SAFETY ANALYSIS OF THE MYRRHA REACTOR

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ABSTRACT

In the framework of the MYRRHA project, an experimental fast-spectrum irradiation facility operated with Lead Bismuth Eutectic (LBE) coolant and able to operate in both sub-critical and critical mode is designed to be built in Mol, Belgium. In addition to material testing, the objectives of the MYRRHA reactor are to prove the feasibility of the ADS technology as Minor Actinides burner and to act as a demonstration plant for future Generation IV heavy metal cooled reactors. SCK•CEN entered the pre-licensing phase for the MYRRHA reactor.

Safety analyses of the MYRRHA system using system codes are currently performed within the European Framework Program project MAXSIMA. The analyses are aimed at the verification of the capability of the safety systems to bring the reactor to a safe shutdown state after a postulated accidental event. The safety analyses focus on three different categories of transient events, including protected transients to analyze the protection system and decay heat removal system capabilities, protected transients to check on the overcooling risk and unprotected transients to define and set up the protection requirements.

This paper describes the results obtained with the RELAP5-3D ver. 4.0.3 system code on three selected transients for the latest design version of MYRRHA: loss of offsite power, overcooling + loss of flow and unprotected loss of flow.

In addition to the deterministic transient studies, the MAXSIMA project foresees the application of uncertainty quantification techniques. The capabilities of the Uncertainty + Sensitivity (U + S) analysis application SUSA is illustrated in this paper for the unprotected loss of flow accident.

List of acronyms

MYRRHA: Multi-purpose hYbrid Research Reactor for High-tech Applications
LBE: Lead Bismuth Eutectic
ADS: Accelerator-Driven System
PHX: Primary Heat eXchanger
PP: Primary Pump
DHR: Decay Heat Removal
RVACS: Reactor Vessel Auxiliary Cooling System
IVST: In Vessel Storage Tank
LOOP: Loss Of Offsite Power
HTC: Heat Transfer Coefficient
LOFA: Loss Of Flow Accident
SRCC: Spearman Rank Correlation Coefficients
PCT: Peak Clad Temperature

1. INTRODUCTION

MYRRHA (**M**ulti-purpose h**Y**brid **R**esearch **R**eactor for **H**igh-tech **A**pplications) is a pool-type Accelerator Driven System (ADS) with the ability to operate also as a critical reactor. It is cooled by liquid lead-bismuth eutectic (LBE). Its main targets can be summarized as:

- Flexible fast-spectrum irradiation facility
- Minor Actinides transmutation demonstration
- ADS demonstrator
- Gen IV Lead Fast Reactor prototype

The MYRRHA project has been recognized as a high priority infrastructure for nuclear research in Europe. Several European FP6 and FP7 projects had, as main target, to finalize a preliminary design of the MYRRHA reactor:

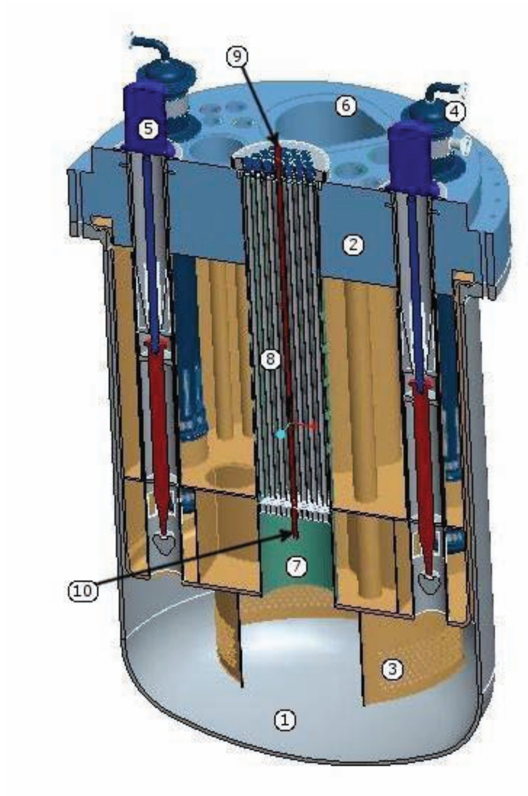
- FP6 IP-EUROTRANS [1], leading to the finalization of MYRRHA/XT-ADS version of MYRRHA in June 2008 [ref to paper on XT-ADS]
- FP7 Central Design Team (CDT) [2], defining the MYRRHA/FASTEF version in March 2012
- FP7 MAXSIMA [3] (started in November 2012, ongoing), more focused on the MYRRHA safety analyses and component qualification

The outcome of these European FP projects has been used to define the latest version of the MYRRHA design (MYRRHA Design Version 1.6), which has been finalized in June 2014 [5] and is currently in the verification phase. Though representing the state of the art, Design Version 1.6 is not definitive: the MYRRHA design is still evolving taking into account results from the parallel R&D program.

