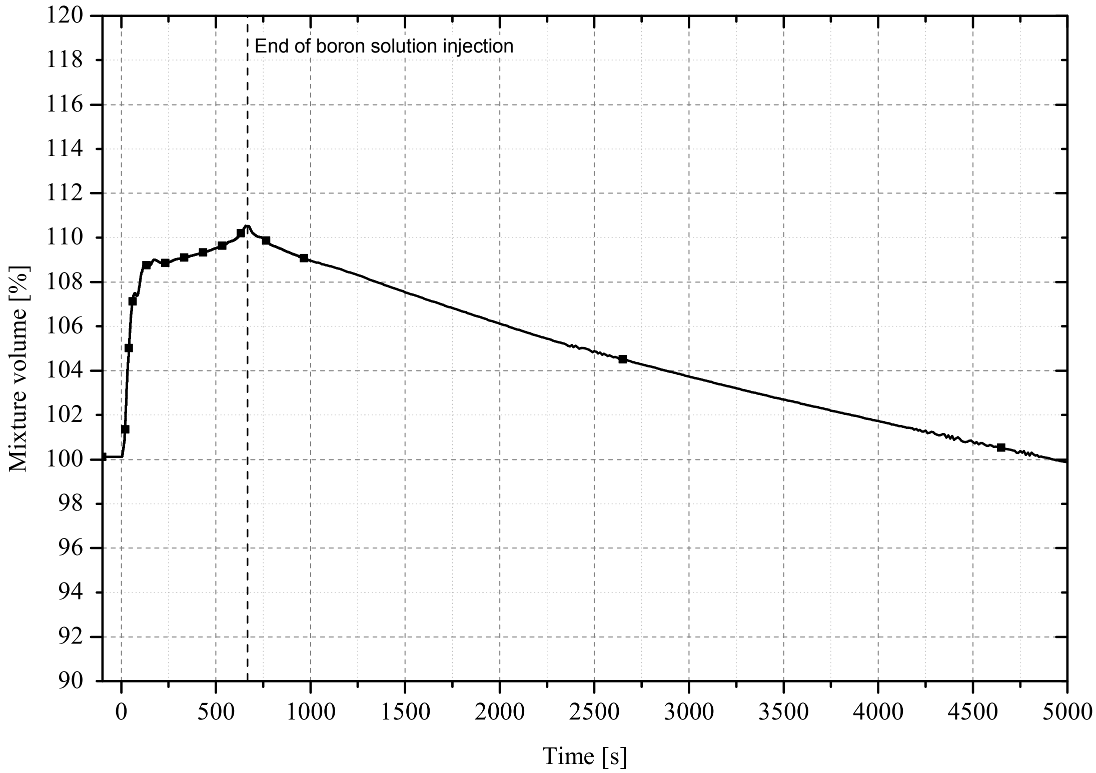


Fig 5: Mixture volume.

