



**D2.2 - Specifications
for the systems
supporting the
management of
incidents and
accidents
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Summary

This deliverable summarises the specifications for the instrumentation developed in FIND to support the management of incidents and accidents in nuclear power plants (NPPs). It will be the reference document for the work performed in work-packages (WP) 3 and 4. Its structure is similar to D2.1.

ASNR is the main contributor with a technical review from VTT and SSTC-NRS, especially to investigate the transposability of the concepts to the reactor designs exploited in Finland and Ukraine (various designs of VVER reactors).

The deliverable presents the benefits for safety of the proposed innovative instrumentation. It discusses similar use-cases that could rely on the same technological bricks. Precise specifications are then presented for each use-case, to give a robust framework for developments and tests. When relevant, the operational return of experience is analysed to highlight the benefits of the proposed systems.

Annexes gather scientific data supporting the conclusions of the main part of the document. Raw data was considered too sensitive to be shared openly.

Data and discussions collected for the preparation of the report confirmed the soundness of the selected use-cases. FIND partners consider they have reached a sufficient level of details to support technology developments, even if the document is not intended to be used beyond technological research. Especially, it cannot be regarded as a qualification standard.

Keywords

Specifications, Accidental instrumentation, Loss-of-Coolant Accident, Water level measurement, Fission product detection

Abbreviations and acronyms

Acronym	Description
CSS	Containment Spray System
CVCS	Chemical and Volumetric Control System
DBA	Design Basis Accidents
DCH	Direct Containment Heating
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
FCVS	Filtered Containment Venting System
LOCA	Loss-of-Coolant Accident
MCP	Main Coolant Pump
NPP	Nuclear Power Plant
NSS	Nuclear Sampling System
PSA	Probabilistic Safety Analysis

PWR	Pressurised Water Reactor
RCS	Reactor Cooling System
RHRS	Reactor Heat Removal System
RWST	Reactor Water Storage Tank
SA	Severe Accidents
SAMGs	Severe Accident Management Guidelines
SBO	Station Blackout
SIS	Safety Injection System
VVER (or WWER)	Water-Water Energetic Reactor
WP	Work Package

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Introduction

FIND aims at developing innovative instrumentation to support decision making during Design Basis Accidents (DBAs) and Severe Accidents (SAs). Deliverable D2.2 defines the functional specifications of the systems to be reached by technology providers. For each use-case, it will remind the general purpose of the instrumentation and explain how it can support decision making. Then, a table will summarise the requirements for the sensors. It will include the performance objectives and the environment the system will have to withstand. D2.2 does not repeat the content of D2.1, especially concerning the operating conditions of the instrumentation during the normal operation of the plant. Indeed, accidental instrumentation must resist to the cumulative effects of normal operating and accidental conditions. Normal operating conditions are less stringent than accidental ones, but they apply for a far larger duration (years or decades, as compared to days or weeks for accidental instrumentation). Therefore, they must be considered for the design of accidental instrumentation.

Three use-cases are addressed: the measurement of the water-level in the sumps, the localisation of a breach during a LOCA (Loss Of Coolant Accident) and the detection of fission products during a SA.

Contrary to D2.1, for which operating return of experience was key, D2.2 will rely more extensively on accident analysis, including results provided by scientific calculation codes. Information was also gathered from scientific literature and interview of experts. Confidential documents were analysed as well, even if they cannot be cited in the report. Data coming from them was summarised in a way to protect confidentiality. Figures in specifications are based on a few representative NPPs of the various considered designs. They cannot fully represent the diversity of situations in all reactors of the same type, but they still give relatively accurate objectives to be achieved by instrumentation.

Use-cases are illustrated by safety significant events collected using the PIREX application (Integrated Platform for Operational Feedback), developed and maintained by the ASNR. PIREX contains all the significant events reported across all French facilities since their commissioning (over 50,000 events to date). Equipped with advanced text mining features, PIREX enabled the identification of relevant operational scenarios to be explored within the FIND project. No relevant situation was identified in PIREX to support the last use-case of the report concerning the detection of fission products in the reactor building.

The instrumentation proposed in D2.2 mostly addresses situations that are already covered by safety demonstrations. As such, it is not strictly necessary for accident management. It must be considered as an improvement of the current procedures, to give additional information in situations that are expected to be highly confusing. Therefore, the qualification of the systems proposed below is desired but not strictly necessary. Nothing prevents operators from using non-classified instrumentation to support their decision, as long as it has no negative effect on safety.

Originally, D2.2 was focused on instrumentation supporting accidents and severe accidents. However, the various sources of information mentioned above showed that the systems initially envisaged could prove useful in less dramatic situations too. In fact, they could support maintenance and incident management as well. Addressing less demanding use-cases with shorter term benefits would help the technologies developed by FIND accessing the market. This first step would help them gain in maturity before being deployed for accident management.

This document will be the basis for the developments of WP3 and tests carried out in WP4.

1. Water level measurement in the sumps

1.1. Rationale

During a LOCA, the water inventory of the nuclear reactor escapes through a primary breach to the reactor building. This leads to a degradation of the core cooling (loss of water inventory and depressurisation of the circuit) and to the pressurisation of the containment. Two main safety systems are activated to manage these phenomena:

- The Emergency Core Cooling System (ECCS) or Safety Injection System (SIS), which injects water into the Reactor Cooling System (RCS),
- The Containment Spray System (CSS), which injects water on the top of the reactor building, to limit its pressure.

The water leaking from the primary breach or injected by the CSS is collected by the reactor building sumps. In the Loviisa VVER-440 reactors with ice condenser containments, the hot steam from a primary breach will cause melting of the ice. The water will then drain from the ice condensers to the sump. In accident management procedures, ECCS and CSS will pick up water from storage tank or reactor building sumps. This water can be cooled down with a heat exchanger. For some reactor designs, external portable systems can be used to inject additional water in the RCS or the sumps.

If a SA leads to the failure of the vessel, the molten core (called the corium) will be poured to the reactor cavity and start to interact with its concrete. In those conditions, some reactor designs can use the water in the sumps for the passive reflooding of the corium. In the Loviisa VVER-440, when the water level in the sump increases, float valves open and drain the excess water to the reactor cavity. The amount of water from the melting ice condensers is so large, and the cavity so small, that the RPV lower head becomes submerged. This is necessary for the in-vessel melt retention strategy.

Knowing the level of water in the sumps is critical for the management of the accident. A minimal level is necessary for the good operation of the recirculation pumps. Enough water is also necessary for the passive reflooding of the corium during the ex-vessel phase of a SA. A complete drying of the sumps can also lead to the volatilisation of soluble fission products to the gas phase, increasing the activity in the gaseous phase of the reactor building and thus the releases to the environment.

On the contrary, an excess of water in the sumps can lead to the flooding of the reactor pit, in case the bellow¹ ensuring the tightness of the reactor cavity has been damaged. This can have dramatic consequences if the vessel fails and releases molten fuel. In this case, the corium-water interaction may lead to a very energetic vapour explosion that would threaten the containment integrity. In addition, debris generated by the accident can clog various equipment of the recirculation system, especially the filters of the sumps, leading its failure despite the presence of sufficient water. A water level measurement could help understanding such

¹ The bellow is usually welded to the reactor flange and ensures the tightness of the reactor pit during refuelling operations and accidental situations.

situations. Knowing the water inventory in the sumps is also important for the long-term management of the accident (post-accidental phases) to manage effluents.

Many nuclear reactors do not have a system to measure the water level in the sumps. Existing water level sensors (manometric, radar or capacitive sensors...) are not always hardened for accidental conditions. They can be used to periodically check that the amount of water in the sumps is in line with the requirements of the safety demonstration². So, the safety demonstration shows that, by design, the level in the sumps should always remain in the acceptable range during postulated accidents. This way, the sump water level measurement system would come as a confirmation and as a support to understand the accident progression. However, some accidental sequences starting with an open vessel have recently been identified, for which the water inventory in the reactor building may not be sufficient to manage the accident in the long term. In such cases, knowing the water level in the sump can become critical, to make appropriate decisions to support the water recirculation function. Operators are now considering adding dedicated sensors qualified to accidental conditions.

The environment faced by the sensor is extremely harsh. Primary water contains several chemicals like boric acid. Caustic soda can be added to the containment spray system to increase the solubility of some radioactive materials released during the accident. Different types of debris can be generated during the accident, like soot in case of fire. The degradation products of thermal insulation can clog the recirculation systems. The CSS injects water droplets in the reactor building, which can interfere with the water level measurement system. The temperature and pressure in the reactor building can increase up to 150 °C and 6 bar for Western-type PWRs, or 250°C and 10.5 bar for VVER-1000 reactors. VVER-440 reactors rely on a different containment strategy and cannot reach such high pressures. Radioactive material dissolved in the sump water produces heat that can produce a vapour saturated atmosphere. The concentration of aerosols in the gaseous phase, of which some are highly radioactive, can reach 10 g/m³. They are mostly composed of silver, caesium, indium, molybdenum, cadmium, iodine and uranium. Dissolved species are also present in sumps water, with a maximum concentration of few hundreds of grams per cubic meter.

The measurement of the water level in the spent fuel pool of a reactor exhibits some similarities with the use-case described above. This application is not detailed here but it would be a natural continuation of FIND.

1.2. Specifications

General information			
Component(s) concerned	Reactor sump		
Safety function	Water collection for the long-term cooling of the reactor and containment building during an accident		
Type of measure	Accidental instrumentation	Reactor designs concerned	All Pressurised Water Reactors (Framatome, Westinghouse, VVER, Konvoi...)

² In some NPP designs, sumps have a minimal water inventory during the normal operation of the plant, to ensure the correct start of the recirculation function if a LOCA was to occur. The regular water level measurements are performed while the reactor is in nominal conditions. Other designs keep sumps dry during normal operation.

Lead contributor	ASNR	Other contributors	Vuez, VTT, SSTC-NRS
Current methods applied	Differential pressure sensors Radar sensors	Limitations of current methods	Usually not qualified for accidental conditions. Differential pressure sensor: manual reading, specific circuit alignment, manual calculation. Radar sensor: lack of reliability due to parasite radar beam reflexion.
Component characteristics			
Water depth	Variation range about 2 m on PWR		
Fluid contained in the sumps			
Type of fluid	Water	Temperature	Western PWR: 20-140°C VVER-1000/V320: 15-250°C
Velocity	Negligeable	Pressure	Western PWR: 1-6 bars absolute Loviisa VVER-440: 1-3 bar absolute VVER-1000/V320: 1-10,5 bar absolute
Chemical composition	1.5 < pH < 9 Dissolved fission products: few 100g/m ³ Boric acid concentration: about 2000 ppm maximum Debris of thermal insulation	Physical state	Liquid, vapour, droplets
Magnitude to be measured for accidental instrumentation			
Magnitude	Water level	Detection range	= height of the sump (e.g., 7m in VVER-1000/V320)
Precision	A few discrete points	Response time	1-10 min
Sources of signal interference			
Presence of droplets	Variable chemical composition	Variable temperature and pressure	Vapour saturated atmosphere
Sensor environment during the accident (see D2.1 for normal operating conditions)			
Temperature	PWR: up to 150 °C in gaseous phase, up to 140 °C in liquid phase VVER: up to 250°C	Humidity	Up to 100%
Pressure	PWR: Up to 6 bar (abs.)	Irradiation	Few MGy for irradiation during several days

	VVER: Up to 10,5 bar (abs.)		
Main safety constraints on hardware	Fire resistant / no corrosive product if burnt	Electricity supply	Low voltage DC power (24-48V)
Data transfer	Analogical with rad-hard cables. Optic fibres without local electronic treatment.		
References			
Ref: National and international standards	RCC-E: Règles de Conception et de Construction des Systèmes et Matériels Electriques et de Contrôle Commande (Standards for the design and construction of electric and I&C systems and materials) IEC 63147:2017. Criteria for accident monitoring instrumentation for nuclear power generating stations		
Product sheet	The Hardline platform developed by Framatome is an example of I&C system specifically suited for severe accidents.		

1.3. Additional operating experience feedback regarding water level measurement in the sumps

Approximately 55 incidents reporting failures in water level measurement in sumps have been recorded in the PIREX database since the 2000s. These incidents did not all necessarily result in the unavailability of the safety injection system, but most highlight either the difficulties encountered by operators in obtaining an accurate water level or the unreliability of current sensors for doing so. Radar sensors have been introduced to secure water level measurements relying previously on a manometric gauge, which proved to be unreliable due to frequent human errors (calculation errors, wrong alignment of circuits). Radar sensors can also be used to check the water level continuously during refilling operations. Unfortunately, radar sensors showed reliability shortcomings as well, as highlighted in the following events. It must be noted that both systems are only used for periodic tests during normal plant operation: they are not qualified for accidental conditions.

Wrong water inventories in reactor sumps have been recently observed in the operating fleet. They are summarized below:

Date: 2021-01-15

NPP: CRUAS

Title: insufficient water level in the sump, rendering Safety Injection System train B unavailable

What happened: In January 2021, during a periodic test to measure the water level in the sump, an insufficient water level was observed in the CSS level column. The radar sensor of the SIS, however, indicated a compliant level. In previous checks, this sensor had provided a constant reading, whereas a **slight decrease should have been observed due to water evaporation**. The value transmitted by the radar sensor was therefore incorrect. The sump was quickly topped up to restore a compliant level. Until July 2021, the water level in the sump was monitored using the Containment Spray System level column. Investigations revealed the presence of significant boron deposits on the Safety Injection System sump crossbar, which was **interfering with the radar sensor beam**.

Consequences: Unavailability of the SIS on a six-month period.

Interest: This event shows **the lack of reliability of the radar sensor which can remain stuck at a high value for several months without the operator detecting a discrepancy**.



Figure 1-1: Histogram of safety events concerning water level measurement in the sumps in France since 2000

Date: 2022-11-01

NPP: CRUAS

Title: Non-compliance with a criterion for water retention of Containment Spray System sump following an erroneous analysis of the availability of the radar level sensor.

What happened: during a periodic test, opening of the CSS sump isolation valve was performed and led to a significant drop in the water level in this sump due to the presence of air in the piping, upstream of the isolation valve. Since early 2021, the operator had doubts about the radar sensor's behavior. A check was carried out following the readings on the CSS level column, and the sump was topped up. However, **due to an error in reading the CSS level column**, the top-up was stopped prematurely, without the discrepancy being detected.

