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D1.3 MYRRHA reference design and initiating events

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Document Summary

The method of Phenomena Identification and Ranking Table (PIRT) is aimed at ranking the importance of the information as a combination of level of knowledge and impact on identified figures of merit, in order to support safety analyses and decision-making towards the prioritization of studies and experiments. In the framework of ANSELMUS WP1 a PIRT panel will thoroughly analyse the phenomena underlying the plant response to challenging safety related events. To support that, the present deliverable summarises the current plant design of the MYRRHA reactor and the initiating events are listed.

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1) Introduction

Phenomena Identification and Ranking Table (PIRT) is aimed at ranking the importance of the information as a combination of level of knowledge and impact on identified figures of merit, in order to support safety analyses and decision-making towards the prioritization of studies and experiments. In the framework of ANSELMUS WP1 a PIRT panel will thoroughly analyse the phenomena underlying the plant response to challenging safety related events. To support that, the current plant design of the MYRRHA reactor is summarised, and the identified initiating events are listed in the present deliverable.

2) MYRRHA reference design general description

This chapter summarizes the main design features of the MYRRHA (Multi-purpose Hybrid Research Reactor for High-tech Applications) reactor design version 1.8 (see Figure 1). The present description focuses on the primary cooling system and its components.

The MYRRHA reactor is a pool-type Accelerator Driven System (ADS) with the ability to operate in critical mode. The design has a MOX-fuelled core cooled by liquid lead-bismuth eutectic (LBE). With the pool-type concept all primary system components are housed within the reactor vessel and inserted from the top penetrating through the reactor cover. The reactor vessel consists of a primary vessel that is integrated and surrounded by a safety vessel. The safety vessel serves as a secondary containment in case of a reactor vessel leakage or break. The reactor cover closes the reactor vessel and supports all the vessel internal components. The reactor core is fully submerged below the LBE coolant, positioned in the central region of the reactor vessel when looking from above, and is surrounded by a shielding jacket and assembled inside the core barrel. The core positions contain the fuel assemblies, control rods, safety rods, the spallation target assembly, in-pile sections, reflector assemblies, instrumentation and surveillance capsules. A core restraint system fixes the radial position of the fuel assemblies. For refuelling, an in-vessel fuel-handling machine is installed permanently in the reactor. The in-vessel fuel storage is integrated into the diaphragm structure.
The diaphragm is a large reactor internal component serving as a structural lateral support function for many of the primary system components, while also guiding the primary coolant flow path through the necessary components, thereby separating high pressure from low-pressure zones. The diaphragm design consists of a perforated circular steel "basket" that is mounted to the primary vessel flange. Within this circular steel structure, a series of vertical chimneys house various components, such as the core unit (CU), the primary heat exchangers (PHX), the primary pumps (PP), the in-vessel fuel handling machine (IVFHM), the fuel transfer channel (FTC), and the in-vessel fuel storage (IVFS). Where dedicated coolant flow paths are necessary for the normal operation of the reactor, piping connects the core to the PHX, and the PHX to the PP.

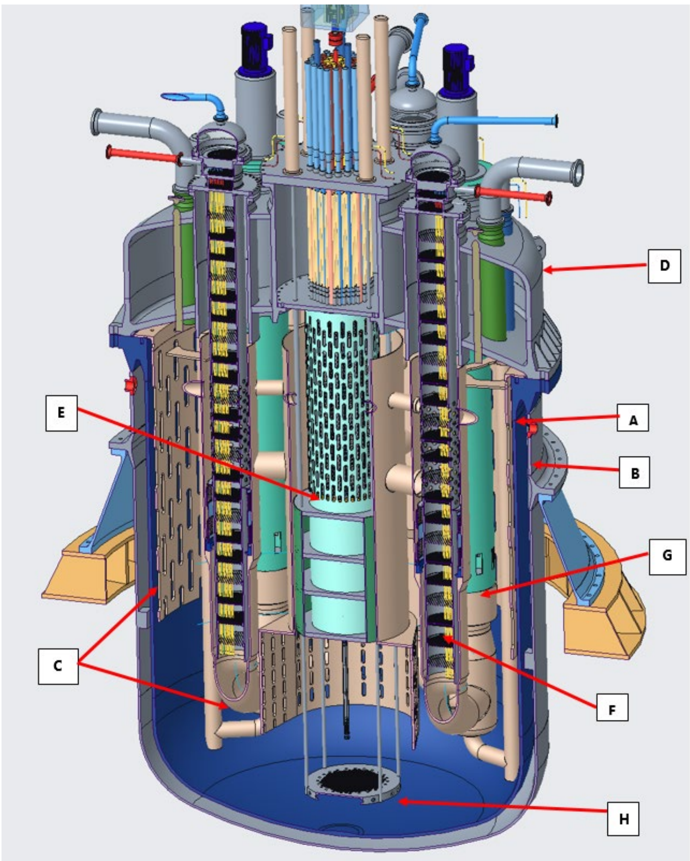

Figure 1: MYRRHA reactor assembly isometric cross-sectional view [1]
A: Primary reactor vessel (RV), B: Safety vessel (SV), C: Diaphragm (DIA), D: Reactor cover (RC), E: Core barrel (CB), F: Primary heat exchanger (PHX), G: Primary pump (PP), H: Core restraint system (CRS), I: In-vessel fuel handling machine (IVFHM)

LBE coolant enters the core from below at a maximum temperature of 220 °C (see Figure 2). This maximum core inlet temperature of 220 °C is selected such that the fuel assembly Peak Cladding Temperature (PCT) does not exceed the maximum threshold of 400°C at full power. Consequently, 220 °C is the maximum LBE temperature of the MYRRHA cold plenum during normal operation and also determines the upper threshold of the dissolved oxygen concentration target. The LBE exits the core unit (assembly of core and the above core structure (ACS)) into the core chimney with a mixed core outlet (maximum) temperature of 270 °C when at full power during normal operation. The LBE free surface of the hot plenum is below the cold plenum free surface level by a distance equivalent to the head loss across the core. The hot LBE exits the core chimney via the horizontal diaphragm pipes that connect the core chimney to the PHX chimneys, where the flow enters the PHXs (four PHX units in parallel with dedicated secondary/tertiary loops). The four primary heat exchangers extract the power of the primary system exchanging it with the secondary system during normal operation. During design basis accidents, dedicated passive DHR1 cooling loops (one loop for each PHX) take over from the secondary/tertiary cooling systems. The LBE outlet flow from two PHXs then combine in a piping duct to enter the primary pump. The number of heat exchangers and pumps was optimised considering the safety redundancy requirements of DHR1 and the space utilisation inside the reactor vessel. This leads to a configuration with two heat exchangers and one pump on each of the two sides of the reactor. The primary pumps, which are vertical shaft machines, pump the LBE back into the so-called cold plenum pool. This ensures sufficient pressure difference to establish the necessary coolant flow through the core and primary system components.

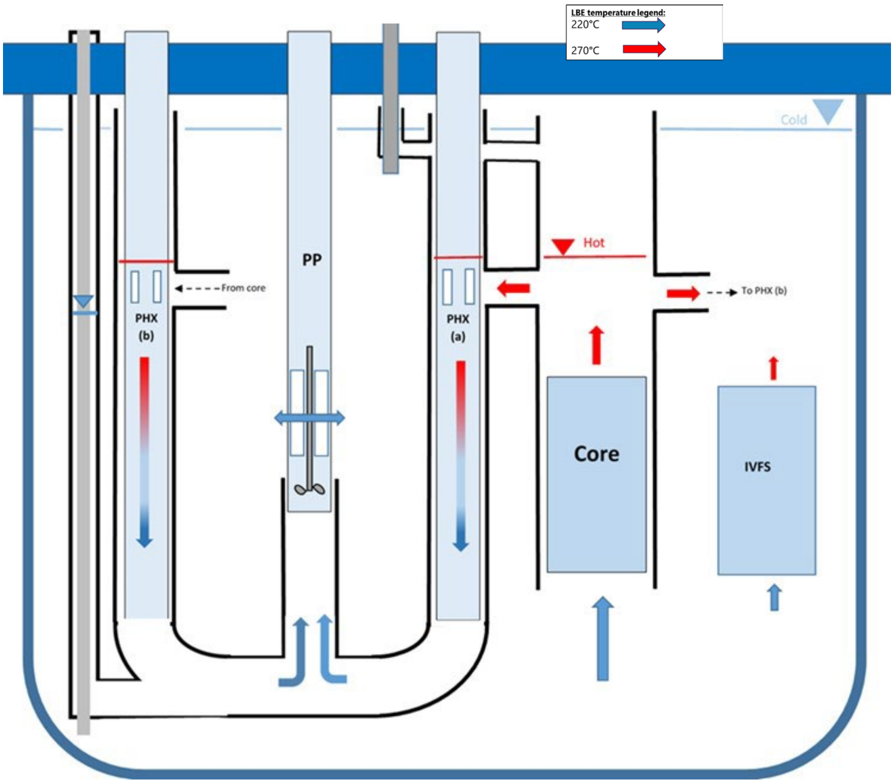

Figure 2: Reactor primary LBE coolant flow paths (normal operation) [1]

Bypass valves (indicated with grey in Figure 2) are also foreseen in the conceptual design to improve the natural circulation when the Reactor Vessel Auxiliary Cooling System (RVACS) is operating, either in Decay-Heat Removal 2 (DHR2) or severe accident cooling (SAC) mode. Heat-removal flow paths will be detailed in Section 2.7.

Primary Cover Gas and Ventilation System (PCGVS, exterior to the vessel) filters and monitors the nitrogen cover gas of the reactor. The cover gas is kept at a pressure slightly below that of the containment. Mitigation of an overpressure event in the primary system is provided by rupture disks placed on the reactor cover.

The main parameters of the reference design are summarised in Table 1.

In the following sub-sections, the main components of the primary system are described in more detail.

Table 1 General design parameters, Myrrha 1.8

GENERAL DESIGN PARAMETERS	UNITS	MYRRHA DESIGN REV. 1.8
Maximum core power	MW_{th}	64
Maximum heat sink rated power	MW_{th}	70
Temperatures		
Shutdown state LBE temperature	°C	200
Maximum core inlet LBE temperature	°C	220
Maximum average hot plenum LBE temperature	°C	270
Fuel assembly		
P/D		1.28
Number of pins		127
Spacer type		Wire spacer
Total pressure drop (NOC conditions)	bar	1.93
Core		
Number of core positions		163
Critical core total mass flow rate	kg/s	9283
Sub-critical core total mass flow rate	kg/s	9641
k_{eff} in subcritical mode		0.93
Maximum core inlet temperature	°C	220
Maximum average core outlet temperature	°C	306
Maximum cladding temperature	°C	400
Maximum core outlet temperature	°C	376
Maximum average hot plenum temperature	°C	270
Maximum average active core coolant ΔT	°C	86
Maximum ΔT between plena	°C	50
Primary heat exchangers		
Type		Tube-and-shell
Number of heat exchanger units		4
Primary side coolant fluid		Liquid LBE
Maximum primary coolant fluid inlet temperature	°C	270
Maximum primary coolant fluid outlet temperature	°C	220
Secondary side coolant fluid		Water-steam mixture
Primary pumps		
Type		Semi-axial flow pump
Number of pumps		2
Overall primary coolant system pressure drop	bar	2.63

2. 1. Primary heat exchanger (PHX)

The MYRRHA PHX (see Figure 3) adopts a shell-and-tube bundle structure, with primary LBE flowing downwards in the shell side and secondary water flowing upwards in the tube side. The tube bundle is enclosed in the cylindrical PHX shroud and held in position by a series of spacer grids.

The component penetrates in the Reactor Vessel through the Reactor Cover, where it is flanged upon assembly. The diaphragm foresees a specific chimney for each of the four PHX units. A bellow is designed to hold the component in position while allowing for thermal expansion and limiting by-pass flow.

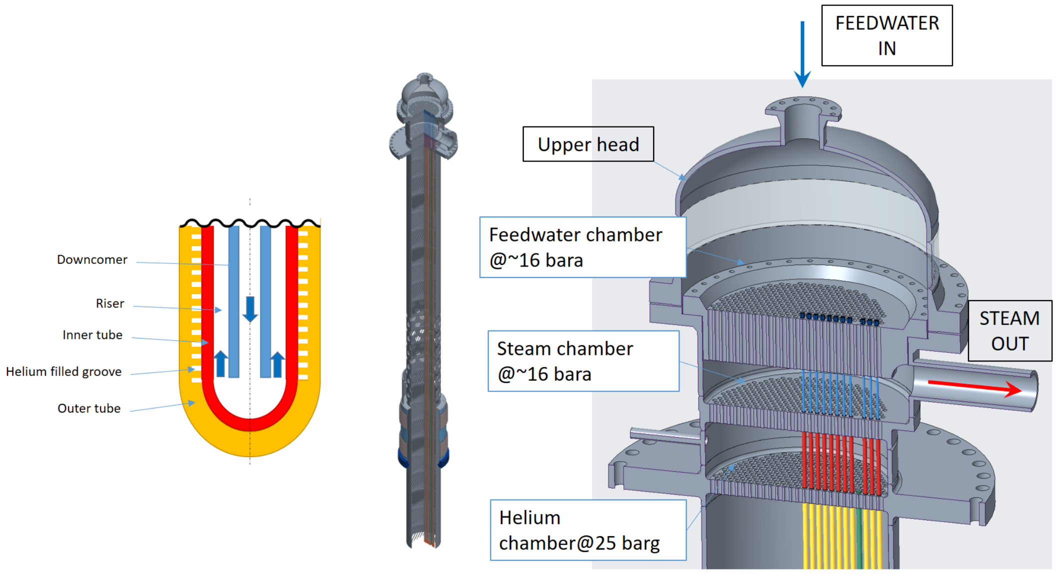

Figure 3: PHX overview: double wall bayonet tube schematic structure (left), overall 3-D view (centre), PHX head structure and flow paths [2 Kennedy: MYRRHA Primary System Design – Revision 1.8, SCK CEN 45203855]

2. 2. Primary pump (PP)

Two identical primary pumps (Figure 4) are located vertically and submerged in the primary LBE generating flow upwards through the pump and exhaling it to the cold LBE volume. The primary pumps are to provide the following main functions:

- Stable hydraulic performance for core cooling during normal operation, while simultaneously allowing a flow path for stable natural circulation of LBE in case of pump shutdown,
- Primary gas confinement. To prevent escape of reactor primary gas into reactor hall,
- Sealing at Diaphragm interface. To separate cold and hot LBE volumes,
- Safety: PPs do not violate safe performance of the reactor,
- Radiological shielding to protect to personnel performing maintenance.

Figure 4 Primary pump general view [1]

2. 3. Core unit (CU) and core components

The core layout has 163 hexagonal channels. The design revision 1.8 core configuration includes seven concentric rings (crowns) of hexagonal channels, of which the outermost ring is designated as a reflector. The hexagonal channels are surrounded by a stainless steel jacket and the core barrel. The sub-critical and critical core configurations are presented in Figure 5.

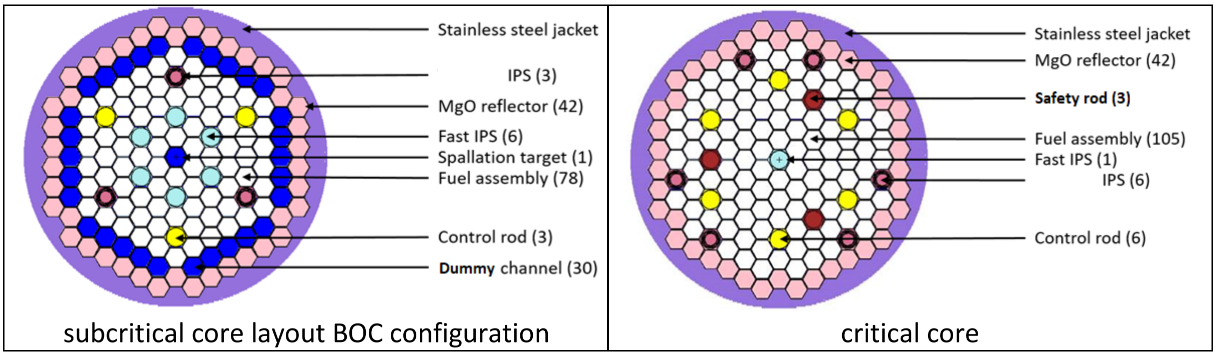

Figure 5: Core layout [1] Note: The current design does not have thermal IPS Radioisotopes will be produced with fast flux.

The fuel assembly (FA) design is similar to a typical design used in fast spectrum sodium-cooled reactors (SFRs). Each FA contains a hexagonal bundle of 127 cylindrical fuel pins surrounded by a hexagonal wrapper. The upper and lower ends of the hexagonal wrapper are connected to the inlet and outlet nozzles guiding the LBE coolant through the bundle. Each fuel pin contains fuel pellets and free space called gas plenum for the helium filling and fission gases retention. Helical wire-spacers wound on the outer surface of fuel pins keep them in position in the bundle.

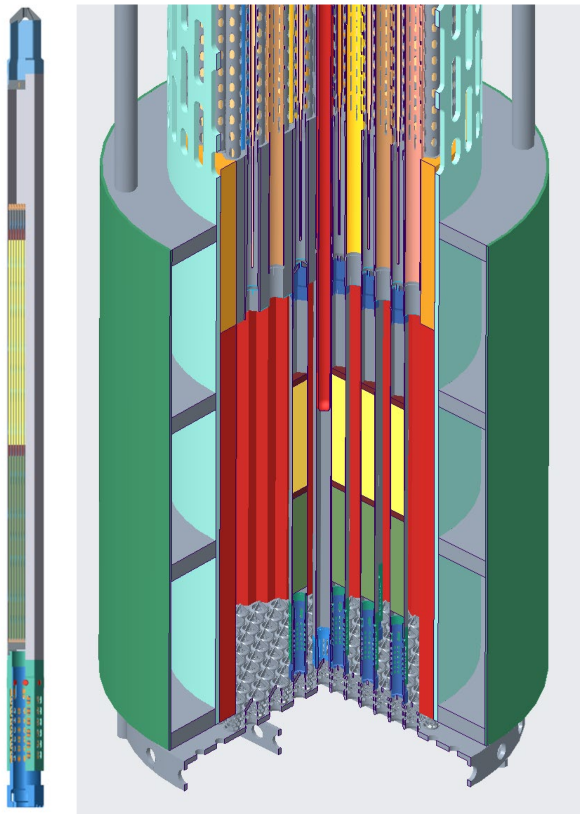

Figure 6: Fuel assembly and full core layout (only 4 FAs shown) [1]

The design of the MYRRHA fuel pin is similar to a typical sodium cooled fast reactor fuel pin. The structure of a fuel pin is presented in Figure 7. The cylindrical cladding is made of 15-15 Ti austenitic steel.

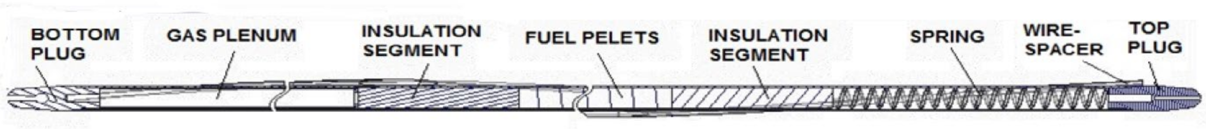

Figure 7: Cross section of a fuel pin [1]

The Core Unit (CU) is a long central structure in the reactor that holds the whole reactor core (core assemblies and IPS) in place (see Figure 8). The core unit is mounted into the reactor cover and runs down the reactor to the flat plate of the diaphragm. The CU allows for bottom core manipulations by the IVFHM when the core restraint grid is lowered.

