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D1.2 ALFRED reference design and initiating events

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Document Summary

This document has the function of describing the ALFRED reactor as necessary for the performance of the Phenomena Identification and Ranking Table (PIRT). The document also rationally reports the accidental transients in object of PIRT and for which it is necessary to provide numerical supporting analyses.

Document Revision History

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Document Approval

The author, WP Leader and Coordinator acknowledge and accept delivery of the work completed for this deliverable.

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Abbreviations

ALFRED	**A**dvanced **L**ead **F**ast **R**eactor **E**uropean **D**emonstrator
BoP	**B**alance **of P**lant
DiD	**D**efence **in D**epth
HLM	**H**eavy **L**iquid **M**etal
LFR	**L**ead **F**ast **R**eactor
OPT	**O**bjective-**P**rovision **T**ree
PIRT	**P**henomena **I**dentification and **R**anking **T**echnique
RCS	**R**eactor **C**oolant **S**ystem
SGTR	**S**team **G**enerator **T**ube **R**upture
UTOP	**U**nprotected **T**ransient **O**ver **P**ower
WP	**W**ork **P**ackage

1 Executive Summary

Within the Generation IV advanced reactors sector, one of the technologies that has found the greatest interest in the scientific community in Europe is liquid Lead-Cooled Fast Reactor (LFR).

In support of R&D activities of LFRs, EURATOM funded the ANSELMUS project, which aims to contribute significantly to the LFR safety assessments. In particular, the research activity focuses on ALFRED, whose design was conceived by Ansaldo Nucleare S.p.A., and MYRRHA designed by SCK-CEN.

One of the main targets of the ANSELMUS project is to perform a "safety evaluation of the designs" which in an initial phase concerning the Work Package 1, is realised through a PIRT on ALFRED and MYRRHA relevant scenarios. To support the initiative, the present document described ALFRED reference design. In particular, the technical specifications of the main components that make up the Reactor Cooling System are reported and deepened, including:

- Core
- Sub-assemblies
- Inner Vessel
- Reactor Coolant Pump
- Steam Generator
- Internal Structure
- Reactor Vessel
- Safety Vessel

The description accounts ALFRED intended operation strategy through the staged approach, and to account for bounding conditions and design development the components are presented at different reactor stages (as example, core design refers to stage 3 while Steam Generator design refers to the operating conditions of stage 2). In addition, the starting point of the PIRT is based on selected initiating events of accidental scenarios, therefore bounding initiating events of relevant interest are identified and described. The outcome provides for 5 enveloping cases:

- Event 1.1.1.1: Inadvertent Control Rod assembly withdrawal
- Event 1.1.2.1: SG tube rupture
- Event 1.1.3: Core compaction
- Event 2.1.3.2: Primary coolant flow blockage
- Event 2.1.3.3: Internal Structure failure
- Event 3.1.1.1: Steam system piping break

Finally, the identification of the initiating events takes place through the so-called OPT methodology which is described in this document.

2 References

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[5] The Risk and Safety Working Group (RSWG) of the Generation IV International, «Basis for the Safety Approach for Design & Assessment of Generation IV Nuclear Systems,» November 2008.

[6] The Risk and Safety Working Group of the Generation IV International Forum, «An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems,» April 2008.

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3 Background and purpose

The present report is issued in the framework of ANSELMUS project which has as its fundamental objective to support the deployment of HLM-cooled advanced reactors. In a first phase, ANSELMUS project focuses on the “safety evaluation of the designs” which is realized by performing a PIRT on ALFRED and MYRRHA.

In general, PIRT is an evaluation method widely used in R&D activities considering a general environment where a process occurs. The process is segmented into phases and for each phase the occurring phenomena are qualitatively ranked depending on importance and knowledge. In the framework of ANSELMUS project, the environment is ALFRED reactor and the process is an accidental event.

PIRT outcomes are exploited to draw up an informed list of physical phenomena of interest having a combination of low level of knowledge or high level of importance to trace the main requirements for an R&D roadmap. As main input data to kick-start the PIRT, it is necessary to provide a comprehensive technical description of the reference reactor as well as of the accidental events object of the PIRT. It should be noticed that the development of a PIRT is a collective iterative process potentially requiring supporting numerical calculations and sensitivities to determine the degree of impact of a given phenomenon in case it is unknown. This process allows to prioritize essential R&D activities that could be later used to validate calculation codes for safety analysis in support of ALFRED licensing.

This report refers to the Work Package 1 – task 1.2 which constitutes the starting point for ALFRED PIRT and provides ALFRED reference design together with selected initiating events for Design Basis Accidents and Design Extended Conditions. For each initiating event, the evolution of the accidental sequence is qualitatively described; a divided into sub-phases is also proposed.

The content of this report is an interface input for all the works of WP 1.

4 ALFRED reference design

ALFRED is a liquid lead-cooled pool-type reactor operating with a fast neutron spectrum and intended as European demonstrator of LFR technology. To de-risk the lack of operating experience and validation (mainly regarding material qualification) a staged approach is adopted for ALFRED operation, i.e., the RCS is used as test bench for proving innovative solutions while operating at reduced power, reduced temperature and increased safety margins. Operating conditions are enhanced along reactor lifetime with increasing operator experience and qualification data. Table 1 reports the main operating conditions of ALFRED in the four anticipated stages [1]:

	Stage 0 Commissioning	Stage 1 Low temperature	Stage 2 Medium temperature	Stage 3 SMR prototype
Core inlet temperature (°C)	390	390	400	400
Core outlet temperature (°C)	390	430	480	520
Core thermal power (MW)	0	100	200	300
Live steam pressure (bar)	/	170	175	180
Live steam temperature (°C)	/	420	435	450

Table 1 ALFRED staged approach

4.1 ALFRED reactor coolant system design

As shown in Figure 1 the main components which constitute ALFRED RCS are the following [1]:

- Core
- Sub-assemblies
- Inner Vessel
- Reactor Coolant Pump
- Steam Generator
- Internal Structure
- Reactor Vessel
- Safety Vessel

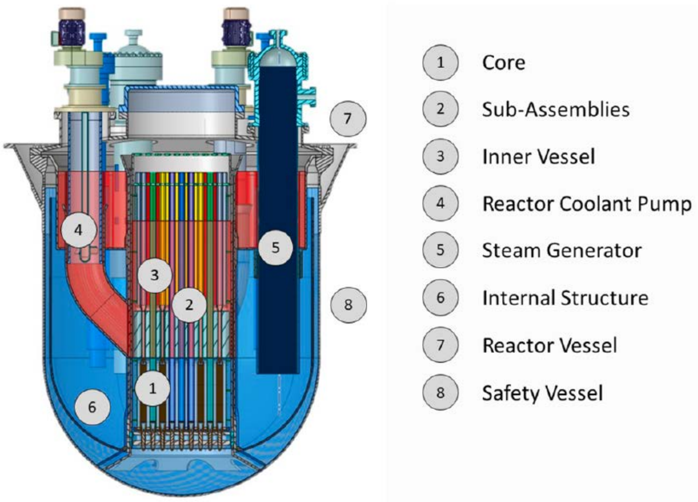

Figure 1 ALFRED RCS sketch

The areas in Figure 1 where the coolant is at high temperature (core outlet) are shown in red while the areas where the coolant is at low temperature (core inlet) are shown in blue. Figure 2 shows the details of the Internal Structure (IS) that guides the motion of the fluid.

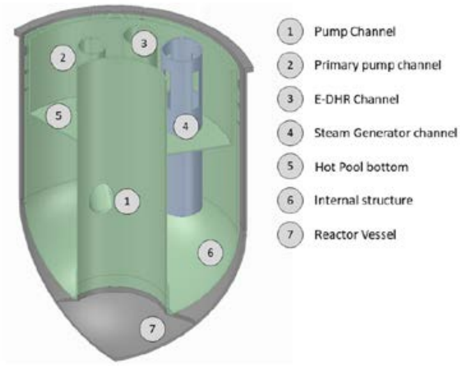

Figure 2 Cross section of the ALFRED RCS

Figure 2 shows one third of the RCS being it symmetrical in the azimuthal direction (every 120°). It is noted that at the outlet of the core the coolant is extracted from the Inner Vessel (IV) by means of the Reactor Coolant Pump (RCP) delivering the fluid to an upper region called the hot pool. The lead goes to the Steam Generator (SG) to release thermal power to the BoP. Liquid lead cools down at the core average inlet temperature and goes towards to the second plenum named cold pool. Hot pool and cold pool are separated by the IS preventing lead from returning directly to the core and forcing it to rise vertically and pass through a set of windows in the upper are at the level of the free surface, before descending in the annular gap between the IS and the RV. Finally, lead crosses the IV radial support and enters the core.
It should be noted that the typical pool-type arrangement of an RCS is subject to three main thermo-hydraulic challenges [2]:

- Thermal stratification at the top of the pool
- Steam entrainment following a Steam Generator Tube Rupture (SGTR)
- Risk of lead-freezing in accidental scenarios

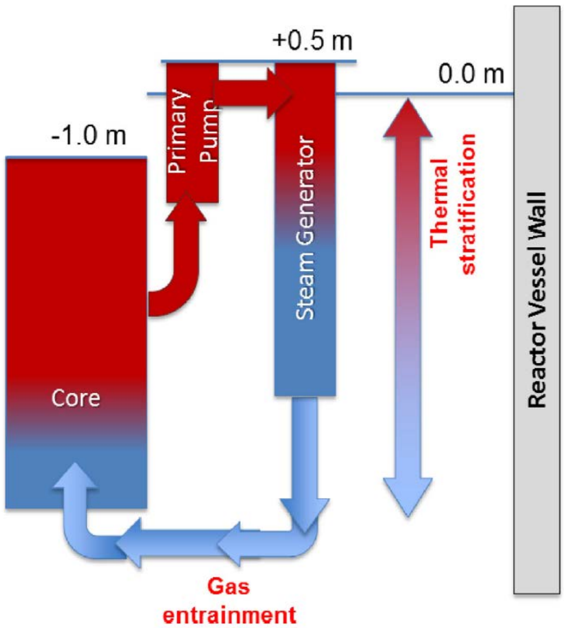

Figure 3 Lead flow path without IS

As regards the first challenge, without the IS (see Figure 3) the upper part of the cold pool is not involved directly in the coolant circulation and progressively heats up by conduction from the hotter regions. Therefore, thermal stratification occurs along the height of the RV and the thermal gradient corresponds approximately to that across the core. As a result of the coolant thermal stratification, the RV wall would be exposed to progressively higher temperature, thereby being affected by creep effect.

Concerning the second challenge, a possible SGTR accident results in direct injection of steam into the lead. Since the water comes from a high-pressure environment compared to the liquid lead system (about 180 bar compared to about 5 bar of lead column), a mass transfer of water to the lead occurs. During outflow, flash evaporation occurs and two-phase mixture/steam is subjected to heat transfer in direct contact with lead, causing the pressurization of the RV. As water interacts with lead entrainment can occur causing water bubbles to enter the core providing positive void reactivity coefficient. Such event is prevented in ALFRED due to IS arrangement.

Finally, the third challenge concerns coolant freezing risk, being lead a high melting temperature element (i.e., 327 °C). Such risk has to be managed in conjunction with decay heat removal function to be performed having the coolant in a liquid state. This challenge has been solved through a particular "self-regulating" decay heat removal system that aligns the thermal power released to the ultimate heat sink to the decay power which decreases in time (see Chap. 4.6).

As far as the former and the latter challenges are concerned, a dedicated configuration of the RCS (depicted in Figure 4) has been conceived that provides the installation of IS separating hot and cold pools that modifies lead flow-path forcing the flow at the outlet of the SG to move upwards towards the cold pool free level.

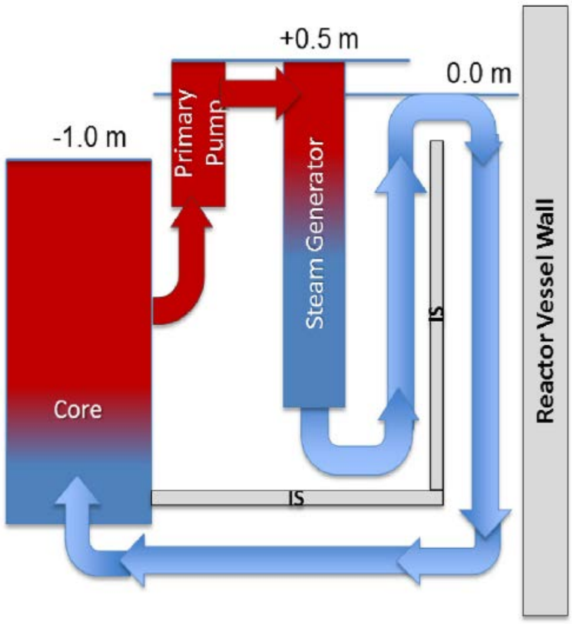

Figure 4 Lead flow path with IS

Therefore, the main outcomes of ALFRED flow-path are:

- No coolant stagnation as main cause of the thermal stratification,
- No water entrainment and transport towards the core.

4.2 Reactor Vessel

The Reactor Vessel (RV) shown in Figure 5 has a cylindrical shell and a hemispherical bottom head. The upper portion connects through a “Y” junction to the reactor support ring by a conical skirt that sustain the whole weight of the RCS, including the reactor cover, on which all internal components are connected through standard circular flanges. The reactor cover is bolted to the RV flange and sealed (e.g. through a metal O-ring). A cone frustum is welded to the bottom head with the function of radial restraint for the internal structure. AISI 316 LN (or AISI 316 L) is considered as preferred materials since offering a good compatibility with the coolant at RV design temperature and pressure and at controlled oxygen in lead.

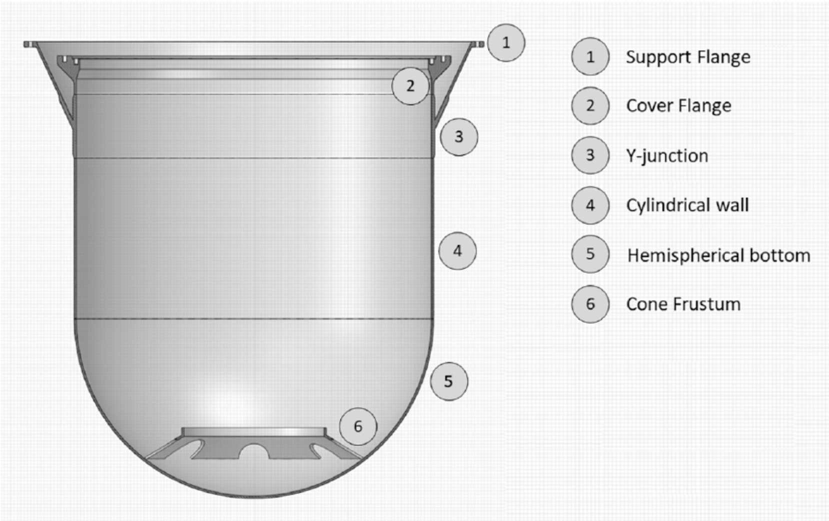

Figure 5 ALFRED Reactor Vessel

The cylindrical shell has a uniform thickness of 50 mm with a localized thickness increment in the triple point zone of the Y junction. Moreover, the cylindrical part and bottom head are conceived to be made of formed plates joined by full-penetration welds. No nozzles are foreseen below the lead-free level (therefore any penetration is through the reactor cover). Table 2 summarizes the main geometrical parameters of the RV [1]:

Parameter	Value	Unit
Inner diameter	8300	mm
Height	10000	mm
Vessel thickness	50	mm
Vessel Y joint thickness	85	mm
Skirt Y joint thickness	50	mm
Skirt straight joint thickness	85	mm
Cover and support flange thickness	200	mm
Material	316LN	-
Corrosion protection measures	Oxygen control	-
Design Code	ASME III Div. 1	-

Table 2 Reactor Vessel main data

4.3 Core Design

The core is housed within the IV, whose functional requirement is to support and restraint the core, maintaining its geometry for criticality purposes. The pin is made of a stack (810 mm tall) of fuel pellets. As shown in Figure 6, fuel pins are hollowed to decrease the thermal gradient and mitigate the maximum fuel temperature while accommodating the fuel swelling expected at the design burnup of 100 MWd/kgHM.

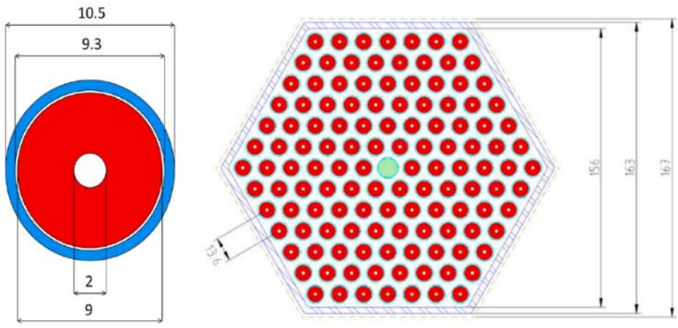

Figure 6 Fuel assembly and fuel pin geometry

126 fuel pins are arranged in a hexagonal AIM1 wrapper to make up the FA, around a central position hosting a dummy pin for additional in-core instrumentation, to get as much as possible experimental information for supporting validation of numerical tools and performance calculation, as well as reduce the local radial peaking factor. The FAs are characterized by elements as depicted in Figure 7 [3]:

- The spike, which allows the correct positioning of the FAs and flow distribution
- The funnel allowing the exit of lead and accommodating the necessary instrumentation such as thermocouples and neutron detectors
- The ballast to prevent FAs from floating in liquid lead (since the average density of FAs is lower than lead)
- The upper shroud allowing the FA to be manipulated outside lead.