Consequences: In the event of a LOCA leading to operating of CSS, an insufficient water seal in the sump could have caused a boiler effect to occur at the valve, which would prevent its operation.

Interest: This event shows **the difficulties in interpreting CSS level column measurements** that can be encountered by operators.

Date: 2024-10-31

NPP: GRAVELINES

Title: Unavailability of SIS due to the level of sump being below the required level.

What happened: The drop in the water level in the sump was caused by an untimely draining of the sump. For several weeks, level checks were carried out using the radar sensor, but this did not allow the discrepancies to be corrected. Indeed, the radar sensor measurement was stuck at a high value, without the operator realizing it.

Consequences: The level of the recirculation sump was below the operating procedures criterion for two months.

Interest: This event shows the **lack of reliability of the radar sensor and the difficulties for operators to be aware of its dysfunction.**

2. Localisation of a breach during a LOCA

2.1. Rationale

The LOCA is a type of accident to which NPPs are designed to resist. They encompass a large variety of scenarios, resulting from the size and the localisation of the primary breach. They result in a loss of water inventory, a depressurisation of the primary circuit and a degraded cooling of the reactor core, which can lead to fuel damage. The hot steam leakage in the containment building pressurises it and can threaten its integrity.

Only a minority of reactors are equipped with systems that can localise a breach on the primary circuit. They exploit existing solutions like Framatome's FLÜS system. It is based on a network of small diameter metallic pipes that run along the pipe of interest. It collects air samples at specific locations, that are pumped to a station measuring their humidity. However, this system is relatively intrusive. In addition, it is designed to detect small leakage and has a relatively long response time (up to 1 hour). Therefore, it is not well suited to support the management of a large break LOCA, whose kinetics is much faster. Other systems have been developed for sodium-cooled reactors, based for example on optic fibres. Other attempts have considered relying on fire detection systems, which may trigger an alarm due to the presence of hot water droplets, detected as smoke. However, such detectors cannot discriminate a fire and a steam leakage. Radiation monitoring systems can also give partial information, but they were not designed for that specific purpose.

In this context, nuclear operators do not rely on leak detection systems in their safety demonstration to manage a LOCA. Emergency Operating Procedures (EOPs) are designed to bring the reactor to a safe state even if the localisation of the breach is unknown. For example, the ECCS will inject water in all the primary loops of the reactor, even if one of them leaks.

Nevertheless, having a reliable information on the breach localisation would make accident management more efficient.

First, some leakages on systems connected to the primary circuit can be isolated thanks to valves, bringing the reactor to a safe state. During the Civaux-1 accident in France in 1998, it took several hours for operators to localise the breach on one line of the Reactor Heat Removal System (RHRS). Once the breach was localised, operators could isolate the leakage and switch to the other redundant line of the RHRS. In most NPPs, the identification of the leakage relies on successive isolation tests. Operators would isolate large portions of circuits and check how reactor parameters evolve, to confirm the localisation of the leakage, which can result from a breach, an untight valve or a wrong alignment of circuits. Operators usually analyse recent maintenance or operational activities to identify the plausible origin of the leakage. However, the process relies a lot on trial and error, with the risk of stopping an operational safeguard system to test its integrity. It can only be done after reactor parameters have been stabilised, delaying diagnosis further. Confirming the success of the mitigation action can take time as well. A reliable leakage detection system would make this process more efficient. Competing systems like FLÜS are already used for that purpose.

Another advantage is to better anticipate the evolution of the accident and adapt the emergency response accordingly, especially for non-isolable large breaks. In fact, a breach located on the cold leg of the reactor can lead to a much faster core uncovering than a breach on the hot leg. As a consequence, emergency responders have far less time to anticipate off-site actions in case of a cold leg LOCA, like the uptake of iodine tablets or evacuation. Several attempts have been carried out to develop methodologies to efficiently localise the breach thanks to the available instrumentation of the NPP. However, none proved to be reliable enough. The instrumentation in FIND should address this shortcoming.

It must be noted that the qualification of the sensor as a safety-relevant system is desired but not strictly necessary. Accident management can benefit from additional information from unqualified sensors, which bring robustness to the safety demonstration.

A structured methodology was employed to define the specifications of the breach localisation system. During a LOCA, several reactor protection systems may automatically isolate the primary circuit, based on specific signals (low pressuriser pressure, high pressure or activity in the reactor building, safety injection...)³. This early disconnection from non-essential safety systems like the Chemical and Volumetric Control System (CVCS) may be sufficient to stop the leakage. So, it was decided to study in priority the systems that are not automatically disconnected from the primary circuit in the early phases of the LOCA, especially:

- The pressurizer and its surge line,
- The RHRS,
- The ECCS,
- The drain and vent systems on the vessel and primary legs.

The complete list is in attachment 3. For each system, the most probable leak sections were considered, based on the “weak points” of the circuit (valves, small diameter welded pipes...). Three representative accident scenarios were calculated for each postulated breach with ASNR’s simulator SOFIA. Only one reactor design was considered (P4 PWR-1300 MWe). The output data included the time evolution of:

- The characteristics of the flow at the breach: enthalpy, velocity, temperature, void fraction, debit.
- The pressure in the room where the breach is located,
- The time at which the CSS is activated.

This basic data can be used by WP3 contributors to characterise the thermal-hydraulic conditions in which their sensors will operate. The CSS is actuated when the pressure in the containment reaches 2.5 bars, which can happen one minute after the start of the accident. The actuation of the spray will lead to complex fluid movements in the reactor building, mixing the atmosphere rapidly. It is assumed that the localisation of the breach will no longer be possible in those conditions, imposing the sensor to respond before the CSS is actuated.

The precision of the sensor is derived from the two objectives defined early: isolate the leak and inform emergency response. So, the system should be able to distinguish the origin of the leakage between the broken pipe and the closest intact pipes. It should also be able to localise the closest isolation valves, if there are any. Isometric plans were analysed to specify the precision of the sensor. Due to the diversity of situations, defining a general localisation

³ Leakages under a certain threshold will not trigger the automatic RCS isolation, even if they require the application of EOPs. In this case, a break on the CVCS must be considered.

accuracy was hard. Nevertheless, determining the localisation of the breach within 10-100 cm seems a reasonable target.

The SOFIA simulator does not calculate local thermal-hydraulic conditions in the containment, so more precise simulations will be necessary to model the jet coming from the breach. Furthermore, the jet can interact with its environment. Most primary pipes are insulated: a large leakage will quickly remove the insulation, but a smaller one can diffuse in it. The jet can also hit a wall and be dispersed, making its origin less clear.

The acquisition system will face an aggressive environment. It must first resist to the temperature and radiations during the normal operation of the reactor (see deliverable D2.1 for more information). The excess radiations received during the accidental transient are probably negligible, because the system must have a short response time and because the activity in the primary coolant is kept at low levels during exploitation. The system is relevant for the early phases of a DBA only. During the accident, the system will have to withstand high-temperature and high-pressure steam (up to 155 bar and 360 °C), but for a relatively short time. The response time of the sensor must be shorter than the delay of actuation of the CSS, as explained above. If the sequence does not lead to the actuation of the CSS, a response time of 15' seems sufficient, considering the typical reaction time awaited from operators.

The system developed in FIND should be as little intrusive as possible. It should use as few containment wall penetrations as possible and minimal energy.

The document focuses on large breaks, which are more important for safety. However, detecting smaller leaks may support plant operation. Leakages between given boundaries (for example, between 230l/h and 2,300l/h on French reactors) require a plant shutdown within a given delay, without applying EOPs. A leak detection system would facilitate the diagnosis of the problem and help fix it faster. Leakages are also detected during plant shutdown and restart operations, thanks to visual inspection by operators. An earlier detection would not only help mending problems faster, but also protect workers against hazards.

2.2. Specifications

General information			
Component(s) concerned	RCS, RHRs, ECCS, pressuriser	Degradation mechanism	Breach leading to a LOCA
Safety function	Core cooling		
Type of measure	Accidental instrumentation	Reactor designs concerned	PWR
Lead contributor	ASNR	Other contributors	Vuez, VTT, SSTC-NRS
Current methods applied	FLÜS system (not safety relevant) Indirect methods (Fire alarms, radiation monitoring)	Limitations of current methods	No safety relevant system.
Fluid to be detected			
Type of fluid	Primary water	Temperature	300-360 °C
Velocity	Sonic jet	Pressure	PWR: 155 bar VVER-440:123 bar

Chemical composition	Boric acid	Physical state	Two-phase under expanded jet
Magnitude to be measured for accidental instrumentation			
Magnitude	Breach localisation		
Precision	10-100 cm	Response time	<1 min
Sources of signal interference			
Jet dispersion by insulation of obstacles		Cooling water injected by the CSS	
The system must discriminate a leakage from a fire.			
Sensor environment during the accident (see D2.1 for normal operating conditions)			
Temperature	Up to 360 °C	Humidity	100%
Pressure	1-2.5 bar	Irradiation	Not significantly higher than in normal conditions
Main safety constraints on hardware	Fire resistant / no corrosive product if burnt	Electricity supply	230-380 V (AC) 24V-48 V (DC)
Data transfer	Postulated accidents usually assume a LOCA is not combined with an SBO (Station Blackout), because the combination of both events is unlikely. So, safety-classified I&C can be considered available. Digital and analogue data transfer 230AC and 24-48V DC power		
References			
Ref: Scientific publications	Development of innovating Na leak detector on pipe, M. Girard, D. Cambet, S. Albaladejo, G. Laffont, O. Carra		
Ref: National and international standards	RCC-E: Règles de Conception et de Construction des Systèmes et Matériels Electriques et de Contrôle Commande (Standards for the design and construction of electric and I&C systems and materials) IEC 63147:2017. Criteria for accident monitoring instrumentation for nuclear power generating stations		
Operating experience	Krško NPP (Slovenia) : - Leakage on the safety injection system 1 m from the vessel of the on 5 th October 2023. - 3 m ³ /h leakage on the primary circuit near the primary pump on 4 th June 2008. See the following cases described in D2.1 of FIND: - Flamanville 2025-03-25 - Civaux 1998-12-05		
Product sheet	The Spinline platform developed by Framatome is an example of I&C system used for reactor protection systems.		

2.3. Additional operating experience feedback regarding location of a breach during a LOCA

It is worth noting that approximately 114 events mentioning a breach on the reactor coolant system are recorded in the PIREX database. Not all these events necessarily led to significant leakage rates nor a LOCA, but their number does not decrease with time.

Situations identified in paragraphs 4&5 of FIND’s deliverable “D2.1 - Specifications for the online damage monitoring systems” (24/10/2025) may be relevant for this use-case too.

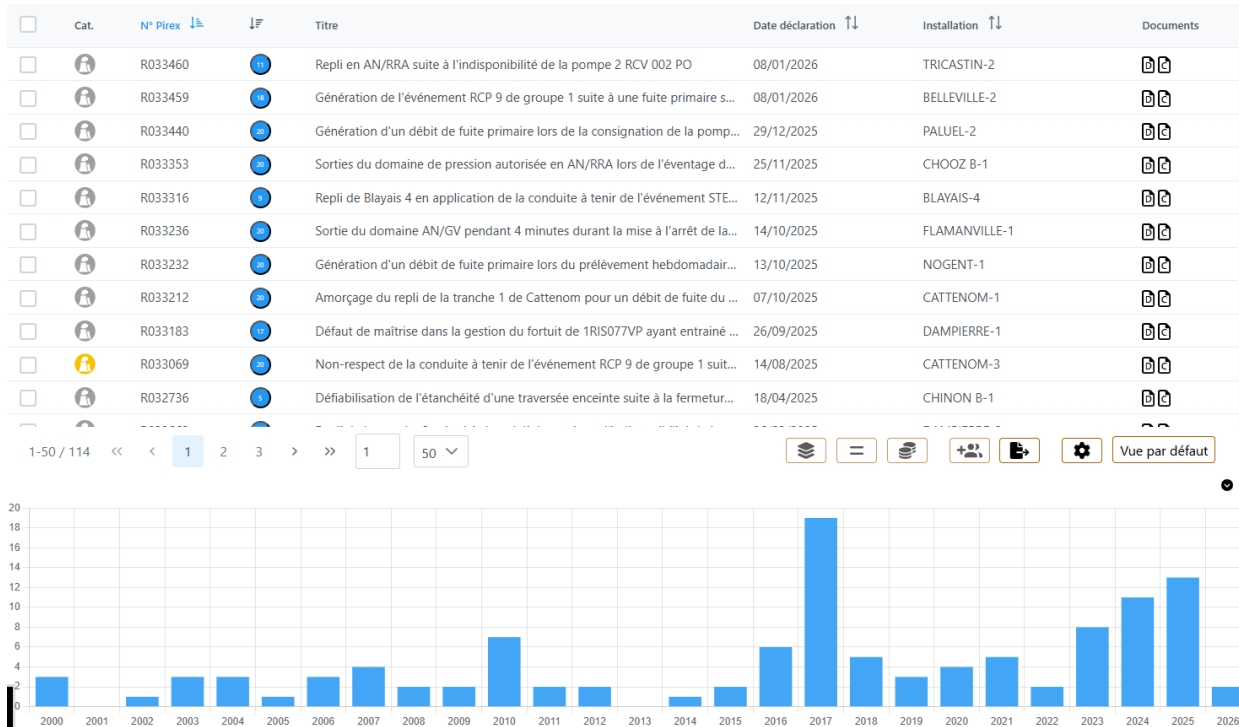


Figure 2-1: Histogram of safety events involving potential primary leakages in France since 2000

Date: 2025-04-15

NPP: CHINON

Title: Defective sealing of a containment penetration following incomplete closure of a containment isolation valve.

What happened: On April 15, 2025, while the reactor of Chinon unit 1 was undergoing pressure and temperature ramp-up in the normal shutdown on steam generators state, a technician detected a steam leak in the Reactor Building. The leak originated from the incomplete closure of a drain valve on a containment penetration of the Nuclear Sampling System (NSS). One month earlier, the field technician had not sufficiently closed this valve due in part to the difficulty of accessing the valve, located behind a support structure itself set back from protective grids.

Consequences: Following the leak, the total volume of primary cooling water spilled was approximately 310 liters. The reactor building was contaminated.