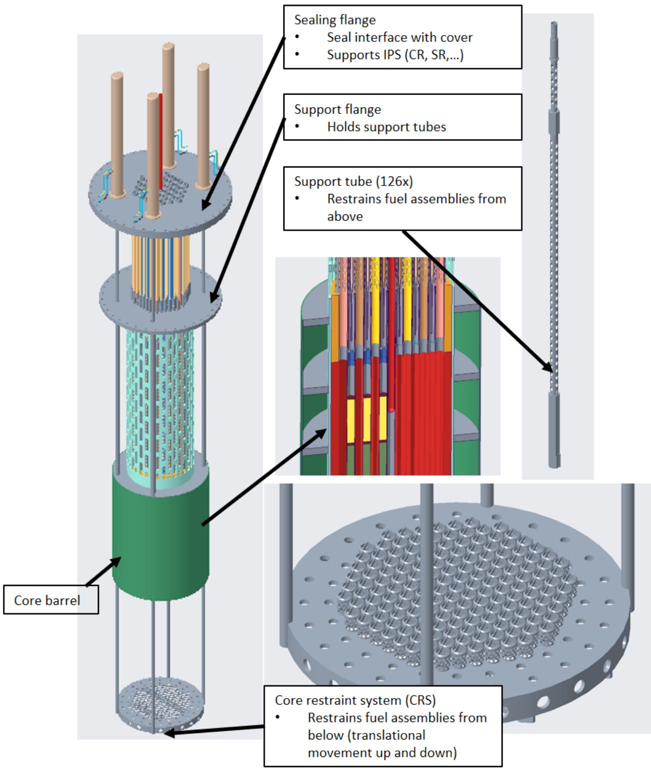

Figure 8: Core unit assembly [2]

2. 4. Diaphragm

The diaphragm separates the cold and hot LBE plena of the MYRRHA reactor pool and creates a hydraulic path for the LBE to flow from one to another plenum. The LBE coming from the cold plenum flows through the core into the hot plenum. The LBE then moves via a system of pipes to the PHXs after which it flows in pipes to the PPs where it is again pumped back into the cold plenum. The walls that define the hot plenum and the system of pipes between PHX and PP are an integral part of the diaphragm. The diaphragm also acts as a structure to support the in vessel fuel storage (IVFS).

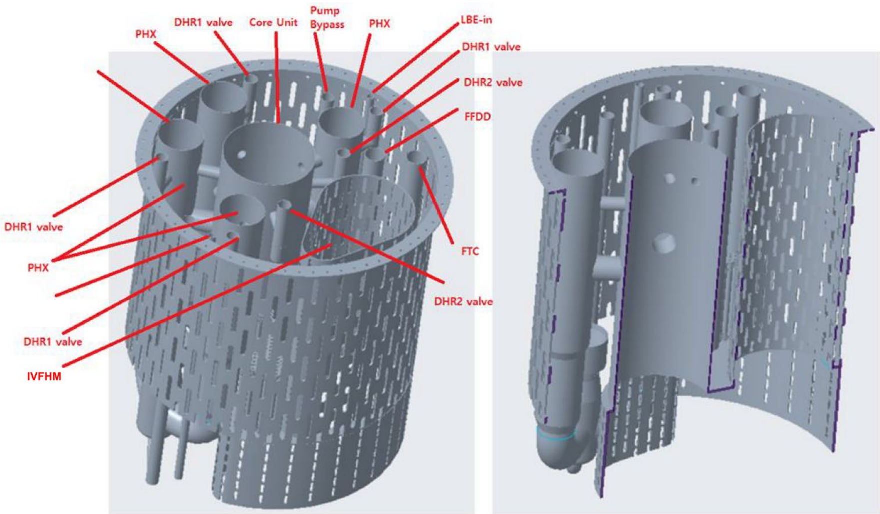

Figure 9: Overview of the diaphragm and its components [1]

2. 5. Reactor vessel assembly

As MYRRHA is a pool type reactor, the Primary Vessel main safety function is the containment of the primary coolant. The Safety Vessel provides redundancy regarding this function. The Reactor Vessel Assembly also includes the supporting elements joining the Primary and Safety Vessels to the civil engineering structure. Regarding the gas plenum containment, the reactor flange allows a gastight connection to the Reactor Cover. The vessel assembly also provides support for the Reactor Cover and the Diaphragm and thereby to all the primary system components.
The Safety Vessel is engineered to contain any potential LBE leak and to survive the guillotine failure and subsequent drop of the Primary Vessel. In this case, large deformation of the Safety Vessel is expected but its tightness regarding LBE is preserved. The Reactor Vessel also fulfils the function of Cover Gas containment. In the event of loss of heat sink via DHR1 (PHX and/or DHR1 loop) or in the event of a severe core disruption, the inter vessel space is used by the Reactor Vessel Auxiliary Cooling System (RVACS) as a cooling jacket for ultimate decay heat removal (so-called DHR2). During RVACS operation, this cavity is filled with water by releasing water from tanks near the top of the reactor building. RVACS operation is precluded if LBE is present in the cavity between the Reactor Vessel and the Safety Vessel.

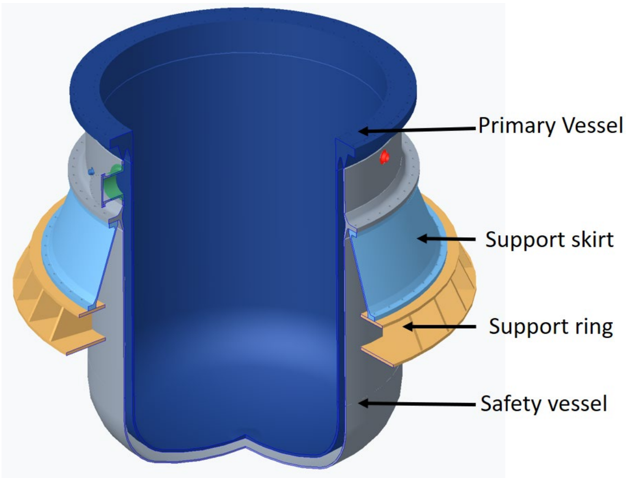

Figure 10: Reactor vessel assembly overview [2]

2. 6. Reactor Vessel Auxiliary Cooling System (RVACS)

The RVACS concept is based on flooding the cavity between the Reactor Vessel and the Safety Vessel with subcooled water originally stored in tanks placed above the Reactor Hall. The water temperature increases if water is in contact with the Reactor Vessel hot external wall. The water temperature eventually reaches the saturation temperature and starts boiling. The steam is then vented out of the containment. The RVACS is organized in four identical open loops. A schematic view of a single RVACS loop and the connection with the intra-vessel cavity is shown in Figure 11.

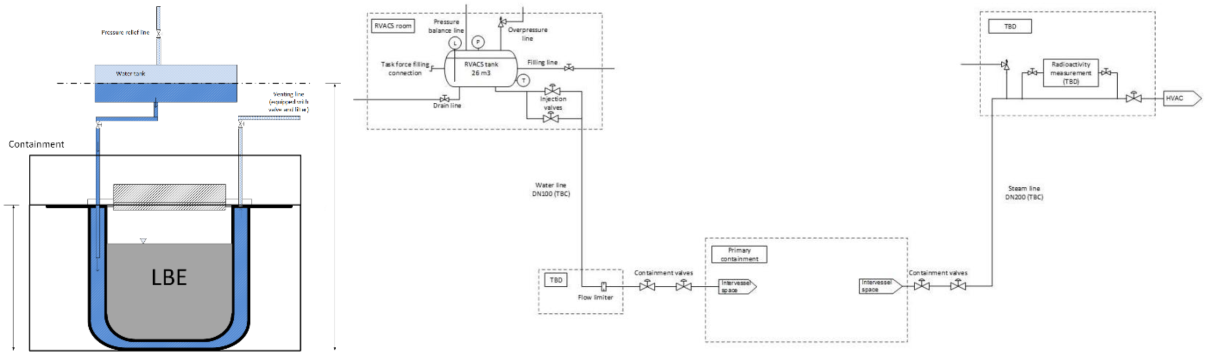

Figure 11: RVACS overview sketch [1] and schematic diagram [8]

2. 7. Cooling principles

MYRRHA is a pool reactor with one free liquid surface. The cover gas above the liquid surface is not pressurized. Main coolant pressure (except for hydrostatic pressure) and fluid motion is created by running the primary pumps. When the PPs are running two different major coolant levels establish, the cold plenum level and the hot plenum level. All the LBE and cover gas is contained within the primary vessel and reactor cover. The different ‘hydraulic regions’ are defined by the PPs, the PHXs and the CU with different pipes and chimneys of the diaphragm hydraulically connecting them.

2. 7. 1. Reactor running in normal operation

During power operation, active cooling is provided by the primary pumps creating pressure, flow and giving rise to different LBE free surface levels (Figure 12). DHR 1 valves and Pump by-pass valves are all closed. Cold LBE flowing upwards from the bottom part of the cold plenum enters the core where it is heated. Hot LBE exits the core into the hot plenum. Hot LBE flows via the lower (main) CU-PHX connecting pipe to the PHX chimney. The LBE flows into the PHX where it is cooled. The PHX chimney is connected to the inlet pipe of the PP. Cold LBE flows from the outlet of the PHX through this pipe to the inlet of PP. The cold LBE is then pumped into the cold plenum and will eventually stream down to the lower part of the cold plenum, returning into the core. The level difference between cold and hot surface corresponds to the pressure drop of the core and the above core structure. There is no LBE in the upper CU-PHX connecting pipes as it is above the hot LBE level. The IVFS is completely immersed in the cold plenum, both inlet an outlet. Cooling of the IVFS is by natural convection only.

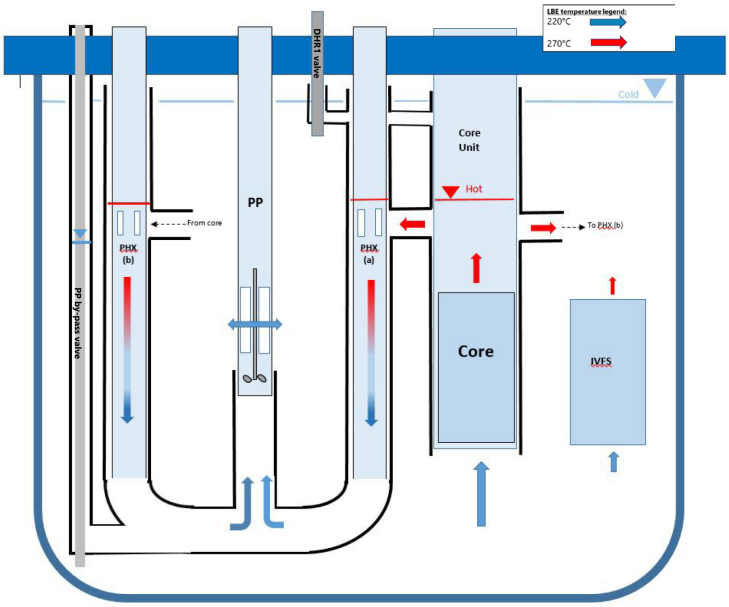

Figure 12: Reactor primary LBE coolant flow paths (normal operation) [1]

2. 7. 2. Decay heat removal through the PHXs (DHR1)

When the reactor is in shutdown and the PPs are off, there is no forced LBE flow and all the LBE free surface levels will equalize at the shutdown level. Decay heat of the core, the IVFS and the activated LBE coolant itself is evacuated through the PHXs via natural circulation. In this mode The PP by-pass valves and the DHR1 valves are open. Part of the heated LBE flowing out of the core will pass through the lower CU-PHX connecting pipe to the PHX. The other part of the heated LBE will flow through the upper CU-PHX connecting pipe to the PHX. The LBE is cooled in the PHX and flows down via the opened PP by-pass valve to the lower part of the cold plenum, by-passing the PP.
Cold LBE, slightly heated by the decay of the IFVS can flow into the PHX chimney via the DHR1 chimney as the DHR1 valve is opened in this state.

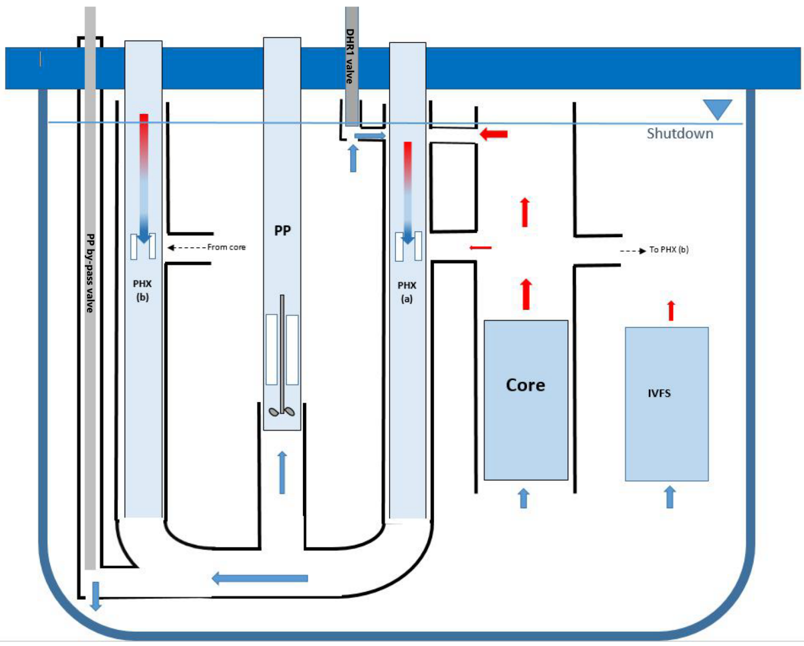

Figure 13: Schematic LBE flow in the primary pool during shutdown conditions [1]

2. 7. 3. Decay heat removal by the RVACS (DHR2)

In a shut-down situation when the PHXs are not operational or DHR1 fails to perform its function, decay heat is evacuated by the RVACS, heat is transferred through the reactor vessel surface. Heated LBE coming form the core can bypass the PHX-PP path by flowing out into the upper connecting pipe through the PHX tube bundle then via the DHR1 valve into the cold plenum. An extra pair of valves, namely the DHR2 valves are opened in DHR2 mode to facilitate the outflow of heated LBE from the hot plenum to the cold plenum.

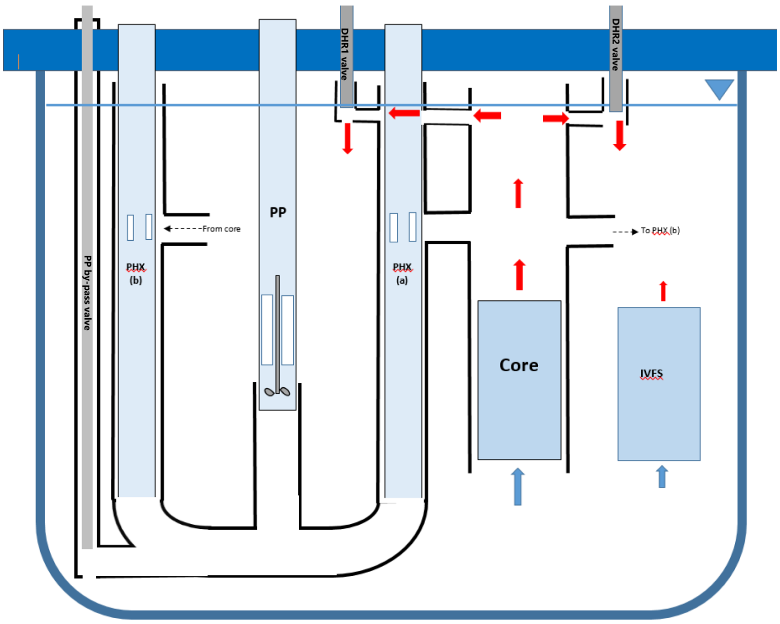

Figure 14: Schematic LBE flow in the primary pool during DHR-2 (RVACS) activation [1]

3) Initiating events (IEs)

3. 1. Methodology for identifying internal initiating events [3]

The following section provides a brief overview of the methodology applied to list the internal initiating events for MYRRHA.

Identifying the initiating events that can have non-negligible radiological consequences is a key element in the safety analysis. Internal initiating events are technical changes in the condition of the installation that are initiated by technical failures of components or by operating errors and challenge the confinement of the source terms of the installation.

Initiating events often originate from the malfunctioning or spurious operation of components of systems. At an early stage of the design the technical details up to the level of the components are not yet developed. So, it is impossible to make a systematic bottom-up analysis of the possible impact of failures of components and analyse their impact (e.g., following a FMEA approach).

On the other hand, performing the initiating event analysis provides the opportunity to optimize the prevention of initiating events that represent a serious threat for safety and for which the mitigation is difficult. In this way, a better balance between prevention and mitigation can be achieved. The preventative part of the safety analysis can form an important part of the design basis of the structures, systems and components (SSCs) of a nuclear installation.

A top-down approach is applied to identify all mechanisms that could challenge the integrity of barriers, including inherent barriers. Mechanisms that are more independent of the technical details of the design and that pose a potential challenge to the confinement barriers of the source terms are identified and examined.

The systematic method adopted for identifying all mechanisms that could produce non-negligible radiological consequences is as follows:

- identify all non-negligible source terms in the installation that potentially can lead to off-site radiological consequences larger than SO1,
- for each non-negligible source term, identify the barriers that contribute to the retention of its radioisotopes,
- for each barrier determine all possible failure states and the physical conditions for which the retention properties of the barrier are lost,
- for each barrier, identify the mechanisms that can produce physical conditions that challenge the retention properties of the barrier,
- determine the potential consequences for the failures modes of the barrier (or multiple barriers) by these mechanisms,
- all this information will be used to develop the prevention and/or mitigation strategy for the identified mechanisms in order to satisfy the safety objectives. This means that for events/mechanisms that meet the exclusion criterion thanks to preventive measures, the design typically does not foresee dedicated mitigating measures. Remark: the master severe accident (MSA) is not the subject of this procedure.

Independently of their presence or not in the actual design, each identified mechanism in this top-down approach can be considered in one or more enveloping form(s) as 'initiating event' for the safety demonstration. It is the safety analysis itself that will determine the envelope character of a case and the SSCs needed to mitigate that case. The following example illustrates this: independently on whether the installation is capable to produce voiding in the core or not, a demonstration that voiding cannot produce positive reactivity injection is a powerful demonstration that initiating events producing this mechanism do not need to be considered in the safety demonstration of reactivity induced events.

3. 1. 1. Basic safety functions

The three basic safety functions are defined as:

- control of reactivity,
- cooling of radioactive material,
- confinement of radioactive material.

The first two safety functions do not explicitly appear in the top-down methodology for the identification of mechanisms. This is because we consider confinement as the primary safety function while both the loss of reactivity control and the loss of cooling are functions that challenge the primary safety function. Reactivity control and cooling safety functions will certainly be addressed by the mechanism in our top-down methodology.
It is clear that the loss of reactivity control and the loss of cooling constitute a special threat to the barriers because if no longer controlled, they can have a destructive influence on several or even all barriers.
The top-down approach in principle captures also events induced by human action. The purpose is to identify those events as well that could be caused by human action. In order to perform a check of the completeness of the barrier failure mechanisms that are identified by our approach, lists of initiating events for other installations and technologies can be used for comparison.