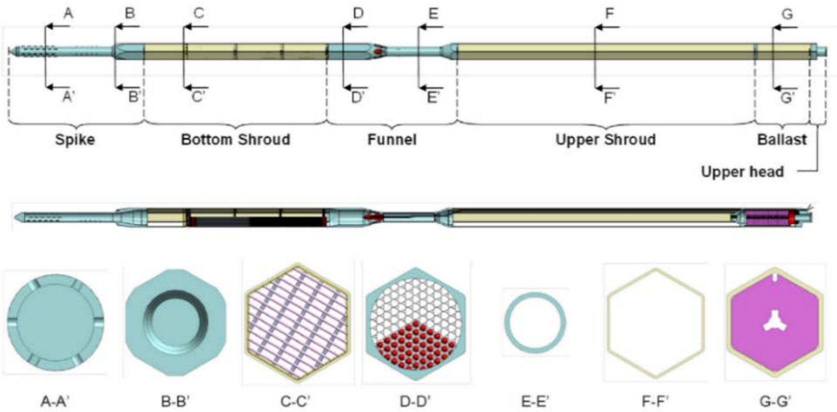

Figure 7 Axial sketch of the Fuel Assembly

A helium gap separates the pellets from the AIM1 cladding (a swelling-optimized variant of the 15–15 Ti class austenitic steels) designed to withstand the irradiation damaging associated to the design burnup and a maximum local temperature of 600 °C (hot spot in stage 3). The chosen nuclear fuel is MOX. The core of the reactor comprises of these regions with different concentrations by weight of Pu as shown in Figure 8: 20.5 wt% in the inner region, made of 56 FAs, and 26.2 wt% in the outer region with 78 FAs. The chosen concentrations and zoning ensure operability in a 5-batches strategy. The central position of the core is not occupied by fuel but has the function of material qualification by test under neutronic irradiation. The core includes also 102 dummy assemblies disposed into two external layers. The former works as a reflector in order to maximize neutron flux. The latter works as a shield in order to minimize radiation damage of structural materials and to limit it below 2 dpa for their design lifetime. Reactivity control is entrusted to two diverse and redundant neutron absorbers. There are a set of 12 control rods (CRs) placed in the outer active region and 4 safety devices (SDs) in the inner zone. The CRs enter from the bottom of the core controlled by an electromagnetic switch for passive exploiting buoyancy force. Vice versa SDs enter from the top thanks to their ballast. The considered absorber material is B_4C (with 90 at. $\%^{10}B$)

Due to the corrosion of lead at high operating temperatures, under certain operating conditions (i.e. after stage 1) the fuel clads and mechanical structures require coating corrosion protection means through Al_2O_3 coating, to be made through pulsed laser deposition (PLD) or equivalent process. Moreover, it is required to adopt coating protection on every surface of the core due to the higher operating temperatures, not compatible with protective measures based on oxygen concentration and passivation of steel surfaces.

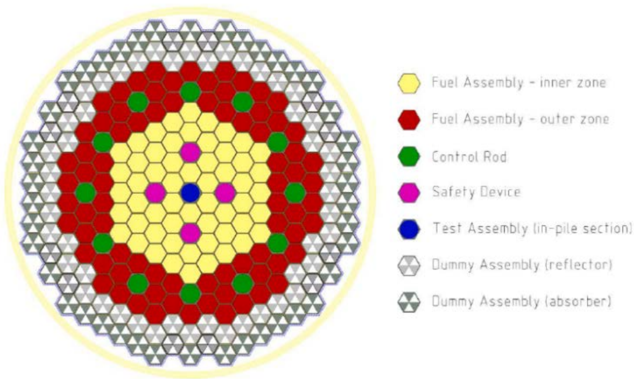

Figure 8 Core layout

4.4 Steam Generator

The system transferring the reactor power to the conventional island is the Steam Generator (SG). The SG is connected to the hot pool by means of a hydraulic or mechanical seal to permit its extraction for maintenance and replacement. The head is made up of two main chambers: on the top there is a hemispherical cap that encloses the feedwater header bolted on the steam header.

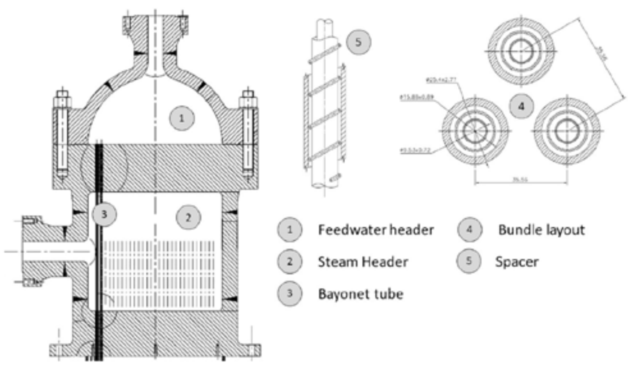

Figure 9 Steam Generator details

As depicted in Figure 9, the SG is of the bayonet type immersed in the primary pool. With a system of coaxial tubes, the feed water coming from the conventional side descends downwards (entering the first hollow tube, which crosses the steam header) and arriving at the bottom (the third tube is closed at its end by welding a small hemispherical bottom) rises to the annular region formed by the concentric second and third tubes. Going up the water removes thermal power from the liquid lead in a pure counter-current configuration. Previous configurations of SGs foresee a double external wall to reduce the probability of interaction between water and lead in the event of SGTR [1] also protecting to the probability of transporting bubbles to the core. As simultaneous break of both barriers of the double wall tube cannot be excluded (even though at a very low probability), the same function is entrusted to the IS baffle which forces the lead to rise up to the free level before returning to the core, thereby ensuring to purge water or steam in the cover gas. In the meanwhile, the elimination of the double walled tubes ensures greater thermal performance of the SG. Table 3 summarizes Steam Generator performance [1]:

Parameter	Value	Unit
N. of SG	3	-
Duty (stage 2)	67	MW_{th}
N° of tubes	880	-
Active length	6	m
Shell inner diameter	1.15	m
Primary side temperature (stage 2)	400/480	°C
Primary side mass flow rate	5615.2	kg/s
Secondary side temperatures (stage 2)	335/435	°C
Secondary side mass flow rate	44.3	kg/s
Live steam pressure (stage 2)	175	bar
Fouling plugging margin	10	%
Design temperature	550/500	°C
Design pressure (primary/secondary)	12.5/200	bar
Design code	ASME Sect III Div. I	-
Material (stages 1 and 2)	AISI 316L	-
Lead protection mean (stage 1)	Oxygen Control	-
Lead protection mean (stage 2)	Oxygen Control	-
Lead protection mean (stage 3)	AL_2O_3 coating or AFA steel	-

Table 3 Steam Generator main data

4.5 Reactor Coolant Pump

The Reactor Coolant Pump (RCP) is hosted in a duct of the IS connecting the core outlet plenum with the hot pool. The RCP does not suffer thermal gradients and it is not subjected to neutron irradiation during normal operation. Moreover, the installation site of the RCP guarantees greater simplicity of extraction in case maintenance operations are required that will be more frequent than the other components since the RCP is subjected to severe operation conditions. Figure 10 shows a lateral view of the RCP.

Figure 10 Lateral view of the RCP

Finally, Table 4 shows the RCP main data [1]:

Parameter	Value	Unit
Nominal flow rate	1908	m^3/h
Nominal head	1.5	m
Specific speed	2.97	-
Minimum/maximum flow rate	900/1980	m^3/h
Pump type	Axial/Mixed flow	-
Total shaft length	5	m
Lead velocity (max/bulk)	10/2	m/s
Bulk material	AISI 316L or AISI 321H	-
Lead protection mean	Alluminization by diffusion coating	-
Electrical supply	380 V/ 50 Hz	-
Rotational speed	289	rpm
Hydraulic efficiency	73	%
Power supply	108.15	kW
Motor power	200	kW

Table 4 RCP main data

4.6 Passive Decay Heat Removal System

A significant challenge concerning heavy liquid metal cooled reactors is to prevent the coolant from freezing (lead solidification temperature is 327 °C). This problem is even more marked in accidental scenarios when it is necessary to remove the decay heat with passive safety systems and having the decay heat a decreasing trend over time there is a risk that the thermal power absorbed by the coolant is lower than the one rejected to the ultimate heat sink. Therefore, it is necessary to include a function to delay reactor coolant freezing, for example by conceiving a particular type of "self-regulating" decay heat removal system, that is, that the thermal power which removes is aligned with the decay heat. The operations implemented by the passive decay heat removal system will be described both in stand-by and in operation. When the reactor is in full power, the safety system is on stand-by since it does not need to remove heat. In the event of an initiating event that causes an accidental sequence, the reactor goes into SCRAM and the passive safety system intervenes. It is important to mention some components that are an integral part of the Balance of Plant (BoP) that intervene both in normal operation and in emergency sequence [4]:

- feedwater isolation valve
- feedwater line
- bayonet tube Steam Generator
- steam line
- steam line isolation valve
- safety relief valve

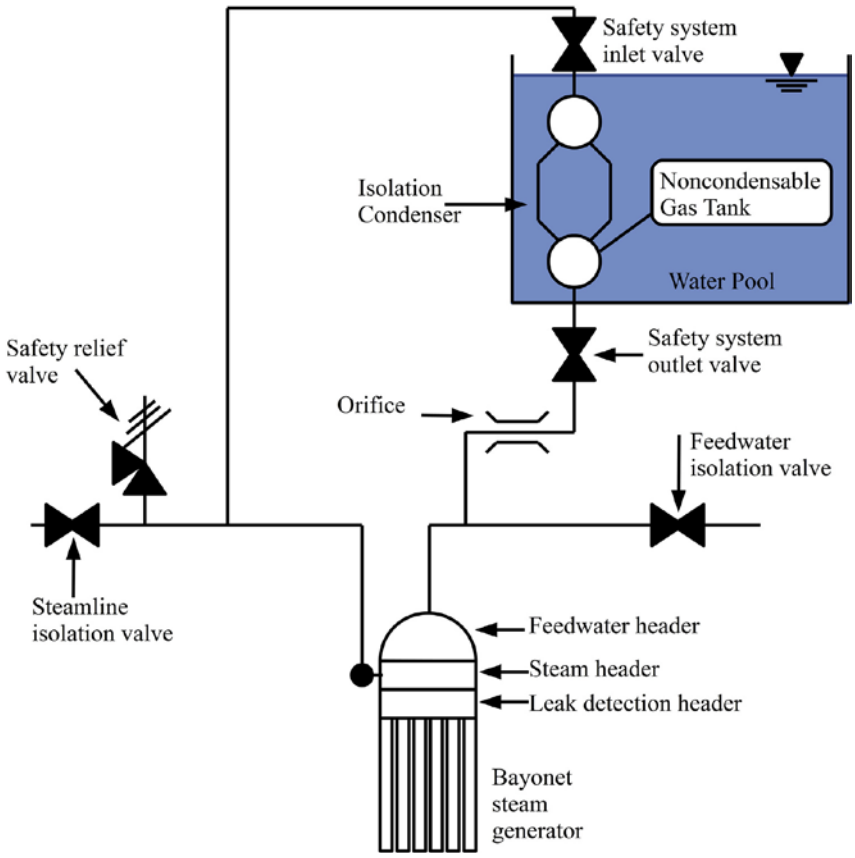

Figure 11 DHR conceptual design

The passive safety system includes:

- safety system inlet and outlet valves
- isolation condenser (IC)
- non-condensable gas tank (nitrogen is chosen as non-condensable gas)
- water pool
- orifice

The inlet of the safety system is connected to the steam-line, whereas its outlet is connected to the feedwater line. The IC, where the heat is transferred to the final heat sink (the water pool), is a shell and tube heat exchanger with a tube bundle of 16 pipes and spherical headers. The IC reference data (stage 2) are reported in Table 5 [4]:

Parameter	Value	Unit
N° of tubes	16	-
Inner diameter	32.1	mm
Outer diameter	38.1	mm
Length	2000	mm
Headers outer diameter	560	mm
Material	Inconel 600	-

Table 5 Isolation Condenser reference data

During the stand-by phase of the safety system (that is reactor normal operation) the valves are in the following positions:

- feedwater and steam-line valves are open
- safety relief valves are closed
- safety system inlet and outlet valves are closed

The fluid coming from the conventional island flows through the feed water line, enters the SG at a temperature of 335 °C and a pressure of 180 bar. Subsequently, after removing thermal power from the liquid lead, the fluid exits the steam generator as superheated steam at a temperature of 450 °C. The steam is subsequently directed towards the turbine through the steam-line. During the stand-by phase nitrogen at the chosen initial pressure and in thermal equilibrium with the water pool fills the control volume between the safety system inlet and outlet valves. The water pool is at ambient conditions. If an initiating event inhibits the normal heat removal from the primary coolant, the state shifts from stand-by to the operation phase and the feedwater and steam-line isolation valves close. Since the trapped fluid is only absorbing heat, it begins to boil and therefore pressure increases. When the pressure reaches about 190 bar the safety system inlet valve opens and the steam vents in the IC. In the first phase of the transient, the pressure of the fluid is greater than the pressure of the non-condensable gases. Then the fluid pushes the nitrogen into the tank of the non-condensable gases (which is connected in the lower head of the IC). Meanwhile, the steam passing through the IC condenses to form a water level. Then the IC downstream valve is opened, and the condensate enters the SG by gravity and natural circulation is established. The natural circulation loop includes the SG where the fluid removes the decay heat from the lead, exits the SG and condenses in the IC. Subsequently, decay heat decreases in time, and the power removed by the IC starts to be greater than the power provided by the SG with consequent depressurization of the system. As pressure reduces, non-condensable gases begin to expand and migrate toward the IC. The non-condensable gases degrade the heat exchange and therefore cause the decrease in the thermal power removed by the IC and the decrease in the depressurization rate of the system.

5 Identification of Initiating Events

One of the challenges facing the development of innovative reactors is the applicability of current codes and standards. The approach to safety is based on recommendation of the Generation IV International Forum (GIF) Risk and Safety Working Group (RSWG) [5]. Of course, the new specific goals of Gen IV nuclear systems are logically following the fundamental safety objectives specified by IAEA that are currently applied to existing reactors. For LFR safety, the GIF RSWG has presented objectives, principles, attributes and tools intended to immediately provide designers of Gen IV systems with concepts and methods that can help their R&D activities in a way that promotes the safety basis and efficient licensing of advanced nuclear technologies.

Risk informed approach (i.e., deterministic and probabilistic analyses) is also suggested by RSWG and other methods defined in the Integrated Safety Assessment Methodology (ISAM) [6] to support the design process in the very early phase. The ISAM consists of five distinct analytical tools listed below:

- **Qualitative Safety Features Review**: a list of recommendations and good practices allowing to verify the design according to different regulators (IAEA, National regulators…)
- **Phenomena Identification and Ranking Table:** the identification of safety-related phenomena and the evaluation of their degree of importance and state of the knowledge for each phenomenon
- **Objective Provision Tree**: for each level of DiD it allows to identify all measures that contribute to the achievement of safety functions as well as their mutual interrelations
- **Deterministic Safety Analysis**: traditional DSAs in which analytical evaluations of the physical phenomena occurring in nuclear plants are performed to demonstrate that acceptance criteria are met
- **Probabilistic Safety Analysis**: according to ISAM, PSA can be applied only to a design that has reached sufficient maturity and detail. Thus, PSA is to be performed, and iterated, beginning in the late pre-conceptual design phase and continuing through the final design stages addressing licensing and regulatory concerns

5.1 The Objective-Provision Tree

The OPT is a practical method used in safety analysis consistently with DiD philosophy. The OPT allows a systematic identification of the provisions which participate to the safety achievement. The OPT is a tool that can help in an iterative approach of definition and design of the safety architecture. As shown in Figure 12, in a hierarchical structure relationship in the form of a tree, the main steps are resumed below [7]:

- **Safety function**: the definition of what is necessary in order to ensure safety (e.g., reactivity control, heat removal control, containment integrity, radiation protection…)
- **Challenge**: it represents the potential issue caused by a safety function fail which in turn can cause the safety objective to fail (e.g., insertion of positive reactivity, abnormal mechanical/thermal stresses on first barrier…)
- **Mechanism**: the initiating event that causes the challenge (withdrawal of a control rod...)
- **Provision**: once identified the initiating events, the provisions with which the challenge is avoided/mitigated must be specified for each event (limiting removal device and associated I&C). The set of provisions implemented for the safety achievement is known as Line of Protection (LOP)