Interest: The breach has been detected by a technician who called to report a steam leak in the Reactor Building. Simultaneously, several fire detectors triggered false alarms related to the steam leak. The breach was located by a field inspection with entry into the reactor building, revealing a steam jet coming from the Nuclear Sampling System piping. **The total time between the detection and the containment of the leak was 4 hours and 20 minutes.**

Date: 2023-12-25

NPP: BELLEVILLE

Title: Reactor shutdown following the detection of an unquantified leak rate exceeding 230 l/h

What happened: On December 25, 2023, an operator detected a drop in the level of the Chemical and Volume Control System (CVCS) tank. An estimate of the leak rate was performed, concluding that the leak rate was approximately 260 l/h. As the leak was not isolated two hours after its detection, the shutdown of Unit 1 was initiated, following the established procedure. Leak detection on the installation identified a leak at the flow meter on the CVCS boremeter sampling line. The subsequent investigation determined that a loose bolt on a sensor of the NSS was the cause of this leak.

Consequences: Shutdown of the reactor in accordance with procedures in order to bring the installation to a state where the residual power is limited in the event of a primary leak escalating into a Loss of Cooling Accident (LOCA).

Interest: The day before the incident, **a drop in the level in the CVCS tank began during the night, but it remained undetected for 12 hours.** No alarm was triggered until the morning operator detected the decrease in the tank level. The leak was located after 3 hours during an on-site inspection and repaired 30 minutes later. **The total time between the onset and the containment of the leak was 15 hours and 30 minutes.**

Date: 2024-10-05

NPP: SAINT-ALBAN

Title: Accumulation of 2 anomalies on the Reactor Coolant System (RCS) resulting in a primary leakage rate exceeding 230 l/h

What happened: On October 5, 2024, Unit 1 was in a hot shutdown state during the recovery phase of a previous outage. The operations team detected a drop in the level of the Chemical and Volume Control System (CVCS) tank with an overall leak rate of 1500 l/h. The apparent cause was a leak in the press-fit valve and RCS valve's leak return tube. The root cause identified was a failure of a motorized valve on the primary circuit following a maintenance failure during a ten-year inspection in 2017.

Consequences: Without detection of the increase in primary leakage rate, the level of the CVCS tank would have continued to decrease, leading to a degradation of the primary water inventory.

Interest: The shift team detected a drop in the CVCS tank level and calculated a primary leak rate of 1500 l/h. **No fire alarm was triggered, although some of the leak was escaping as steam.** It is not possible to precisely date the appearance of the puncture that led to this steam leak. **The total time between detection and isolation of the leak was 5 hours with approximately a total volume of primary cooling water of 1500 liters.**

3. Analysis of fission products in the containment building

3.1. Rationale

A SA is characterised by an extensive damage of the reactor core, leading to the release of large quantities of radioactive fission products from the fuel. Only three SAs have occurred in nuclear power reactors: The Three Mile Island and Fukushima Daiichi accidents. The real situations, emergency drills and hypothetical SA scenarios studied by the nuclear community confirm one point: SAs are confusing situations characterised by a high level of uncertainty. Indeed, SAs are associated with extreme situations, caused by a highly improbable combination of circumstances or cataclysmic external conditions. Only few instruments are designed to

operate in such circumstances. It is very difficult to infer the state of the core from the information they provide, complicating onsite and offsite decision making.

Most NPPs are equipped with the following sensors to monitor the progress of a SA:

- The core exit temperature sensors: they can detect when the core reaches a temperature of 1,000 °C, indicating the start of the fuel cladding runaway oxidation, an exothermic chemical reaction accelerating core melting,
- The containment dose rate sensors, indicating the level of radiation in the containment,
- Temperature sensors in the reactor pit, indicating if the primary vessel has failed and if corium has escaped from it (not all designs),
- The containment pressure sensor, indicating if the design pressure of the containment is reached and if a filtered containment venting is necessary to ensure its integrity.

To a certain extent, the measure of the dose rate in the containment can be correlated to important phenomenological times of a severe accident: the extensive failure of fuel claddings or the extensive core melting. However, this information is indirect and difficult to interpret. In addition, the dose rate does not inform about the composition of the atmosphere in the reactor building.

FIND proposes to develop a sensor able to detect in the reactor building some chemicals of major importance for the management of a SA.

The 1st chemical of interest is molecular iodine (I_2). This volatile chemical can have dramatic offsite radiological consequences. Its gaseous form makes it hard to trap by the different plant mitigation systems: CSS, Filtered Containment Venting System (FCVS). In particular, FCVS using sand-bed filters can only trap the aerosol forms of iodine. In addition, molecular iodine has a high biological affinity with thyroid, meaning that the human body will trap and keep this radioactive element for a long time and possibly causing thyroid cancers. To mitigate this risk, emergency responders can recommend the population to take iodine tablets and confine itself or decide an evacuation. Considering the risks associated with a too broad or too narrow evacuation, more precise information about the source term is highly desirable. Unfortunately, current simulation tools used by emergency response centres are unable to accurately predict the iodine content in the reactor building during a SA. First, the exact stage of the accident progression is difficult to diagnosis precisely. Second, the chemistry of iodine in the reactor core, the circuits and the containment building is very complex and can only be modelled with large uncertainties. This is why measuring the quantity of iodine in the reactor building is so crucial.

Methyl iodide (CH_3I) has some similar characteristics with molecular iodine. It will also be considered in FIND.

The fission reaction in the nuclear reactor produces large quantities of heavy noble gases, especially xenon. Since it is a chemically inert gas, it is released very rapidly when fuel claddings start to fail. It is also impossible to trap, so it will be directly transferred to the reactor building and remain there until the containment is vented. In this case, it will be largely transferred to the environment without any retention. Xenon has no affinity with biological tissues and cannot deposit on the soil or in the water. Hence, its contribution to the radiological exposure comes only from external irradiation, through the phenomenon known as “the cloud shine”. In addition, the characteristics of xenon make it a good indicator of the progression of the accident. The first clad breaks will release the xenon in the fuel-clad gaps and in fuel cracks. The fuel melting will free the xenon that is trapped in the fuel matrix, leading to much higher releases. In most

cases, the released xenon will rapidly be transferred to the containment building⁴ the pressuriser relief valve². So, the concentration of xenon can relatively easily be correlated to the state of the core. ²²²Rnall of them². Hence, FIND proposes to develop a method to quantify xenon in the reactor building during a SA.

Calculations detailed in appendix 1 were used to determine the useful measurement range of the SA sensor to fulfil the previously mentioned objectives. They are expressed in terms of volumetric activity and mass concentration in the table below. It must be noted that iodine and xenon produced during the accident are composed of both stable and radioactive isotopes. The mass concentrations to be detected can be extremely small, whereas the activity is huge, due to the high specific activity of the short-lived fission products.

The SA conditions and practical constraints for the sensor implantation have been described in section 1.1. Two main types of measurement methods can be envisaged: chemical and radiological analysis. The two options face different challenges. The containment building contains a large variety of radioactive elements during a SA. This results in very complex radiological spectrum to analyse. They have been characterised by the DECA PF and PHEBUS PF projects for example. On the other hand, the signal to be measured is huge. On the contrary, the mass concentration of chemicals can be very low. Many chemicals are in aerosol state, which can perturbate the sensor. The purely gas phase has a relatively simple composition, dominated by air, steam, hydrogen, carbon monoxide, NO_x and noble gases. Iodine and bromine are the most prominent gaseous trace elements.

These objectives would complement the results of previous research projects. DECA PF developed various measurement methods to analyse the source term escaping through the FCVS during a filtered release. The conditions faced by the sensors were less aggressive than those considered for FIND. However, the system would be blind before the actuation of the FCVS, limiting its support to anticipation and decision-making. MITHYGENE developed a system to measure combustible gases in the containment building, based on a relatively complex Raman spectroscopy system. The necessary detection limit for iodine is several orders of magnitude lower than those considered in MITHYGENE.

Apart from the reactor building atmosphere, other sensor localisations are of interest. Molecular iodine can also be dissolved by water and collected in the reactor sumps. Measuring its concentration in the liquid phase will support the long-term management of effluents. If the sensors have sufficiently low detection limits, they could be used out of the reactor building to detect uncollected contaminated gaseous or liquid leakages, as happened during the Three Mile Island accident. This would confirm the tightness of the containment building and the good operation of the automatic containment isolation systems.

A sensor able to detect traces of iodine in a harsh environment and with a low electric consumption may interest other types of nuclear installations. Iodine is a volatile and extremely mobile chemical that is hard to confine. The sensor could be used in fuel reprocessing facilities or in nuclear waste interim storages or geological repositories. In this context, only stable and long-lived iodine isotopes can be measured, because of radioactive decay. For countries relying on direct storage of fuel assemblies, stable xenon can indicate a loss of fuel cladding integrity, even if it is not dangerous. Dedicated specifications need to be defined for such use-cases,

⁴ Severe Accident Management Guidelines (SAMGs) recommend depressurising the primary circuit as soon as SA conditions are reached. The objective is to avoid a high-pressure vessel failure that can lead to a containment failure and the dispersion of the corium, a phenomenon known as “Direct Containment Heating” (DCH).

which are not considered in this document. The targeted detection limits would probably be significantly different.

3.2. Specifications

General information			
Localisation	Dome	State of the reactor	SA, especially early phases (from clad failure to fuel melting)
Type of measure	Accidental instrumentation	Reactor designs concerned	All
Lead contributor	ASNR	Other contributors	VTT, SSTC-NRS, CEA
Current methods applied	Core exit temperature Dose rate measure	Limitations of current methods	Indirect measurement, no information on the composition
Magnitude to be measured for accidental instrumentation			
	Iodine as gaseous I ₂	Iodine as gaseous CH ₃ I	Xenon
Volumetric activity (Bq/m ³)	10 ⁷ to 10 ¹²	10 ⁷ to 10 ¹¹	10 ¹¹ to 10 ¹⁴
Mass concentration (g/m ³)	10 ⁻⁷ to 10 ⁻²	10 ⁻⁷ to 10 ⁻³	10 ⁻² to 10 ¹
Precision	Factor 2 between measurement and real value	Factor 2 between measurement and real value	Factor 2 between measurement and real value r
Response time	10 min according to SAMGs		
Sources of signal interference			
Chemical compounds (gas phase)	Air, steam, H ₂ , CO, Xe, Kr, NOx	Aerosols	Concentration: 10 g/m ³ Various composition (1/3 metals, 1/3 metallic oxides, 1/3 more complex compounds) Ag, U, In, Cd, I, Mo, Cs dominate the composition of aerosols (depending on the type of control rods used) Size: micrometric
Ionising radiations	Very complex signal, mixing alpha, beta and gamma rays of various energy ranges.		
Sensor environment in accidental conditions (see D2.1 for conditions in normal operation)			
Temperature	PWR: up to 150°C in gaseous phase	Humidity	Up to 100%

	VVER: up to 250°C		
Pressure	PWR: Up to 6 bars (abs.) Up to 3 bar (abs.) in Loviisa VVER-440 VVER: Up to 10.5 bar (abs.)	Irradiation	Few MGy for irradiation during several days
Main safety constraints on hardware	Fire resistant / no corrosive product if burnt	Electricity supply	Low voltage DC power (24-48V)
Data transfer	Analogical with rad-hard cables. Optic fibres without local electronic treatment.		
References			
Ref: Scientific publications	<p>G. Ducros, P.-G. Allineï, C. Roure, C. Rozel, N. Blanc de Lanaute, et al.. On-line Fission Products measurements during a PWR severe accident the French DECA-PF project. ANIMMA 2015 - 4th International Conference on Advancements in Nuclear Instrumentation Measurement Methods and their Applications, Apr 2015, Lisbon, Portugal. (cea-02509258)</p> <p>Sylvain Magne, Ahmed Bentaib, Leroy Matthieu, Emmanuel Porcheron, Etienne Studer, et al.. Fiber-coupled nuclearized Raman probe for remote gas monitoring in nuclear containment: proof-of-concept for h2-risk management of design and beyond-design accidents. <i>Specialist Workshop on Advanced Instrumentation and Measurement Techniques for Nuclear Reactor Thermal Hydraulics and Severe Accidents</i>, Jun 2024, Dresden (GERMANY), Germany. (irs-n-04858123)</p> <p>Sylvain Magne, Simon Nehr, Xavier Buet, Ahmed Bentaïb, Emmanuel Porcheron, et al.. “Fiber optics remote monitoring of gas concentrations by Spontaneous Raman Scattering - Application to the H2-risk management in nuclear containment during a severe nuclear accident”. <i>International conference on Advancements in Nuclear Instrumentation Measurement Methods and their Applications - ANIMMA 2019</i>, Jun 2019, Portorož, Slovenia. pp.17 - 20. (hal-02413772)</p> <p>M.P. Kissane, On the nature of aerosols produced during a severe accident of a water-cooled nuclear reactor, <i>Nuclear Engineering and Design</i>, Volume 238, Issue 10, 2008, Pages 2792-2800, ISSN 0029-5493, https://doi.org/10.1016/j.nucengdes.2008.06.003.</p>		
Ref: National and international standards	<p>RCC-E: Règles de Conception et de Construction des Systèmes et Matériels Electriques et de Contrôle Commande (Standards for the design and construction of electric and I&C systems and materials)</p> <p>IEC 63147:2017. Criteria for accident monitoring instrumentation for nuclear power generating stations</p>		

Product sheet	The Hardline platform developed by Framatome is an example of I&C system specifically suited for severe accidents.
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Conclusion

D2.2 gives detailed specifications for the acquisition systems to be developed in FIND. It will be the basis for the research activities carried out in WP3 and WP4. In addition, it can be used by other technology providers to propose alternative solutions, leveraging the principles of open science.

The deliverable highlights the benefits of innovative instrumentation to improve accident management beyond existing safety demonstrations. Other use-cases will be proposed in D2.3. The characteristics of the systems significantly vary depending on their objectives: rapid or low response time, high or medium accuracy, more or less stringent environmental operating conditions... All constraints must be discussed thoroughly with specialists before envisaging any technological developments. In many cases, these requirements can drastically affect the choice of the sensor's physical principle. They also call for dedicated experimental test facilities.