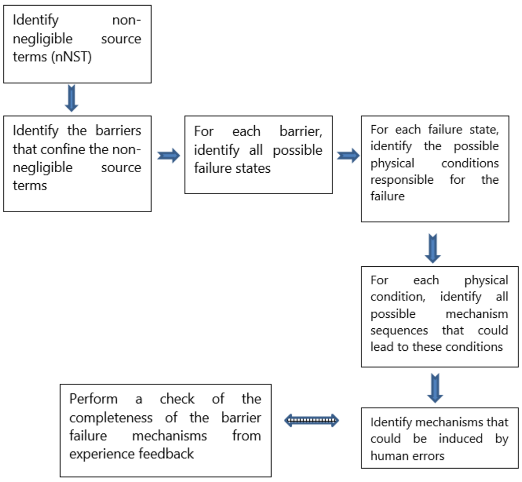

Figure 15: Overview of the top-down approach [3]

3. 2. Barrier integrity [4]

The following sections will provide the list of identified internal initiating events for MYRRHA that represent an important challenge to the confinement barriers of radioactive source terms in the installation.
First, the source terms and associated barriers in the installation are identified. In a second step, the physical conditions that can endanger the integrity of the barriers for the considered source term will

be identified, together with mechanisms that can produce these physical conditions, independent of the details of the installation. In a last step, the events in the installation that can produce the mechanisms that endanger the barrier will be identified.

3. 2. 1. The fuel matrix barrier

The fuel matrix is an 'inherent' barrier. The retention properties of this barrier during normal operation determine the volatile source term inside the next barrier, the clad. Since the clad provides a full confinement for all fission products that leave the fuel matrix (except for tritium) the failure of the fuel matrix alone does not represent a safety threat. IE identification for fuel matrix integrity is not applied but the retention of the fuel matrix is incorporated in the safety analyses of the initiating events that challenge the clad.
The principal physical parameters that determine the retention of the fuel matrix are burn-up and temperature of the pellet. Retention properties deteriorate at higher burn-up and at higher temperatures.
An important source term is the part of nuclei that in normal operating conditions has already left the fuel matrix. According to conservative calculations for MYRRHA, the release fractions from oxide fuel pellets during normal operation are significantly larger than for PWR operating conditions. This is partly due to the conservatism of the calculations. Based on these calculations it has been decided to consider for the time being the following conservative (corresponding to EOL burn up) composition for the gap release during normal operation:

- Group 1: noble gases: 100%
- Group 2: halogens: 100%
- Group 3: alkali metals: 100%
- Group 4: tellurium group: 100%
- Group 5, 6, 7 and 8: 0%

These groups are the ones usually adopted for LWR source term characterization. As long as the fuel does not melt, we do not expect other elements than the ones of the conservative gap release to leave the fuel matrix. The elements of the non-volatile groups (groups 5, 6, 7 and 8) should show more retention in the LBE and in the containment. Small amounts, a few percent of release for the isotopes in these groups would not have an important impact on dose calculations. Therefore, we can assume that the release of elements from groups 5, 6, 7 and 8 can be neglected as long as the fuel does not melt.
For scenarios in which the fuel matrix reaches melting temperature, more elements might leave the molten fuel. For this condition, the source term should be re-evaluated.
The present assumptions concerning the retention properties of the fuel matrix are very conservative and might be reviewed if necessary.
Due to the high effectiveness of the clad barrier, the failure of the fuel matrix retention alone will have no consequences and no initiating event analysis will be developed for this inherent barrier. Conservative retention properties of this barrier covering all transients except those involving melting of the fuel matrix have been defined. Melting conditions should not occur due to the high boiling point of the coolant and the mitigation strategy for loss of coolant events.

3. 2. 2. The fuel clad barrier

Failed states of the clad and the physical conditions that can produce these failed states are listed in Table 2. The different failed conditions in which the clad no longer fulfils its barrier function are identified as:

- molten clad,
    - failure by melting: melting point of the clad,

- cracks in the clad,
    - failure by creep induced strain,
    - failure by PCMI induced strain,
    - burst by high material stress (due to internal pressure),
- material loss of the clad,
    - loss by corrosion/erosion,
- loss of retention properties (transparency) of the clad.

These failed states are of course closely connected to the physical limits for the barrier, meaning the physical/chemical conditions that will give rise to a failed state.
Stochastic failure mechanisms of the cladding by definition will only affect one fuel pin at a time and should not have an impact on the other barriers of MYRRHA. Such stochastic failure mechanisms will only lead to events with a single pin failure and have a very limited radiological impact and they can be considered as part of normal operation.

In Table 2 and in the subsequent tables the failure conditions and the failure mechanisms leading to those are listed in detail. Reasoning is summarised in notes in the respective sections. The tables include sequence paths that can be reasonably excluded and are indicated with strikethrough.

Table 2: Physical conditions that can lead to failure of the clad barrier

Failed state		**Failure condition**						
Molten clad (Note 1)	←	High clad temperature	←	Fuel melting (Note 2)	←	***Fuel overpower***		
			←	~~High coolant temperature~~ (Note 3)				
Ruptured clad	←	High clad stress (Note 4)	←	***Clad overtemperature***				
			←	***High internal pin pressure***				
			←	~~Low external pin pressure~~ (Note 5)				
	←	High clad strain (Note 6)	←	~~Creep~~ (Note 6)				
			←	Excessive internal force (Note 7)	←	PCMI	←	***Fuel overpower***
			←	Excessive external force	←	***Coolant freezing*** (Note 8)		
					←	External pressure (Note 9)	←	~~Static overpressure~~
							←	~~*Pressure transient (shock wave)*~~
					←	~~Impact objects~~ (Note 10)		
	←	~~Fatigue~~ (Note 11)						
Material loss	←	~~Corrosion~~ (Note 12)						
	←	~~Vibration wear~~ (Note 13)						
	←	~~Foreign objects~~ (Note 13)						
	←	~~Erosion/Corrosion~~ (Note 13)						
Transparent clad (Note 14)	←	~~Diffusion through the clad~~						

Note 1: Molten clad

Molten clad can only be the result of a high coolant temperature or extremely large heat flux. The LBE coolant has a boiling point far above the clad melting temperature and as long as the coolant does not boil the thermal coupling between clad and coolant is very good and their respective temperatures remain close, even for a large heat flux. A MYRRHA fuel pin always has significant internal pressure during normal operation, already at BOL, and the outside pressure will always be low compared to the internal pressure. Therefore, a high temperature of the clad will always have to be combined with high mechanical stress and thus burst failure will occur before melting.

Note2: Molten fuel.

A different mechanism is the case of a very strong heat flux on the clad, by contact with molten fuel that relocates from the centre of the pellet to the clad during a power transient. Here, fuel melting occurs by completely different mechanisms than high coolant temperature.

Although the failure mode by contact between clad and molten fuel is not clearly established, we will consider (until further clarification) the criterion that there is no fuel melting as a decoupling criterion for clad damage. Molten fuel can only be the result of a fuel overpower phenomenon.

Note 3: Clad melting by high coolant temperature.

The temperature of the coolant can increase in case of increase of core power, decrease of power removed from the primary system and decrease of flow through the core.

Because of the large thermal inertia of the primary system, a total, unprotected loss of the power removed from the primary system leads to a slow increase of the coolant temperature (300°C/h, without reactivity feedback). This implies that for scenarios with loss of heat removal from the primary coolant there is sufficient time to stop the reactor by the reactivity control safety function. If, however, after a scram of the reactor, the decay heat is still larger than the heat removed, the coolant temperature can continue to increase in an uncontrolled manner. Since four lines of defence for decay heat removal are present, such sequence can be excluded.

In case of an uncontrolled increase of coolant temperature in the whole primary system, the clad is certainly not the barrier that is going to fail first. The maximum allowed temperature for the primary system is much lower than for the fuel pin such that mechanisms affecting the primary system barrier due to a high temperature cover any initiating event analysis for the clad by this mechanism. In all these conditions, the clad will fail due to bursting long before melting. So, we can certainly cancel this condition as a possible initiator for clad melting.

Note 4: Clad high stress.

Rupture of the clad due to high stress in the clad material can be characterized only by a maximum high clad temperature and differential pressure over the clad reached during the transient (and not the dynamics). The temperature of the clad itself does not increase the stress but decreases the threshold for failure by stress.

Since the differential pressure over the clad depends on the internal pin pressure and external pressure, high stress is obtained through high internal pressure and low external pressure.

In case of burst by high stress, there is no clear information on the possible size of the rupture and whether fuel particles large enough to block flow channels of the pin bundle can escape from the pin. Therefore, at present failure by rupture is not used as decoupling criterion for any DBC2 up to DBC5 transients.

Note 5: Clad stress by low external pressure.

The external pressure at the position of the fuel assemblies corresponds to the hydrostatic pressure at that depth in the reactor, which is about 4-5 bar (including the cover gas static pressure). This pressure is comparable to the initial filling pressure of the fuel pins (5 bar) and lower than the initial (BOL) internal operational pressure inside a fuel pin (about 10 bar). The external pressure will only decrease in case of a loss of coolant accident. Depending on the size of a hypothetical reactor vessel break, the pressure can temporarily drop to a low value during the transient phase. This will have to be considered for this transient but the influence of the small relative rise of pressure difference on the burst correlation is much lower than the influence of the temperature increase. A fast LOCA is the only transient during which the external pressure would reduce but this IE will already be identified as a possible mechanism or increasing the clad temperature.

Note 6: Clad high strain.
When excessive strain is imposed on the cladding, the material will develop cracks and loose its leak tightness. The strain limit at which failure of the clad occurs depends on the strain mechanism. Creep is a typical phenomenon that induces strain. For short-term transients, creep failure is covered by high stress failure. Creep is a phenomenon that is taken into account for normal operation and for the long-term integrity after unrecoverable events, but it is not a phenomenon that is important for protected transients (transients with a maximum duration a few seconds, the time needed to take protection actions). Two conditions can lead to high strain: excessive internal force and excessive external force. An excessive external force can be the consequence of coolant freezing, high external pressure or mechanical impact by objects.

Note 7: Clad internal mechanical force.
Excessive internal force can be caused by pellet-clad mechanical interaction (PCMI). During transients that increase the power and hence the temperature of the fuel pellets the thermally expanding pellet can come into contact with the clad and impose strain on the clad. If the strain is too high, the clad will develop cracks and lose its leak tightness. During normal operation, the pellet also expands due to burn up and PCMI can develop. The process is however so slow that the strain is relieved by irradiation and does not lead to crack development.
Flow reduction events can have an impact on the pellet temperature, but the cladding temperature will increase simultaneously and expand more. For that reason, these types of transients will not lead to PCMI. Only fuel overpower transients can lead to PCMI.

Note 8: Clad mechanical stress by freezing.
Freezing of the coolant (that happens around 125 °C) could by shrinking or by expansion during the phase transition induce strain/stress on the clad and endanger its integrity. Very small density changes are expected during the freezing of LBE, but experiments indicate that after solidification a slow recrystalisation occurs with a more important density change. Freezing can also block the flow and lead to a temperature increase for heat sources in the flow blocked region. This is a complicated mechanism, very dependent on the geometry of the system and the geometry of the freezing. Freezing in this case is a mechanism for clad overheating. At present, we have insufficient knowledge to exclude that freezing can have unacceptable mechanical or heat removal consequences and the present strategy is to prevent freezing of the core and of the flow path for decay heat removal.

Note 9: High external clad pressure.
The external static pressure on the clad is limited by the overpressure protection of the reactor vessel. The overpressure protection is set at a low value of a few bars and the hydrostatic pressure of the coolant (4 bar) must be added to this. A pressure of less than 10 bar outside the clad will induce only low (and compressive) stresses in clad. Moreover, the minimum internal pressure in the pin is already of the same order of magnitude.
Transient pressure waves in the primary system are not necessarily limited by the overpressure protection of the primary system. The only identified mechanism for explosive phenomena in the

primary system is the sudden contact of water with LBE at a much higher temperature than the vaporization temperature of water at the static pressure in the core.
The places where there is possible contact between the LBE and water are the primary heat exchangers and the beam line (IPSs with water are currently not considered in the design).
The PHX pressure envelope is designed with a double wall to eliminate events and sequences that could cause the release of water in the primary system. In case of a heat exchanger tube rupture, the water flow is rather limited and impact on the core of the pressure waves generated in the PHX bundle is very unlikely.
For what concerns possible ingress of water through the accelerator beam line, this will be avoided by the design requirement that there must be a double barrier between any water source around the accelerator beam line and the beam line itself. Fire extinguishing of the accelerator will always include the closure of the isolation valves between the accelerator and the reactor.

Note 10: Foreign object mechanical interaction.
To be able to damage the clad of a fuel pin an object must enter the flow channels between the fuel pins. Only very small objects (size of maximum a few mm) can enter the flow channels. Such objects will never be massive enough to inflict mechanical damage to the clad by one or a limited number of collisions with the clad surface. Fretting (wear) due to fast periodic movement of the object or the pins can lead to local material loss of the clad and eventually penetration of the clad. Experience feedback in PWRs demonstrates the ability of small objects blocked in the fuel pin bundle to inflict clad damage. The mechanism is further discussed as a material loss mechanism in Note 13.

Note 11: Clad fatigue.
Fatigue is a phenomenon that leads to failure by repeated application of a stress cycle. Since accidental transients (DBC3 and higher) will represent at most one cycle during the lifetime of a fuel assembly, fatigue is not a possible mechanism for general clad failure during these transients.
Fatigue due to normal operation transients and anticipated operational occurrences (AOOs) will be considered to determine the initial condition of the clad for accident analysis. This agrees with for instance the ASME and RCC-M codes. Fatigue cycles of normal operation transients and AOO can be monitored and compared to the number that is anticipated.

Note 12: Clad corrosion.
Corrosion is a phenomenon that can generate material loss up to an amount that the clad internal stress in the affected zone reaches burst stress and ruptures. Localized corrosion (corrosion cracks, pitting, …) can also lead to pin failure but in corrosion tests with LBE on the structural material chosen for the primary system of MYRRHA only general loss of material is observed. During normal operation, corrosion is managed by controlling a minimum amount of oxygen in the LBE and by limiting the temperatures in the reactor, i.e., by limiting the maximum temperature on the cladding.
The low probability of clad failure due to corrosion during normal operating conditions is demonstrated by the clad corrosion qualification programme, a precise chemical conditioning and monitoring of the coolant, and the quality of clad manufacturing. The large number of fuel pins present in the core does not allow excluding stochastic clad failures due to corrosion, however this will only concern one pin at a time and the consequences of such event are below SO1. Stochastic failures of a fuel pins are considered part of normal operation and are accounted for in radioactive releases during normal operation.
Corrosion experiments also show that short term transients (protection time of the order of 3 seconds) within the temperature range that prevents rupture of the clad, will not lead to failure by corrosion.

Note 13: Clad external mechanical attack
Vibration wear is a mechanism that can jeopardize the leak tightness of the clad barrier. The mechanism is however so slow that the simultaneous loss of leak tightness of many fuel pins by this mechanism is not credible. Individual fuel pin failures are considered in the design to be part of normal operation. Single and even several fuel pin failures can be mastered by the normal operating systems and this phenomenon can be considered not to be able to produce off-site consequences larger than SO1. Since at present, for normal operation, a 'no fuel leak strategy' is adopted, the phenomenon might increase the frequency of fuel pin failures and require a more frequent shutdown of the reactor.
Mechanical attack of the clad by a foreign object will create only isolated single pin failures with limited radiological consequences that can be mastered by the normal operating systems. The phenomenon might increase the frequency of single pin failures and require a more frequent shutdown of the reactor. Foreign objects in the pin bundle can have a different origin. During manufacturing quality assurance and controls should limit the probability to insert and leave a foreign object within the pin bundle.
The assumption that phenomena that only lead to individual pin failures are non-critical for safety (not being able to produce events with important releases), presumes that there is no mechanism for fast extension of single pin failures. In sodium-cooled reactors, the chemical interaction between the coolant and the fuel causes a relatively fast mechanism for extending single pin damage to neighbouring pins and beyond. Stopping this mechanism requires an automatic scram of the reactor on a relatively short time scale (minutes). For LBE coolant, there is no evidence for a coolant/fuel interaction and at present, we suppose there is no such interaction. Even in the absence of any coolant/fuel interaction, release of fuel particles from a single failed fuel pin cannot be excluded and such fuel particles might get stuck in the cooling channels of the fuel bundle and considerably reduce the flow locally. Calculations of the thermal impact of blockages in flow channels demonstrate that local blockages of flow channel will not lead to clad temperature at which short term failure can occur. The localized increase of clad temperature can however lead to increased corrosion speed, and this could provide a possible extension mechanism for the initial failure. The extension by corrosion is supposed to be so slow that the adoption of a no leaking pin strategy can prevent extension of a single pin failure. This still has to be further confirmed by corrosion tests at higher temperatures.

Note 14: Clad material intrinsic retention properties
Even if no material loss or loss of mechanical integrity occurs, one can imagine that in certain conditions (some) fission product are capable of diffusing through the stainless steel barrier of the fuel pin. This actually happens with tritium in normal operating conditions and at present we consider that during normal operation all tritium escapes from the fuel pin and contributes to releases and waste production of normal operation. Diffusion through the barrier is a slow process and even at high temperatures the diffusion of important quantities of fission gasses during the duration of a transient is negligible. We therefore consider that this phenomenon will not play any role in transient analysis.

Taking into account the actual physical properties of the clad and some well-established characteristics and design options of the installation, a very limited number of failure mechanisms remains for the clad. Particular for the clad is that a stochastic failure of the barrier itself (not correlated with a transient that affects many or all pins) will be limited to one pin simultaneously and have only minor radiological consequences that can easily be mastered within normal operating limits. The possible extension of such event due to flow blockages by released fuel fragments cannot be totally excluded at present and will have to be prevented by monitoring fuel pin integrity and adopt a strategy of no continued operation with leaking fuel pins. Because of the expected very slow extension mechanism (by corrosion) no automatic protection needs to be associated with this mechanism.

The remaining failure conditions for the clad taken from Table 2 are:

- fuel overpower,
- high coolant temperature,
- high internal pin pressure,
- coolant freezing,
- pressure transient (shock wave) in primary system.

For each of these conditions in the following, a tree of mechanisms is further developed to identify events that can produce these conditions.

3. 2. 2. 1. Fuel overpower.

Since core neutron dynamics is different for the subcritical core and the critical core, different damage mechanism sequences need to be developed for each operating mode.
The power of the subcritical core is determined by the reactivity of the core, the intensity of the proton beam on the spallation target and the coupling between the source neutrons and the core. The latter parameter, the coupling between source and core, is optimized during normal operation and events involving only this parameter cannot increase the general power of the core. It cannot however be a priori excluded that they could have an impact on local overpower.

<u>Fuel overpower event mechanisms for critical mode (Table 3)</u>

Note 1: Local overpower.
In a fast reactor core, the average distance (in a straight line) for a neutron to the next fission is about 50 cm, such that local overpower by a local increase of reactivity is not a viable mechanism for short term clad failure. The situation changes when matter with a thermalizing effect is locally introduced in a fast core. Due to the very high flux in a fast reactor and the much-increased fission cross section for thermal neutrons, neutrons that are thermalized locally can lead to very strong local power effects.
Local overpower could also be due to FA misplacement (Note 5) or control rod misalignment (Note 6) but no strong effects are to be expected in these cases. Since the most reactive fuel assemblies are already in the centre of the core, misplacing a fresh fuel assembly will lead to a lower local power than the one on the central fuel assembly. Control rods can only decrease local power flux and can only have a small indirect overpower effect fuel assemblies the rod.
The matter present in the reactor vessel or connected systems that could have a thermalizing effect are water in the PHX and cooling water of accelerator components. Only the former sources can lead to water ingress from the PHX to the core (Note 4).

Note 4: Water ingress in core from PHX.
If water leaks into the primary system at the level of the PHX, only small bubbles can be transported with the flow to the core. Moreover, due to the temperature and the pressure that reigns in the lower plenum of the primary system, the vapour will have a very low density and will not be able to thermalize neutrons.

Note 5: FA misplacement.
Since for experimental reasons the core of MYRRHA is designed for maximum peaking in the middle of the core, it is probably impossible to create larger power densities by misplacement of fuel during refuelling. The misplacement of a fuel assembly can however create local power densities that where not anticipated in the safety analysis. In such a case, however, two events (the FA misplacement and the initiating overpower event) need to occur simultaneously in order to have a problem. Moreover, all fuel assemblies will have a unique error coded identification code that can be read out during each

in-vessel manipulation. This approach allows excluding FA misplacement by errors during the handling operations.