SCK•CEN has actively participated in these FP6 and FP7 projects focusing on the safety analysis through use of system codes by performing code-to-code comparison of steady-state and transient calculations on the MYRRHA reactor operating in sub-critical and critical mode.

2. MYRRHA CURRENT STATUS DESIGN CONFIGURATION

In Figure 1 it is possible to recognize the main primary system components of MYRRHA Design Version 1.6.



1. Reactor vessel
2. Reactor cover
3. Diaphragm
4. Primary heat exchanger
5. Pump
6. In-vessel fuel handling Machine
7. Core barrel
8. Above core structure
9. Core plug
10. Spallation window

Figure 1 - Overview of the MYRRHA reactor (in ADS mode)

The primary system is completely enclosed in the primary vessel (pool-type system). The primary LBE coolant flows from the lower plenum into the core (T ~ 270 °C) to remove the core power (100 MW in critical mode) and, from there, into the upper plenum where it mixes with the cold by-pass flow. The average upper plenum temperature is 325 °C. Four Primary Heat eXchanger (PHXs) units receive the LBE from the upper plenum, which then flows into two Primary Pumps (PPs) (one PP serving two PHXs). From the PPs the LBE is reinserted into the lower plenum.
The cold lower plenum is separated from the hot upper plenum by the Diaphragm, an inner vessel structure supporting the core barrel and the penetrations for the PHXs and the PPs. Above the LBE free surface level an inert gas layer (Nitrogen) separates the primary coolant from the reactor cover.

The primary system is linked to four independent secondary systems through the four PHX units. Each secondary system is operated in a forced-flow regime with a two-phase water mixture at 16 bar (~200 °C): the water enters the PHX in almost saturated conditions and exits with a quality ~0.3. The moisture is then separated in a steam drum, from where the steam is directed towards an air condenser (one per secondary loop) and the water is recirculated to the PHX. In normal operation the secondary water temperature is kept constant by the control system, letting the primary LBE conditions to change in function of the core loading.

The steam dissipates the heat to the external environment through the tertiary system air condenser and is then recirculated into the steam drum. Each tertiary system contains an air fan operated in forced circulation and logically connected to the steam drum pressure for power removal balance (Figure 2).

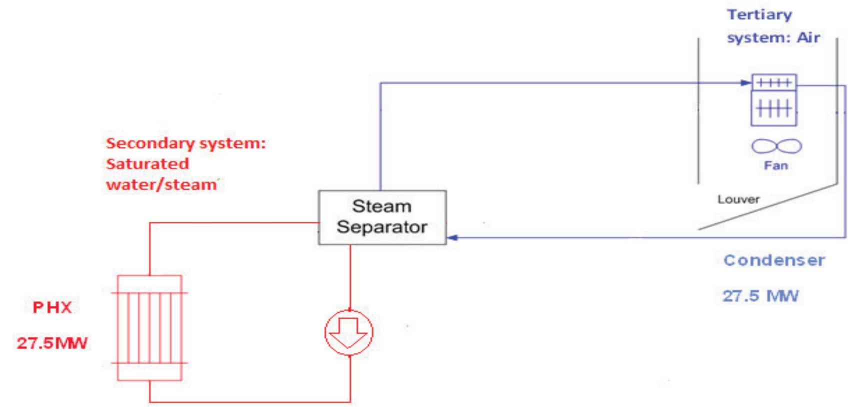


Figure 2 – Secondary system (single loop) schematic concept

All three systems are designed to operate in forced circulation (active mode) during normal operation. Nevertheless, the plant must also be able to remove the decay heat in accidental conditions in full natural circulation (passive mode), thus operating in Decay Heat Removal (DHR) mode. Two systems are devoted to decay heat removal in accidental conditions: the DHR1 system is composed by the secondary and tertiary system themselves, but operating in passive mode. The DHR2 system is the Reactor Vessel Auxiliary Cooling System (RVACS): it floods the reactor cavity with liquid water in order to remove the heat from the vessel external surface and passively delivers it to a series of heat exchangers placed ~50 m above the reactor.