Appendix 1: Severe accident calculations supporting the definition of specifications

1. Scenarios considered

To characterize sensor environment for water level measurement in sumps and for fission products analysis in the containment atmosphere, 5 accidental sequences performed with ASTEC code have been selected from a level 2 PSA (Probabilistic Safety Analysis) study performed by ASNR. The sequences have been calculated with ASTEC versions 2.2. The main criterion of selection is the core degradation kinetics. Indeed, it has an important effect on which fission product will be measurable in the containment atmosphere, due to differences in radioactive decay half-lives. No specific criterion has been highlighted for water measurement because sensor environment (pression, temperature, humidity ...) in the sumps is quite similar whatever the severe accidental sequence is. One additional sequence, calculated with ASTEC version 3.1, has been selected because of FCVS opening. This additional sequence has also been calculated on a longer duration. The sequences selected are the following:

- Power_Slow: loss of all voltage electrical power (resulting in the loss of SIS) and small breach at primary pumps gaskets at power state, which leads to a slow core degradation;
- Power_Average: loss of coolant accident at power state, with a small hot leg breach and loss of SIS in the recirculation mode, which leads to an average kinetics for core degradation;
- Power_Fast: loss of all voltage electrical power (resulting in the loss of SIS) and large breach at primary pumps gaskets at power state, which leads to a fast core degradation;
- Power_FCVS: large primary breach and loss of SIS in the recirculation mode, which leads to a fast core degradation, with FCVS (Filtered Containment Venture System) opening;
- Shutdown_Slow: loss of electrical power at shutdown state with safety injection partially available (SIS lost 24h after the start of recirculation mode), which leads to a slow core degradation;
- Shutdown_Fast: loss of electrical power at shutdown state (resulting in the loss of SIS), which leads to a fast core degradation.

The 6 sequences have been calculated on a 4-loop PWR.

The following table gives delays between the beginning of fission products release and vessel failure, which is representative of core degradation kinetic.

Sequence	Delay (h)
Power_Slow	26
Power_Average	5
Power_Fast	2
Power_FCVS	2
Shutdown_Slow	43
Shutdown_Fast	3

Table 3-1: SA scenarios considered to evaluate fission product concentration in the reactor building

2. Figures of merit

For each of the 6 sequences, the figures below present:

- The gaseous dose rate in the containment for four zones:
 - Sumps (SUMP_G),
 - Reactor pit (CAV_G),
 - Top of the containment (DOME_G),
 - Annular space, which is a free zone under the top of the containment near walls in interface with the environment (ANH_G);
- The total activity (sum of all isotopes) in all the containment for three phases: suspended aerosols (Aerosols), gaseous phase (Gas) and liquid phase (Liquid).

Releases in the containment of fission products from the gap between fuel claddings and fuel pellets have been calculated in those sequences, but the activity in the containment due to fuel claddings rupture is only available for the sequences with a slow core degradation. For sequences with average and fast degradation, core melting begins too early and it is not possible to separate activity due to fuel claddings rupture and activity due to core melting.

Those data are presented from the beginning of the fission products release (t=0) until the vessel failure and include two key moments: the beginning of the core melting and the moment at which 50% of the core is dislocated. The table below presents the characteristic moments of each sequence with the start of fission products release as initial time. The beginning of core melting time and the moment at which 50% of the core is dislocated could not be calculated for the sequence Power_FCVS.

Sequence	Time (h)			
	FP release	Core melting	50% dislocated	Vessel failure
Power_Slow	0	5	19	26
Power_Average	0	0.3	2	5
Power_Fast	0	0.1	1	2
Power_FCVS	0	/	/	2
Shutdown_Slow	0	1	38	43
Shutdown_Fast	0	0.1	1	3

Table 3-2: Main phenomenological times of the considered SA sequences

Raw data is available upon request to ASN. Due to its sensitive nature, it could not be released openly.

2.1. Sequence Power_Slow

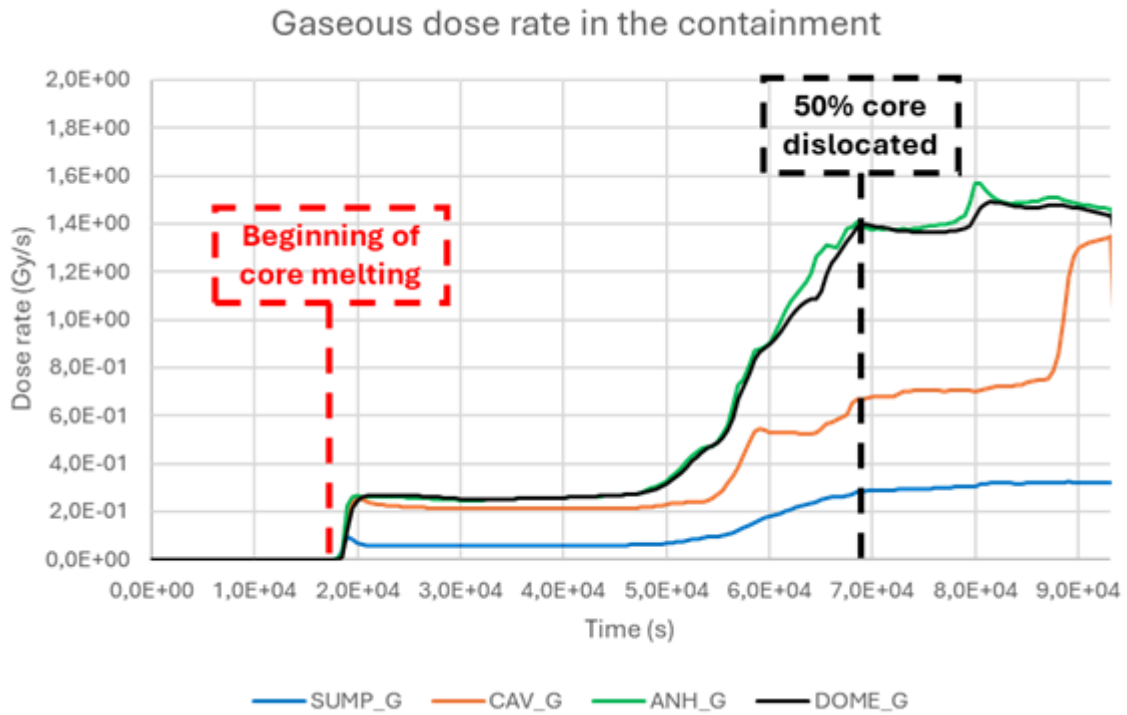


Figure 3-1: Gaseous dose rate in the containment for the sequence Power_Slow

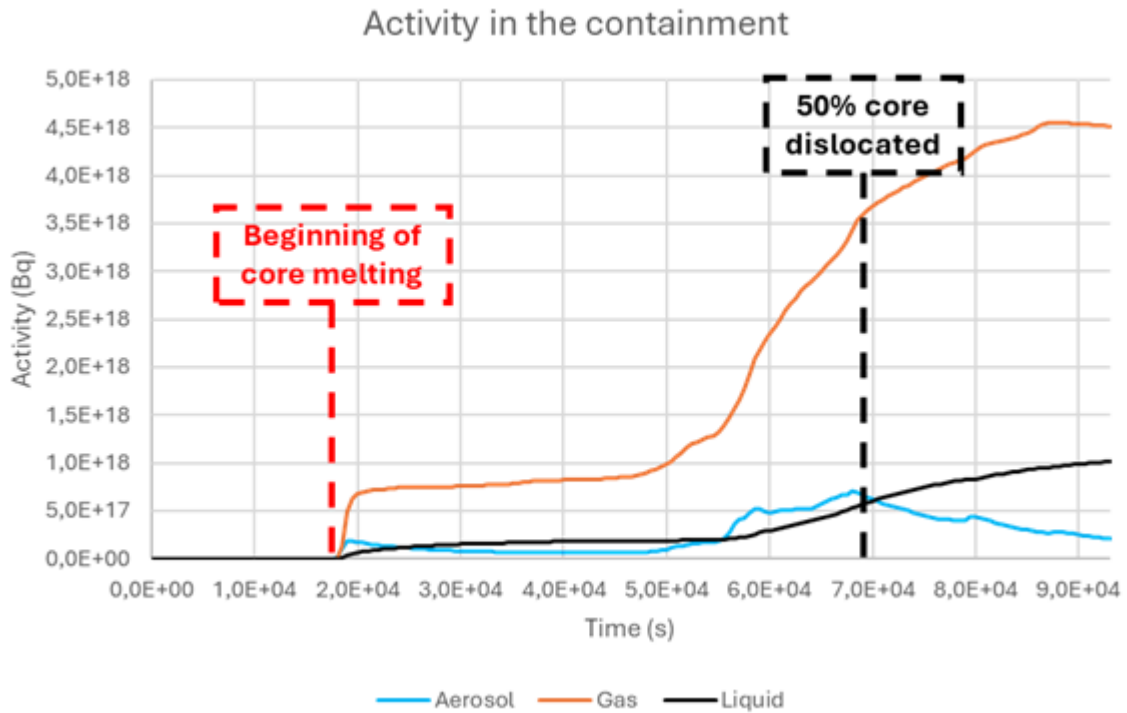


Figure 3-2: Activity in the containment for the sequence Power_Slow (linear scale)

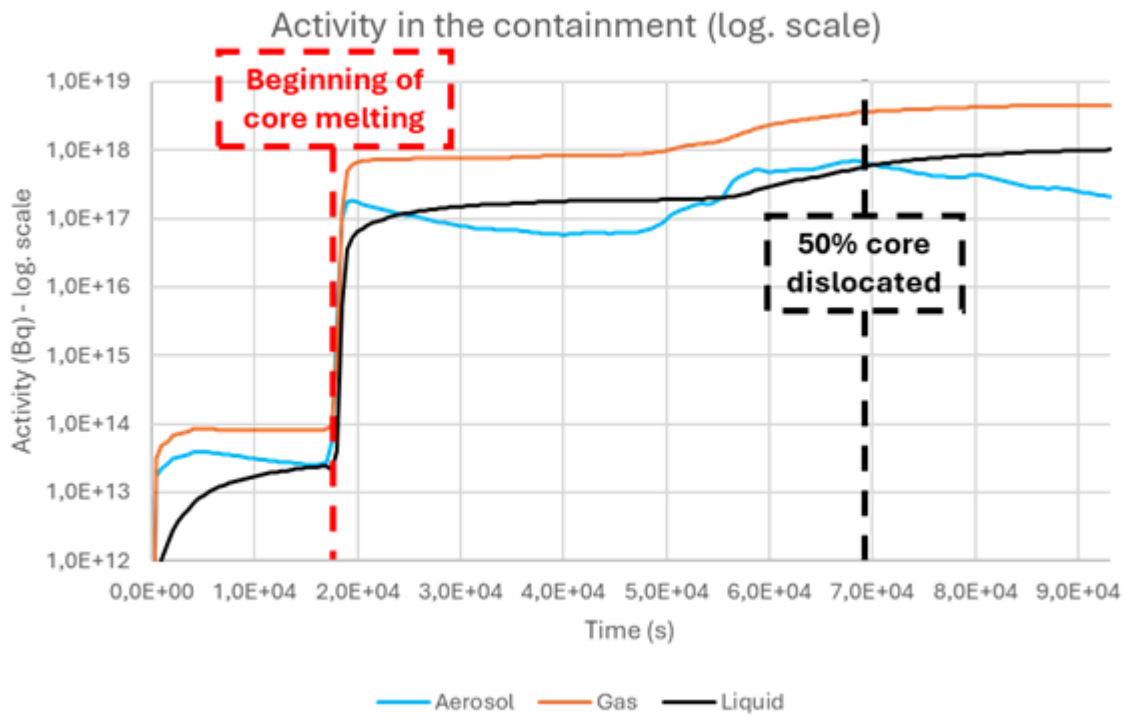


Figure 3-3: Activity in the containment for the sequence Power_Slow (logarithmic scale)