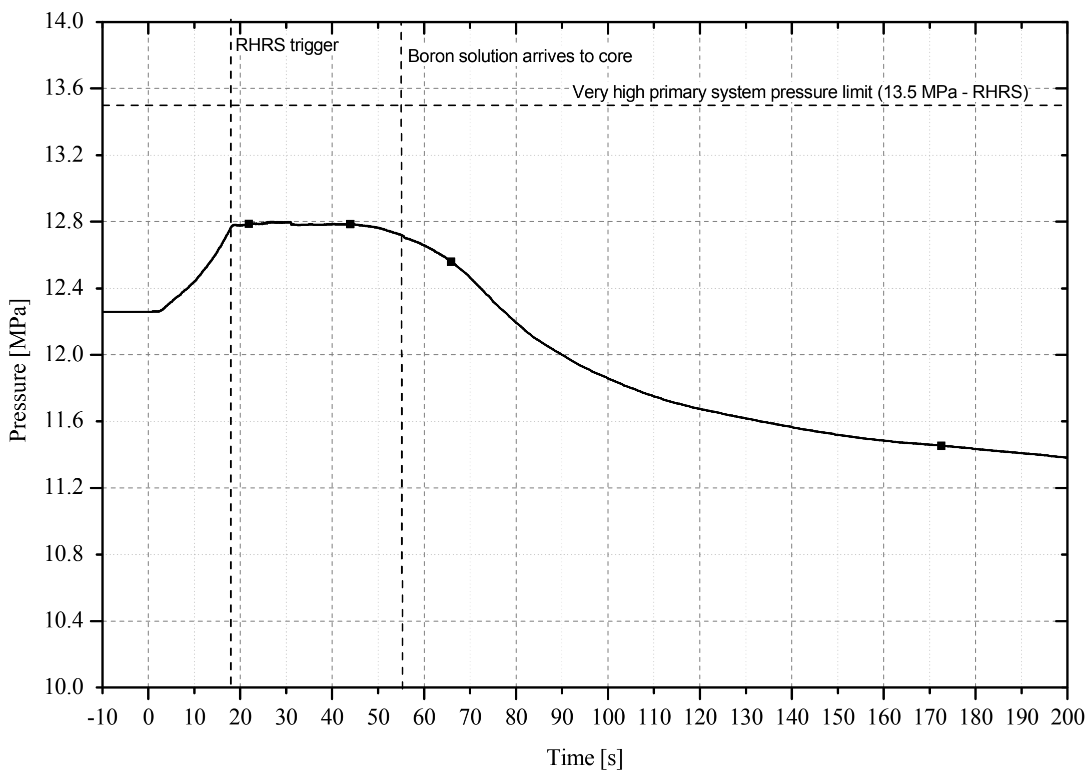


Fig 6: Primary system pressure.

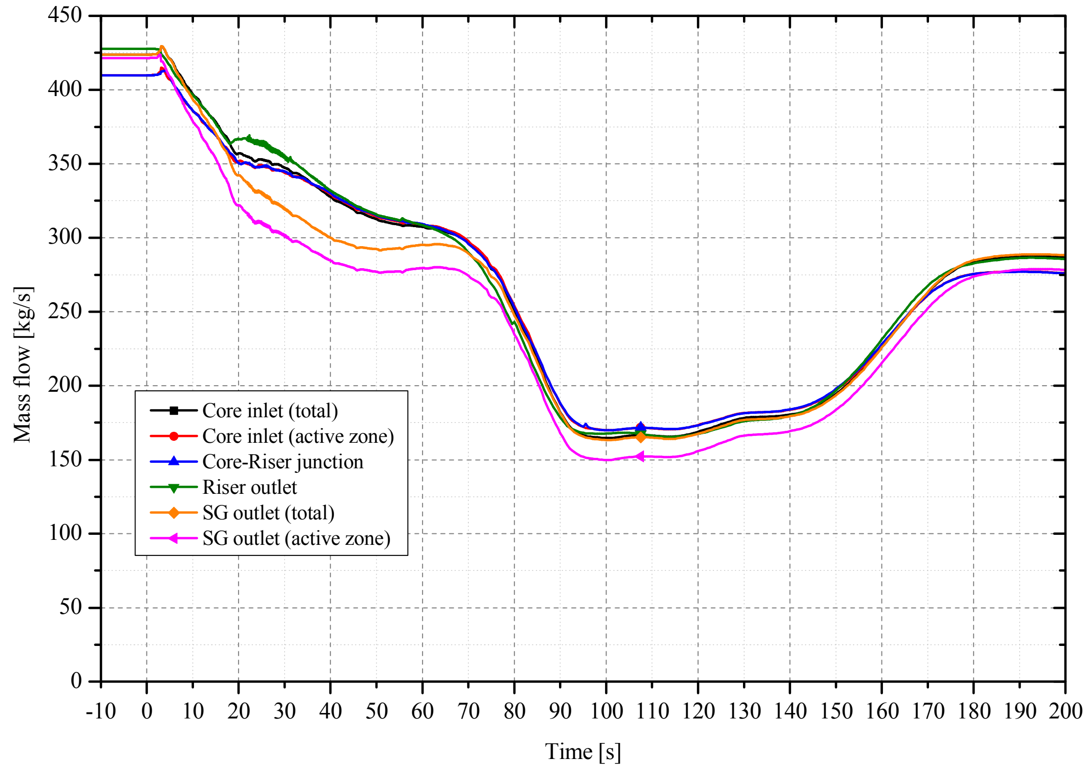


Fig 7: Primary system mass flow at different locations.