The main differences between the critical and sub-critical operating modes can be summarized as:

- Presence of a closed vacuum tube guiding the proton beam provided by the accelerator towards the LBE spallation target into the central position of the core
- k_{eff} (1.0 in critical mode vs. 0.95 in sub-critical mode)
- Core maximum power level (100 MW vs. 75 MW)
- Core design (108 vs. 72 fuel assemblies)

Despite the maximum core power of 100 MW, the plant has been designed with a maximum nominal power of 110 MW: 100 MW from the critical core plus 10 MW to take into account all additional heat sources, such as In Vessel Storage Tank (IVST), pump power, Po-decay heat, γ-heating, spallation target power, etc…). As a consequence, all the plant equipment with power exchange functions has been designed for 110 MW.

3. MODEL DESCRIPTION

In order to simulate the steady-state and the transient behavior of the MYRRHA facility, a RELAP5-3D [4] nodalization has been modeled in detail by SCK•CEN. In this paper, only the critical mode is studied.

RELAP5-3D is a best-estimate deterministic system code able to represent and simulate the behavior of the complete plant (from core to tertiary fans) through the use of:

- 1-D or 3-D volumes and junctions (domain for the mass, momentum and energy balance equations)

- 2-D heat structures (simulating the solid parts of the system where heat generations and/or exchanges take place)
- A point neutron kinetics module to represent reactivity feedbacks

RELAP5-3D has been validated for the use of LBE as coolant: specific physical properties and heat transfer coefficient correlations have been implemented.

The model has been built according to the design data specified in [5] for what concerns the mechanical, thermal-hydraulic and neutronic design. It simulates the primary, the secondary and the tertiary cooling system with a total of 2518 volumes and 2590 junctions [6]. An extended use of cross-flow junctions has been done in the primary system simulation, in order to attempt a more realistic simulation of the three-dimensional velocity and temperature fields in lower and upper plenum. The tertiary system (final heat sink) is connected to the external environment at a temperature of 37 °C. This presents a conservative boundary condition with respect to clad and LBE temperatures in the primary system (but is not conservative for freezing).

A preliminary system regulation has been implemented using a control device simulating the tertiary fan velocity control driven by secondary system PHX pressure: from a nominal pressure value of 16 bar at the bottom of the PHX, the air flow rate is linearly modified.

The core has been modeled by 4 channels simulating the hot channel, the average channel, the dummy channel and the inter-wrapper flow channel (see Figure 3). The model applies the RELAP5-3D point neutron kinetics module in order to simulate the power generation within the core active zone and to represent the effect of the neutronic feedback, which is particularly important during unprotected transient analysis.

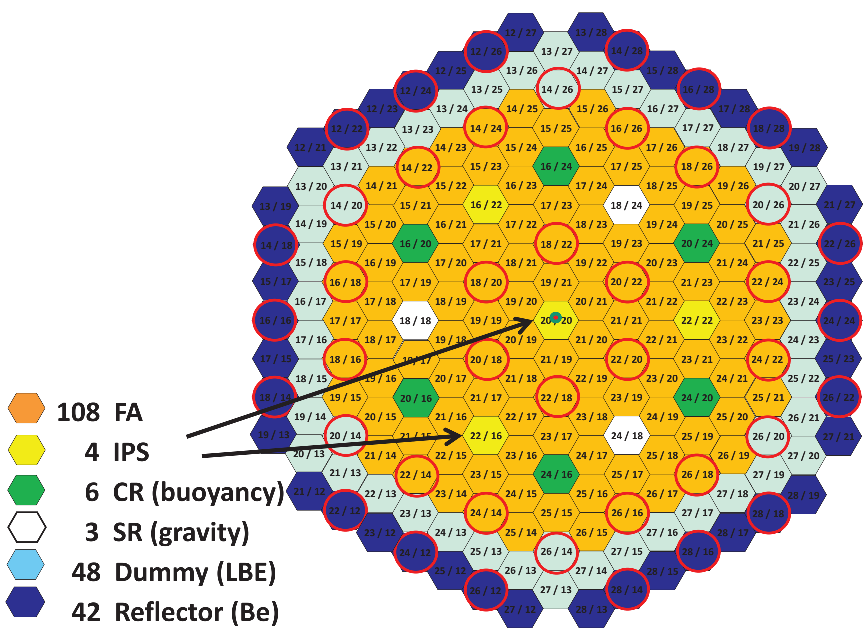


Figure 3 – Critical mode reference core configuration [7]

In addition to what concerns the "active" components (core, PHX and air condenser), solid heat structures are also used to model the core barrel, the diaphragm and the vessel. The 10

MW considered for additional heat sources has been inserted as power generated into the dummy channel.