2.2. Sequence Power_Average

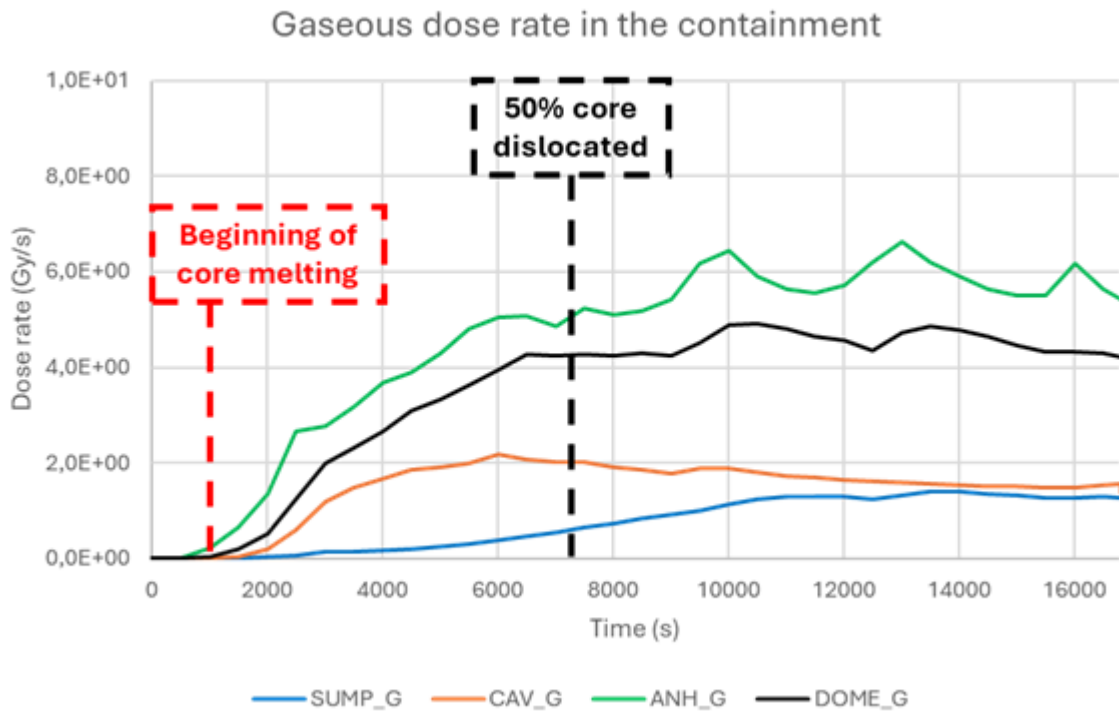


Figure 3-4: Gaseous dose rate in the containment for the sequence Power_Average

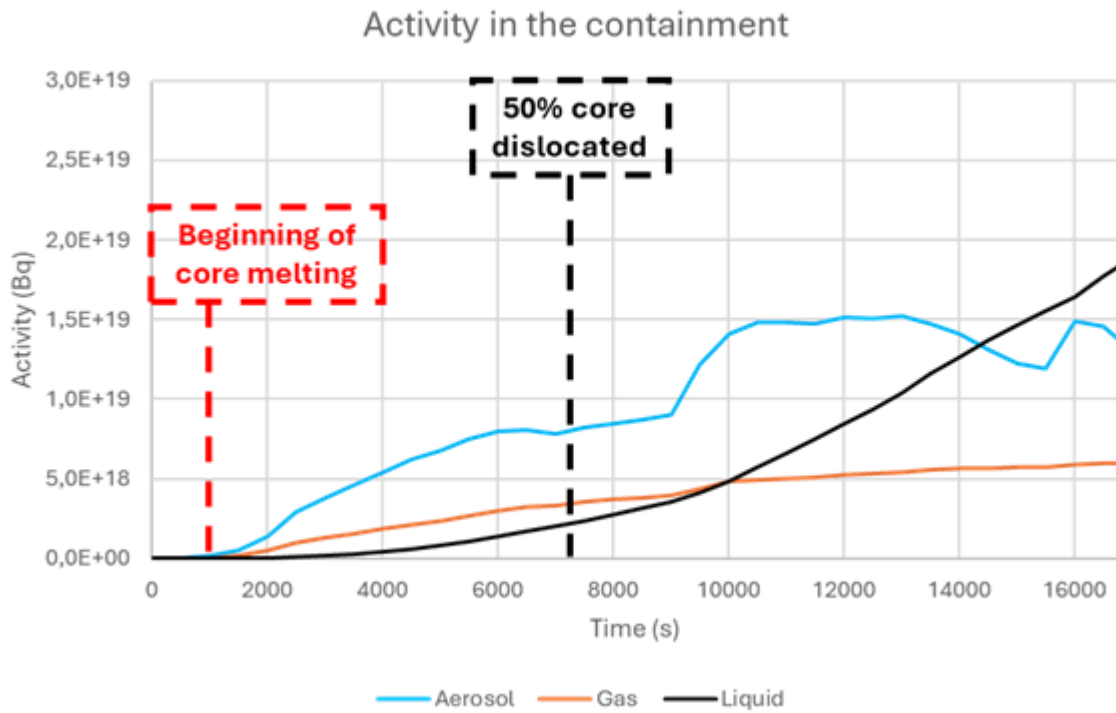


Figure 3-5: Activity in the containment for the sequence Power_Average

2.3. Sequence Power_Fast

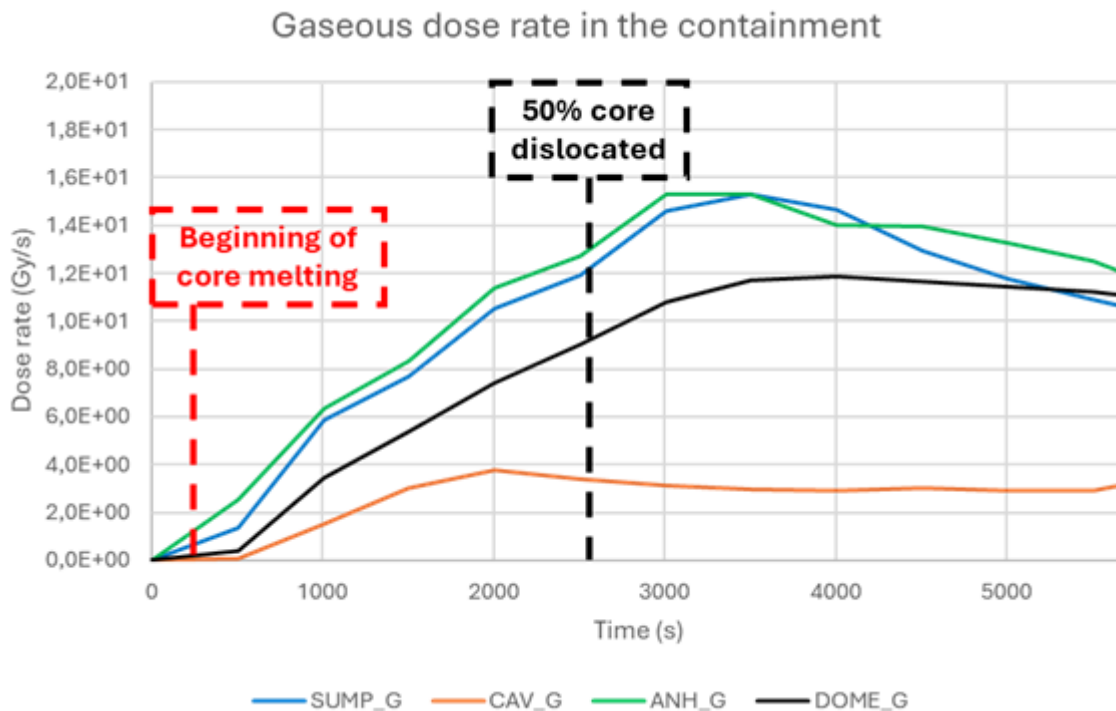


Figure 3-6: Gaseous dose rate in containment for the sequence Power_Fast

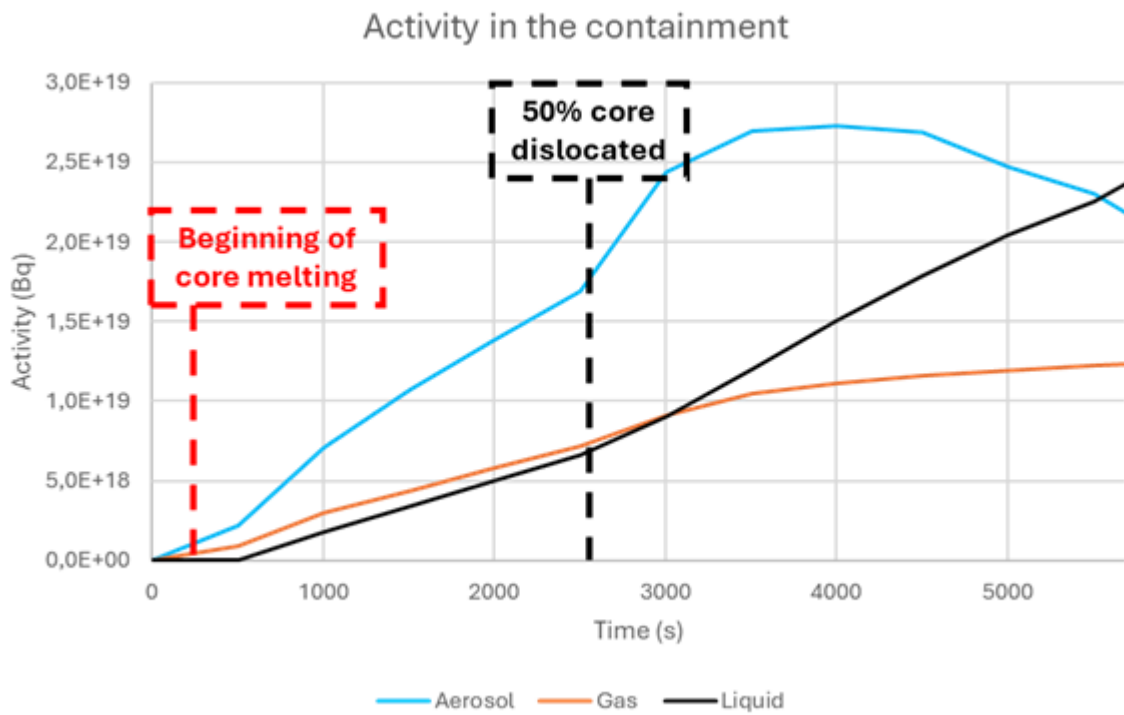


Figure 3-7: Activity in the containment for the sequence Power_Fast

2.4. Sequence Power_FCVS

For this sequence, the beginning of core melting time and the moment at which 50% of the core is dislocated are not presented because they could not be calculated.