Note 6: Control rod misalignment

Control rods will be equipped with a continuous position monitoring and associated alarms. Control rod misalignment calculations will be performed to determine the impact on local power density and to assess the potential safety consequences [17]. Because the local flux in a fast reactor is much less susceptible to local reactivity changes, the impact is expected to be small. Moreover, the control rods are located at positions in the core with relatively low power density.

Since safety rods at power operation will always be in the fully extracted position, misalignment of safety rods increasing the local power density can be excluded.

Note 7: Fuel compaction

In contrast with a LWR core, the core of a fast reactor is not in an optimized geometry for reactivity and a compaction of the fuel can lead to an important increase of reactivity.

Two mechanisms are identified that can lead to the relocation of fuel in the core.

First there is the possible failure of the structures that position fuel pins and fuel assemblies. If fuel pins inside a fuel assembly would move this is not expected to have an impact on the reactivity of the core (it would have an impact on coolability but this mechanism is treated elsewhere). If a single fuel assembly moves, this can also be expected to have negligible impact on core reactivity. If, however, many fuel assemblies would move inward in a coherent way, this has an important positive impact on reactivity. Since the upper core support structure is composed of individual tubes, no credible failures can be imagined that would lead to a coherent compaction displacement of the top of fuel assemblies. The lower core support structure is however one single structure and a failure by which the support of all fuel assemblies is lost simultaneously can not a priori be excluded. The worst possible impact of the complete loss of the lower core support has been calculated in [10] and this value of reactivity insertion will be used as an enveloping event for all possible fuel displacement events.

A second mechanism for relocating fuel is when the clad of many fuel pins in one or more fuel assemblies fails. The simultaneous failure of the clad of many fuel pins is not an initiating event but could be the consequence of an initiating event. No event sequences producing such conditions have been identified until now. Hypothetical accident conditions including fuel relocation that have been studied are: the single fuel assembly internal blockage and the full core blockage (MSA). In these conditions, we observe fuel relocations that lead to compaction with positive feedback on reactivity [11]. For the unprotected single fuel assembly blockage, the impact on reactivity is small and the power increase due to it is limited to a few 10% of full power. It even can provide an extra (diverse and redundant) means to detect the event and protect it. The MSA is a protected event. Strong compaction and reactivity increase is observed during the MSA scenario but at the moment not larger than the anti-reactivity inserted by the protection system.

Note 8: Removal of anti-reactivity

Mechanisms of reactivity changes due to IPS inventory changes are highly dependent on the IPS configuration and will be considered separately. Most of the core is arranged to maximize reactivity and the use of materials that have a negative impact on reactivity is avoided, except for the control of reactivity (control rods and safety rods) and the content of some IPS.

In contrast to sodium, removal of LBE coolant from the core (voiding) has only a small positive reactivity effect, comparable to steam release in the core. Boiling of LBE happens at a higher temperature than melting of the clad, such that ingress of gas (such as water vapour) is the only way to remove LBE from the core.

Structural materials of the core have a small negative impact on reactivity, but except for a core degradation scenario, there are no physically credible mechanisms by which these materials can be

removed from the active part of the core. For MYRRHA all core degradation scenarios will be excluded and only in the master severe accident this mechanism might be important.
Control rods are designed to introduce a large negative reactivity for reactivity control of the core. Ejection of a control rod is not possible as for a PWR since the system is not pressurized. A credible scenario of this kind would be the failure of the control rod travel stop mechanism in case of rod drop. If such failure happens during the drop of all rods, the reactivity effect is largely compensated by the other rods. So, a control rod ejection for MYRRHA actually corresponds to an event (single rod drop) plus the very unlikely failure of the rod travel stop. The control rod travel stop will be designed such that its failure can be excluded.
Safety rods are always fully extracted before the core is made critical. Nevertheless, hypothetical ejection of a safety rod injects a reactivity of 250 pcm. The safety rod ejection will also be eliminated by design. Uncontrolled control rod withdrawal remains the only reactivity injection event coming from the reactivity control systems. The speed of the control rods will be physically limited to about 30 pcm/sec which is largely sufficient for the normal reactivity control of the reactor.

Note 9: IPS reactivity insertion.
The mechanism is intentionally limited to events by which supplementary extra fissile material is deposited in the IPS. The option is chosen to introduce new experiments only during the shutdown of the reactor. In that case, the reactivity insertion by the IPS insertion (or removal) is anticipated and sufficient anti-reactivity will be present to compensate for the intended reactivity insertion and possible insertion errors.

<u>Fuel overpower event mechanisms for subcritical mode (Table 4)</u>
The general power of the subcritical core is determined by the reactivity of the core, the intensity of the proton beam on the spallation target and the coupling between the source neutrons and the core. Coupling between source and core is optimized during normal operation and events involving only this parameter cannot increase the general power of the core. It cannot however be a priori excluded that they could have an impact on local overpower.

Note 1: Local overpower increase.
The mechanisms for local overpower in a critical core are valid for the subcritical mode of operation too.
Water ingress in the beam tube from the accelerator could theoretically lead local overpower for the fuel pins around the beam tube but such scenario can be excluded because the water would immediately boil at the low pressure in the beam tube. If unmitigated such event would probably destroy the beam line and provide a direct connection between the LBE/cover gas source term and the non-nuclear part of the accelerator that is not provided with a leak-tight barrier.

Note 2: Accelerator beam misalignment.
A misalignment of the accelerator beam will displace the source neutron generation closer to the fuel pins in the direction of the misalignment. The source neutrons have an even longer mean free path than the fission neutrons and their displacement by maximum a few centimetres (until the beam reaches the beam tube) will certainly not have an important impact on local power.
The protons of the accelerator beam and of secondary particles generated by these primary protons in nuclear reactions have ranges that do not allow excluding them from directly reaching fuel pins. Such particles, by the stopping power released in the fuel could increase local power density.
MCNP calculation results indicate that beam misalignment cannot result in local overpower by increased fission only by charged particles.

Note 3: Reactivity increase.
In a critical core, the mechanism for global overpower is a positive reactivity injection. Many of these events remain valid for the subcritical core and will have a similar reactivity effect.
Because of the large subcritical level of the MYRRHA core (keff = 0.93), reactivity injection events that are challenging for the critical core will only have a minor effect on the power of a subcritical core. Since the subcritical mode of operation does not introduce any new reactivity events, the safety analysis of the critical core for reactivity events should largely envelope a demonstration for the subcritical core.

Note 4: Beam power increase.
A beam power increase has a direct proportional impact on the core power without any substantial time delay (order of microsecond). A detailed study of the maximum beam power increase is not available at present. Therefore, the maximum allowed value will be determined and will serve as a design basis for the accelerator.

Table 3: Fuel overpower event mechanisms for critical mode

Failure cond.		Mechanism sequence						
Fuel over-power	←	Local overpower (Note 1)	←	Water ingress	←	~~Water ingress inside IPS *~~		
					←	~~Water ingress in core from IPS~~ *		
					←	~~Water ingress in core from PHX~~ (Note 4)		
			←	~~FA misplacement~~ (Note 5)				
			←	Control rod misalignment-(Note 6)				
			←	~~Safety rod misalignment~~ (Note 6)	←	Fuel compaction (Note 7)	←	Core degradation (MSA)
	←	General overpower	←	Reactivity increase	←	Fuel compaction (Note 7) Removal of anti-reactivity (Note 8)	←	Core restraint mechanism failure
							←	Coolant voiding
					←	Removal of anti-reactivity (Note 8) IPS reactivity insertion (Note 9)	←	Core degradation (MSA)
							←	~~Rod ejection~~
							←	Control rod withdrawal
							←	Mo-target insertion
					←	~~IPS anti-reactivity extraction *~~	←	~~IPS loss of water inventory*~~
					←	~~IPS anti-reactivity extraction *~~	←	~~IPS water boiling *~~

*IPS with water is removed from the design

Table 4: Fuel overpower mechanisms for subcritical mode

Failure condition		Mechanism sequence				
Fuel over-power	←	Local overpower (Note 1)	←	Water ingress	←	Water ingress inside IPS
					←	~~Water ingress in core from IPS~~
					←	~~Water ingress in core from PHX~~
					←	~~Water ingress from accelerator.~~
			←	~~FA misplacement~~		
			←	Control rod misalignment		
			←	Accelerator beam misalignment (Note 2)		
	←	General overpower	←	Reactivity increase + (Note 3)		
			←	Beam power increase (Note 4)		

+: case enveloped by the critical mode event analysis

3. 2. 2. 2. Clad overtemperature

Clad temperature increase during a transient would eventually lead to melting of the clad (at 1680 °C). Because of the internal overpressure in the fuel pin that is already present in a fresh pin and increases by burn up, burst of the clad occurs at a much lower temperature and is the enveloping failure mode for fast transients involving a temperature increase of the clad.
At high temperature but still below burst temperature the clad can also fail by creep due to primary stress. For the timescale of protected transients, the failure is determined by the burst correlation. For situations in which the cladding can be subjected to high temperatures during a longer time period, the creep criterion also has to be verified.
The LBE coolant has a boiling point far above clad failure temperature and since clad temperature and coolant temperature are closely linked as long as the coolant does not boil, only transients in which the coolant heats up will represent a danger for temperature increase of the cladding.

Note 1: Fuel overpower.
In case of fuel overpower transients, the coolant temperature will rise proportionally to the overpower and very high overpower is required to make the clad fail. Overpower is limited by PCMI and fuel melting to values much lower than those that can produce coolant temperatures at which the clad fails. This means that because of the high boiling temperature of LBE coolant overtemperature of the clad can only occur by changes in the heat removal characteristics outside the clad. Namely in case of general or local loss of coolant, general or local loss of flow of the coolant, degradation of heat exchange or high coolant inlet temperature.

Note 2: General loss of coolant.
The mitigation strategy for general loss of coolant events allows to exclude the loss of coolant at the position of the core and to preserve decay heat removal capability by the secondary cooling system after such events. A loss of coolant event with uncovering of the core is excluded by design by the presence of a safety vessel. However, during the transient phase a loss of coolant event can have an important impact on the flow through the core.
Since the coolant boundary represents a barrier itself, the loss of coolant events is further developed in Section 3.2.3.

Note 3: Loss of coolant outside the vessel.
Loss of coolant events on other components than the reactor vessel (conditioning system, …) by design cannot drain the reactor vessel lower than the LBE suction in the vessel and will also not have an influence on flow through the core in the transient phase.

Note 4: Local loss of coolant.
By local loss of coolant, we mean conditions like the boiling of the coolant or ingress of gas in the coolant. In these conditions heat transfer from the cladding will decrease dramatically. At full power such condition could lead to clad failure if it persists during more than a second.

Note 5: Boiling of LBE
Boiling of LBE is a mechanism for locally losing the coolant. Boiling of LBE happens at a temperature above the melting temperature of the clad and is therefore not an initiator for high clad temperature but could be a consequence of other initiators for which the clad has failed long before LBE boiling occurs. LBE boiling can therefore only occur in core degraded conditions.

Note 6: Ingress of gas in the core.
There are no gasses present in the LBE and for the gas above the LBE surface we see no mechanism able to transport it to the level of the core.
There is water present in the PHX and this can turn into gas when it comes in contact with hot LBE. Water envelope parts of the PHX (have been designed double walled (bayonet tubes).
Another potential source for gas ingress is the nitrogen in the safety vessel. In case of a very large break of the vessel ingress of nitrogen in the vessel during the early, very dynamic phase of the transient cannot be excluded and might be difficult to calculate. It is also investigated whether large breaks of the vessel can be excluded by periodic inspection or by a leak before break demonstration for the vessel. If this turns out to be impossible, there is always the possibility to prevent large flow sections in case of a rupture of the vessel by mechanically restricting the displacements after the break.

Note 7: Homogeneous loss of flow in the core.
'Homogeneous loss of flow' groups the mechanisms that lead to a loss of flow that is identical for all fuel assemblies in the core and for which the origin is therefore located outside the core.

Note 8: Two primary pumps stop.
The simultaneous failure to run of both primary pumps requires a common cause factor for their failure. The design option today is not to provide emergency power for the primary pumps such that a loss of offsite power represents a common cause for the simultaneous loss of forced flow by both primary pumps. This transient however is inherently protected because upon loss of offsite power the CR and SR insert mechanisms will de-energize and the rods will insert. In case the initiating event is the loss of forced flow by one pump, the other pump will certainly experience this as a very important impact on its operating conditions. The choice is today not to make the primary pumps safety class such that the loss of the second primary pump needs to be considered as a consequential failure if aggravating. The delay of the failure of the second pump might depend on when the IE transient is felt by the pump.

Note 9: Locked rotor.
The instantaneous locked rotor is considered to be the envelope for mechanical events on the pumps with influence on flow. This event itself is certainly not a DBC2 but it will be considered as the envelope for all mechanical events. To avoid a detailed analysis of the probability of such events we will at present verify whether the envelope transient satisfies DBC2 safety objectives. Remark that in case of a locked rotor event on one pump, the other pump will be considered to fail as a consequential aggravating failure. The overall envelope case to be studied will be the locked rotor with aggravating failure of the other primary pump.

Note 10: General blockages.
This mechanism considers homogeneous loss of flow in the core by blockages of flow paths outside the core region or by random blockages of flow channels inside all fuel assemblies simultaneously. We can thereby distinguish blockage outside the core region, namely blockage of one flow path and blockages in the core region through blockages at the entrance of the core or homogeneous internal core blockage.

Note 11: Blockage of one flow path.
Blockages outside the core can be single obstruction blockages by a large object or slowly evolving blockages by small, suspended particles or by some chemical deposition mechanism.
Mechanical blockage by small particles is most likely in the core where flow sections are smallest. If the blockage is however produced by a chemical mechanism, the heat exchangers, being the coldest place in the primary system, could be the preferential place for blockage.

The impact of a gradual blockage of the PHX on the flow through the core can be easily detected long before the mechanism can produce an important increase of clad temperatures. We have at present no mechanism sequence up to an initiating event leading to the blockage of the PHX. Even if such events would exist, the resulting blockage would be slow and easy to detect, such that the problem reduces to decay heat removal. At present we consider this mechanism can be excluded. Analysis supported by research evaluates this potential mechanism.
Blockage of a flow path by a large object (foreign or due to a structural failure somewhere in the primary system) can be supposed to be a single blockage event (no simultaneous blockage of several flow paths). It is not straightforward to postulate the worst size for such blockage because the blockage also represents an obstruction for the inverse bypass flow through that flow path once the pumps have stopped. The locked rotor is an example of a flow blockage event, but it is already treated in Note 9 as a loss of forced flow event. It is not necessarily an envelope case for all possible flow path blockages. To analyse this mechanism, we will perform transient studies for a few hypothetical instantaneous blockages of single flow paths outside the core.

Note 12: Blockages at the entrance of the core.
A blockage at the entrance of the core, below the lower core support plate, will have a homogeneous influence only if the object of the blockage covers a large part of the inlet channels to the core. Most structures are in the hot plenum and have to pass through the pump to reach the entrance to the core. Such objects will be limited in size and will only be able to cover a small part of the core (see Note 15 on non-homogeneous loss of flow). The only structures below the inlet of the core that are capable to limit flow to a large part of the core is the baffle, the structure that is installed below the diaphragm to avoid the loss of a fuel assembly outside the region covered by the IVFHM, and the FA themselves during reloading operations.
The baffle below the diaphragm is an open structure with holes. The precise design basis for the baffle has not yet been established yet but the holes could certainly be made large and numerous enough to let pass sufficient flow to scram timely and to continue decay removal without jeopardizing cladding integrity. There will be a study for this in licensing phase.

Note 13: Homogeneous internal core blockage.
A homogeneous blockage of the core is possible by a random clogging of the flow channels between the fuel pins that represent the smallest flow sections in the primary system. An imaginable mechanism is the deposition of material on the cladding by an unanticipated chemical effect or the gradual blockage of flow channels by small particles suspended in the coolant. Both mechanisms are slow (compared to scram delay) and easy to detect. However, if the blockage continues increasing after scram this might eventually lead to loss of decay heat removal by natural circulation.
It is an important goal of the experimental LBE loops build and operated at SCK CEN to provide the experience feedback to demonstrate the absence of any (chemical) effect that might lead to material deposition on the clad or the creation of particles (precipitates) in the coolant during normal operation and in accident conditions. A detailed study will be performed on the mechanisms that can create blockages and how blockage by these mechanisms is or can be excluded. If such mechanism cannot be physically (chemically) ruled out they have to be excluded with sufficient confidence and reliability or stopped by mitigating actions. Possible mitigating actions could be to stop the primary pumps to avoid particles from being forced through the core, or to change the temperature to avoid further precipitation or deposition on the clad.
At present the experience feedback on the loops demonstrates the absence of clogging mechanism during normal operation. In order to exclude such mechanism in accident conditions, the research also covers the justification of a long-term safe state without any chemical conditioning of the primary coolant. Experience feedback shows that no clogging/deposition mechanisms should be expected in a long-term low temperature state (above freezing) without primary pumps functioning and without

conditioning of the LBE and the cover gas. This demonstration continues throughout the licensing phase.
We are convinced that there is and will remain a strong safety case to demonstrate the prevention of a homogeneous blockage scenario to a level that decay heat can no longer be removed. Because of the complexity of the chemical mechanisms in the primary system we consider such scenario as the one with highest residual uncertainty for the prevention core damage in MYRRHA. The fourth level of defence to mitigate severe accidents will therefore be designed on a Master Severe Accident scenario that is based on a homogeneous blockage scenario. As an envelope case the scenario of a protected instantaneous full core blockage is being considered. The scenario might still be relaxed (to a less unrealistic, slower developing blockage) if the mitigation of the postulated (unrealistic) envelope case turns out to be difficult to mitigate.

Note 14: Bypass flow of the core.
The high- and low-pressure parts of the primary system are separated by the diaphragm.
If a break occurs in this structure, flow will bypass the core and the flow in the core will decrease, depending on the characteristics of the primary pump.
The diaphragm is a complicated structure that is locally subjected to relatively important stresses. The diaphragm will also be very difficult to inspect because of low accessibility and because of its complex geometry and location in the LBE. Thermohydraulic calculations (1D and 3D) demonstrate that even large bypass failures can be protected fast enough to prevent clad damage. The static stress analysis of the structure during normal operation demonstrates that the stresses are relatively localized and that it is therefore unlikely that if a crack develops in one part of the diaphragm it will be able to propagate unstable to a large part of the structure. This would imply that large breaks for which core cooling could be jeopardized can be excluded. The possibility to design the diaphragm in such a way that large breaks (that could jeopardize core cooling) can be excluded (if possible, physically) will be analysed and is the preferable solution for this event group.
The transient phase of a vessel break will also represent a sort of bypass flow of the core (Note 2).

Note 15: Partial blockages at core entrance.
Partial blockages at the entrance of the core by objects much smaller than the core diameter are more probable than whole core blockages.
If the object comes from parts of the primary system, we have to distinguish again between those coming from the hot plenum that have to travel through the pump to reach the core entrance and those originating from parts in the cold plenum that can more easily and in any size migrate to the core entrance.
During power operation the only part present in the hot plenum below the core entrance is the baffle. This part will be designed such that it cannot reduce flow to the extent that cladding damage can occur during power operation (forced flow) and decay heat removal (natural circulation).
During reloading operations also parts of the IVFHM and full fuel assemblies could come loose and travel to the entrance of the core. The bottom nozzle of a fuel assembly is provided with side openings that allow inflow if the main flow opening of the bottom nozzle would be blocked by an object The diverse flow inlets of an individual fuel assemblies have to guarantee that if one entrance is blocked, the other entrances provide sufficient inflow to limit the temperature of the cladding to a value it can survive during a time necessary to detect the temperature increase in the affected fuel assembly.
If the blocking object extends over the inlet of several fuel assemblies, there will be diverse detection of high temperature at the outlet of several fuel assemblies, and this will produce diverse scram signals of for the reactor. After scram sufficient flow should remain to remove decay heat. This must be analysed in more detail, but we are convinced that it is always possible to put spacers on the lower core support plate that prevent an object from totally blocking core entrance. This scenario can be further analysed in the licensing phase.