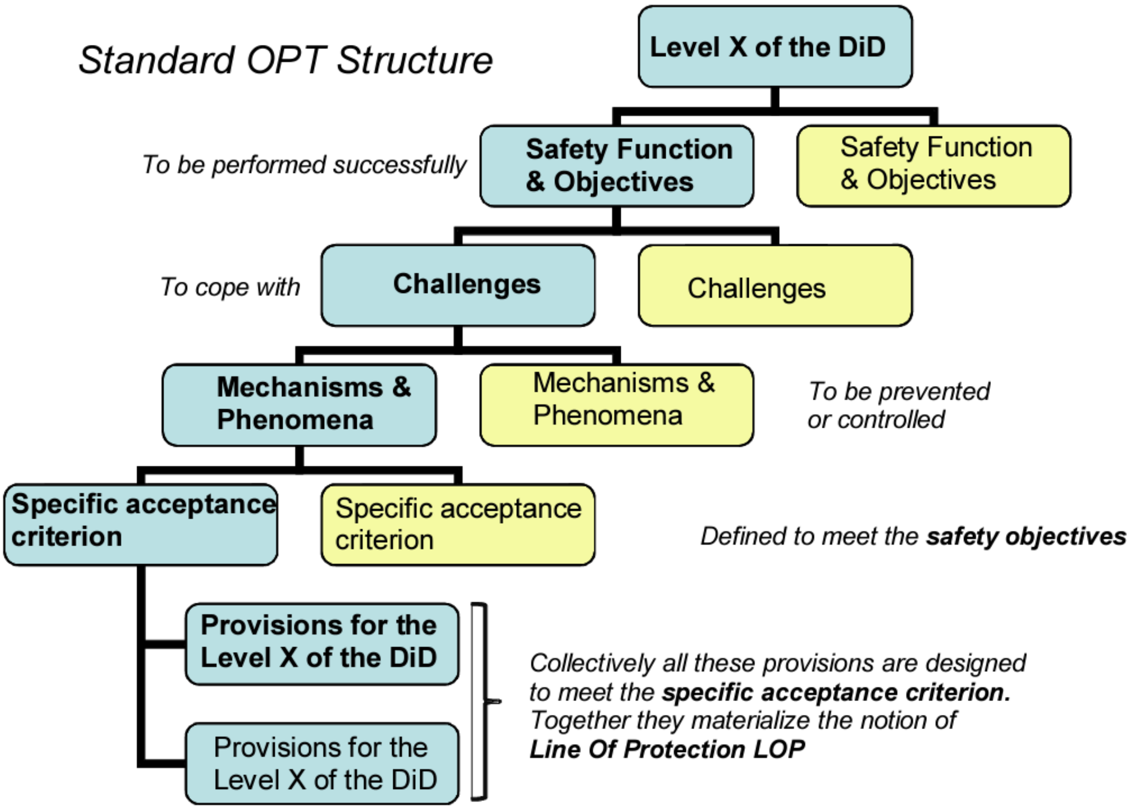

Figure 12 OPT standard structure

The main advantages of the OPT method are the following:

- It is applicable to those designs which are at preliminary stage (e.g., LFR)
- It allows to identify the sets of measures needed to ensure safety
- It allows to select the best design option in case in which several sets of provisions can provide similar degree of safety
- The graphical representation can be used as an input of PSAs models

5.2 Application to ALFRED design

The identification process of ALFRED initiating events is based on the challenges affecting the three physical barriers separating radioactive material from the external environment:

- Fuel cladding challenges
- RCS boundary challenges
- Containment challenges

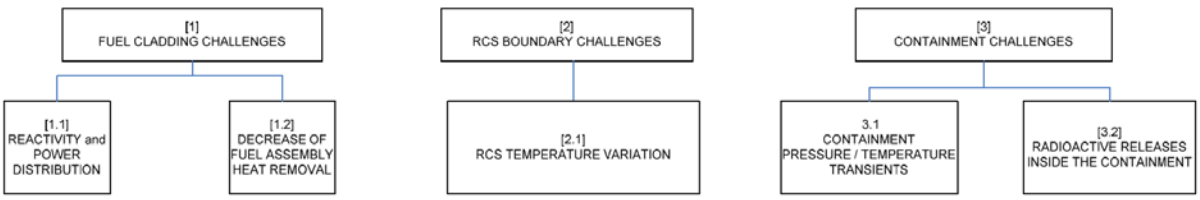

Figure 13 MLD development

To support the process of identifying initiating events, the use of Master Logic Diagrams (MLDs) allows to highlight the root cause of a challenge, being each event specified at an N level is the cause of the event reported in level N-1.

5.2.1 Fuel cladding challenges

Figure 14 shows the initiating events which affect the barrier "Fuel cladding".

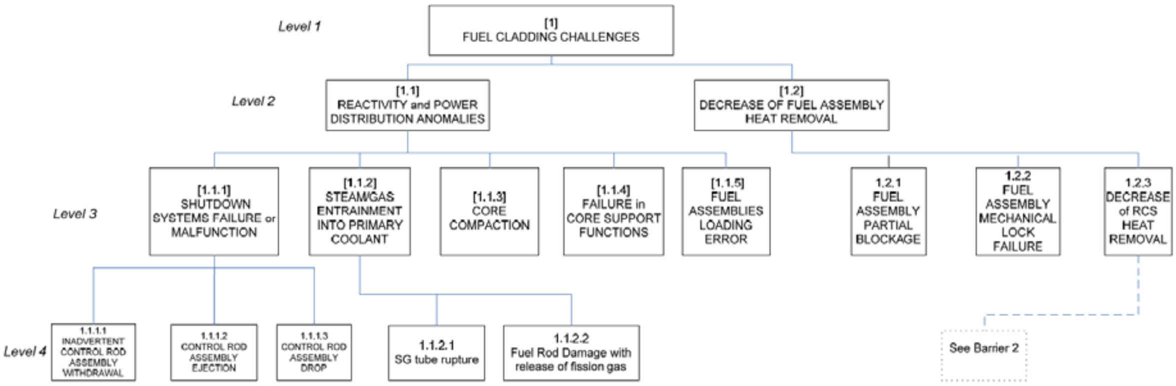

Figure 14 MLD for fuel cladding challenge

The events marked as 1.1.1.1, 1.1.1.2, 1.1.1.3 are caused by possible failures of the mechanism of the control rods handling.

The event marked as 1.1.2.1 refers to the insertion of reactivity in the core due to the moderating properties of the water and the positive void coefficient due to steam bubbles trapped in the liquid lead. Even though it is mentioned in section 4.1, there are no sufficient evidences to support the claim that the water inlet into the core is excluded by design.

With regard to event 1.1.2.2, reactivity increase is due to the positive void coefficient resulting from fission gas release into the coolant.

Concerning the events marked as 1.1.3 and 1.1.4, they can be attributed to possible events which change the core geometry such as overpressure transients, shock wave, earthquake; Moreover, the event 1.1.5 can occur also because of improper positioning of fuel assemblies.

Regarding the event marked as 1.2 it refers to anomalies in the core flowrate which in turn decreases the core heat transfer resulting in clad overtemperature. Events marked as 1.2.1 and 1.2.2 are both caused by a mechanical failure of Fuel Assemblies which determines modification to hydraulic characteristics and reducing coolant flow rate.

The event 1.2.3 is described in section 5.2.2.

5.2.2 RCS boundary challenge

Figure 15 shows the initiating events affecting the barrier "RCS boundary".

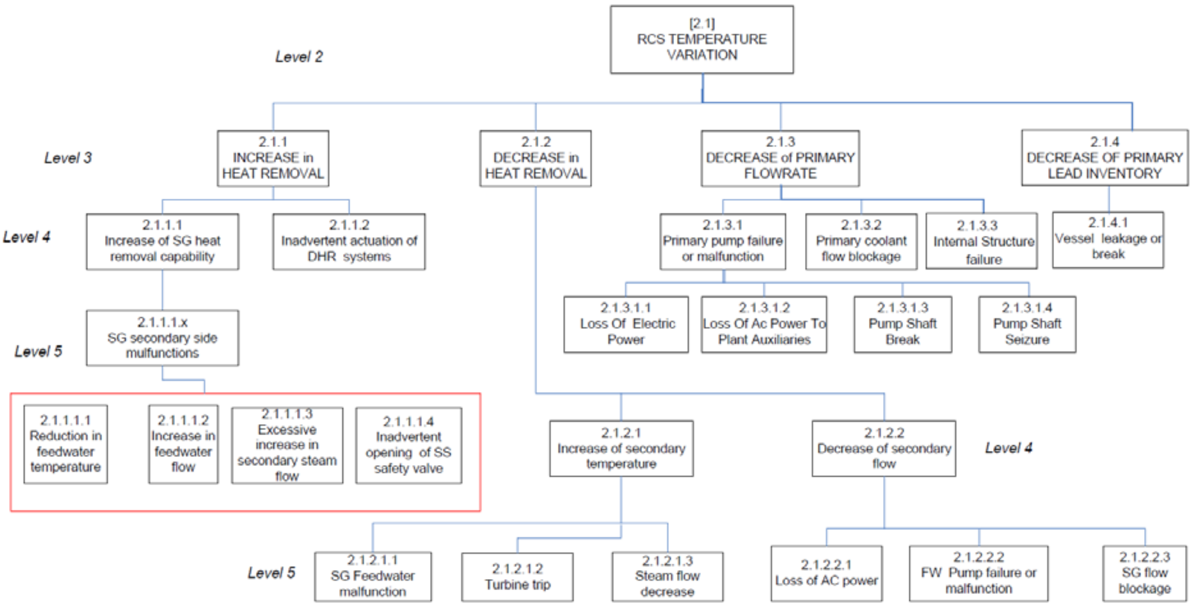

Figure 15 MLD for RCS boundary challenge

The event marked as 2.1.1.1 refers to any anomaly involving the secondary loop that causes the increase in heat removal in the SG with a consequent decrease in temperatures of the lead (see Figure 15 where the red box provides relevant examples such as reduction in feedwater temperature, increase in feedwater flow…).