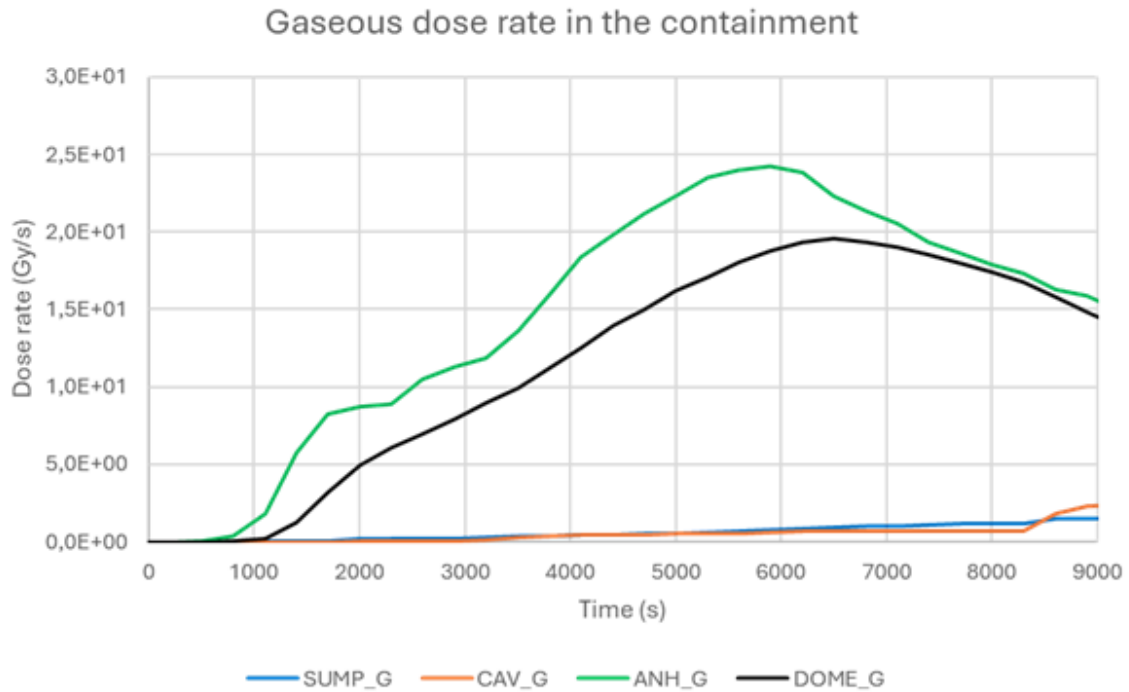


Figure 3-8: Gaseous dose rate in the containment for the sequence Power_FCVS

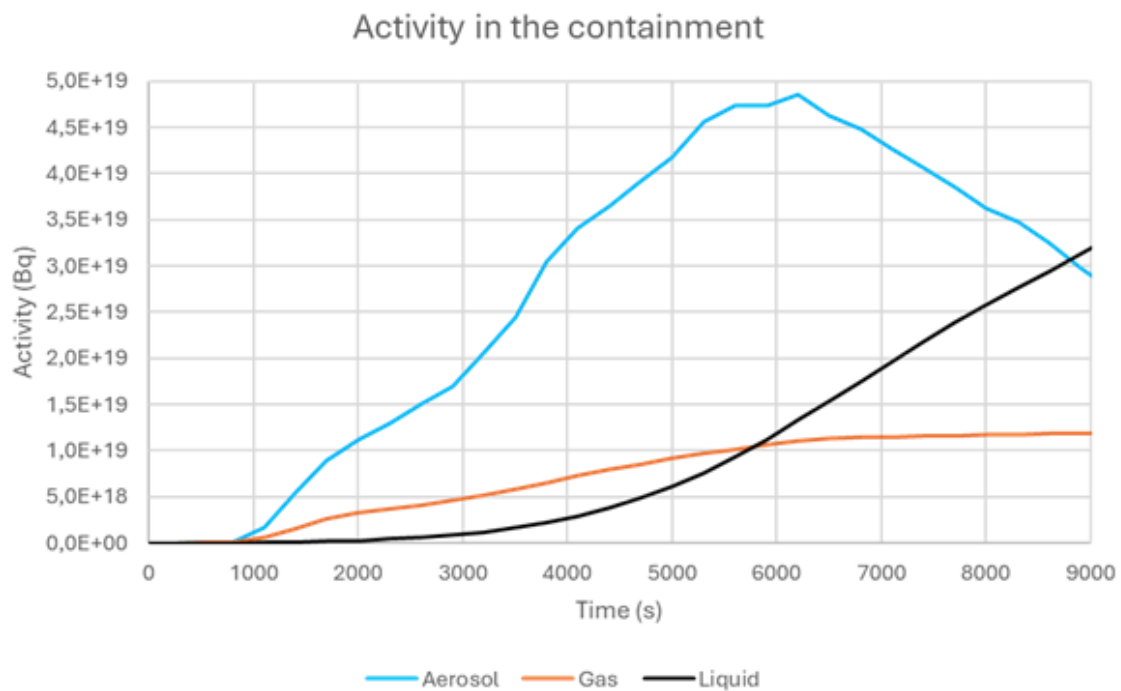


Figure 3-9: Activity in the containment for the sequence Power_FCVS

2.5. Sequence Shutdown_Slow

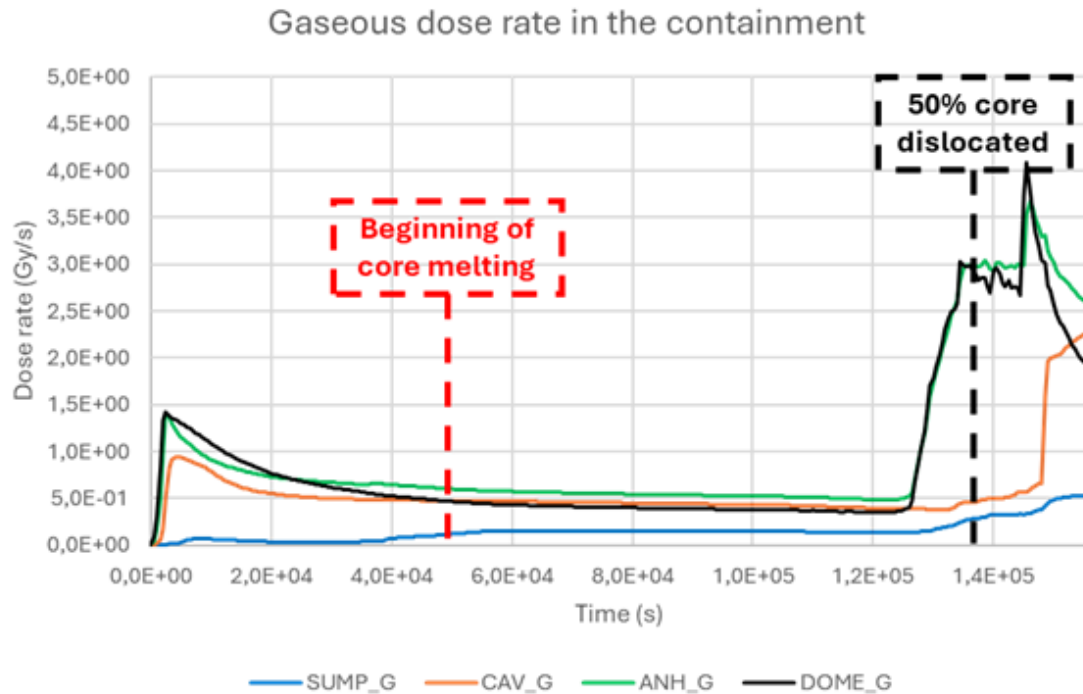


Figure 3-10: Gaseous dose rate in the containment for the sequence Shutdown_Slow

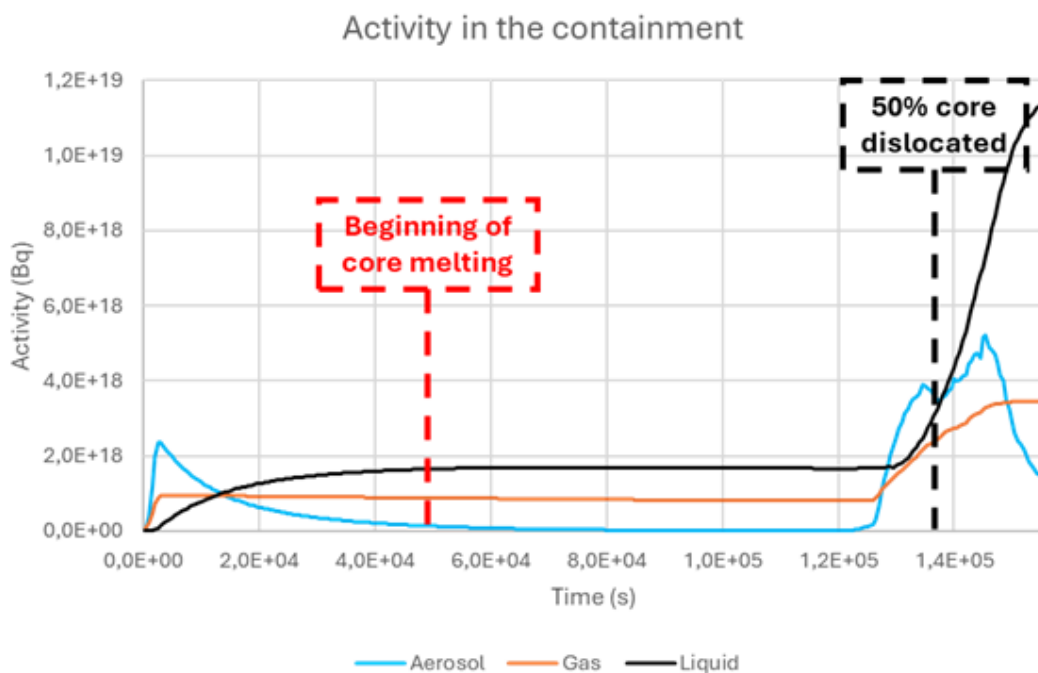


Figure 3-11: Activity in the containment for the sequence Shutdown_Slow

2.6. Sequence Shutdown_Fast

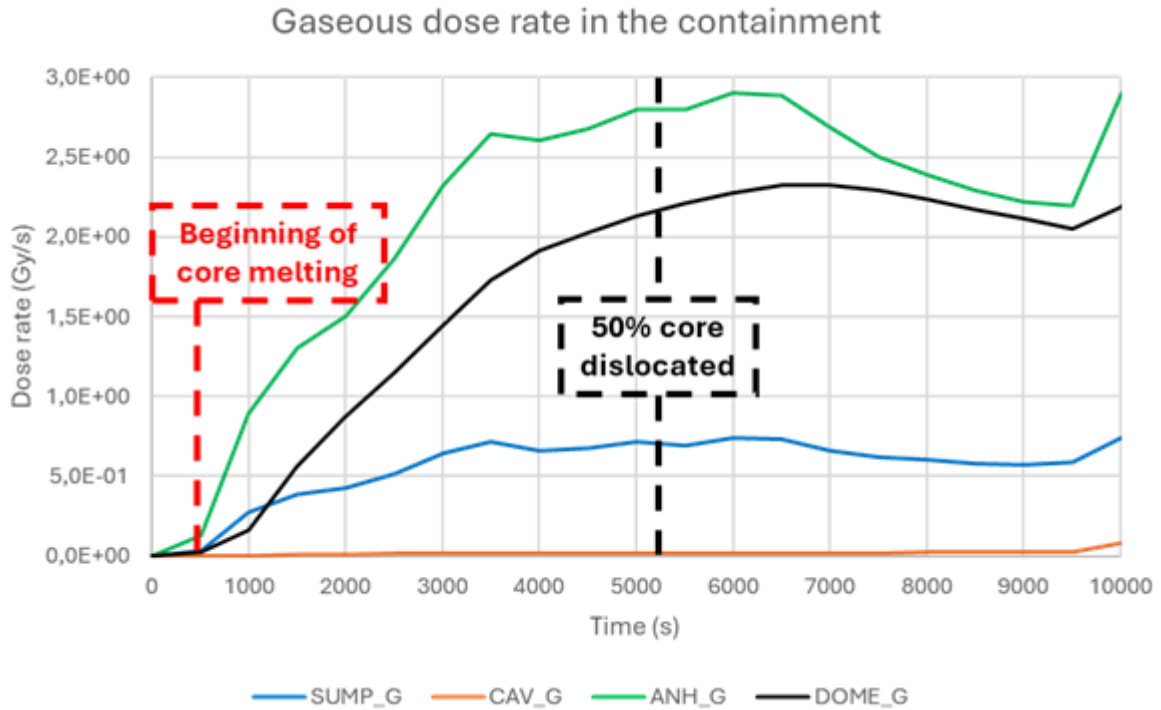


Figure 3-12: Gaseous dose rate in the containment for the sequence Shutdown_Fast

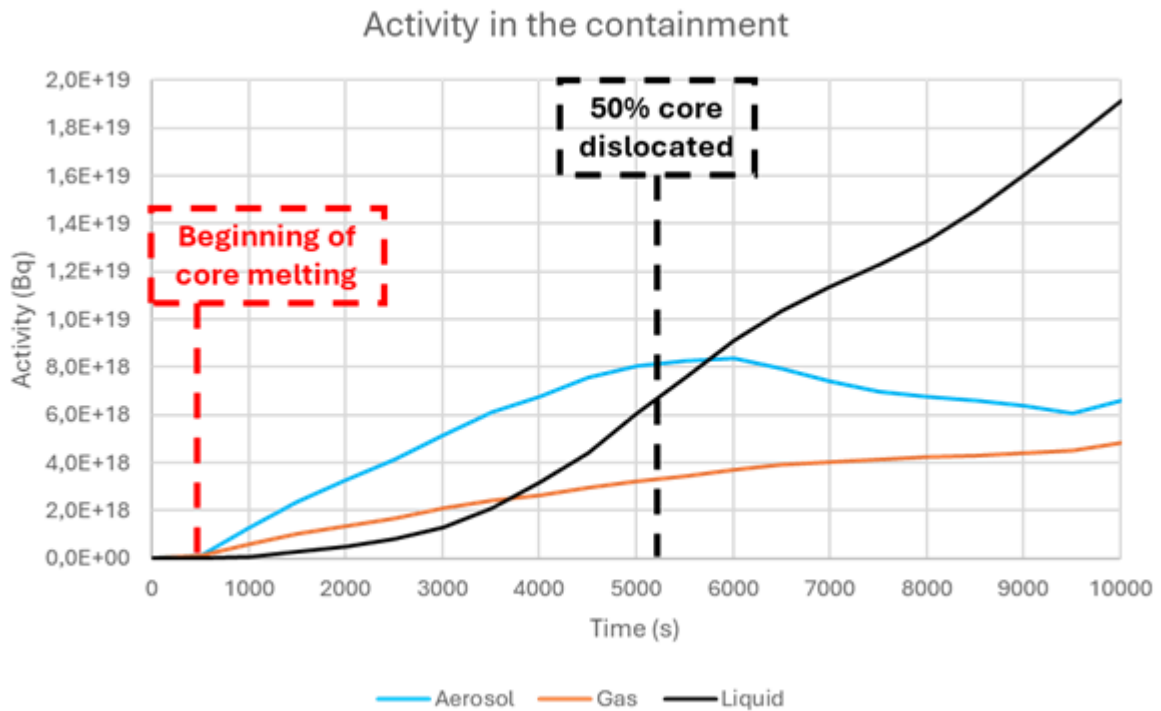


Figure 3-13: Activity in the containment for the sequence Shutdown_Fast

3. Iodine concentration in containment gaseous phase

Before the beginning of the core degradation, with about 10% of fuel claddings rupture, iodine concentration in the atmosphere of the containment should be at the lower bound range given in paragraph 3.2 of the main document.

During the core degradation, I_2 and CH_3I concentration in the containment atmosphere depends on iodine form released from the primary circuit, and especially the percentage of iodine released as gaseous I_2 , which is an uncertain parameter. An important iodine release rate as gaseous I_2 (about 5%) can lead to the following iodine concentration in the containment atmosphere:

- 10^{12} Bq/m³ or 10^{-2} g/m³ for gaseous I_2 ;
- 10^{11} Bq/m³ or 10^{-3} g/m³ for gaseous CH_3I .

After the degradation phase, and due to chemical reaction on iodine species, iodine concentration in the containment should be about:

- 10^{10} Bq/m³ or 10^{-4} g/m³ for gaseous I_2 ;
- 10^{10} Bq/m³ or 10^{-4} g/m³ for gaseous CH_3I .

Appendix 2: LOCA calculations supporting the definition of specifications

1. Introduction

This appendix presents the work done to identify the possible break locations on a French PWR of 1300 MWe and the characterization of three different breaks, which differ, because of their size and their locations.

This work was performed with help of ASNR's simulator SOFIA ("Simulator for Observation of Functioning during Incident and Accident"). This tool is used at ASNR to carry out studies related to the expertise in PWRs. With SOFIA, ASNR owns a representative platform of 900 MWe, 1,300 MWe, 1,450 MWe reactor types, as well as the EPR. SOFIA allows simulating a wide range of situations from cold shutdown states to full power state. Thanks to the modern and modular data-processing architecture of SOFIA, the configurations in conformity with the actual state of the nuclear power plants are easily kept up to date, by introducing the latest modifications carried out on French NPPs.

First, the work aimed to identify the different possible locations for a break is presented. After, three possible breaks are characterized, in particular in terms of impact on the containment conditions.

2. Possible break locations

This table shows the different connections of the primary circuit with systems, which are represented in the CATHARE 2⁵ dataset of the SOFIA simulator for the PWR1300. These connections are therefore the most likely LOCA locations.

Thus, it is possible to simulate a break in the mesh where this circuit is connected. This table also indicates the initial pressure and temperature of the circuit at Full Power Steady State (100%NP) in the mesh where the break will be simulated. The different potential break locations are represented on the figure below, which represents the modelling of the primary and secondary circuits with CATHARE code.

The reactor building of an NPP is separated in different rooms surrounded by concrete walls. Rooms in the reactor building do not have a confinement function, so they are often connected to one another by small openings, making it possible for steam to flow to the reactor dome (main volume of the reactor building) relatively quickly. Therefore, in SOFIA, the different rooms of the reactor building are not modelled individually. At a more local scale, the steam jet coming from a break would impact the walls of the room, modifying its trajectory. A reactor building is separated in several levels, spaced by 3 to 5 meters. Grating floors are often used, enabling a fluid circulation between the different levels.

⁵ The CATHARE code (CEA, EDF, FRAMATOME, ASNR) is dedicated to best-estimate modelling of thermal-hydraulic transients in nuclear reactors, using a combination of 0D, 1D and 3D components to represent the cooling systems. The code is based on a two-fluid, six-equation model, including equations for non-condensable gases and additional equations for the transport of radiochemical components. It also includes thermomechanical and neutronic modelling capabilities.

Three of these possible breaks have been simulated with the SOFIA simulator. It corresponds to:

- Break on the pressurizer surge line – transient n°1 (break n°1)
- Break on the Nuclear Island vent system on vessel head - transient n°2 (break n°11)
- Break on the connection of Pressurizer relief valves, situated on the top of pressurizer – transient n°3 (break n°12)

These 3 examples represent a diversity of situations, with small and large breaks, situated at different localisations, with different states of the coolant (subcooled water for the two first cases and steam for the last one). As already mentioned, the configuration of the room where the break occurs is not modelled in SOFIA. The section of the break is calculated as the area of a disc having a diameter corresponding to the size of the break.

Only breaks in the reactor building have been considered. Indeed, check valves prevent a break happening out of the reactor building to initiate a LOCA, preventing a direct containment bypass. A break on a safeguard system out of the reactor building (for example on the safety injection system in the safeguard auxiliaries building) is not considered here. Such break would most likely involve liquid water and not steam, calling for a different detection system. In this sense, we do not consider an aggravating factor. Similarly, all systems automatically isolated during a LOCA by the reactor protection system are ignored.