The case in which the inlet of only one fuel assembly is blocked by an object, has no diverse detection because only a 2/3 high temperature detection is foreseen at each FA outlet. This case is mitigated by the large flow that remains available through the side holes in the inlet nozzle. A conservative estimate of this flow has been calculated for a hypothetical design of the inlet nozzle. The analysis demonstrates the possibility to implement this inherent property in the bottom nozzle of the FA.
The required residual flow in case of main flow entry blockage is based on the prevention of the clad damage by burst. We consider that the unprotected condition after the blockage event will only persist for a limited time, such that clad failure by corrosion can be prevented by stopping the reactor. This will be guaranteed by an LCO (Limiting Condition for Operation) that limits the continuation of the operation at power in case failures of thermocouples are detected at the outlet of a FA. The justification is that the undetected failure of 3 thermocouples at the outlet of a fuel assembly can be excluded. Such situation could only go undetected if the thermocouples all would indicate similar wrong outlet temperatures that roughly correspond to normal operating temperatures. Such common mode failure is physically impossible for thermocouples. To prevent a common cause failure in the protection system, the signals of the thermocouples of each FA will also be processed by both control and the protection system.

Note 16: Partial blockages inside a fuel assembly.
Blockages inside a fuel assembly have a completely different mitigation strategy than for blockages at the entrance of the fuel bundle or blockages inside the fuel bundle.
Blockages at the entrance of the fuel bundle are inherently mitigated by the design of the inlet nozzle of a fuel assembly. The flow entry openings in the inlet nozzle are sized such that an object that can pass through can only block a fraction of the inlet surface to the fuel bundle that doesn't lead to excessive cladding temperatures. Calculations show that more than 80 % of the inlet surface of the tube bundle can be blocked before the clad temperature reaches the burst limit. Objects of such size will not be able to pass the inlet nozzle opening. On the longer term the event is either protected or the reactor is manually shutdown in case the protection is not operational. Decay heat removal in natural circulation mode also provides sufficient cooling for long term integrity of the clad.
Blockages inside the pin bundle are a different story. Blockages inside de pin bundle can be produced by objects small enough to enter the pin bundle but having a probability to get blocked somewhere in a flow channel. A possible strategy to prevent or lower the probability for such events is to provide an entrance grid to the pin bundle that only allows passage of objects that have a very low probability to block in the fuel bundle. But such measure cannot prevent blockages by objects that could be left in the pin bundle during manufacturing or blockage in the pin bundle by fuel escaping from a stochastically failed fuel pin.
Enveloping calculations have been performed for isolated blockages of flow channels. These demonstrate that for blockages limited to one flow channel, the temperature increase of the cladding will not lead to fast failure (by burst or creep). However, events of isolated blockages inside the pin bundle will not lead to a measurable impact on the outlet temperature of the affected fuel assembly. Such event can therefore not be detected and protected by reactor shutdown. Therefore, increased clad corrosion could on the long-term lead to clad failure of the affected pin(s). The event of a few clad pin failures will not lead to worrisome radiological consequences. But the pin failure by corrosion can be accompanied by the release of fuel particles that produce blockages in neighbouring channels. This potentially leads to a progressive extension of the problem to the whole fuel assembly.
The ingress of particles in different neighbouring channels of a same fuel pin is considered extremely unlikely. Such an event is stochastically only possible with the presence of many blockages and will be detected by a general increase in pressure drop over the core and of the outlet temperature of the fuel assemblies.
A stochastic pin failure or a pin failure by an initial blockage can possibly release fuel particles in the flow channels. The blockages that could be formed by such particles represent a specific threat

because the blocking particles themselves will produce heat. On the other hand, blockages by irregular particles will have a porous character and will still allow some flow in the blocked channel(s). Calculations of local maximum cladding temperatures in case of blockages by fuel particles will be performed to determine the possible propagation of the open failure of one pin.
The strategy today is to prevent the extension of local blockages by shutdown of the reactor in case a failed fuel pin is detected and by replacement of the FA with the failed pin. This very conservative approach might be relaxed in the future. If there is possibility to make a distinction between fission gas release and fuel particles release from a failed pin, reactor shutdown could be decided only in case of detection of fuel particles release. If analysis demonstrates that blockage by fuel particles only lead to a slow and gradual extension, shutting down the reactor is only necessary after detection of multiple pin failure.
The pin failure propagation for MYRRHA is, at least in the first phase of the event, probably determined by corrosion, a slow phenomenon. This propagation mechanism is very different from the ‘pin failure propagation mechanism’ in a sodium reactor. In a sodium reactor the propagation of an individual pin failure is due to a chemical interaction between sodium and fuel. The reaction compound has a lower density than the fuel and expands the failed pin such that the flow channels around the affected pin are completely blocked (the flow channels in an SFR-type fuel assembly are smaller than for MYRRHA). The increased temperature in the blocked channels will soon lead to local sodium boiling and the failure of the clad of the neighbouring fuel pins. Tests on the propagation speed of a pin failure in sodium coolant have been performed and have shown that this mechanism has to be stopped within a timeframe of 15-30 minutes to avoid so much damage that even decay heat removal becomes compromised. In a sodium reactor the open pin failure (failure with entrance of sodium into the pin) is detected by detecting fuel release into the primary system.
The propagation by corrosion for an LBE cooled reactor is much slower than propagation driven by the chemical interaction in a sodium cooled reactor. Since at present we have no method to selectively detect an open pin failure (one with fuel release), the strategy is to stop the reactor in case of detection of fission gas release by a single pin failure.

Note 17: Degradation of heat exchange between fuel pin and coolant.
Deposition of a layer of material with low thermal conductivity on the fuel clad would increase the temperature of clad. An oxide layer of maximum 0.5 µm thickness is considered for normal operation and for transient analysis. This oxide layer is based on experience feedback.
If the oxide layer is thicker than anticipated, the temperature of the clad will also be higher than anticipated. To produce creep or burst failures the temperature has to increase al lot and the oxide layer has to become unrealistically thick. A thicker than anticipated oxide layer might lead to increased corrosion of the clad, but a thicker oxide layer will also better protect the clad metal against contact with LBE. Even if the oxide layer would be thicker than anticipated, the impact would be very stochastic and can lead to an increased rate of individual pin failures. The mechanism by itself cannot lead to multiple simultaneous pin failures and radiological consequences for the public above SO1.
Other depositions that might lead to a deceased thermal conductance between the clad and the coolant will have the same stochastic behaviour and will not lead to simultaneous multiple pin failures but might lead to earlier and more pin failures during transient.
The safety analysis will consider a maximum oxide layer thickness of 0.5 µm. In the research program experiments are foreseen to confirm the maximum thickness of the oxide layer and the absence of any deposit on the cladding that might appreciably deteriorate the conductance.

Note 18: Coolant high inlet temperature to core.
In case there is an imbalance between the heat produced in the primary system and the heat removed from that system, the coolant temperature and with it all the structures in the primary system will gradually heat up until balance is re-established or until primary structures start failing.

In a state in which the reactor produces fission power, a first action in case of power imbalance is always to stop the chain reaction. Since 2 diverse SLOD are present to execute this action, the failure of eliminating fission power can be excluded. Because of the very large inertia of the primary system (only 400 °C/h temperature increase for maximum power imbalance) there is always ample time to scram the reactor.

Decay heat cannot be switched off and in case of insufficient decay heat removal the temperature in the primary system will continue to increase until structural failures occur. Since the primary system will fail much earlier than the clad in case of overall temperature increase in the primary, and the clad will fail due to the loss of the primary barrier, this failure mechanism will be further discussed in the development of the primary barrier failure.

Table 5: Mechanisms for clad overtemperature

<table>
<tr><th>Failure condition</th><th colspan="5">Mechanism sequence</th></tr>
<tr><td rowspan="14">Clad overtemperature</td><td>← ~~Fuel overpower~~ (Note 1)</td><td></td><td></td><td></td><td></td></tr>
<tr><td rowspan="2">← General loss of coolant (Note 2)</td><td>← ~~Vessel breaks~~</td><td></td><td></td><td></td></tr>
<tr><td>← ~~Loss of coolant outside the vessel~~ (Note 3)</td><td></td><td></td><td></td></tr>
<tr><td rowspan="3">← Local loss of coolant (Note 4)</td><td>← ~~Boiling of LBE~~ (Note 5)</td><td></td><td></td><td></td></tr>
<tr><td rowspan="2">← Ingress of gas in the core (Note 6)</td><td>← ~~Ingress of water vapour~~</td><td></td><td></td></tr>
<tr><td>← Ingress of Nitrogen</td><td>← Vessel breaks</td><td></td></tr>
<tr><td rowspan="8">← Loss of flow</td><td rowspan="8">← Homogeneous loss of flow in the core (Note 7)</td><td rowspan="3">← loss of forced flow</td><td rowspan="2">← Two primary pumps stop (Note 8)</td><td>← LOOP</td></tr>
<tr><td>← Failure of one pump plus aggravating failure</td></tr>
<tr><td>← Locked rotor (Note 9)</td><td></td></tr>
<tr><td rowspan="3">← General blockages (Note 10)</td><td>← Blockage outside the core region</td><td>← Blockage of one flow path (Note 11)</td></tr>
<tr><td rowspan="2">← Blockages in the core region</td><td>← Blockages at the entrance of the core (Note 12)</td></tr>
<tr><td>← ~~Homogeneous internal core blockage~~ (Note 13)</td></tr>
<tr><td rowspan="2">← Bypass flow of the core (Note 14)</td><td>← Diaphragm failures</td><td></td></tr>
<tr><td>← Reactor vessel breaks</td><td></td></tr>
</table>

Table 6: Mechanisms for clad overtemperature (continued from Table 5)

<table>
<tr><th>Failure condition</th><th></th><th colspan="9">Mechanism sequence</th></tr>
<tr><td rowspan="8">Clad overtemperature</td><td rowspan="5">←</td><td rowspan="5">Loss of flow</td><td rowspan="5">←</td><td rowspan="5">Partial loss of flow</td><td>←</td><td>Partial blockages at core entrance (Note 15)</td><td>←</td><td>Blockages at FA entrance</td><td>←</td><td>lost objects (e.g. fuel assembly)</td></tr>
<tr><td rowspan="4">←</td><td rowspan="4">Partial blockages inside an FA (Note 16)</td><td>←</td><td>Blockages at the pin bundle entrance</td><td></td><td></td></tr>
<tr><td rowspan="3">←</td><td rowspan="3">Blockages inside the pin bundle</td><td>←</td><td>Foreign object ingress during operation</td></tr>
<tr><td>←</td><td>Foreign object during manufacturing</td></tr>
<tr><td>←</td><td>Stochastic pin failure with fuel release</td></tr>
<tr><td rowspan="2">←</td><td rowspan="2">Degradation of heat exchange between fuel pin and coolant (Note 17)</td><td>←</td><td>Deposition of layer on the clad</td><td colspan="6"></td></tr>
<tr><td>←</td><td>Oxide layer on the clad</td><td colspan="6"></td></tr>
<tr><td>←</td><td>Coolant high inlet temperature to core (Note 18)</td><td>←</td><td>Insufficient heat removal from primary system</td><td>←</td><td>Insufficient decay heat removal capability (see Table 8)</td><td colspan="4"></td></tr>
</table>

3. 2. 2. 3. Pin internal pressure increase

Note 1: Heating of the lower fuel pin gas plenum.
The internal pressure in the pin is mainly determined by the temperature of the lower gas plenum because most of the gas volume of the pin is located there (90 %). This hypothesis supposes that the gas can freely flow in all parts of the pin, also during a fast transient, such that the pressure inside the pin is always homogeneous.
The lower gas plenum part of the fuel pin does not generate any power and since the nuclear power is generated downstream the gas plenum, the lower gas plenum will only be heated by transients in which the LBE temperature at the entrance to the core increases or in case of a flow inversion in the core.

Note 2: LBE temperature increase at core entrance.
Due to the large thermal inertia of the lower (cold) plenum of the primary system the temperature in the lower plenum will only increase in case of unprotected overpower transients and in case of loss of long-term decay heat removal. Sequences of that type will be excluded by design.
A strong heating by the secondary system is impossible during power operation because the only power source in the secondary system (the heating system) does not have sufficient power for this and also because the secondary would fail at temperatures/pressures required for a temperature inversion at the PHX.

Note 3: Flow inversion in the core.
Flow inversion would require an inverse pressure difference over the core. Possible physical mechanisms that can produce an inverse pressure over the core are a strong heating by the secondary cooling system, a large break in the lower plenum or a transient level inversion in the primary system after a loss of forced flow event.
Temporary flow inversions have been observed in safety analysis for large core bypass events (large diaphragm failures), due to a damped oscillating behaviour of the levels in the low and high temperature plena of the primary system.
Temporary flow inversion has also been observed in the safety analysis of very large vessel ruptures.
We can conclude that pressure increase inside the fuel pin can only happen in case of excluded sequences or in case of scenarios with flow inversion. These scenarios will also induce a temperature increase of the cladding and will be treated in this event group (see Table 2, Note 8). The pressure increase in the fuel pin will be part of the calculation and taken into account for the burst failure criterion.

Table 7: Mechanisms for pin pressure increase

<table>
<tr><th>Failure condition</th><th></th><th colspan="11">Mechanism sequence</th></tr>
<tr><td rowspan="5">Pin internal pressure increase</td><td rowspan="5">←</td><td rowspan="5">Heating of the lower fuel pin gas plenum (Note 1)</td><td>←</td><td rowspan="2">LBE temperature increase at core entrance (Note 2)</td><td>←</td><td>~~Unprotected overpower transients~~</td><td colspan="6"></td></tr>
<tr><td></td><td>←</td><td>~~Loss of decay heat removal~~</td><td colspan="6"></td></tr>
<tr><td rowspan="3">←</td><td rowspan="3">Flow inversion in the core
(Note 3)</td><td>←</td><td>~~Strong heating from the secondary side~~</td><td colspan="6"></td></tr>
<tr><td rowspan="2">←</td><td rowspan="2">Transient loss of flow</td><td>←</td><td>Large break in lower plenum</td><td>←</td><td>Vessel breaks (See Table 8)</td><td colspan="2"></td></tr>
<tr><td>←</td><td>Loss of flow in the core</td><td>←</td><td>Large core bypass events</td><td>←</td><td>Diaphragm ruptures</td></tr>
</table>

Table 8: Mechanisms for coolant freezing

<table>
<tr><th>Failure condition</th><th></th><th colspan="3">Mechanism sequence</th></tr>
<tr><td rowspan="3">Coolant freezing
(Note 1)</td><td>←</td><td>Freezing by PHX overcooling (Note 2)</td><td>←</td><td>Simultaneous freezing of more than one PHX</td></tr>
<tr><td>←</td><td>Freezing by RVACS overcooling (Note 3)</td><td colspan="2"></td></tr>
<tr><td>←</td><td>Long-term freezing by heat losses (Note 4)</td><td colspan="2"></td></tr>
</table>

3. 2. 2. 4. Coolant freezing

Coolant freezing might lead to blockage of flow paths in the primary system and the inability to effectively remove decay heat from the core. Coolant freezing could however also lead to structural failures in the primary system barrier by induced mechanical stresses. These mechanisms will be discussed in the paragraph dedicated to failures of this barrier.

Note 1: Loss of primary flow by freezing.
There are three mechanisms by which heat can be withdrawn from the primary system and these are the inappropriate functioning of heat removal systems (2 mechanisms) and the passive heat losses of the primary system. There are to our knowledge no other mechanisms to remove heat from the primary system.

Note 2: Freezing by PHX overcooling.
There are many failure modes for the secondary system that can lead to an increase of the power removed from the primary system. A design requirement is to design the four secondary loops independent such that there is no single event that creates a cooling transient in two loops simultaneously. This for instance requires an independent control system for each loop and a physical separation to avoid that a break of one loop could induce a break in another loop. This way we can exclude simultaneous freezing of multiple loops. If we can exclude the simultaneous freezing of multiple loops, we can exclude the freezing of both flow paths in the primary system. Freezing of a PHX can also lead to a loss of the integrity of parts of the PHX and the release of large quantity of water in the primary system. Such overpressure mechanism could at the same time lead to loss of the integrity of the RVB and release of source term from the LBE. The mechanism is further developed in Table 8/Note 3.

Note 3: Freezing by RVACS overcooling.
The goal is to limit the heat removal capacity of the RVACS such that the first 72 hours of the automatic operation of the RVACS only limited freezing, without any safety consequences (mechanical or on flow) can occur. We suppose that after 72 hours RVACS condensers in the pools can be taken out of service to equalize heat produced and heat removed from the primary system.

Note 4: Long-term freezing by heat losses
After heat removal by the RVACS is reduced to a minimum, the inevitable heat losses of the system will ultimately (at very low decay heat) lead to a freezing of the LBE. Some calculations on how this freezing evolves have been carried out but have not led to a conclusion whether freezing at such a low decay heat can still lead to temperatures in the core at which clad damage can occur. The research program to find this out continues.
If freezing remains a safety problem at the low power of unavoidable heat losses, we will implement a safety class heating system. The power requirements will be maximum 50 kW (heat losses of the present reactor design) and the time after which such heating needs to be deployed will be much longer than 72 hours.

3. 2. 3. The LBE coolant barrier

The LBE forms a second barrier for the fission products confined by the clad. The radiological consequences of events leading to clad failures might be importantly influenced by the impact of the event on the LBE retention properties. The initiating events for fission product release are identified by the IE analysis for the clad barrier. The behaviour of the LBE barrier during these events will be considered if necessary (in case of clad failures). The goal is however to use decoupling criteria that guarantee clad integrity.

In normal operation the LBE will itself contain an important source term. This source term mainly comes from the activation of the coolant, and to a lesser extent from the activation of the impurities in the coolant and from fission product leaking from the fuel pins. The last group is due to stochastic pin failures that can be expected to occur during normal operation. Since at present the operation strategy is to immediately replace all FA with a failed pin, the contribution of fission gasses to the LBE source term will remain small. A specific case is tritium produced in the pin that diffuses through the clad in normal operating conditions.

Some of the activation and fission products are very volatile and badly retained by the LBE. Those end up in the cover gas during normal operation.

The LBE retention properties are element/compound specific and strongly temperature dependent in the sense that retention degrades with increase of temperature. The LBE retention properties for elements at different temperatures have been calculated.

- The retention for radioactive isotopes in LBE can therefore degrade in two conditions: high temperature of the coolant and formation of compounds that have a bad retention in LBE. For the first phenomenon envelope calculations can be performed for the temperature conditions that cannot be excluded. The second phenomenon is more difficult to develop because of the almost countless possible chemical interactions.