The RELAP5-3D model has been validated against a validation matrix [8] proposed for qualification of thermal hydraulic codes nodalizations. Furthermore, all results obtained through this model are subject of comparison within the ongoing FP7 MAXSIMA project with other RELAP5 models and different system codes (TRACE), showing a good agreement between the results [2].

4. STEADY STATE RESULTS

Before proceeding with the MYRRHA transient safety analysis, the steady state has been checked to match the nominal design conditions.
The steady state has been run in End of Cycle (EoC) conditions. This choice includes a series of design assumptions to be taken, mainly regarding the form factors for the neutronic flux (radial, axial, S/A) and the oxide layer thickness on the fuel pin [7].
The steady state evaluation by RELAP5-3D code has shown a number of differences with the nominal design data. These differences, mainly identifiable in the temperature differences in the core and in the PHX, are caused by the differences in LBE physical properties considered. In particular, a specific heat difference of ~4% at MYRRHA operation temperatures causes a proportional reduction in temperature differences [10].
The main steady state parameters obtained from RELAP5-3D model are compared with the reference design values in Table 1.

Table 1 – Main MYRRHA steady state parameters

Parameter	Unit	RELAP5-3D value	Reference value
Lower plenum temperature	°C	270.1	270
Upper plenum temperature	°C	322.9	325
Maximum core outlet temperature	°C	424.6	430.7
Primary flow rate	kg/s	13829	13800
Core flow rate	kg/s	7716	7711
Secondary water pressure	bar	16	16
Secondary water PHX inlet temperature	°C	198.2	200
Secondary water PHX outlet quality	-	0.30	0.3

At EoC, the maximum clad temperature is ~450 °C, which is definitely within the safety limits assumed for normal operation conditions [5].
For what concerns the maximum fuel temperature, the peak value is found to be ~1600 °C. This value does not represent a real challenge for the fuel and is mainly due to the reduced fuel pin linear power (peak value: ~230 W/cm) with respect to light-water power reactors (LWR).

5. TRANSIENT ANALYSIS

The main purpose of the analysis is to verify the capability of the safety systems to bring the reactor to a safe shutdown state after a postulated accidental event. Several transients have been proposed for the preliminary safety analysis and can be divided in three main categories:

- Protected transients to study the response and the capabilities of the DHR systems
- Protected transients to analyze the risk connected to the primary LBE overcooling
- Unprotected transients to understand the system physical limits and determine Instrumentation & Control (I&C) response times

In the protected transients, the reactor safe shutdown system is supposed to be triggered (SCRAM) and only decay heat power has to be removed. In the case of unprotected transients, the failure in the reactor safe shutdown system will cause the reactor power to be driven by reactivity feedbacks on fuel (Doppler, fuel expansion), LBE, core structures, etc… All transients have been run in EoC conditions because this status proved to be more challenging for clad integrity.

In order to simulate the SCRAM signals for transient events different than Loss Of Offsite Power (LOOP), a series of triggering parameters has been studied at SCK•CEN and discussed with other participants. The SCRAM logic consists having the SCRAM signals linked to the reactor parameters activating them in case one parameter is found to be outside the expected range. The reactor protection is thus granted by choosing parameters covering all possible transient events. The time delay between the SCRAM signal and the SCRAM actuation is assumed to be 1 s. Safety rods are assumed to be fully inserted after 2 s from the actuation.

A series of enveloping transients has been discussed and set up. The detailed analysis of the most challenging events for each category is described in this paper, as an example of the transient behavior of the MYRRHA reactor.

5.1 Loss Of Offsite Power (LOOP)

In the simulation of a Loss of offsite power (LOOP) all active components (primary pumps, secondary pumps, tertiary fans, secondary and tertiary control systems) are supposed to become unavailable. The reactor SCRAM signal will be triggered at time 0 s, meaning the actual Safety Rods insertion will commence after 1 s from the signal. In the reactor primary pool and in the DHR1 system (secondary and tertiary cooling system) a natural circulation regime will be established, and the decay heat power will be removed through the PHXs and the condensers operating in natural circulation.