Break n°	Potential Break locations (1300MWe connections)	Size and location of the break	Size of the room where the break happens	Pressure (bar)	Temperature (°C)
1	Pressurizer surge line (406,4mm)	12 inches on Cold Leg 1	Level: 16.3m Area: 69.33m ² Height: 5.22m	156	324
2	RHRS inlet (Hot leg) (323,9mm)	12 inches on Cold Leg 3 or 4	Level: 13.32m Area: 53.78m ² Height: 13.50m	156	324
3	Safety Injection System a(Cold Leg) (273mm)	10,8 inches on Cold Leg 1/2/3/4	Level: 10.26m Area: 15.8m ² Height: 3.06m Ceiling level: 13.32m	159,6	289
4	RHRS outlet (Cold Leg) (273mm)	10,8 inches on Cold Leg1 or 2	Level: 10.26m Area: 52.12m ² Height: 13.50m	159,6	289
5	Safety Injection System (Hot leg) (273mm)	10,8 inches on Hot Leg1 or 2	Level: 10.26m Area: 52.12m ² Height: 21.84m Ceiling level: 32.10m	156	324
6	Pressurizer Spray (Cold Leg) (114,3mm)	4,5 inches on cold leg	Level: 32.76 Area: 19.14m ² Height: 4m	159,6	289

Break n°	Potential Break locations (1300MWe connections)	Size and location of the break	Size of the room where the break happens	Pressure (bar)	Temperature (°C)
		1 or cold leg 4			
7	CVCS inlet (Cold Leg) (114.3mm)	4,5 inches on cold leg3	See break n°3	159,6	289
8	CVCS outlet (U Leg) (88.9 mm)	3,5 inches on cold leg4	Level: 10.26m Area: 52.12m ² Height: 13.50m	153,7	289
9	Reactor Coolant excess letdown (CVCS) (size=1,46.10 ⁻³ m ²)	1,7 inches on cold leg3	No information	153,7	289
10	Nuclear Island drain system on U cold leg (size=1,46.10 ⁻³ m ²)	1,7 inches on cold leg1/2/3/4	Level: 16.02m Area: 47.51m ² Height: 13.50m	153,7	289
11	Nuclear Island vent system on vessel head (size=3,46.10 ⁻⁴ m ²)	0,83 inches on Vessel Head	Level: 21.24m Area: inner surface of the reactor dome (48m diameter) Height: 14.86m	157,2	289
12	Connection of Pressurizer relief valves (size=1,584.10 ⁻³ m ²)	1,77 inches on top of pressurizer	Level: 37.26m Area: 35.67m ²	155	344 (steam)

Table 3-3: Characteristics of main primary breaches

The size of the room where the break happens includes:

- The room where the break happens,
- Upper and lower levels connected to it by gratings (to calculate height).

Rooms connected by a small opening were considered separated, because it was considered unlikely for the jet to flow exactly in its direction. So, it is unlikely to have an inertial jet flowing into it. The opening may still be sufficient for steam to diffuse to the adjacent room, with a much smaller velocity. In fact, concrete walls are designed to support components and not to confine gases. So, a steam leakage would not be confronted with clearly separated rooms, but rather with a labyrinth of walls with complex arrangement.

Many systems are connected to the cold leg of the primary circuit, in the SG casemate, which has an elongated shape. The SG casemate is a high and narrow space, whose several floors are separated by gratings. A steam jet resulting from a leakage in this space would most probably hit a wall.

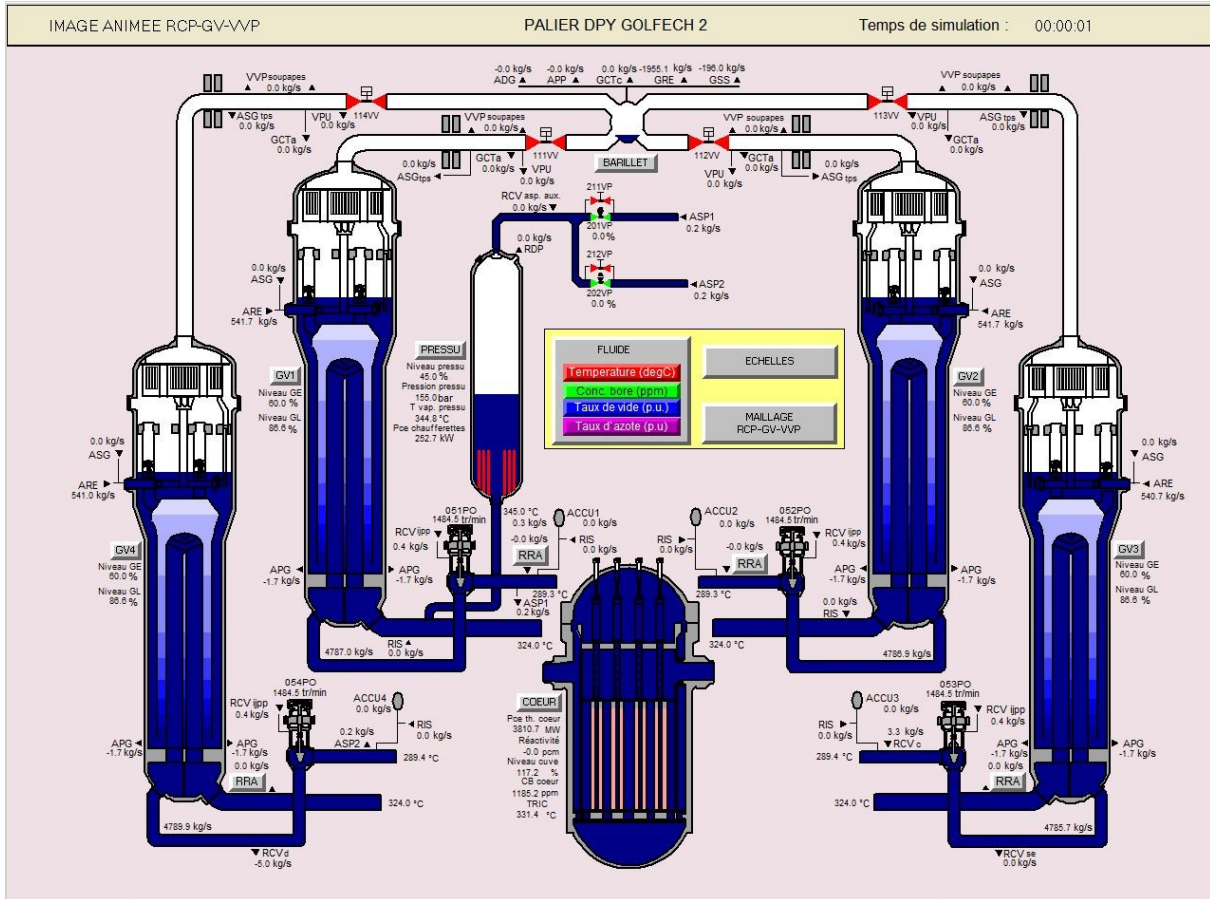


Figure 3-14: Main display of the SOFIA simulator

It can be noted that several systems are located in the same rooms, which can even contain systems connected to the hot or cold leg of the same loop (like the hot-leg SIS and the cold-leg RHRS extraction pipe). The minimal distance between the systems could not be found in documentation.

In addition to the above-mentioned break localisations, leakages can appear in lower levels of the reactor building, for example on the RHRS. Circuits are located in larger rooms (>100m² area and 5m high). The RHRS can only be connected to the primary circuit when the temperature is below 180°C, which is still sufficient to have a steam leakage.

Connections to the primary circuits may be relatively close, separated by a few tens of centimetres only. The exact geometry of piping in the reactor building is complex, so it is difficult to determine the distance separating two systems in all configurations.

Several objectives can be assigned to the detection system. For small breaches that do not lead to containment pressurisation, confirming that the main coolant leakage is due to a primary breach and not an internal leakage or wrong alignment of circuits would bring added value. It

would make diagnosis faster even if the localisation is not known precisely. For small and large leakages, identifying the room where the leak occurs would bring additional insights, to give a short list of the systems potentially concerned. Identifying the optimal isolation system is probably not necessary, since operators will isolate the entire flawed system and rely on redundant lines. The most difficult objective for the sensor is certainly to localise a non-isolable leakage on the primary circuit, to inform accident management. Indeed, the concerned legs of the primary circuit and associated connections can be close to one another. The debit of steam can be extremely large as well, making detection even more difficult.

3. Simulations performed with SOFIA French PWR 1300MWe

Three transients of LOCA at Initial Condition 100% nominal power have been simulated. The main thermalhydraulic parameters are presented in the excel tables given for the specifications, with a recording of them at each timestep, i.e. 100ms.

They correspond to:

- containment pressure value (bar)
- containment temperature (°C)
- liquid phase flow rate at the break (kg/s)
- steam flow rate at the break (kg/s)
- total flow rate (liquid and gas phases) at the break (kg/s)
- enthalpic flow rate at the break (unit: J/s)
- void fraction ratio at the break (1.10^{-5} for water, 1 for steam)

Only the main interesting values according to the specific transient are represented below. By convention, leakages from the primary circuit are represented by negative values for mass and enthalpy fluxes. Raw data is available upon request to ASNR. Due to its sensitive nature, it could not be released openly.

3.1. Transient n°1 : Large break on hot leg

The break is open at 1 second. Its size is 12 inches and located on the pressurizer surge line. It corresponds to the largest break that SOFIA can model.

The evolutions of the main thermal hydraulic parameters are presented below. It must be noted that containment spray system is actuated automatically at 1'02", after the pressure in the containment reached 2,6 bar.