Table 9: Mechanisms for LBE barrier degradation

Failure condition	Mechanism sequence					
High LBE temperature (Note 1)	←	~~Fission overpower~~ (Note 2)				
		Loss of decay heat removal	←	Loss of core decay heat removal (Note 3)	←	~~Loss of 4 LOD for the DHR function~~
					←	~~Loss of flow through the core~~
			←	Loss of IVFS decay heat removal (Note 4)	←	~~Loss of 4 LOD for DHR~~
					←	~~Loss of flow through the IVFS~~
			←	Loss of Po decay heat removal (Note 5)	←	~~Loss of 4 LOD for DHR~~
					←	Loss of flow in the primary system
					←	LBE losses outside the primary coolant system
	←	Beam window leak/break (Note 6)				
Volatile compounds formed (Note 7)	←	Water ingress in LBE	←	~~Large PHX rupture~~ (Note 8)		
			←	~~PHX tube rupture~~ (Note 9)	←	~~Single rupture~~
					←	~~Single rupture with open reactor~~
					←	~~Multiple rupture in one PHX~~
					←	~~Double rupture in different PHX~~
			←	~~Rupture of the LBECS cooling heat exchangers~~ (Note 10)		
			←	TCS injection part untimely actuation (Note 11)		
			←	~~Water ingress from the accelerator beam line~~ (Note 12)		
			←	Secondary line break with open reactor (Note 13)		
	←	Other chemical interactions				

Note 1: High LBE temperature.
Calculations have been performed on the potential radiological impact of the escape of radiotoxic isotopes from the LBE at higher-than-normal temperatures. Very volatile isotopes or compounds of isotopes will already have escaped from the LBE during normal operation and will have to be handled in the cover gas and by the barriers that confine the cover gas. At higher-than-normal temperatures, more and other isotopes are released from the LBE. It is clear from source term radiotoxicity calculations that 210Po is the main radiological concern and that other isotopes that might be released from the LBE at higher-than-normal operating temperatures do not constitute a SO3 size source term for the public. Increase of LBE temperature can be due to loss of removal of heat added by: fission power (Note 2), core decay heat (Note 3), IVFS decay heat (Note 4) or Po decay heat (Note 5). Local increase of LBE is possible by heat deposited by the beam not being removed in case of a beam window leak/break (Note 6).

Note 2: Fission overpower.
Temperature of the coolant can increase when the heat added is lower than heat extracted from the coolant. The condition is only problematic with respect to evaporation of isotopes from the LBE in case of temperature increase at the surface of the LBE where the radioactive isotopes can escape. A local temperature increase in the core will not lead to the release of source term. Since fission overpower transients are quickly protected by the reactivity control system, this mechanism cannot lead to a temperature increase of the bulk of LBE.

Note 3: Loss of core decay heat removal.
Loss of decay heat removal scenarios that lead to high temperature in the vessel are discussed in Table 11/Note 5.
The conclusion is that the 4 LODs present allow excluding loss of decay removal as a possible mechanism for general heating of the LBE and the increased release of isotopes dissolved in the LBE.
Loss of flow through the core scenarios are treated in Table 5. In case of partial loss of flow, the LBE bulk temperature will not increase as long as decay heat continuous to be removed from the primary system. The case of a full core blockage has been excluded but is to be studied in the MSA.

Note 4: Loss of IVFS decay heat removal.
The IVFS decay heat is removed by the same system as the core decay heat. Sequences with the loss of the 4 LOD present for DHR can be excluded.
Total flow blockage in the IVFS can be excluded on the same grounds as for the core. The IVFS is however designed with a provision that provides continued external cooling of a fuel assembly is case of complete internal blockage within the pin bundle. This external cooling is provided in order not to involve IVFS degradation in case of the MSA.

Note 5: Loss of Po decay heat removal.
The decay heat released by radioactive isotopes in the LBE coolant is dominated by 210Po decay. It is about 300 kW and it is distributed homogeneously in the coolant. Although it is caused by many isotopes, we (by definition) call it Po decay heat.
Evaluations show that with sufficient decay heat in the core, the LOD foreseen for decay heat removal provide sufficient flow everywhere in the primary system to extract the decay heat of the Po without substantially increased temperatures anywhere in the system. If no fuel is present in the core or in the IVFS, there is no clear hot source/cold source geometry to generate flow in the primary system transporting heat to the cold source (PHX or RVACS). Calculations indicate that nevertheless heat is effectively (without excessive temperatures) transported to the cold sources. These calculations are being continued.

If LBE is spilled outside the normal confinement boundaries of the primary coolant (primary system and LBE conditioning system) due to a leak, break, … of these systems, 210Po can evaporate from the spilled fluid and potentially contaminate the atmosphere of the containment and leak to the environment. The potential for large releases to the environment by such a scenario is low because of the high retention of Po in LBE (even up to high temperatures well above 500 °C), because of the high binding of Po on stainless steel metallic surfaces of SSC in the containment such as the liner of the containment and because of the low expected leak rate for an event that doesn't pressurize the containment.

If, however a large amount of LBE is spilled, self-heating by the activation product could increase the temperature and the evaporation of Po and even heat up and pressurize the containment. This largely depends on the amount of spilled LBE and on the geometry and composition of the structures that collect the LBE. Calculations will be performed to see if options to prevent important self-heating, up to temperatures with large 210Po release, need and can be implemented in the design. Possibilities are the limitation of the amount of LBE that can be spilled, spill collection structures that promote passive heat removal and limit LBE temperature increase. As already pointed out, we do not expect LBE spill scenarios to be able to lead to radiological consequences for the public but contaminating the reactor hall with Po might importantly complicate waste management, radiation protection and decommissioning afterwards.

Note 6: Beam window leak/break.

A beam window leak/break is a mechanism by which LBE can enter the beam tube and be heated to (due to the insufficient cooling) very high temperatures at which Po is readily released and even to a temperature at which LBE evaporates.

Because the beam window is a very stressed component (high temperature gradients, high stresses, high radiation damage, …) its failure will be regarded as a very likely (DBC2) initiating event and as an aggravating failure for transients during which the working conditions of the window appreciably change.

The beam line will be provided with fast closing isolation valves that will receive a closing signal upon detection of a beam window leak or break, after the accelerator beam itself has been shut down.

Shutting down the accelerator beam will remove heat input and terminate the release, but a fast closure of isolation valves might be necessary to prevent contamination of the accelerator (unacceptable for a DBC2 event) by the already evaporated activity and to limit release (of 210Po, …) to the environment.

Depending on the nature and size of the leak/break, the physical development of the transient can be very different, the release rate of activity can vary orders of magnitude and detection possibilities can change importantly. If for instance LBE leaking into the beam tube would be continually evaporated the detection possibilities are substantially different than in case of a large break of the window.

At present two possibilities to protect the event are being investigated.

A second window as a structural barrier would suffer the similar uncertainties as the main window and would never qualify as an XLOD and maybe not even as a SLOD. Moreover, a structural window would have to be placed outside the reactor vessel (to provide cooling fluid) and will probably have an unacceptable influence on beam emittance. A very thin second window could however provide an excellent detection means and could provide absolute assurance that no activity has escaped before the isolation valves have closed.

The second possibility relies on the high adsorption probability and long residence time of elemental polonium on the stainless-steel surface of the beam tube. Because of this physical property the beam tube is a very effective filter for 210Po and the isotope would only escape after a very long time or in case of extremely strong leakage. The adsorption properties of Po have been determined and a model of the escape process is being developed.

Note 7: Volatile compounds formed.
This mechanism concerns the release of radiotoxic isotopes from the LBE by formation of a volatile compound due to the interaction with a substance that is normally not in contact with the coolant. There are not that many substances for which accidental contact with LBE is possible. All substances that are during normal operation continuously or regularly in contact with LBE should not produce a massive release of isotopes dissolved in the LBE. If chemical release already happens during normal operation, these volatile compounds must be dealt with by the condition system of the cover gas.
Substances that are present but normally separated from the coolant could in abnormal conditions make contact with the coolant and volatilize isotopes by chemical interaction. Examples of substances that can be accidentally released in the primary system are the boron carbide in the CR and SR, the fuel in the cladding, water in different components and systems. A systematic study will be performed of the substances that can come into contact with LBE and their possible chemical interactions with radiotoxic isotopes in the LBE.
At present only the release of 210Po by contact of the coolant with water has been identified as a safety relevant chemical release mechanism. This mechanism is particularly dangerous because it can simultaneously generate overpressure for the two remaining barriers.

Note 8: Large PHX rupture.
The PHX is situated inside the primary system and is partly submerged in the LBE. It contains large quantities of water in the feedwater line and in the lower collector providing entrance to the heat exchange tubes. Since pressure and temperature are those of a high energy system, without taking special measures breaks of these parts cannot be excluded. Breaks on these parts would lead to very dynamic and strong overpressure transients in the primary system that would be difficult if not impossible to mitigate. Therefore, the option of bayonet tubes that would allow excluding physically large volumes of water in the primary system was adopted.

Note 9: PHX tube rupture.
Heat exchanger tubes by their operational function unavoidably form a barrier between the two coolants: water and LBE. Double walled tubes allow to mitigate a tube rupture.
Because of the large number of tubes and because there is at present insufficient experience on possible degradation mechanisms of the tubes, a PHX tube rupture is considered as a DBC2 event. Since a DBC2 event will not be allowed to contaminate the primary containment, an overpressure suppression system is designed to collect the steam from the primary system and contain the contaminated water.
If the PHX tube rupture event would take place while the reactor is open for maintenance, the steam from the rupture might not be prevented from being released inside the primary containment. However, with the double-walled bayonet tube design a rupture of both internal and external tube may only occur during operation as LBE temperature is not high enough to cause tube pressurization during shutdown.
Multiple tube ruptures are a potential mechanical consequence of the freezing of a PHX by a secondary line break. It is very difficult to prevent freezing by a secondary line break. The exclusion of damage propagation is ensured by the double-walled design.
The simultaneous failure of PHX tubes in different heat exchangers is excluded on the basis of the low likelihood, because there is no common cause for such failures to occur simultaneous. The exclusion of a double independent tube rupture imposes a maximum frequency smaller than 10-3/y for the individual tube rupture (stochastic combination of independent events). This value must be justified by the inspection program.

Note 10: Rupture of the LBECS cooling heat exchangers.
The LBECS contains heat exchangers for cooling the LBE. Contact between the coolant and LBE will be excluded. Different means are possible: double walled pipes, intermediate LBE loop, etc. The ultimate solution will be chosen by the design engineer, but one design option is to exclude chemical contact between cooling agents and LBE.

Note 11: TCS injection part untimely actuation.
The TCS was installed to mitigate severe accident conditions. The safety function of the system is to preserve the upper part of the primary barrier by spraying water on the surface of the LBE after a severe accident in which fuel could be relocated to the surface.
The TCS is not designed to sufficient detail to be able to judge the probability for its untimely actuation. Moreover, the detailed actuation mode determines the mechanisms involved. If for instance only the injection part of the TCS is actuated there is also a risk to overpressure the primary system. The primary overpressure suppression system will however be able to take the capacity of the TCS injection that is limited to 1kg/s which corresponds to the PHX tube rupture flow rate.
The Design Engineer of the TCS will be given the task to minimize the likelihood for untimely actuation and if possible, exclude the event.
In the meantime, it will be supposed to be a DBC3 and the radiological consequences for a full actuation will be calculated.

Note 12: Water ingress from the accelerator beam line.
The potential for water ingress in the primary system by the beam line has been discussed in Table 3 and Table 4. The design basis for the accelerator is to exclude ingress of water by double barriers.

Note 13: Secondary line break.
In case a secondary line break occurs with the aggravating condition of open reactor, the surface of the LBE coolant will be exposed to water and Po can be released from the LBE. This mechanism will be analysed, and the radiological consequences of such event will be analysed.
The radiological consequences of such a scenario will probably be enveloped by PHX tube rupture scenarios. The scenario will be studied to determine whether this scenario becomes determining for the primary containment design in case the PHX tube rupture would be excluded. A secondary line break can also result from the freezing of the LBE on the primary side of the PHX. The potential mechanical impact of this mechanism on the integrity of the PHX tubes is a concern that is addressed in Note 9.

3. 2. 4. The reactor vessel barrier (RVB)

The RVB is a complicated barrier consisting of different parts with different safety functions and very different possible failure states.
The vessel part of the barrier is a simple metallic shell without any penetrations. The possible failure conditions are determined by mechanical and chemical conditions on the stainless-steel shell. A possible failure state to consider is the presence of cracks in the material leading to breaks or leakages. Crack propagation is related to mechanical stress and the physical properties of the material. The other possible failure state is the removal of material by a chemical interaction (e.g., material dissolution). The most probable failure state of this mechanism is a leak if the loss of material is local but can also be break if the chemical interaction results in a uniform thinning of a large surface of the shell.
The reactor cover part is a very complicated component with many penetrations for instrumentation, conditioning systems, heat removal, etc. Because it is not in contact with a chemically aggressive substance (only highly inert cover gas) we suppose that only cracks of mechanical origin leading to

break or leaks are applicable failure states for the shell/piping parts. All penetrations of the barrier are closed systems (instrumentation lines) or are equipped with a redundant set of isolation valves (cover gas conditioning, LBE conditioning, …). During normal operation, these valves are open. They are isolated in case of events on the systems they isolate and in case of events on the primary system that require isolation of the RVB.

The RVB also serves operational purposes to contain the coolant and control the oxygen content of the coolant (in the containment the oxygen concentration is still much higher than the concentration that is allowed in the LBE for normal operation). These two functions are preventative safety functions related to the integrity of the clad barrier and have been discussed in Note 12 of Table 2.

As a barrier of source terms, the RVB contains all the major source terms that are confined by the other barriers discussed before in this document. The RVB is the first barrier for the very volatile source term of the cover gas.

3. 2. 4. 1. RVB as a confinement for cover gas

The cover gas source term during normal operation is composed of volatile spallation and activation products of the LBE and occasionally volatile fission products of a leaking fuel pin (though this might be considered as a DBC2 event in case a no leak strategy is adopted for normal operation). The unmitigated full release of this source has consequences below SO2 and if the source term is released in the primary containment, the consequences are even below SO1.

Almost the whole barrier for the cover gas, including the systems connected to the primary system that also carry cover gas, remains within the primary containment, there is no reason to detail the mechanisms for cover gas release, since for all possible scenarios of cover gas release are mitigated to consequences below SO1 by the primary containment. The only reason to detail the mechanisms is for scenarios that would at the same time deteriorate the primary containment safety function. The only events on the RVB that also challenge the primary containment are those that release mass and energy in the primary containment. This concerns failures with water release in the primary containment. The IPS will never contain an amount of water that can jeopardize the integrity of the primary containment. Water release scenarios from the SCS can challenge the primary containment and need to be examined.

The only cover gas barrier bypassing the containment is the beam tube. IPS systems could also provide a direct connection between the inside of the RVB and the outside of the primary containment. At present it is decided that IPS systems that need to penetrate the primary containment will in any case remain enveloped by a containment barrier that is equivalent (in leak rate) with the primary containment. The reason to go outside the primary containment with IPS systems is the inaccessibility of the primary containment (nitrogen atmosphere).

At the height of the cover gas, the beam line is a simple metallic shell and a break as an initiating event can be considered to be a DBC4. At present, there are no parts of or connections with the beam line at the height of the cover gas for which a DBC2 failure of the leak tightness need to be considered. The beam window is a weak part of the beam line, but it is situated in the LBE and loss of integrity will not lead to escape of cover gas.

Another possible mechanism that can threaten the integrity of the beam line at the height of the cover gas is a beam misalignment. Beam misalignment events are probable events (DBC2) on an accelerator and they are protected by a very fast protection system to prevent the loss of the integrity of the beam line. It is possible to protect the beam line at the height of the cover gas against this mechanism by installing a collimator upstream in the beam line that physically prevents the beam from hitting the beam line in that region. The advantages (excluding a cover gas release mechanism) and disadvantages (beam losses on collimator and radiation protection consequences) of such solution are being examined.

Table 10: RVB failure mechanisms as a barrier of the cover gas

Failure condition	Mechanism sequence	
Leak/break collected in the primary containment not challenging the primary containment	←	Unspecified. Full instantaneous release considered as DBC2
Leak/break of the beam tube at cover gas	←	Leak/break as an initiating event
	←	Leak/break by beam misalignment
Release of water into the containment	←	SCS breaks inside the containment (See Table 12/Note 4)
	←	SCS breaks inside the reactor vessel (See Table 11/Note 3)

3. 2. 4. 2. RVB other safety functions

Apart from confining the volatile source term present in the cover gas, the RVB serves many other safety functions. These are safety functions that in case of loss do not directly lead to radiological release. Loss of these safety functions will only lead to radiological consequences if they simultaneously affect other barriers. Such mechanisms should have been identified in the analysis for the other barriers. We will perform the analysis for the RVB to confirm no new mechanisms are exposed.
The most important safety function of the RVB is to contain the coolant that is necessary for fission heat and/or decay heat removal. Decay heat removal is already discussed in Section3.2.2 on the clad barrier.
Another safety function is to contain the source term in the LBE. This source term has already been discussed in Section 3.2.3 concerning the LBE barrier. We will however repeat this analysis from the point of view of this barrier to verify that indeed no other mechanisms are exposed. The RVB at some places prevents the ingress of water in the LBE in order to avoid release of Polonium and overpressure of the RVB. This concerns the parts of the RVB that separate the secondary cooling system from the primary system. It might also concern IPS containing water (for cooling or for moderation). The release of water from an IPS in the centre of the core might also have an unacceptable mechanical impact on the fuel assemblies.
A particular part of the RVB is the accelerator beam tube, the only part that bypasses all containment functions that envelope the RVB. The loss of integrity of the beam window will expose LBE to the vacuum of the beam tube.
The RVB in contact with the LBE is composed of the following parts:

- The vessel part in contact with the LBE;
- The beam tube part in contact with the LBE;
- The IPS parts in contact with the LBE;
- Other cover penetrations in contact with the LBE (e.g., the LBECS.)

The failure conditions for these parts of the RVB and the corresponding mechanisms are discussed below.

Table 11: RVB failure mechanisms as a barrier for coolant loss, radioactive isotopes escape and oxygen ingress

Failure condition	Mechanism sequence					
Leak/break of the vessel (Note 1)	←	As an initiating event by crack development due to fatigue. (Note 2)				
	←	High pressure inside the primary system (Note 3)	←	Water ingress	←	~~Large PHX rupture~~ (see Table 9/Note 8)
					←	PHX tube rupture(s) (see Table 9/Note 9)
					←	~~Rupture of the LBECS cooling heat exchangers~~ (see Table 9/Note 10)
					←	TCS injection part untimely actuation (See Table 9/Note 11)
					←	~~Water ingress from the accelerator beam line~~ (See Table 9/Note 12)
			←	Compression of cover gas (Note 4)	←	Inappropriate level control in the reactor vessel
			←	~~Pressure control cover gas~~		
	←	High temperature in the primary system (Note 5)	←	~~Fission overpower~~ (See Table 9/Note 2)		
			←	~~Loss of decay heat removal~~ (See Table 9/Notes 3, 4, 5)		
	←	High pressure outside the primary system (Note 6)	←	~~PHX tube rupture~~		
			←	RVACS untimely actuation		
			←	Secondary system break		
	←	~~Low pressure outside the primary system~~ (Note 7)				
Leak/break of an IPS barrier	←	~~IPS overpressure/overtemperature events~~ (Note 8)				
	←	~~IPS break/ leak as initiating event~~ (Note 9)				
Leak/break of the beam tube (Note 10; Table 9/Note 6)						
Leak/break at SCS interface (Note 11)	←	PHX tube ruptures (see Table 9/Note 9)				
	←	~~PHX ruptures~~				

Note 1: Leak/break of the vessel.

Breaks/leakages of the structural surfaces of the RVB parts can occur as initiating events (IE) due to an unanticipated degradation of the barrier or can occur due to an overload on the barrier beyond its design limits caused by other initiating events not directly affecting the barrier.