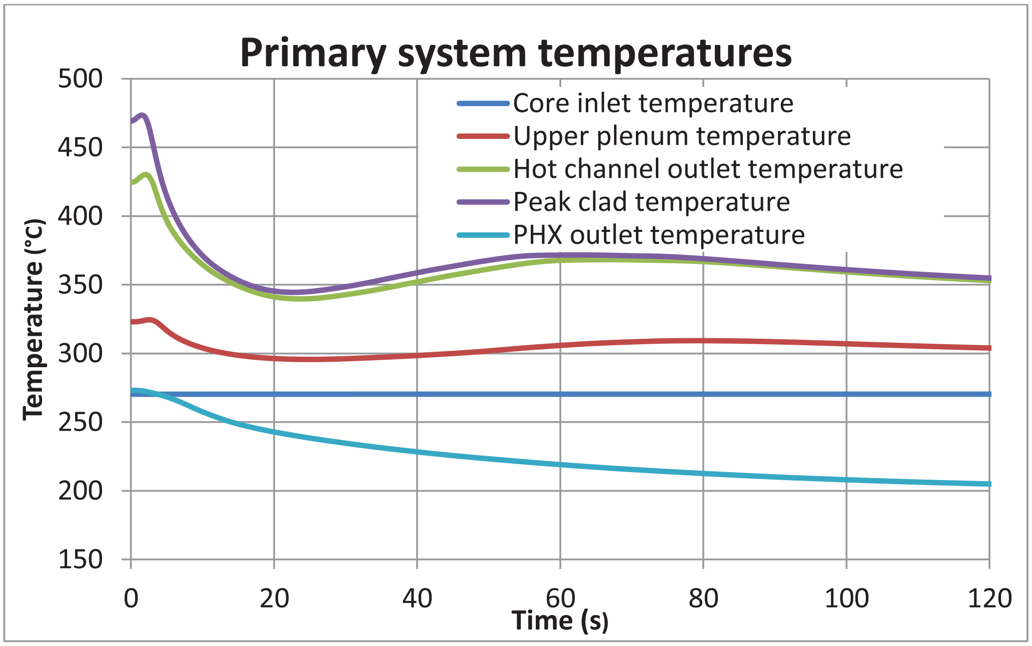


Figure 4 – Primary system temperature evolutions in LOOP transient

In Figure 4 the evolution of the reactor temperatures is shown. After the primary pumps shutdown, the LBE flow through the core and the PHX-PP group will reduce towards the natural circulation regime. After ~70 s the mass flow rates are stabilized to the natural circulation values, which will slowly decrease following the decay heat. It is interesting to note how the backward flow taking place in the PHX-PP group is reduced by a factor two compared to previous MYRRHA versions [2]: this is mainly due to the new, more realistic pump design available for the MYRRHA design, which resulted in a more detailed implementation in the MYRRHA RELAP5-3D model [5].

Due to the PHX design, the heat removal efficiency is not affected too much by the flow reduction in both LBE and water side. The water side remains in the two-phase regime, thus the water side heat transfer coefficient (HTC) remains high, while the LBE side HTC is also not decreasing considerably. The power delivered to the secondary water side remains higher than the decay heat. This property of the system is positive for multiple aspects: the primary system temperatures show no peaks (after the initial peak due to the delayed SCRAM), but the LBE is cooling down towards a safe shutdown condition: after ~6 s, no temperature exceeds 400 °C.

Moreover, the air condenser is also not suffering a big HTC decrease. As a result, there is no noticeable pressure peak in the steam drum: the pressure decreases steadily. The pressure in the secondary system is always kept at low values (< 17 bar) thanks to the tertiary system natural circulation. No appreciable pressure peaks can be noticed in any transient event.

The natural circulation behavior of primary, secondary and tertiary systems is stable, without noticeable oscillations and instabilities: after 1000 s, the steam flow towards the condenser has become stable with a value of ~1 kg/s and it remains stable for the complete transient. The natural circulation mass flow rate in the tertiary side decreases in function of the decay heat, as well as the secondary two-phase flow through the PHX. The PHX outlet steam quality, after an initial increase due to the reduced mass flow rate in the secondary pump and in the

PHXs, decreases to a value of ~0.035 kg/s after 750 s and to subcooled nucleate boiling after ~16000 s.

On the downside, the LBE freezing risk is unavoidable after a certain time, since the tertiary condenser keeps extracting power through the air natural circulation flow, thus decreasing the temperature in the secondary system. The LBE at the exit of the PHX will reach freezing temperature after ~20000 s (> 5 hours).

5.2 Overcooling + Loss Of Flow Accident (LOFA)

The Overcooling + LOFA simulates the failure of one out of four tertiary fans control system: the fan in the failed loop remains active at 100% of the rotational speed without being controlled by pressure error anymore. The remaining cooling systems are supposed fully operational (including the control system). Both primary pumps are also supposed to fail, leaving the primary system to be cooled by LBE natural circulation (LOFA conditions). The reactor trip is activated at time 0 s.