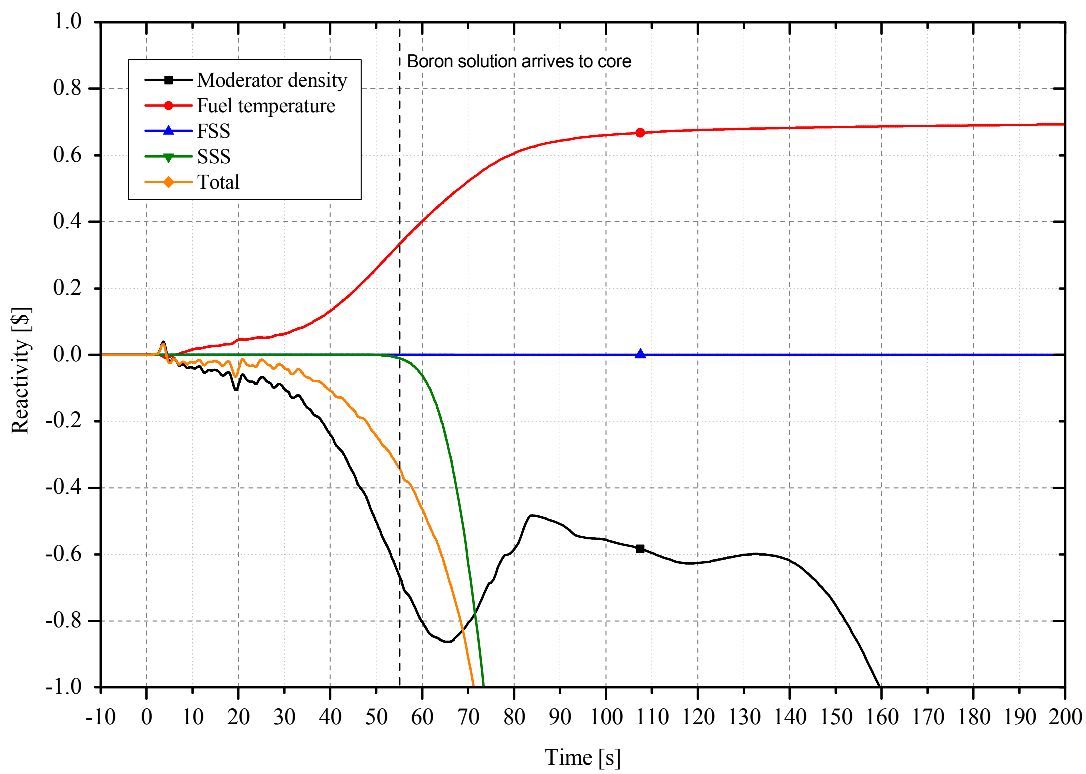


Fig 8: Reactivity, short therm.