The event 2.1.1.2 refers to a spurious actuation of ALFRED DHR caused by an operator error or a false actuation signal. This event will not be taken into account since it has been studied in previous European projects and through numerical simulations it has been verified that the spurious actuation of the ALFRED DHR has a negligible impact on the normal operation of the RCS.

On the other hand, other anomalies which affects the secondary loop that causes a decrease in the heat removal in the SG (e.g., increase of feedwater temperature, reduction of feedwater mass flow…) and therefore an increase of primary temperatures are gathered in events 2.1.2.1 and 2.1.2.2.

Another main reason for the reduction in heat removal is due to the decrease in the primary flow rate. It is caused by a primary pump failure due to electrical failure (events marked as 2.1.3.1.1 referring to the motor and 2.1.3.1.2 referring to a station blackout) or mechanical failure of the pump itself (events marked as 2.1.3.1.3 and 2.1.3.1.4).

In addition, the event marked as 2.1.3.2 assumes obstructions of the flow area. These obstructions increase the pressure drops of the primary loop and depending on the typology of the RCP installed, in the case in which it is a direct supply pump, at the same absorbed electrical power, the flow rate of the primary coolant decreases. In case in which the pump is also powered by an inverter in order to keep the flow rate of the primary coolant constant, there is an increase in the electrical power absorbed by the pump. These obstructions may be due to unknown parts accidentally introduced during refuelling or maintenance, or to the failure of structural elements located within the vessel reactor, or alternatively to prolonged oxide precipitation and deposition in cold areas.

A further initiating event within 2.1.3 category is the failure of the IS. Following a modification of internal geometry involving leak bypass through the IS, thermal stratification can occur and the RV would be affected by thermal creep causing a mechanical stress and component life-time reduction. Additionally, if the leakage is not detected under normal operating conditions a SGTR can result in steam entrainment and transport towards the core leading to the same phenomena described for event 1.1.2.1.

Finally, the event 2.1.4.1 assumes a reactor vessel leakage causing lead discharge in the safety vessel.

5.2.3 Containment challenge

The third pathways depicted in Figure 16 aims to identify the initiating events affecting the barrier "Containment".

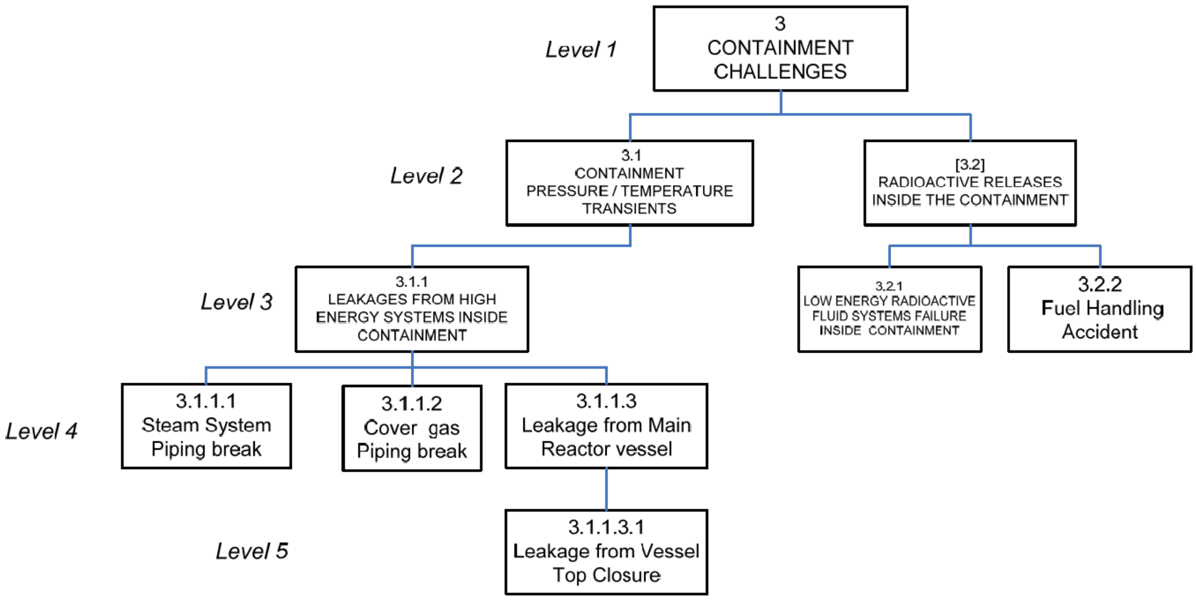

Figure 16 MLD for Containment challenge

It should be noticed that the event marked as 3.1 refers to accidental events with pressure and/or temperature increase in the reactor containment. On the other hand, the event marked as 3.2 refers to accidents that release radioactivity in the reactor containment with no or negligible mass and energy release. Due to the fact that it has been verified that the accidental events caused by external hazards that could affect the containment integrity (e.g., airplane crash, extreme weather conditions, earthquakes…) are excluded by design. Therefore, these initiating events are not described in this report.

The event marked as 3.1.1 refers to the failures of high pressure and temperature systems that result in the release and release of fluid (potentially radioactive) into the containment.

Finally, initiating events can involve the direct release of radioactive material due to piping failure inside the reactor building (marked as event 3.2.1) or as an accident due to incorrect handling of spent fuel leading to the breakdown of clad and therefore to the release of fission gases (marked as event 3.2.2).

6 Selection of initiating events

Among the events proposed in Chapter 5, the bounding initiating events were selected on the basis of engineering judgement so to include in the PIRT discussion most of the expected physical phenomena, possibly simplifying the identification of those having lower knowledge. To this end, the selection was carried out using the challenges proposed in the OPT cataloguing program as a guideline. It should be noticed that for the selection of initiating events, it has been assumed that the initial condition of the accidental transient is the normal operation even though it may not represent the most conservative hypothesis.

As far as the initiating events belonging to the category "Fuel cladding challenges" concerning, all those events under the event 1.2 have not been taken into account since they gather phenomena in common with the events belonging to the category "RCS temperature variation", the considerations on these events will be performed later on. Events 1.1.4 and 1.1.5 have been discarded since the former considers an event that is a subset of event 1.1.3 being that it assumes a seismic event, in addition to the deformation of the fuel assemblies there can also be a breakdown of them. The latter has not been considered since the initial condition is not normal operation. The 3 events involving a possible failure of the control rods handling system group the same phenomena caused by a local positive reactivity insertion with different time scale.

Regarding the initiating events belonging “RCS temperature variation”, all those events concerning potential anomalies affecting the secondary loop (i.e., 2.1.1 and 2.1.2) have been discarded since the main target of this report is to focus on the phenomena that occur in the primary loop. Moreover, all those events that assume a loss of flow accident of the primary coolant have not been considered since this accidental event has been studied in previous European projects. Finally, the event 2.1.4 has not been taken into account because it assumes an initiating event with such a low frequency of occurrence (it is estimated that it is lower than 10^{-6}) that it is to be considered as "practically eliminated".

Concerning the initiating events of “Containment integrity”, all those events that do not involve the primary have not been considered. At a later phase, those excluded events that consider the release of radioactive material will be deepened on. For all those events subordinate to event 3.1.1 that assume a release of energy inside the containment, the event that causes the release of the greatest energy has been selected, being that the most burdensome situation.

Therefore, according to these considerations, the selected initiating events are the following:

- Event 1.1.1.1: Inadvertent Control Rod assembly withdrawal
- Event 1.1.2.1: SG tube rupture
- Event 1.1.3: Core compaction
- Event 2.1.3.2: Primary coolant flow blockage
- Event 2.1.3.3: Internal Structure Failure
- Event 3.1.1.1: Steam System piping break

Event 1.1.1.1 and event 1.1.3 both challenge the first basic function of nuclear safety (i.e., reactivity control). In addition, both events consider the failure of the reactor shutdown system. Despite this, the two transients differ for the following reasons: the former is a slow UTOP with a local effect on the core, the second is a fast UTOP with global effect on the core. Therefore, the choice of these two events allows to study the same transient (the UTOP) but in two different time scales and two different regions of influence.

The event 1.1.2.1 has been taken into account since the resulting transient gathers a set of expected phenomena of relevant interest both of neutronic and thermo-hydraulic nature (detailed in section 6.2).