The following mechanisms for overstress of the structural parts are identified: high pressure inside the primary system (Note 3), high temperature in the primary system (Note 5), high pressure outside the primary system (Note 6) and low pressure outside the primary system (Note 7).
The safety vessel is designed to mitigate the consequences of leak/breaks of the vessel. If the safety vessel is to be regarded as a SLOD, the probability of leak or break should be lower than DBC4. This is accomplished by the qualification of the material of the vessel for compatibility with LBE at the operational temperatures, by the choice of the appropriate QA for design and construction and by an appropriate in service inspection and/or monitoring (leak before break) program for the vessel wall. If a leak of the vessel is more probable than DBC4, the safety vessel needs to be stronger than an SLOD. Since the safety vessel in normal operation is not subjected to any degradation mechanisms, it might not be too difficult to attain the required reliability for an XLOD.
At present we still have difficulties to calculate the transient phase for very large breaks of the vessel. If these difficulties persist or if such breaks really present a safety problem, they will have to be prevented. Possibilities to achieve the required reliability is to increase the safety class of the reactor vessel, examine a leak before break approach or exclude large breaks of the vessel by an appropriate mechanical design of the safety vessel.

Note 2: Leaks/breaks as initiating events.
The commonly considered degradation leading to leaks or cracks in metallic structural barriers is fatigue growth of existing cracks. For nuclear safety class equipment, such degradation is commonly attributed a DBC4 frequency of occurrence. Another potential mechanism is corrosion. It is the purpose of the LBE conditioning and the reactor vessel inspection program to reduce probability of failure by corrosion to a DBC4 frequency.

Note 3: High pressure inside the primary system.
The normal operation conditions for the RVB are well defined and determine the first choice for the design pressure. The purpose of these events is to establish the DBC2-5 design pressure requirements for the RVB and/or the overpressure protection of this barrier.
There is a lot of energy stored in the coolant of the primary system but because of the thermal and hydraulic properties of the coolant this heat cannot generate high pressure. The RVB will fail due to high coolant temperature before substantial overpressure can be created by LBE boiling.
The only fluid present in the primary envelope that can generate high pressure is water. The internal energy of the water of some sources (secondary cooling system) is sufficient to generate a dangerous overpressure in the primary system. Moreover, if the water comes into contact with LBE, its internal energy can importantly increase and a higher pressure will be reached. Notes 8, 9, 10, 11 and 12 in Section 3.2.3 on LBE as a barrier (Table 9) deal with the possible causes for water ingress in the primary system. The possible overpressure issues of the RVB after such initiating events will be treated in the safety analysis of these water ingress events.
Temperature increase of the cover gas can increase the pressure within the RVB. This mechanism is further developed in Note 6.
Compression of the cover gas by a faulty operation of the LBE level control is possible but the maximum overpressure transient that can be expected from such event will be slow and therefore easy to protect by an overpressure protection.
Inappropriate operation of the cover gas (under)pressure control could create overpressure. The cover gas conditioning system will be designed such that it cannot physically create a dangerous overpressure in the reactor vessel.

Note 4: Overpressure generated by the cover gas.
Overpressure generated by the cover gas is a much slower and weaker mechanism than overpressure created by the ingress of water in the primary system, which is already a DBC2 for the PHX tube rupture. We develop it here for completeness. In case the PHX tube rupture would be eliminated it could become the DBC2 design basis for the overpressure protection of the reactor vessel.
Overpressure by the pressure control of the cover gas can be inherently protected by choosing a control system that is not physically capable of producing dangerous overpressure.
Since there is no heat source inside the cover gas, the heating of the cover gas must come from an increase in temperature of the LBE. Since an increase of LBE temperature will compromise the RVB integrity directly long before the cover gas pressure will substantially increase in temperature and pose an overpressure threat for the RVB, we can neglect this mechanism for RVB failure.

Note 5: High temperature in the primary system.
There is so much thermal inertia in the primary system that overheating by fission power is slow and can be easily protected. The mechanisms for generating high temperatures in the primary system are loss of decay heat removal from the core, the IVFS or the Po.

Note 6: High pressure outside the primary system.
Overpressure outside the RVB is possible if an event creates an overpressure in the primary containment or in the space between the safety vessel and reactor vessel.
The largest overpressure in the containment is caused by the PHXTR. Since the PHXTR originates in the reactor vessel, it will not create a pressure difference over the vessel.
A break of a secondary cooling system outside the vessel will create a pressure difference over the vessel.
In general, it is appropriate to design the RVB for an external overpressure that corresponds to the design pressure of the primary containment.
In case of the untimely actuation of the RVACS, water could be injected in the safety vessel without extraction of the produced steam. If unprotected, such event will inevitably lead to an overpressure outside the RVB for which the RVB cannot be designed (too high temperature heat source). The design of the RVACS will be optimized to prevent untimely actuation and the mitigation will probably consist of an overpressure protection of the RVACS/RVB.

Note 7: Low pressure outside the primary system.
We did not identify a possible mechanism sequence that could cause a low pressure in the primary containment or in the safety vessel.

Note 8: IPS overpressure/overtemperature events.
Depending on the design of the IPS, there can be several mechanisms that can create overpressure/overtemperature in an IPS.
Currently for MYRRHA 1.8 IPSs with water is not part of the design, thus such events where IPS cooling water comes into contact with hot surfaces (potentially leading to overpressure) can be excluded.

Note 9: IPS break/ leak as initiating event.
A leak/break of an IPS will not directly jeopardize the coolant retention by the RVB and will not immediately release large amount of radionuclides from the LBE. The indirect consequences depend on the content of the IPS. As a general principle, the design option is to preserve in all circumstances

a physical barrier between the IPS internal parts and the core. For an empty IPS or an IPS that cannot produce mechanisms that can threaten the core an exception can be made and one barrier or no barrier might be sufficient.

Note 10: Leak/break of the beam tube.
Failure of the beam tube within the LBE will have a negligible influence on the coolant inventory. The beam window part of the beam tube is supposed to be the least reliable part of the beam tube and the consequences of a beam window failure will envelope the consequences of a failure of the beam tube elsewhere within the LBE. The beam window failure is extensively discussed in Note 6 of Table 9.

Note 11: Leaks/breaks at the interface with the SCS.
Parts of the secondary cooling systems are part of the RVB. These parts of the RVB are of particular interest because of the following features:

- The SCS penetrates the primary containment. Leak/break events on these SCS parts make a direct connection between the LBE and the SCS part outside the containment. This might require that the SCS part outside the primary containment acquires a primary containment safety function.
- LBE will enter the SCS and be exposed to the vapour that remains in the SCS after the break. This provides a mechanism to release Po in the SCS.
- Release of water in the primary system will lead to overpressure in the RVB and (if relieved) into the containment. This means two barriers are endangered by one event.

The potential threat of leaks and breaks of the SCS inside the RVB was already identified by mechanisms leading to release of activity from the LBE (see Section 3.2.3).

3. 2. 5. Primary containment related initiating events.

During normal operation, the primary containment will not contain a volatile source term that will be able to produce SO1 consequences. The under pressure of the gaseous volumes in the reactor vessel and in the LBECS system guarantee the absence of release of contaminated gas from the primary systems to the primary containment during normal operation.
Clean up and refreshment of the containment atmosphere will have to guarantee that small amounts of volatile radioisotopes leaking into the primary containment will not concentrate to an amount being able to produce SO1 consequences. This is part of the design basis for normal operation and for the clean-up and refreshment systems of the containment atmosphere.
The normal operation strategy therefore allows excluding that initiating events jeopardizing only the leak tightness of the primary containment (containment failure as initiating events) can lead to sequences with unacceptable radiological consequences for the public.
In the case of MYRRHA, the containment also has the normal operation function to avoid ingress of oxygen that can disturb the oxygen conditioning of the primary coolant. Too high oxygen concentration in the LBE could lead to the formation of lead oxides and this is a potential mechanism for full core blockage, an event for which it is the intention to exclude it by design.
Since lead oxides form at lower oxygen concentration for lower temperatures, we will consider as an enveloping event the instantaneous exposure of the LBE surface to air during operation with open reactor. This analysis is also linked with the demonstration of a long-term safe state without any cover gas and LBE conditioning.
The main safety function of the primary containment is as part of the LOD to mitigate the consequences of the failure of other barriers.

Table 12: Primary containment failure mechanisms

Failure conditions	Mechanism sequences			
Leak/break of the primary containment (Note 1)	←	Leak/~~break~~ of the primary containment as IE (Note 2)		
	←	Overpressure	←	PHX tube rupture (see Table 9/Note 9) (Note 3)
			←	Secondary system rupture (Note 4)
	←	Overtemperature (Note 5)	←	Pressurized water system breaks (see overpressure above)
			←	Loss of decay heat removal
	←	Local mechanical loads (Note 6)	←	High momentum Projectiles
			←	High energy line breaks

Note 1: break/leak of the primary containment.
We make a difference between the appearances of leaks/breaks on the primary containment as initiating events and the aggravating failure of the primary containment during other events.
The following mechanisms producing the aggravating failure of the containment are identified: overpressure, overtemperature and local mechanical overloads.

Note 2: Leak/break of the primary containment as initiating event.
During normal operation, the loads on the primary containment are low and a structural failure (break) of the SSCs of the containment is so unlikely that it can be excluded. It is possible during normal operation that cracks develop in the concrete shell and slowly degrade the leak-tightness of the containment. Such phenomenon will be slow and measures for repair or avoiding ingress of air are possible.
What cannot be excluded is a faulty operation of a penetration of the primary containment leading to a large leak (e.g., simultaneous opening of both doors of an airlock).
To envelope possible scenarios, we will consider an instantaneous complete replacement of the nitrogen in the containment by air and assess the influence on the coolant. During operation with closed reactor, the influence on the oxygen concentration at the LBE surface will be slow because of the low leakage of the RVB. Operating conditions with open reactor (for maintenance) will be the most aggravating condition to be considered. We therefore suppose a postulated event of instantaneous replacement of nitrogen air above the LBE surface as an envelope for this mechanism.

Note 3: Overpressure by PHX tube rupture.
The PHX tube rupture with closed reactor but failure of the pressure suppression system is the scenario that produces the highest pressure in the primary containment. Since this event degrades the LBE barrier and reactor vessel barrier, the containment needs to remain functional in this scenario.

Note 4: Overpressure by secondary line break.
The secondary system confines a source term that consists of activation of the secondary water by the neutron flux at the PHXs in the reactor vessel. This source term has not been quantified yet, but the intention is to make it so low that it can be released into the environment without unacceptable exposure of the public. This is necessary anyway, to allow the event of a secondary line break outside the containment.
The containment therefore shouldn't be necessary to mitigate the radiological consequences of a secondary line break.

However, with the aggravating condition of an open primary system, the break of a secondary line can lead to the release of Po from the LBE surface exposed to a high concentration of water. This event is already identified as a release mechanism for the LBE barrier: see Table 9/ Note 13.

Note 5: temperature increase in the containment.
Overpressure events in the primary containment will also generate high temperate for which the containment will have to be designed. The values will result from thermal/hydraulic analysis of the events. Today, the unprotected PHX tube rupture at nominal power with failure of pressure suppression system releases the most energy in the primary containment and determines both the design pressure and temperature of the primary containment.
Because of the 4 LODs present for fission control and for decay heat removal, we don't have to consider a situation in which the heat removal from the heat sources is lost and the fission or decay heat is fully released in the containment. Only the heat losses of the systems that contribute to decay heat removal will end up in the containment and these can be minimized by lowering the temperature of the long-term safe state.
To prevent freezing, the decay heat is removed at a relatively high temperature of 200 °C and the heat losses of the decay heat removing systems will end up in the containment. It is not clear whether in the long-term safe state a safety system is necessary to maintain the temperature in the containment below the design temperature of the systems that need to operate in the long-term safe state. This can only be evaluated when the heat losses of both the heat carrying systems and of the containment building are known. For this problem, the optimal choice is an as good as possible isolation of the decay heat removal systems (this also improves the freezing issue) and a bad isolation of the primary containment barrier (small but negative effect on long term freezing). If we fail in limiting the temperature inside the primary containment by control of heat losses, a containment heat removal system might be necessary for the long-term operation after some events.
Temperature and pressure load on the primary containment can also originate from internal fires. These events are not considered here because a different approach is followed for the initiator definition and safety demonstration for this hazard.

Note 6: Local mechanical loads.
Mechanical aggressions on the primary containment mainly come from external hazards like earthquakes and plane crashes. These are not treated here.
Projectiles of high momentum components or breaks of high energy lines (HELB) can also exert mechanic loads on the containment and are considered internal hazards. HELB is of particular importance because it can be at the same time a hazard for the containment and an internal event that requires the containment. A specific HELB analysis will be foreseen covering both aspects.

3. 2. 6. Other source terms and their confinement barriers

The barriers considered in the former paragraphs are the barriers of the major source terms composed of the fission products and the LBE activation and spallation products. Some of the barriers also confine non major source terms such as the cover gas. Apart from these source terms there are and will be other source terms in the installation. Most of them are not clearly defined at present and their confinement has not been determined either.
The only other source terms expected to be a major source term that also possesses a dangerous decay heat are the spent fuel storages. The short-term storage is done in the reactor vessel and a specific

initiating event analysis can already be developed for this source term that possesses the same barriers as the fission products in the core. Many events for the core will be applicable to the in-vessel fuel storage. This will be done in a next revision of this document. The intermediate and long-term storage are at present not designed to a sufficient (conceptual) detail to allow an initiating event analysis. For all storages, it is the intention to be able to exclude massive fuel pin disintegration sequences (affecting more than one fuel assembly) based on the safety approach followed for the core. Generation III+ pressurized water reactors follow the same goal and to our knowledge no severe accident (massive pin failure events/sequences) involving spent fuel have been postulated for mitigation.
Many source terms will relate to IPS experiments and productions. At present, two IPSs are being developed to sufficient detail to allow a safety analysis. These are the 99Mo medical isotope production and irradiation of a bundle of a few pins with minor actinide fuel. The safety analysis of these IPS will probably be enveloping and allow to set design basis requirements for the IPS in general. IPS will eventually be brought outside the primary containment and be processed in hot cells that represent a completely different 'barrier' environment. There is at present insufficient detail on the design of the hot cells to perform a safety analysis.
Another important source term might be the beam dump of the accelerator. Only some exploratory calculations have been performed that demonstrate that with the right choice of material (low Z material) the beam dump will not be a major source term and will possess such low decay heat that with an appropriate design decay heat can be removed inherently.
A very peculiar source term with respect to confinement will be the activation products in the water of the secondary systems. It is peculiar because in the present design, the secondary water is sent directly to air condensers outside without any form of radioactive confinement apart from the SCS barrier itself. The hypothesis today is that the impurities in the secondary water can be reduced to a level for which the activation source term can be released without reaching SO2, see paragraph 3.2.6.1.

3. 2. 6. 1. Secondary water source term

The confinement of the secondary water is the secondary system. Outside the reactor building (on the roof of the building), this is the only barrier that separates this source term from the environment and it is clear that a failure of the barrier outside the primary containment will have enveloping consequences. The leak/break of the secondary system inside containment also must take the impact on the LBE source-term into account, but this mechanism is already treated in Table 9.
To determine the prevention/mitigation strategy for a break/leak of the secondary system outside the containment, we need to determine the source term. The secondary water source term will depend on the material choices for the system and the purification strategy.
We are convinced that it will be possible to reduce impurities such that the source term is not capable of producing SO2 consequences. In that case a break of the system outside the primary containment needs to be a DBC4. DBC2 leaks will have to rely on detection and controlled discharge of the system, except if even the SO1 limit cannot be reached by this source term.
The hypothesis that we will be able to limit the SCS source term below SO2 is based on the similarities with the Primary/secondary system of a BWR and the much lower quantity of water that is irradiated and that can be released compared to a BWR. The SCS is must simpler and operates at a much lower temperature than the primary/secondary system of BWR such that impurities should be easier to control. The thermal neutron flux at the PHX is less than the flux in the core of a PWR.

3. 3. Current status of IEs for MYRRHA 1.8 [5]

Table 13 is an excerpt of the MYRRHA Safety Database [5] that lists all the IEs that are currently considered for the safety demonstration. It has to be noted that the term Initiating Event (IE) is used in a broad sense. In some cases, the IE identification exercise presented in the previous sections stops at the level of source term release mechanisms. In other cases, enveloping scenarios are identified that do not correspond to a specific IE but rather represent conservative bounding cases for the safety analysis. Such envelope scenarios (range of events) at the level of mechanism are defined with the intention to cover all mechanisms that cannot be practically eliminated. Goal of the safety analysis with such an envelope case is to demonstrate that the system withstands such event or that the envelope can be eliminated (the envelope event does not threaten the radiological barriers).
The Safety Database is a ‘live’ database continuously updated according to the ongoing development of safety analysis and the system design. To be able to follow the evolution of the list the database contains excluded or cancelled events (for example cancelled due to design modifications) as well, however for the current project those are not presented.

Table 13: List if initiating events for MYRRHA 1.8 [5]

Initiating Event ID	Initiating Event name	reactor mode	origin
3	Hypothetical core degradation resulting from the full core blockage	critical (subcritical)	internal event
6	Control Rod withdrawal	critical (subcritical)	internal event
10	Accelerator beam misalignment causing local overpower	subcritical	internal event
11	Beam power increase	subcritical	internal event
12	Vessel break	critical (subcritical)	internal event
13	LOOP	critical (subcritical)	internal event
15	Mechanical failure of one primary pump	critical (subcritical)	internal event
16	Blockage of one flow path outside the core region (excluding primary pump inlet/outlet pipe)	critical (subcritical)	internal event
17	Blockages at the entrance of the core	critical (subcritical)	internal event
18	Large diaphragm break	critical (subcritical)	internal event
19	Objects blocking FA entrance	critical (subcritical)	internal event
20	Blockages at the pin bundle entrance	critical (subcritical)	internal event
21	Foreign object ingress inside the pin bundle during operation	critical (subcritical)	internal event
22	Foreign object ingress inside the pin bundle during manufacturing	critical (subcritical)	internal event
23	Stochastic pin failure with fuel release	critical (subcritical)	internal event
28	Common Cause Failure of DHR1 as postulated event	critical (subcritical)	internal event
29	LBE losses outside the primary system due to LBECS failure	critical (subcritical)	internal event
30	Beam window leak/break	subcritical	internal event
34	Spurious actuation of TCS injection	critical (subcritical)	internal event
35	Secondary line break with open reactor	critical (subcritical)	internal event
38	Inappropriate LBE level control in the reactor vessel causing cover gas compression	critical (subcritical)	internal event
39	Spurious actuation of RVACS injection	critical (subcritical)	internal event
40	Leak/break of the beam tube	subcritical	internal event
41	Accelerator beam misalignment causing tube failure at cover gas level	subcritical	internal event
43	Leak of the primary containment as IE	critical (subcritical)	internal event
44	Full instantaneous release of the cover gas into the primary containment	critical (subcritical)	internal event
73	Ex-vessel mechanical failure of a Fuel Assembly	critical (subcritical)	internal event
74	Leak of LBECS bubble column reactor at gas level	critical (subcritical)	internal event
75	Load drop into open vessel	critical (subcritical)	internal event
76	Secondary line break with closed reactor	critical (subcritical)	internal event
79	PHX tube rupture by freezing due to secondary line break	critical (subcritical)	internal event
81	ACS sudden displacement as IE	critical (subcritical)	internal event
82	Operating Basis Earthquake	critical (subcritical)	external hazard
83	Design Basis Earthquake	critical (subcritical)	external hazard
84	Design Extension Condition Earthquake	critical (subcritical)	external hazard
85	Small diaphragm break	critical (subcritical)	internal event
88	IPS-related IE	critical/subcritical	internal event
92	PHX tube rupture affecting the internal tube	critical (subcritical)	internal event
93	PHX tube rupture affecting the external tube	critical (subcritical)	internal event
94	Secondary system overpressure	critical (subcritical)	internal event
95	Primary pump failure with coast-down	critical (subcritical)	internal event
96	Stochastic pin failure without fuel release	critical (subcritical)	internal event
98	Fuel Assembly wrapper failure	critical/subcritical	internal event
99	Aircraft Crash Level 1	critical/subcritical	external hazard
100	Aircraft Crash Level 2	critical/subcritical	external hazard

3. 4. Selection of Initiating Events subject to PIRT

In the framework of ANSELMUS WP1 'PIRT on HLM systems' and Task 1.2 'Reference design and initiating events' two loss of flow type events of the MYRRHA 1.8 ADS system are suggested for the PIRT exercise:

- Phase II and Phase III of a loss of forced flow (LOFF) transient,
    - namely onset of natural circulation and long-term decay heat removal,
- diaphragm break worst break scenario.