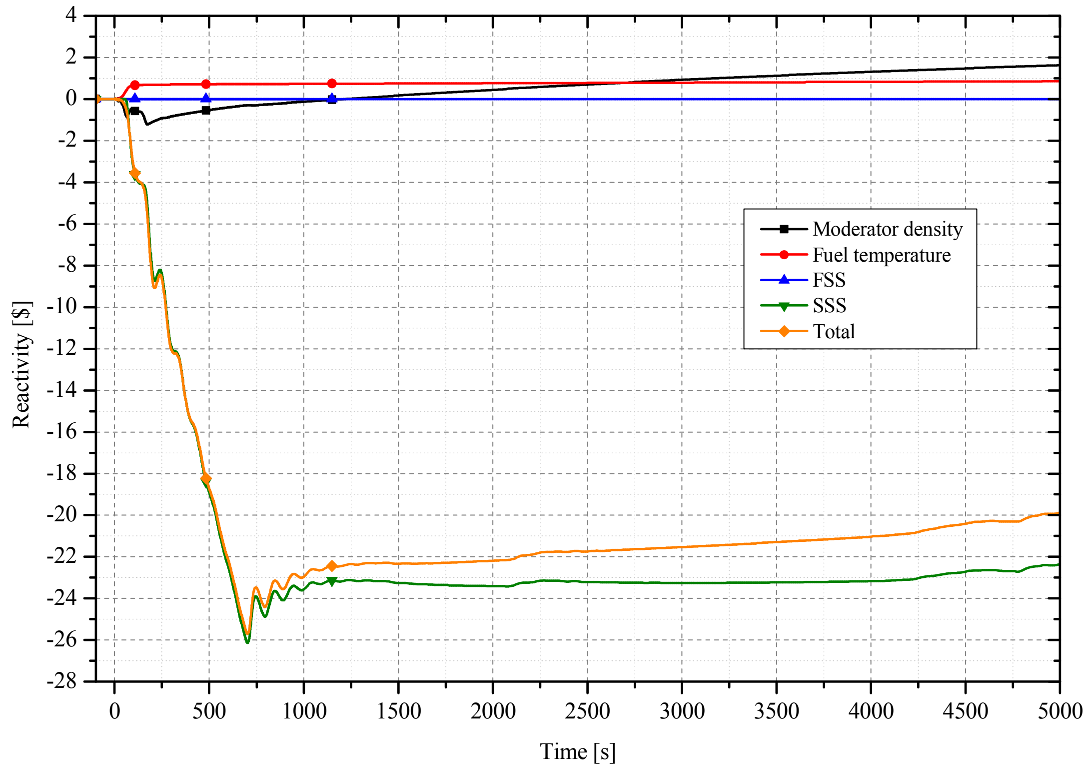


Fig 8_a: Reactivity, long therm.