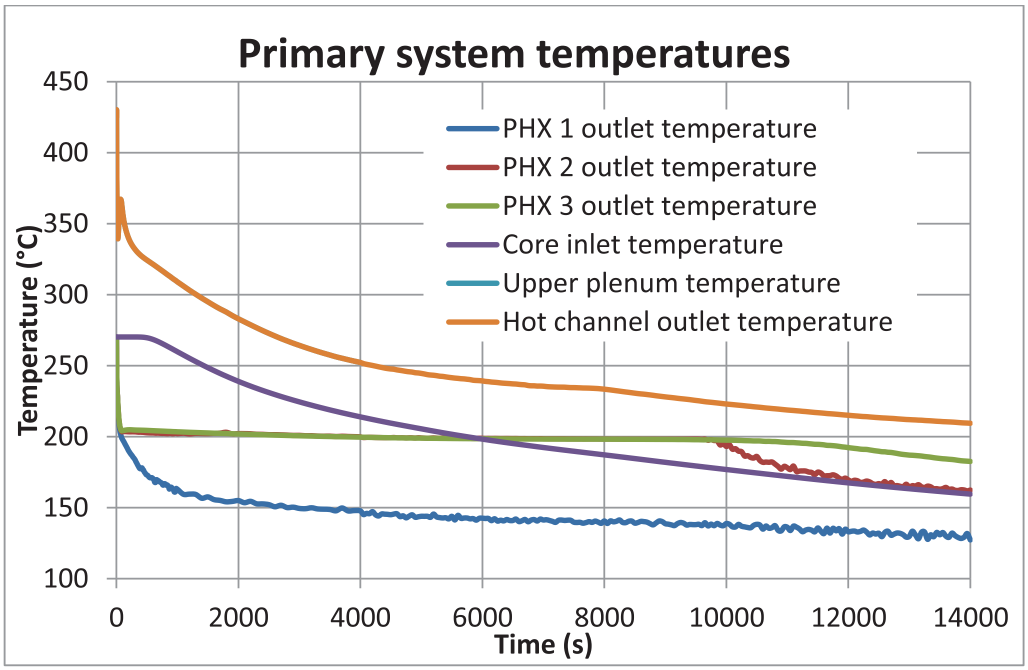


Figure 5 – Primary system temperature evolutions in Overcooling + LOFA transient

The power removed by the PHXs is considerably higher than the decay power in the reactor pool. Because of this, the LBE temperature is decreasing. The LBE in the PHX linked to the failed DHR1 system will freeze after ~14000 (~3.5 hours) s. The water pressure and temperature in the failed line will keep decreasing, despite the 3 remaining air fans to stop after the scram due to the control system logic.

5.3 Unprotected LOFA

The unprotected LOFA simulates the unprotected trip of both primary pumps. No additional failures are supposed to intervene during the transient evolution (i.e. secondary and tertiary controls remain active). The reactor temperatures evolution is shown in Figure 6.