The event 2.1.3.2 challenges the second basic function of nuclear safety (i.e., heat removal). This initiating event has been selected because it is considered to be burdensome compared to the others that challenge heat removal and also this initiating event is the envelope of the others.

The event 2.1.3.3 challenges the second basic function of nuclear safety (i.e., heat removal). This event has been selected as induced by a recently introduced design change therefore the phenomena that follow this PIE require dedicated investigation.

The event 3.1.1.1 was selected because it has the dual function of challenging the third basic function of nuclear safety (i.e., containment integrity) and investigating the possible freezing of the primary coolant.

6.1 Event 1.1.1.1 - Inadvertent Control Rod assembly withdrawal

This initiating event refers to an accidental withdrawal of the control rod having highest worth from the core and challenges the first basic function of nuclear safety (i.e., reactivity control). Therefore, it is an insertion of positive reactivity which in turn causes the increase of power generation in the core. Phenomena of interest for this transient involve:

- Reactivity feedback from coolant and fuel temperature,
- Primary coolant temperature behaviour,
- Coolant-materials interaction.

It is assumed the failure of the emergency shutdown system (the UTOP), thus investigating a slow transient with a local influence in the reactor core, compared to other reactivity induced accidents.

Figure 17 shows the qualitative event tree of this transient. The actuation signal of the reactor protection system is still under development, either based on neutron flux measurement or rate of change, or on core outlet temperature measurement.

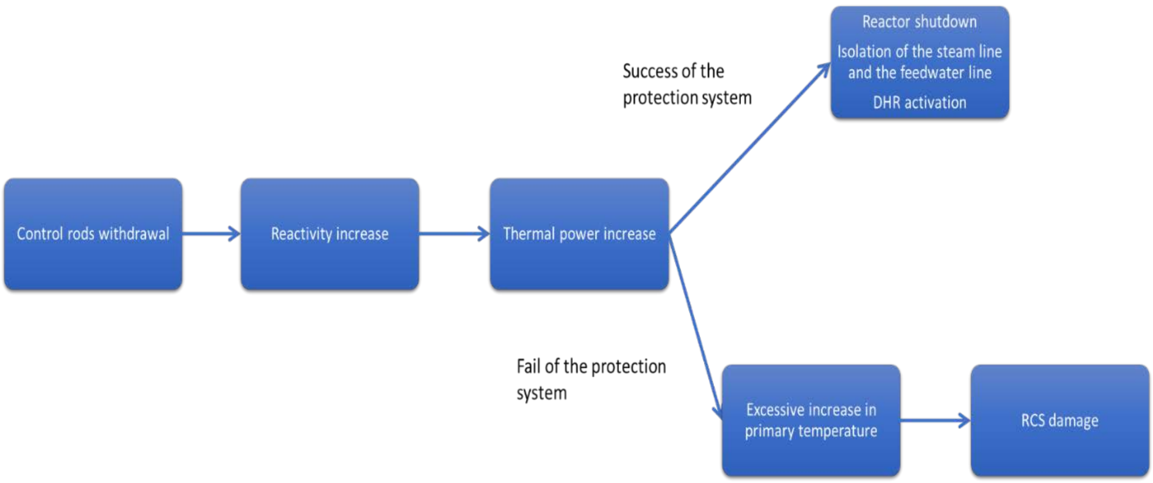

Figure 17 Event tree of inadvertent control rod withdrawal

6.2 Event 1.1.2.1 - SG tube rupture

This event involves a SG tube rupture causing steam / two-phase mixture release into the coolant. This event may challenge both the first basic function of nuclear safety (i.e., reactivity control) and the second (i.e., heat removal). Steam entrainment and subsequent transport in the core region causes a positive reactivity insertion due to the positive void coefficient and the moderating properties of water. As far as ALFRED is concerned it has to be demonstrated whether the IS can prevent the ingress of water into the core.

In addition, it is necessary to investigate whether or not a tube rupture involves a single tube or also the neighbouring ones due to jet impingement and whipping and the potential SG fail due to the rise and propagation of pressure waves.

The evolution of the transient is shown in Figure 18. The possible quantity to be measured in order to detect a possible SGTR and then implement the protection system is the concentration of steam in the gas cover. In case of failure, the transient continues with the same phenomena as event 1.1.1.1.

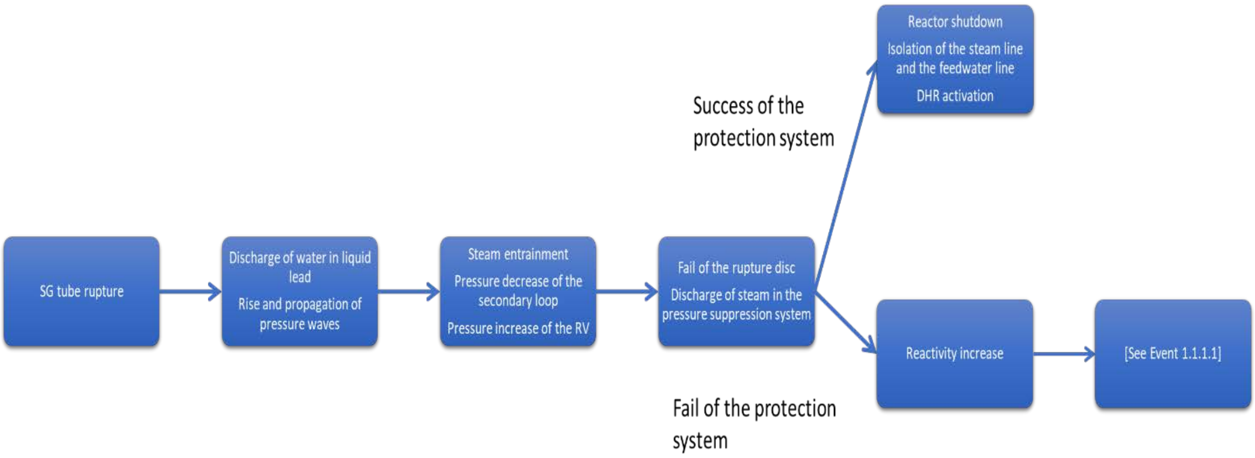

Figure 18 Event tree of SG tube rupture

6.3 Event 1.1.3 - Core compaction

This event refers to a global change in core geometry resulting from a seismic event. The emergency shutdown system can perform the function or fail. It is assumed that the earthquake causes deformation of fuel assembly. This deformation in turn causes a change in the geometry of the core that leads to the increase of reactivity. Then, it causes an increase of thermal power generation in the core.

This transient involves variation of reactivity due to fuel and coolant temperature feedback. Also, fission gas release into the liquid lead following clad burst is an expected outcome, with the formation of complex interaction between radioisotopes and lead.

The transient evolution is depicted in Figure 19. The same considerations about the reactor protection system dealt for the event 1.1.1.1 are applicable.

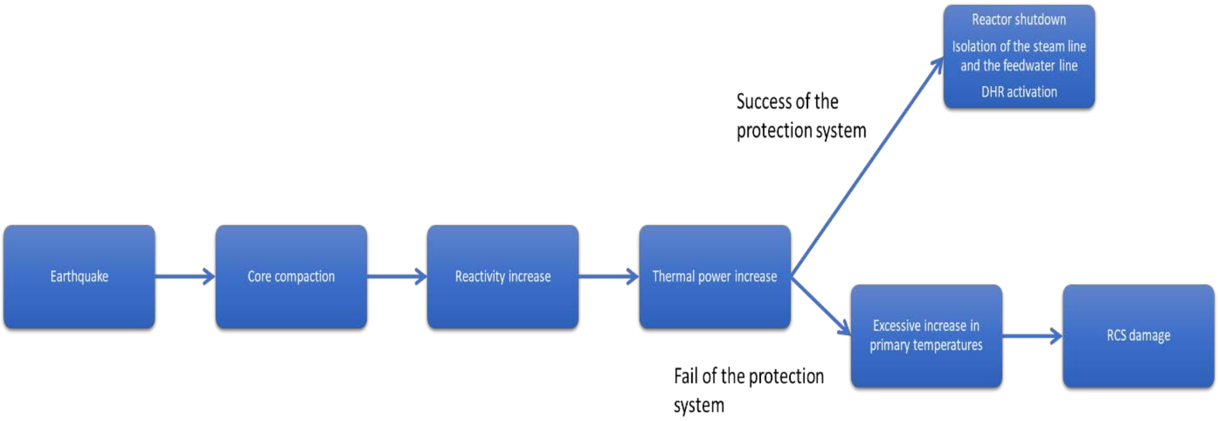

Figure 19 Event tree of core compaction

6.4 Event 2.1.3.2 - Primary coolant flow blockage

This initiating event refers to a primary coolant flow blockage and challenges the second basic function of nuclear safety (i.e., heat removal). This event can be caused by a foreign object inadvertently introduced into the primary system (e.g. during refuelling, in-service inspection or maintenance) or to deposition of lead oxides in cold areas or resulting from build-up of metal oxide debris on the spacer grids of the fuel assemblies.