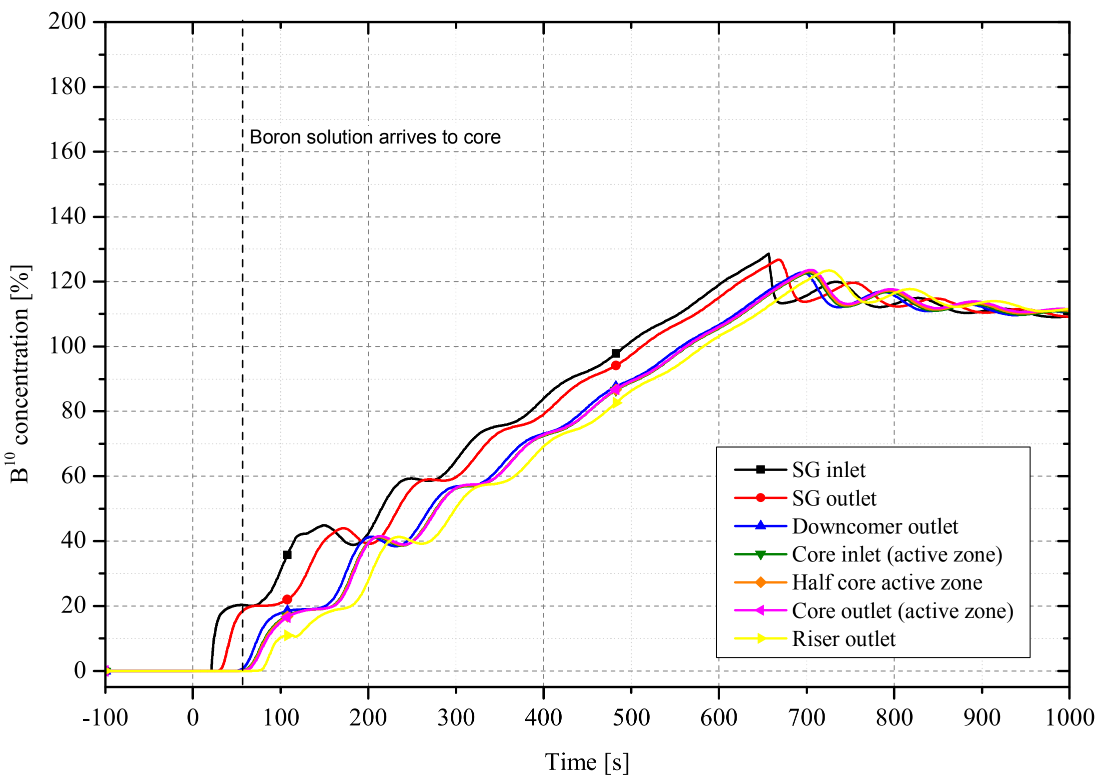


Fig 9: Primary system boron concentration.

2.2. Further studies

Some studies were carried out regarding possible trips for RHRS during this Event. In the previous section, the triggering of RHRS jointly with SSS signal is shown. In this section an alternative is studied, triggering both systems by an alternative parameter, like high primary system level or very high primary system pressure.
Phenomenologycally, this event has also three phases. In the first phase, the initial 12 s are indentically to section 2.1.2 event. Then, as long as the RHRS is not triggered 5 s after the SSS signal, the power imbalance (Fig 10) in the primary system continues to increase both the mixture level and the pressure of the primary system during this phase (Fig 11 and Fig 12). The level increment is due to coolant expansion by density primary system coolant decrease. At t = 31 s, the primary system pressure reaches 13.5 MPa so the RHRS is triggered.
In the second phase the boron front reaches the core region a second later than the section 2.1.2 simulation. Then the event is similar to what was observed in the previous section.
The representative figures of the mentioned changes are shown below.
It can be seen an improvement of 10.5 % in the maximum level when RHRS is triggered jointly with SSS; therefore this is the preferred alternative.