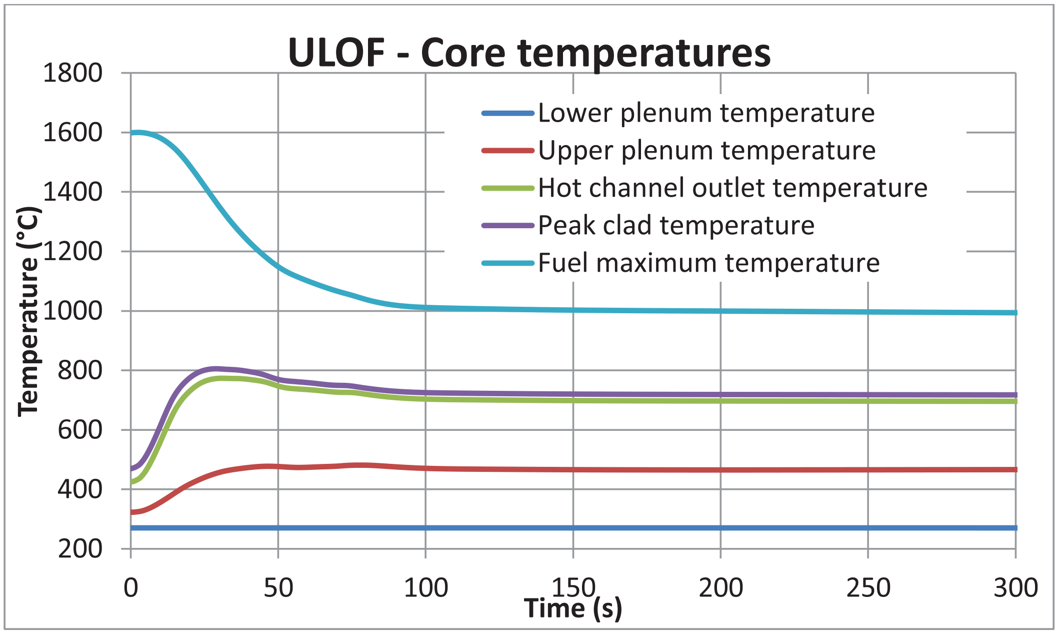


Figure 6 – Core temperature evolutions in Unprotected LOFA transient

Similarly to what has been seen for the LOOP transient, the reverse flow is redirected through the PHX-PP group to a lesser extent. The new PP design is providing enough hydraulic resistances so that the flow generated from the level decrease of the cold free surface will be mostly redirected through the core instead of by-passing it through the PPs. The core mass flow rate, after ~80 s, is stabilized at the natural circulation value of about 12% of the nominal mass flow rate.

Since no SCRAM is foreseen for this event, the reactivity feedbacks are supposed to provide a power reduction after the initial peak due to the loss of flow through the core: again after ~80 s, the core power level is stabilized at a value of ~40 MW. The maximum clad temperature stabilizes at a value of ~720 °C, after a peak of ~800 °C occurring in the first 30 s. The maximum fuel temperature shows no peaks and it stabilizes at a value of ~1000 °C.

The reactivity feedbacks drive the system towards the new steady state. The negative contribution provided by the LBE density is partially counterbalanced by the Doppler and the fuel axial expansion reactivity coefficients, which contribute with a positive insertion due to the decreasing fuel temperature. The total reactivity experiences a minimum on a value of -57 pcm after ~40 s from the transient beginning.

6. UNCERTAINTY + SENSITIVITY ANALYSIS

A first test of the Uncertainty + Sensitivity (U + S) methodology was performed making use of the Software for Uncertainty and Sensitivity Analysis (SUSA) [9] coupled with RELAP-3D. The procedure was applied to the Unprotected LOFA transient case.

6.1 Methodology description

The purpose of the U + S analysis consists in the quantitative evaluation, through multiple system code runs, of the variation of a certain safety-relevant output parameter caused by a certain input variation (to be inserted in a system code input deck) using statistical methods on a number of output parameters.

Every input parameter considered for the U + S analysis has been selected and identified through its reference value and a suitable variation range, estimated on the basis of engineering judgment. A Gaussian distribution (± 3σ), centered on the reference value, has been considered for each parameter. The variations around the reference value are expressed in percentage or in absolute values (see Table 2).

A list of relevant output parameters has been defined. The choices have been made on the basis of the outcome of the deterministic analysis, trying to select the safety-relevant parameters whose variation appears to be more pronounced. Eventually, 17 relevant output parameters have been selected (see Table 3).

Table 2 – Input parameters and the associated variations considered for U + S analysis

Parameter	Unit	Value	Variation
Average clad oxide layer thickness	μm	5	± 2
Hot clad oxide layer thickness	μm	10	± 2
Clad oxide layer conductivity	W/(m.K)	1	± 10%
Peak pin gap conductivity	W/(m.K)	0.067	± 0.02
Peak pin gap width	mm	0.012	± 8%
Fuel conductivity	W/(m.K)	2.4	± 10%
Fuel heat capacity	$J/(m^3.K)$	3.34E+06	± 10%
PHX LBE side oxide layer thickness	μm	40	± 2
PHX water side oxide layer thickness	μm	10	± 2
PHX oxide layer conductivity	W/(m.K)	1	± 10%
Core inlet pressure drop form factors	-	0.5	± 20%
Secondary pressure setup	bar	16	± 1
SCRAM set-point (DT core)	°C	168	± 5%
Control rods insertion delay	s	1	0 ÷ 1

In order to achieve a 95%/95% statistical accuracy, 100 RELAP5-3D input decks have been run. In principle, 96 runs would have been sufficient. The 100 outputs generated by the 100 input decks and representing 100 different transient evolutions have provided, through statistical processing by the SUSA tool, two main comparative parameters summarizing the influence of the input variations on the output:

- Mean values and standard deviation associated to each output parameter considered
- Spearman Rank Correlation Coefficients (SRCC: assessment of how well the relationship between two variables can be described using a function) for all input parameters in function of time (< 0.2: statistically negligible)

Table 3 – Output parameters considered for U + S analysis

Parameter
Active core flow
Active core power
PHX LBE flow
PHX water flow
PHX power
Core coolant inlet temperature
Maximum core coolant outlet temperature
Hot plenum temperature
Cold plenum temperature
Maximum fuel temperature
Maximum clad temperature
PHX LBE inlet temperature
PHX LBE outlet temperature
PHX water inlet temperature
PHX water outlet temperature
Steam drum pressure
Total reactivity feedback effect

6.2 Comparative U + S analysis results

The variations induced by the selected input parameters on the Unprotected LOFA transient evolution are not wide. The Peak Clad Temperature (PCT) shows maximum deviations within ± 20 °C from the best estimate value. In general, the Unprotected LOFA transient shows limited sensitivity to the considered input parameters, as can be seen by Figures 7 and 8 where the SRCC coefficient lies below the value of 0.2 for the most part.