In particular, for the analysis of the primary coolant flow blockage, the following accidental scenario are assumed:

- Blockage caused by build-up of metal oxides debris
- Blockage caused by deposition of lead oxides in the SG outlet

The former assumes a partial / total occlusion of the flow area of the primary coolant in the spacer grids of the fuel assemblies leading to a degradation of the core heat transfer causing a significant increase of temperatures determining a potential burst of the clad.

In this case, it is necessary to investigate the duration of oxides build-up and transient, whether any local damage to a FA does not involve other FAs extending the damage and what percentage of minimum blockage can cause clads damage.

Figure 20 shows the main events of this transient. If the temperature set point of the Protection System would be exceeded, this would cause the initiation of reactor trip. Therefore, outlet temperature of each fuel assembly is measured.

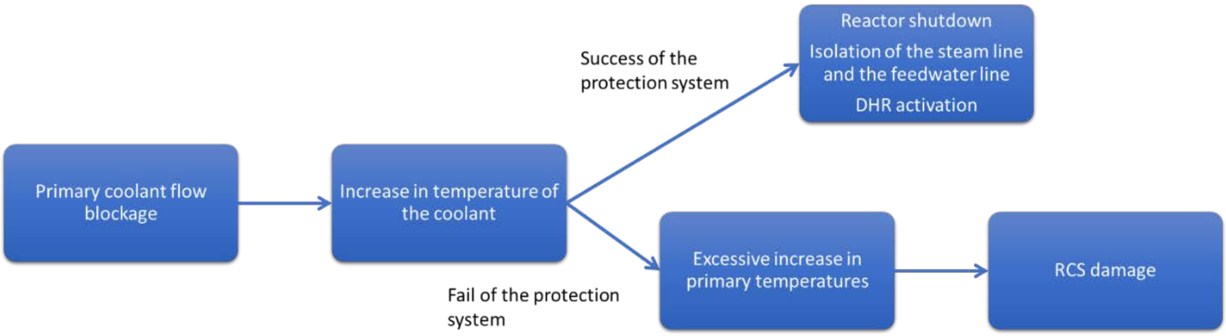

Figure 20 Event tree of primary coolant flow blockage

The latter scenario assumes a loss of oxygen control / loss of the nominal thermal field along the SG. Both are necessary in order to ensure stable passivation of structural materials and thus limit corrosion of materials. The loss of one of the two factors can promote the formation of unstable layer of metal oxides that breaks down and the oxides are dragged by the coolant. The oxides are deposited at the exit of the SG being the coldest area of the RV thus reducing the flow area leading to the same qualitative phenomena of the previous case.

6.5 Event 2.1.3.3 – Internal Structure failure

This initiating event refers to the mechanical failure of the IS. This event leads to a change in the nominal flow path of the primary coolant causing lead stagnation in the upper part of the cold pool which may eventually heat up due to conduction from the hot pool and cause thermal stratification.

Under such conditions, the RV is exposed to an axial thermal gradient involving creep effects and related loads.

In addition, in the event that this break is not detectable, steam would have a simplified path to the reactor core in case of a SGTR. Steam transport and migration to the core induces a positive reactivity insertion.

Eventually, the event tree for this PIE is depicted in Figure 21. In this scenario the alarm signals are still to be determined as well as the mitigating actions.

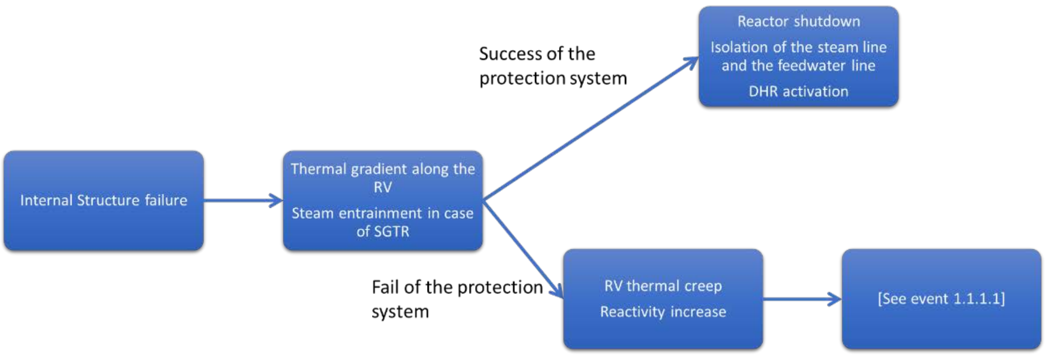

Figure 21 Event tree of IS failure

6.6 Event 3.1.1.1 - Steam System piping break

This initiating event is a guillotine break on the main steam line of one SG. The event results in a release of water coming from a high energy system (450 °C and 180 bar referring to ALFRED stage 3) which can damage the containment due to pressurization and pipe whip if the break occurs upstream the main isolation valve (challenging the third basic function of nuclear safety i.e., containment integrity).

At first, the accident provides an increase of the SG heat removal ability determining the decrease in temperature of the primary loop, potentially inducing localized freezing or excessive precipitation of metal oxides.

The evolution of the transient depends on the position of the steam line break. If the break occurs upstream of the isolation valve, the loop goes out of service and the actuation of the DHR loop will not be possible (other two loops are still available).

If the rupture occurs downstream of the isolation valve of the steam line, the Reactor Protection System intervenes. The Protection System would shut down the reactor, isolate the steam lines and the feedwater lines and initiate DHR systems. In addition, due to the high mass flow rate of steam, the event of the closure of the isolation valves can induce hydraulic loads due to water hammer.

Finally, the event tree for this transient is depicted in Figure 22. The reactor protection system actuation in this case depends on the steam pressure in the SG.

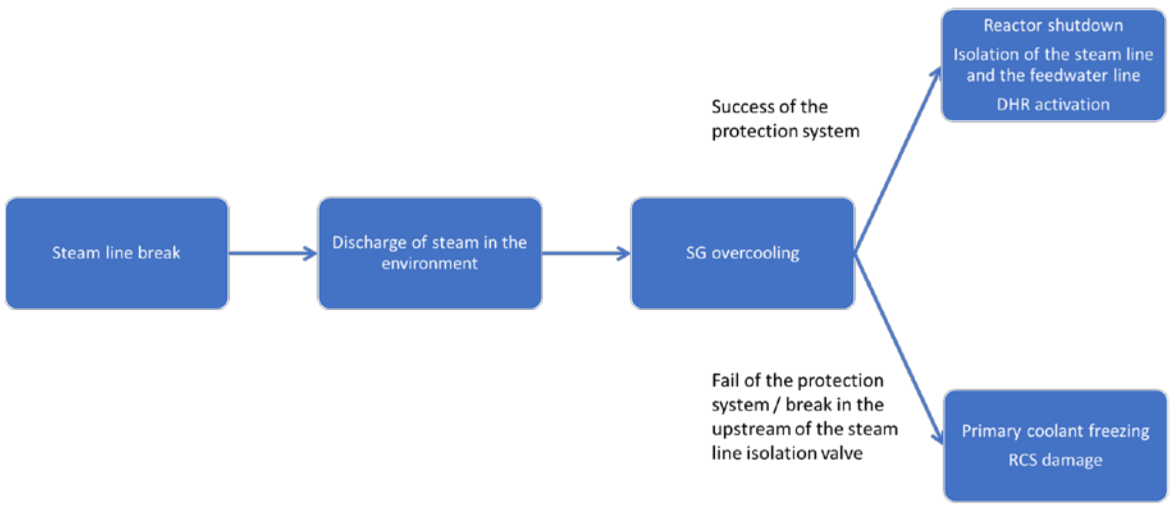

Figure 22 Event tree of Steam Line Break

7 Conclusions

The present deliverable refers to the WP 1 – task 1.2 in the context of the ANSELMUS project. WP1 aims to perform the safety evaluation of the design of ALFRED and the starting point is to elaborate the PIRT. The data required for the PIRT are the reference design of ALFRED and accidental events which could occur in the reactor.

Section 4 is devoted to the description of ALFRED reactor. In section 5, a brief description of the main methodology to implement DiD in LFRs is presented and in particular section 5.1 describes the Objective Provision Tree as recommended by the RWSG.

Later in chapter 5 there is an exhaustive review of initiating events listed in MLDs and briefly described.

Subsequently, chapter 6 collects and describes the reference initiating events for PIRT:

- Event 1.1.1.1: Inadvertent Control Rod assembly withdrawal
- Event 1.1.2.1: SG tube rupture
- Event 1.1.3: Core compaction
- Event 2.1.3.2: Primary coolant flow blockage
- Event 2.1.3.3: Internal Structure failure
- Event 3.1.1.1: Steam system piping break