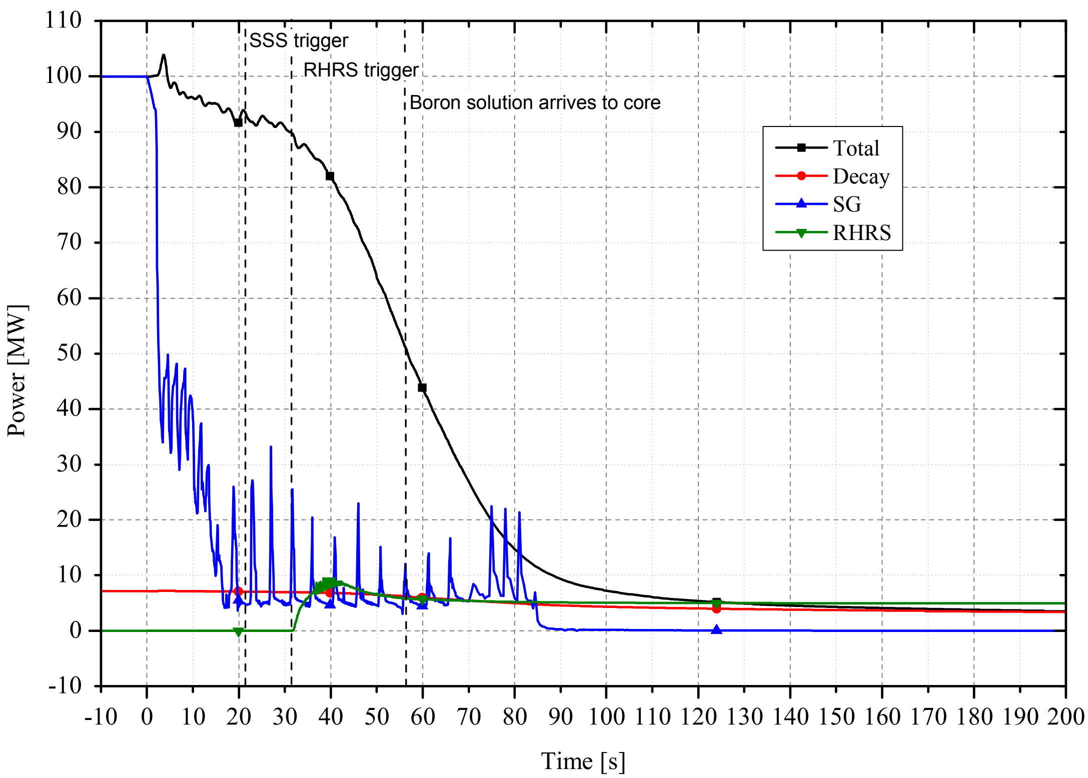


Fig 10: Power.

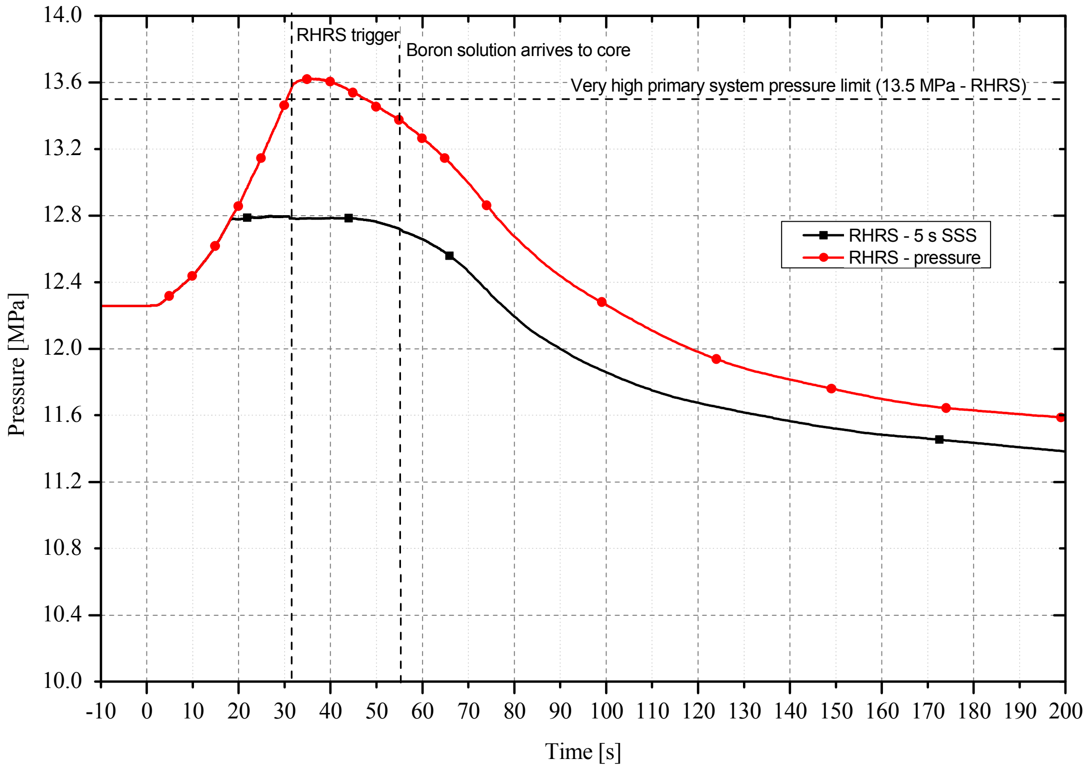


Fig 11: Primary system pressure.