The fuel conductivity, the fuel heat capacity and the oxide layer thickness appear to be the most influent parameters over the transient duration, especially for what concerns the core parameters (PCT, LBE outlet temperature, etc…). The reactivity feedback evolution shows a stronger variable input parameters dependence: the core inlet pressure drops and the oxide layer thickness on the PHX (both influencing the temperature in the core, especially during the natural circulation following the accidental event) are proved to be relevant for the beginning and the end of the transient, respectively.

Limited importance is coming from input parameters related to the secondary system (water pressure, water temperature, water mass flow rate). This result confirms how the secondary cooling system has a low feedback action on the primary system parameters, which is mainly due to the notable difference in thermal inertia between the two systems.

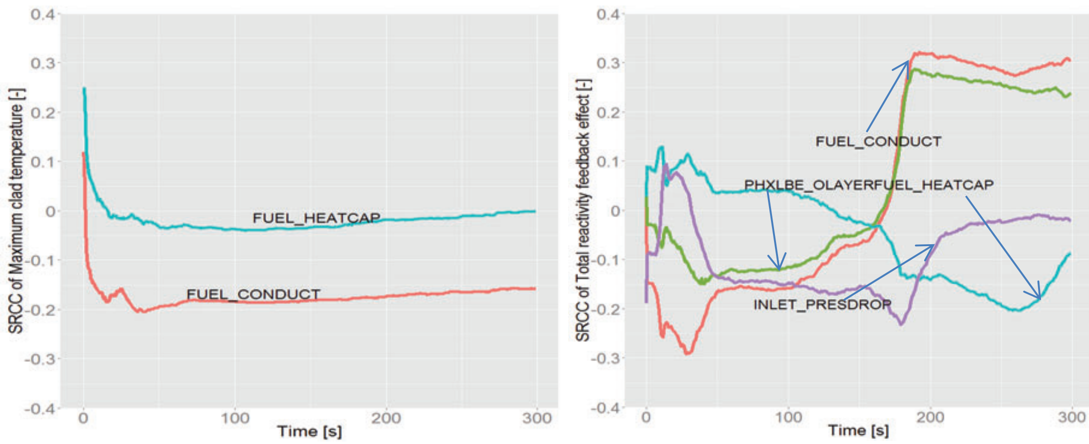


Figure 7 and 8 – Parameter sensitivity for Unprotected LOFA – PCT and reactivity feedbacks

The U + S methodology applied to Unprotected LOFA transient gives interesting first insights in a statistical approach for the MYRRHA safety analysis, but it requires some improvements in the definition of the parameters to be considered as most important for the reactor safety limits. More notable and definite sensitivity results could be obtained by extending the input parameters range and by adapting the selected statistical distribution.

7. CONCLUSIONS

SCK•CEN has performed a steady state calculation and a number of transient analyses of the MYRRHA plant in its critical configuration. Evaluations have been performed through use of the RELAP5-3D system code. The transient events analyzed had as main objective to check the plant behavior for what concerns the Decay Heat Removal (DHR) systems, the overcooling risk and the response to unprotected transients.

The MYRRHA reactor proved to have a satisfactory response, with the clad temperature exceeding 700 °C only during the Unprotected LOFA.

In case of LOOP, the pressure in the DHR1 system (secondary water system) is always kept at low values (< 17 bar) thanks to the tertiary system natural circulation behavior, allowing the power to be evacuated without accumulating into the water system. No appreciable pressure peaks can be noticed. The LBE freezing risk due to overcooling will occur after more than five hours.

The most challenging of the transients simulating the overcooling risk includes Overcooling + LOFA transient where the LBE reaches freezing temperature after ~3.5 hours.

The Uncertainty + Sensitivity analysis has been applied to the Unprotected LOFA transient using SUSA. The variations induced by the selected input parameters on the ULOF transient evolution are not wide. The PCT shows maximum deviations within ± 20 °C from the best estimate value. In general, the transient shows limited sensitivity to the considered input parameters. Further optimization of the parameter selection is however necessary.

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