3. 4. 1. Loss of forced flow analysis for MYRRHA 1.8 [6]

In the framework of MYRRHA safety analysis loss of forced flow transient is investigated for the design revision 1.8 [6]. Loss Of Forced Flow (LOFF) accidents originating from primary pumps (PP) malfunction are MYRRHA design basis postulated events. Such events induce flow reduction in the primary system and in particular, in the core region, with consequent core temperature increase.

To cover all events related to pump malfunctioning, a conservative envelope event is developed based on the results of a Phenomena identification and ranking Table (PIRT) and a Conservative Analysis Setup Table that determine conservative input data and modelling choices to carry out a conservative safety analysis. The conservative model covers both MYRRHA operating modes (critical and subcritical).

This envelope event is analysed to demonstrate compliance with the safety objective 1 (SO1) considering the most stringent decoupling criterion, which was defined for C2 events, i.e., "no consequential clad failure of fuel pins". The Figure of Merit (FoM) monitored is the Peak Clad Temperature (PCT) such that no clad burst failure occurs. In this way, there is no need for discussing the DBC class of the initiating events.

A complete loss of primary pump internals (following a mechanical failure event) is considered as the envelope case, combined with non-locked rotor conditions for the other pump as an envelope for all possible coast down behaviour. In this way, the envelope is completely independent of the PP mechanical failure modes.

Considering a conservative reactor shutdown delay time of three seconds (3 s), it is shown that the clad failure is prevented and the SO1 is always fulfilled for any PP failure initiating event. The three seconds reactor shutdown delay time is a safety design requirement for the design basis of the reactor protection safety function for MYRRHA design version 1.8. It is a requirement for both (diverse) scram systems. This scram delay time of 3 sec certainly also covers the proton beam cut-off delay time in the subcritical operating mode which can happen in a few milliseconds.

3. 4. 1. 1. Definition of the postulated initiating event (PIE) [6]

This section concerns events that cause a homogeneous (identical loss of flow for all fuel assemblies in the core) LOFF in the core and for which the origin is located outside the core as defined in Section 3.2. The scope is limited to transients induced by the abnormal behaviour of one or both primary pumps (PPs). Other events (such as loss of flow caused by blockages or diaphragm breaks) are not discussed here.

Homogeneous LOFF are induced by failure of one or both pumps and will result in core flow reduction and core temperature increase. If not detected and protected in a sufficiently short time fuel pin failure will occur; therefore, detection is needed to protect the reactor.

Different IEs can induce PP abnormal operation. These IEs are grouped considering their impact on the PPs behaviour and consequently on the core flow. In this respect, one can group all PP failures IEs into two main categories:

- Mechanical failure events (loss of pump performances and function),

- IEs related to loss of power supply events (which induce a PP trip)

These two groups are discussed hereafter with the objective of postulating an envelope initiating event (PIE). The process of events grouping and selection of the envelope is summarized in Figure 16.

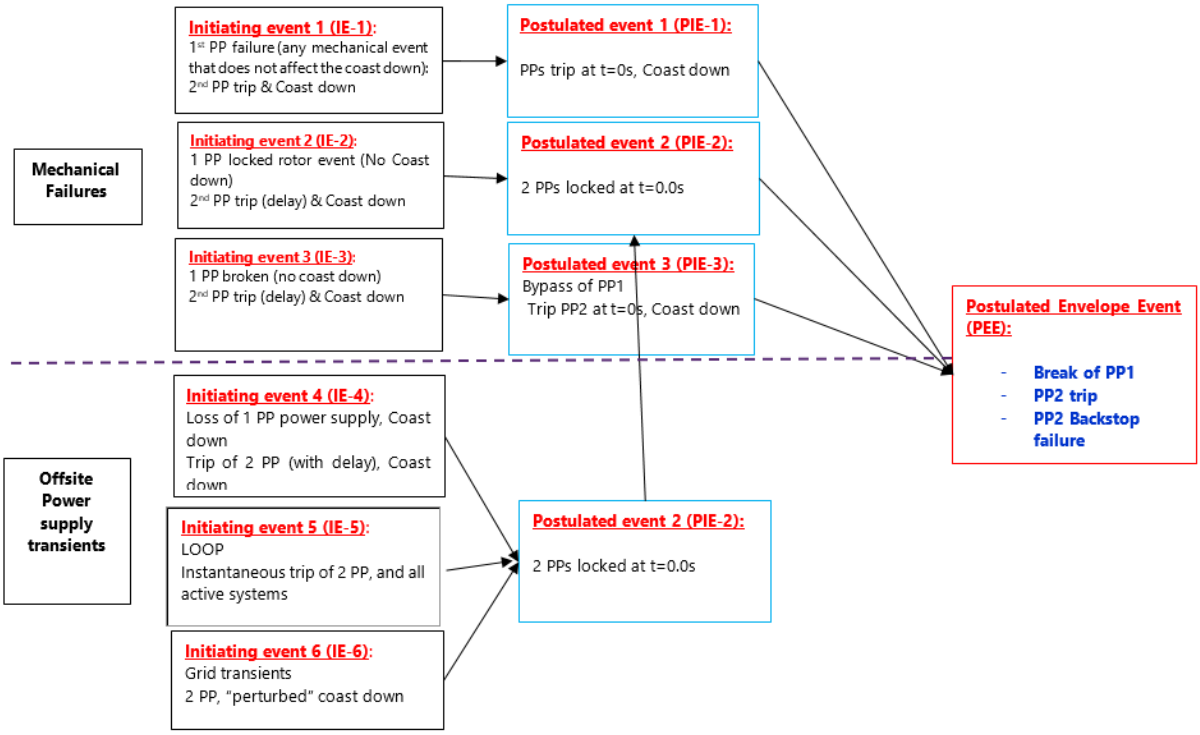

Figure 16: Initiating event for PP abnormal operation conditions and development of the envelope event [6]

According to MYRRHA safety approach for LOFF no clad failure is the decoupling criterion to demonstrate that SO1 is fulfilled. All radiological safety objectives will be covered by this approach.
The main failure mechanism of the fuel pin cladding in case of LOFF events is the cladding burst failure caused by:

- Weakening of the clad material at high temperatures,
- Increase of the pin internal pressure due to temperature rise.

Other barriers are not affected.
The conservative cladding burst criterion depends on the peak cladding temperature and pressure difference over the clad reached during the transient which means that the Figures of Merit (FoMs) for the LOFF events are the internal pin pressure and the pin peak clad temperature (PCT). In order to decouple this dual criterion of pressure and temperature, it is considered that the pin that has the highest cladding temperature or PCT (i.e., hottest pin) in the core has maximum internal pressure (i.e., the highest burn up fuel pin internal pressure). Of course, such assumption does not reflect reality because the hottest pin in the core has always the lowest burnup and therefore the lowest pin internal pressure. This assumption is considered to simplify the analysis and it is conservative.
For MYRRHA design revision 1.8 the burst limit corresponds to T = 950°C as long as no flow inversion occurs in the core. In case of flow reversal, the burst limit has to be evaluated over the course of the transient, considering temperature variation in the FA lower and upper plenums, which impacts the pin internal pressure. Table 14 summarizes the safety limit for the PCT, which is the Figure of Merit (FoM) to monitor.

Table 14: Figures of Merit and safety criteria, LOFF, MYRRHA 1.8 [6]

Figures of Merit	Safety criterion	Numerical limit	Comments
Peak cladding temperature (PCT)	PCT< burst failure criterion limit	950°C (1)	Applied to the hottest pin which we assume having the highest burnup (i.e., maximum pin internal pressure) and no flow inversion.
Peak cladding temperature (PCT)	PCT< burst failure criterion limit	variable	Evaluation of pin internal pressure and burst limit in case of flow inversion, as function of FA Lower/upper plena gas temperature.

For the determination of the typical system behaviour during a LOFF event, a simultaneous trip of both PPs (PIE-1) is postulated. To appreciate the margin to the burst limit, PIE-1 is run unprotected. It is referred to as best estimate case (BE) in the following sections, although conservative considerations were adopted to develop the model.
The following conditions are considered in the definition of the MYRRHA conservative configuration used as an envelope to study the LOFF transient:

- Maximize pin power:

The MYRRHA core must comply with the corrosion limit for normal operation (PCT≤ 400°C) [10], [11], [12] which is the same for both operating modes. The clad temperature limit defines the maximum admissible pin power. Investigation of MYRRHA design revision 1.8 cores performances showed that the subcritical mode has higher power density [13] [14]. The maximum pin power is reached for the BOL subcritical core. This maximum pin power represents the worst pin conditions for the analysis of LOFF.

- Maximize core inlet temperature:

The Secondary Cooling System (SCS) conditions (in terms of mass flow rate and temperature) are kept constants for every event and for both modes (Pressure at 16 bara), the average primary pool temperature varies with the core power level. The LBE core inlet temperature of 220 °C is therefore the maximum core inlet temperature to consider for the LOFF.
The behaviour of the secondary cooling system during the transient has no impact on the transient. Because of the large volume of the lower plenum, the inlet temperature to the core will remain constant at normal operation value during the whole transient phase (until steady, natural circulation is reached).

- Reactivity feedbacks neglected to cover both operation modes.

Reactivity feedback has a positive impact on LOFF transients. To envelope the subcritical mode of operation, in which reactivity feedbacks are minimal, no reactivity feedbacks are considered during the transient.
A sub-critical core with the hotspot from all investigated configurations is considered. In addition, the temperature reactivity feedback effects are not considered in this analysis, i.e., the power is kept at the nominal level during the course of the event.
At the occurrence of the initiating event, the two PPs are tripped simultaneously. Flow starts to decrease, and the free surface levels start to move towards a new equilibrium level. It should be noticed that the PPs coast down is very fast because of their very low inertia. After the PPs have stopped, the still remaining LBE levels difference induces flow reversal through the PPs and the PHXs to fill up the CB and ACS (because the core pressure drops are higher). As a consequence of filling up the ACS space the core pressure difference increases which induces decrease of flow through the core with potential core flow reversal. The sequence of the transient is summarized in Table 15 and development of free surface level changes and core mass flow rate are demonstrated in Figures 17-18.

Table 15: Sequence of events during ULOFF [6]

ULOFF phases	Time (s)	Models/Remarks	Phase
Postulated Initiating Event	0.0	PIE	I
Primary pumps trip/failure	0.0	Instantaneous trip of 2 PPs	I
Reactor shutdown	N/A	No scram (unprotected event)	I
End of pump coast-down	~2.0	BE Pump characteristics	I
LBE levels equilibrium	~20.0	Potential energy driven	End of I
Natural circulation	>20.0	Determined by buoyancy forces and system pressure drops.	II
Long term decay heat removal		Determined by buoyancy forces and system pressure drops.	III

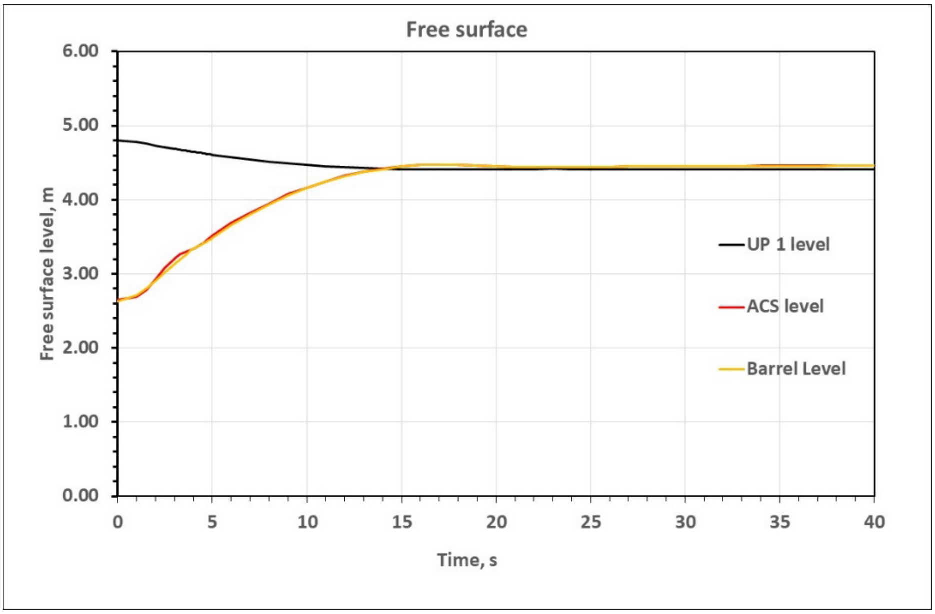

Figure 17: LBE surface levels following a postulated ULOFF event [6]

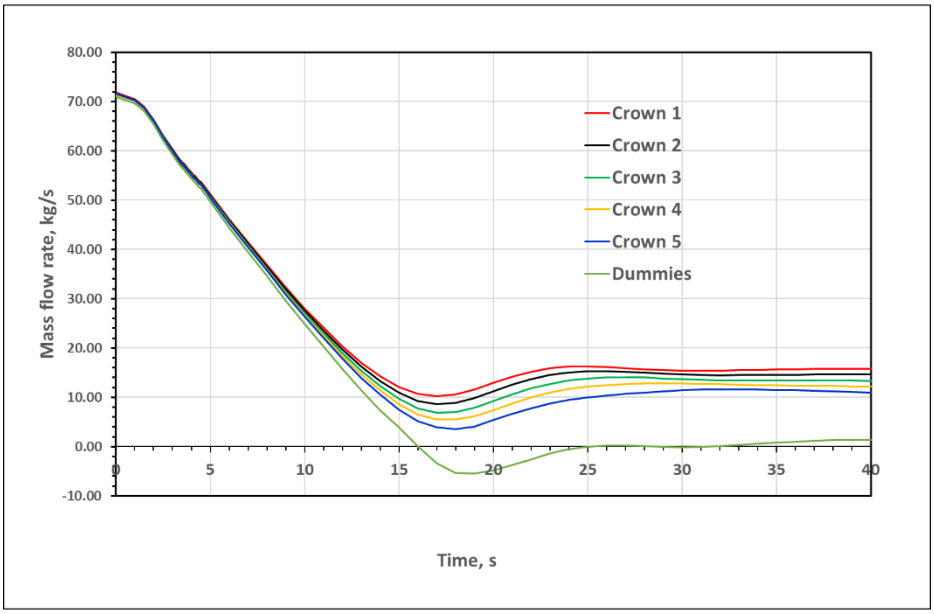

Figure 18: Mass flow rate in the active core FA and dummy assemblies [6]

3. 4. 2. Diaphragm break scenario for MYRRHA 1.8 [7]

Five postulated diaphragm breaks events were considered with breaks located at different parts of the diaphragm (see Table 16).

Table 16: List of diaphragm break locations

	Break location
1	Lower horizontal pipe, lower half/complete cylindrical surface
2	PP chimney bottom surface (PP by-pass connection)
3	PP inlet piping horizontal section
4	Upper horizontal pipe, lower half/complete cylindrical surface
5	Break on surface on core unit chimney

Comparison of exploratory STH calculations are shown in Figure 19 presenting the cases of different breaks. For certain cases flow reversal in the core can occur and also oscillations of the free surfaces can be observed for a prolonged period (depending on the break location and size). Free surface level equilibrium is reached around t = 20 s. It can also be seen that a more violent transient can be expected compared to the loss of flow without break case (indicated as LOFA in Figure 19). When natural circulation is established, flow is determined by the pressure drops in the system (including the break). Based on the preliminary calculations large break on the core barrel is proposed for the PIRT analysis.

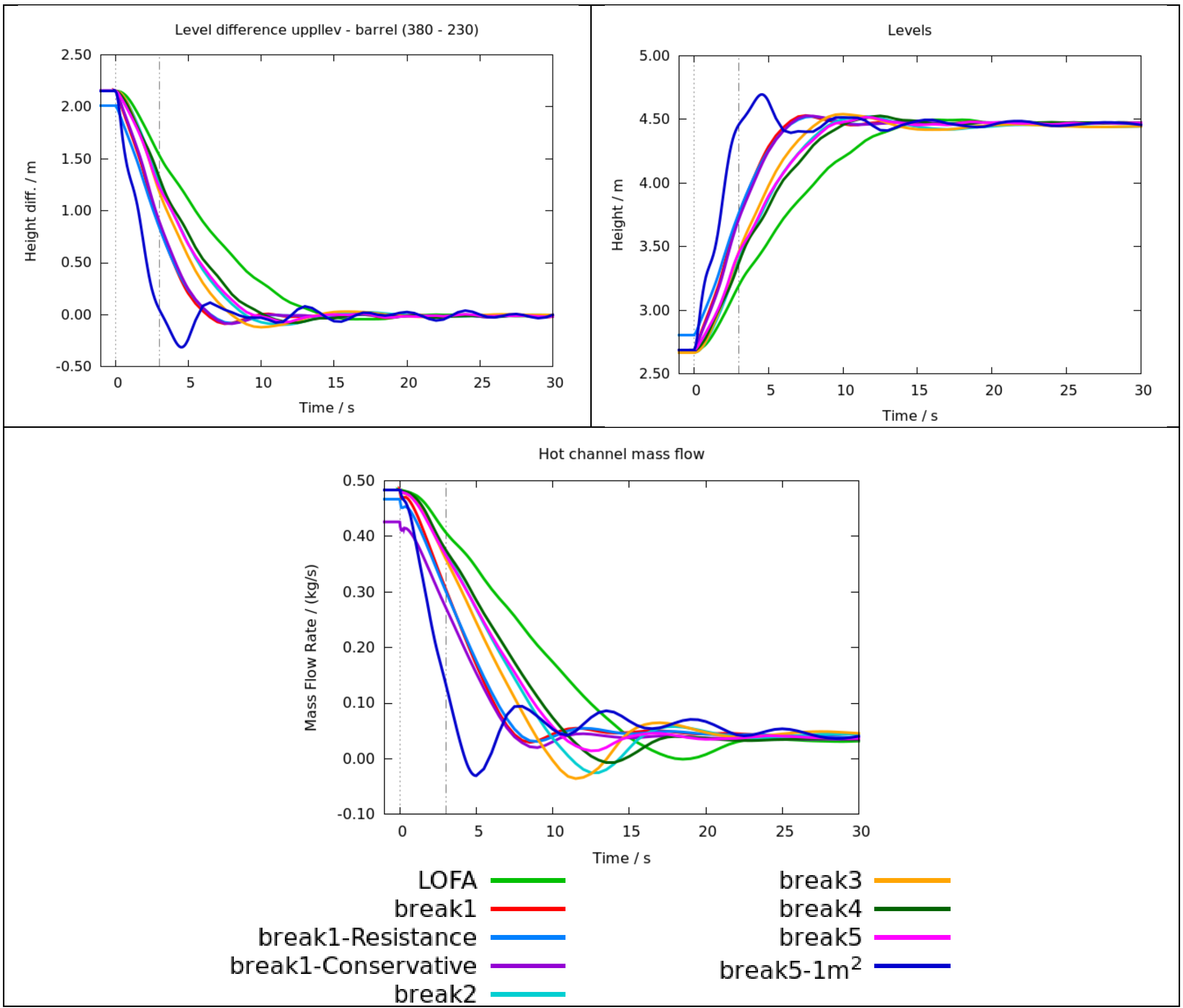

Figure 19: Comparison of different diaphragm break locations and sizes and loss of flow case without break (LOFA) [7]; top left: Free surface level difference (cold plenum - core barrel); top right: core barrel LBE free surface level; bottom: hot channel mass flow rate

4) References

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