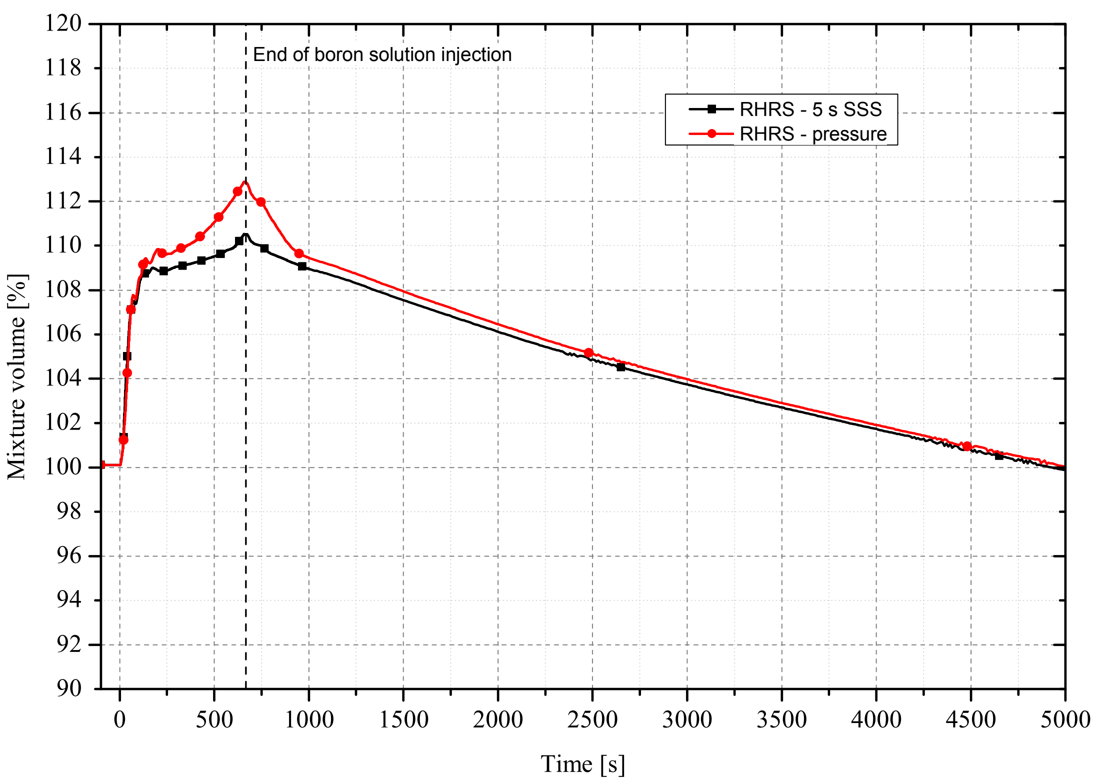


Fig 12: Mixture volume.

2.3. Conclusions

In this paper it was analyzed the PMFE "Station Black Out with FSS failure and success of SSS with RHRS conditioned to SSS actuation, and discharge of both SSS tanks". It was observed that before the FSS failure, the SSS is able to fulfil the fundamental safety functions for which it was design.

In this simulation, an important variable, due to physical characteristics of the studied nuclear reactor, is the growth of primary system coolant mixture volume. In this event, this growth is due to three contributors: the imbalance between the core power generation and the removed from the primary system, the void generation by action of the RHRS, and finally, by the incoming volume of borated solution from the SSS tanks. It is assumed that, by simulation hypothesis, the RHRS actuation was conditioned to SSS signal with a delay of 5 s after SSS discharge valves were opened.

Under this hypothesis the coolant mixture volume increases about 10.5 %; comparatively, if the RHRS is triggered by an alternative parameter (ie: high pressure), this increase would have been of 12.8 %. Therefore, the adopted strategy shows an improvement in the level increment margin, so the level will be far of the inlet to safety systems, like the SSS or the RHRS, avoiding a possible poor performance.

Finally, it is recommended to analyze the feasibility of having a RHRS trigger conditioned to the SSS demand and/or others ways of minimizing the power imbalance between the generated and removed from the primary system, in order to reduce the maximum mixture level reached.

2.4. References

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