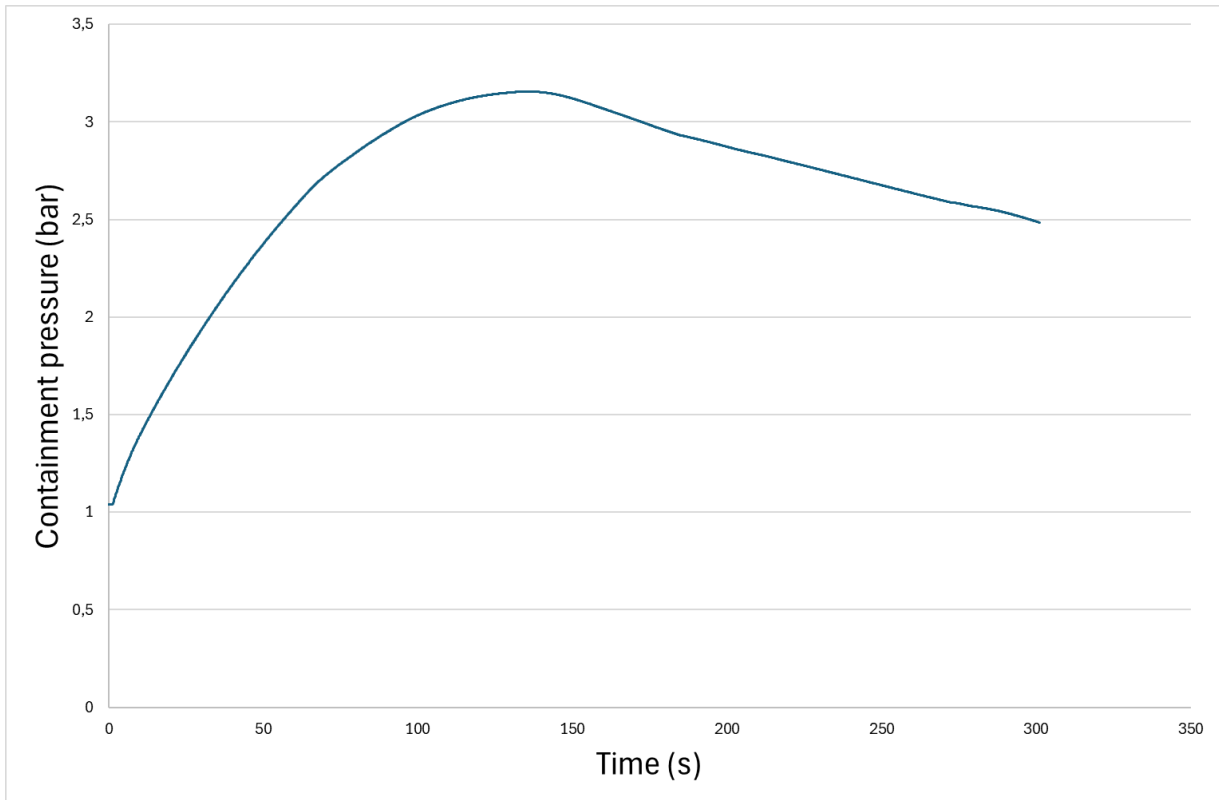


Figure 3-15: Evolution of containment pressure for transient n°1

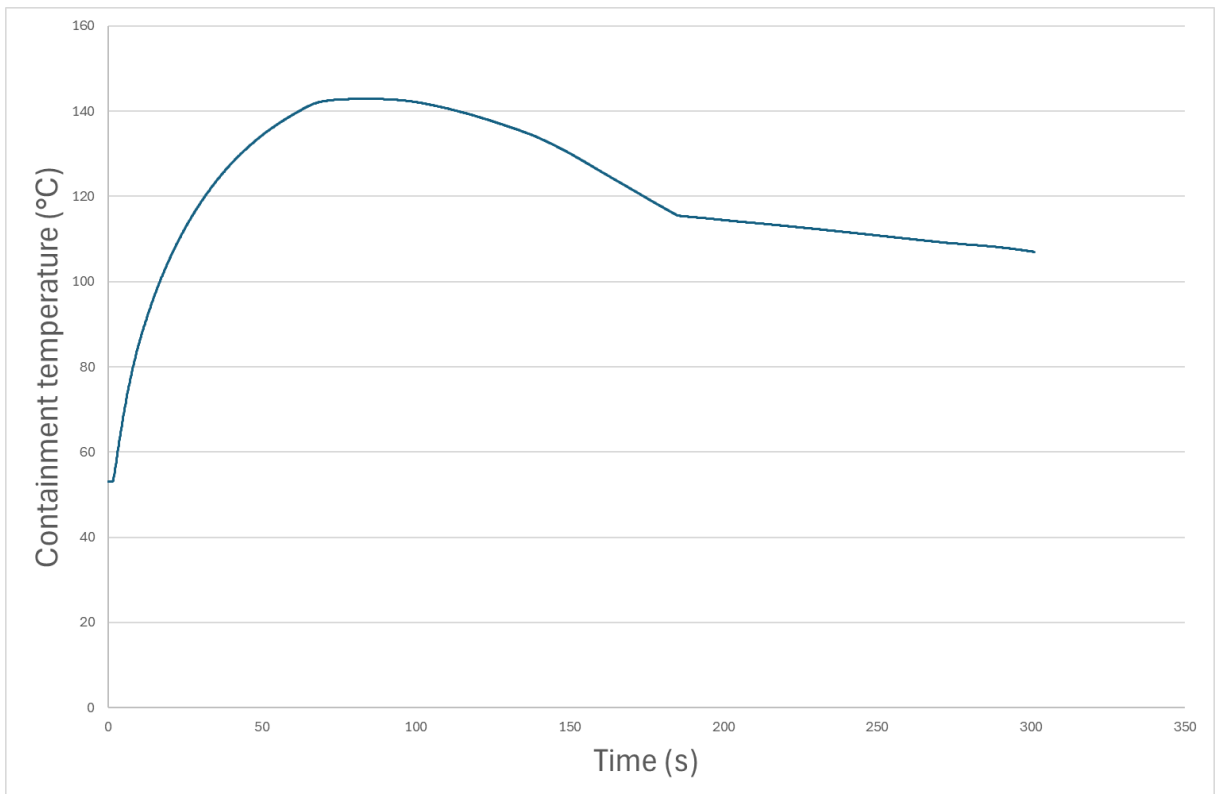


Figure 3-16: Evolution of the containment temperature for transient n°1

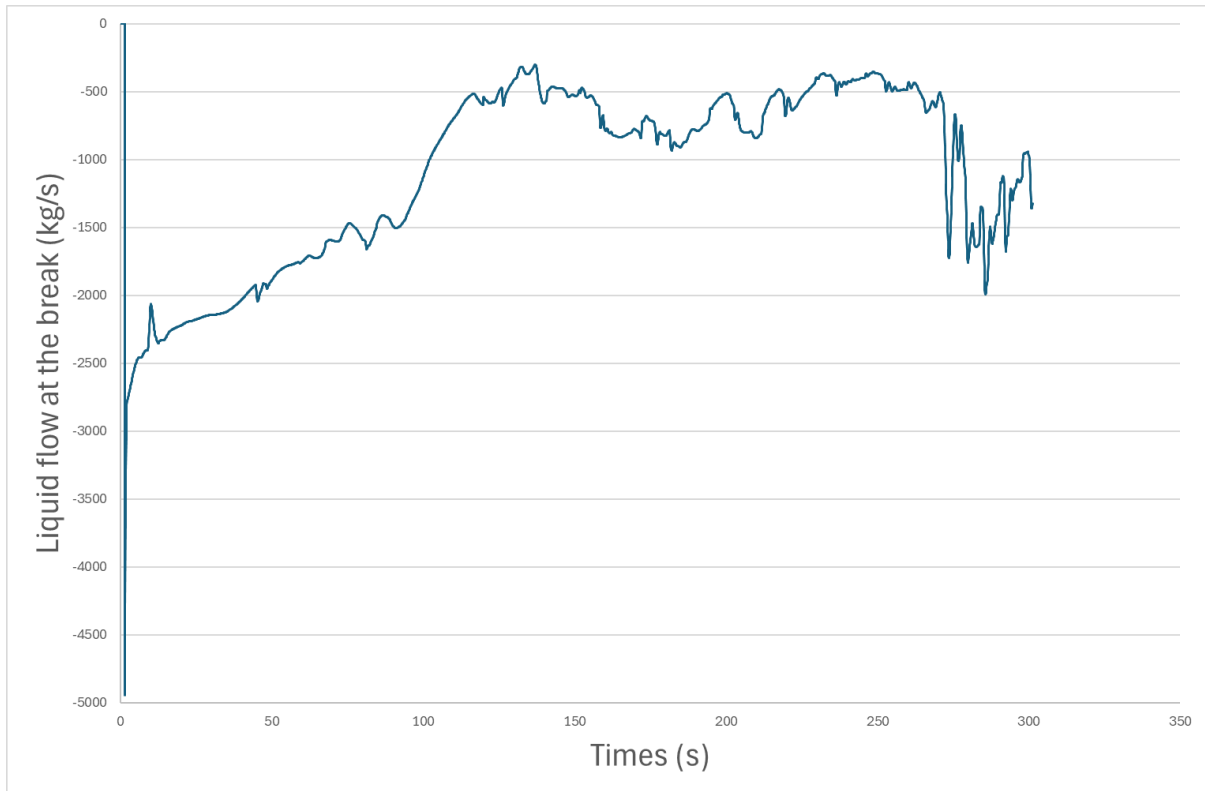


Figure 3-17: Evolution of the liquid flow at the break for transient n°1

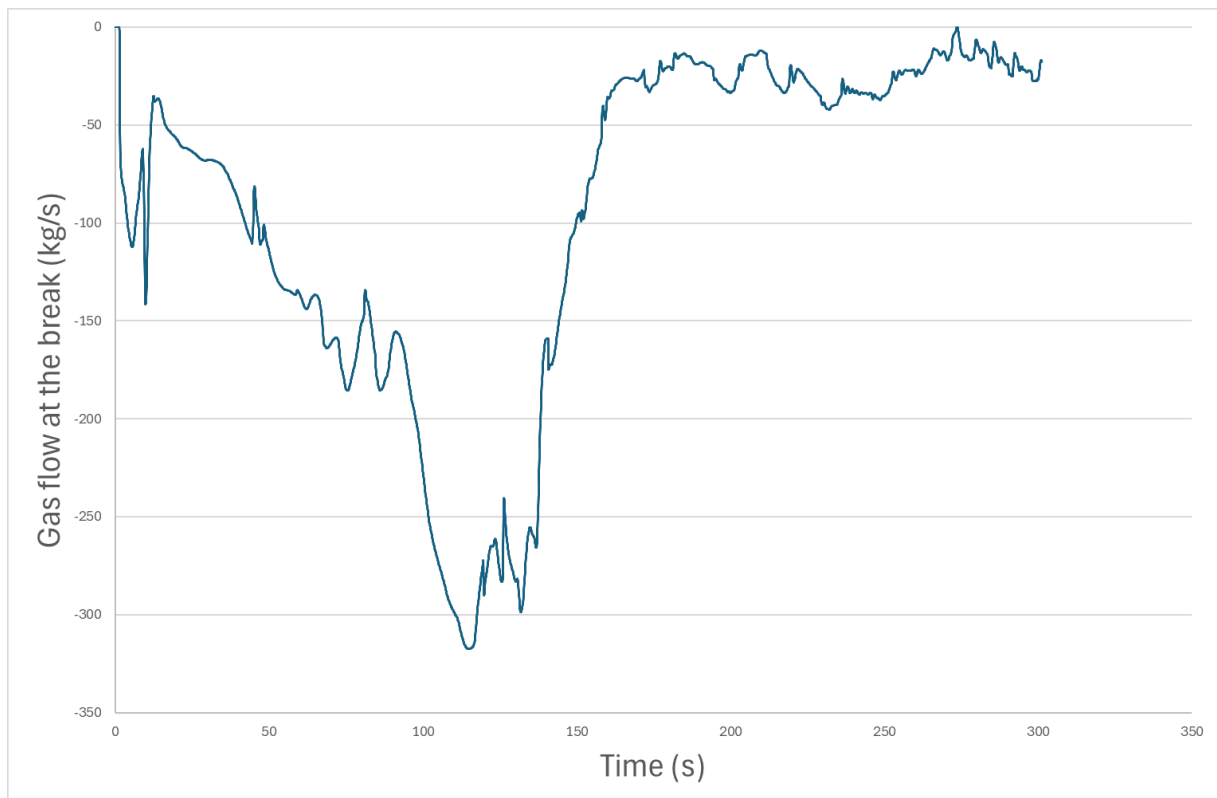


Figure 3-18: Evolution of the gas flow at the break for transient n°1

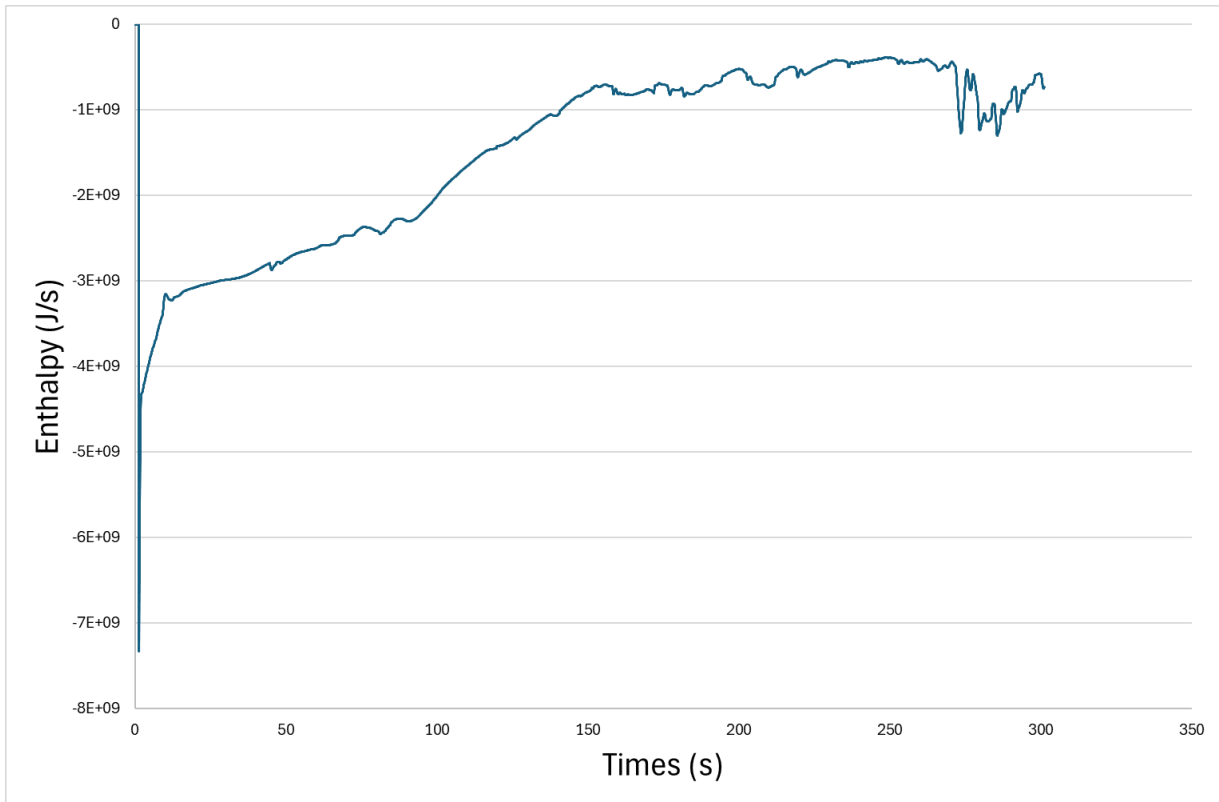


Figure 3-19: Evolution of the enthalpy flow at the break for transient n°1

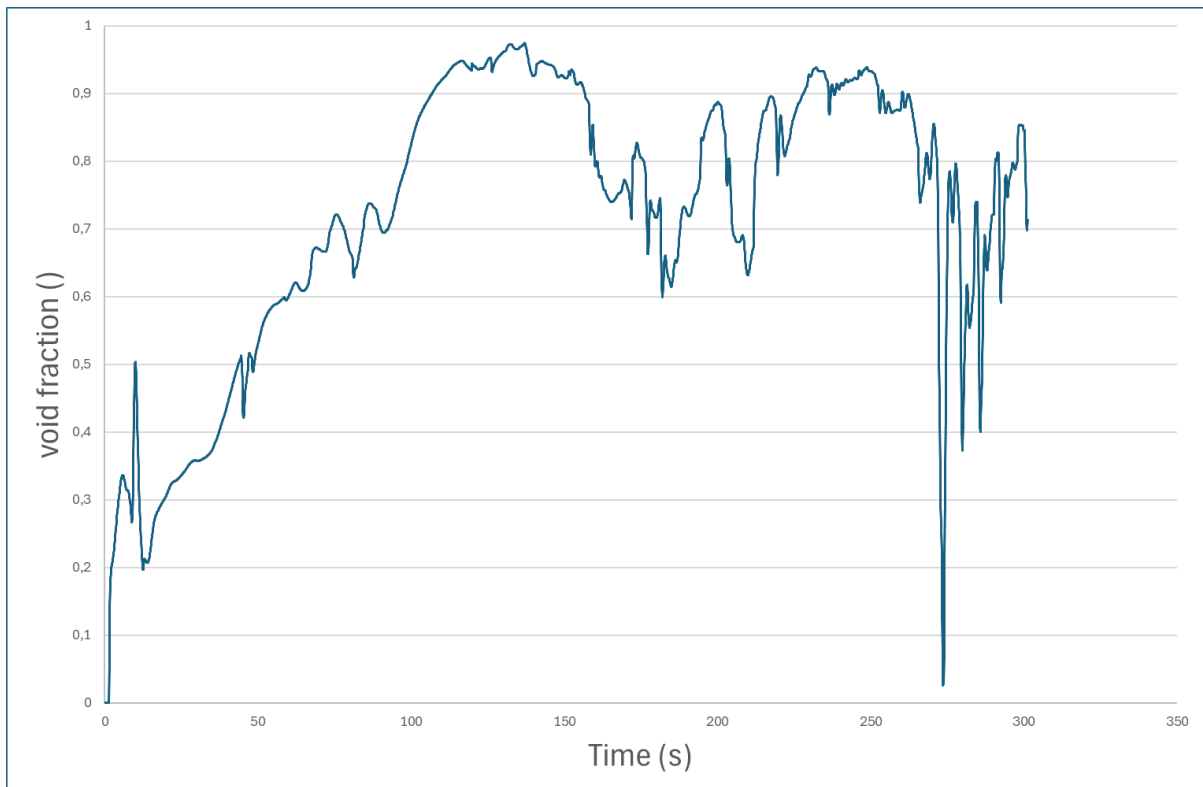


Figure 3-20: Evolution of the void fraction at the break for transient n°1

3.2. Transient n°2: small break on the vessel head

The break is open at 1 second. Its size is 0,83 inches and located on the vessel head. This corresponds to a small break LOCA. The mass flow rate at the break must be compared to the maximal charging capacity of the CVCS (8kg/s) and the criterium to enter EOPs (approx. 0.5kg/s of uncollected leakages).

The evolutions of the main thermal hydraulic parameters are presented below. It must be noted that containment spray system is not actuated, because of the small size of the break and that the flow at the break remains liquid.

The simulation is realized for 30 minutes.

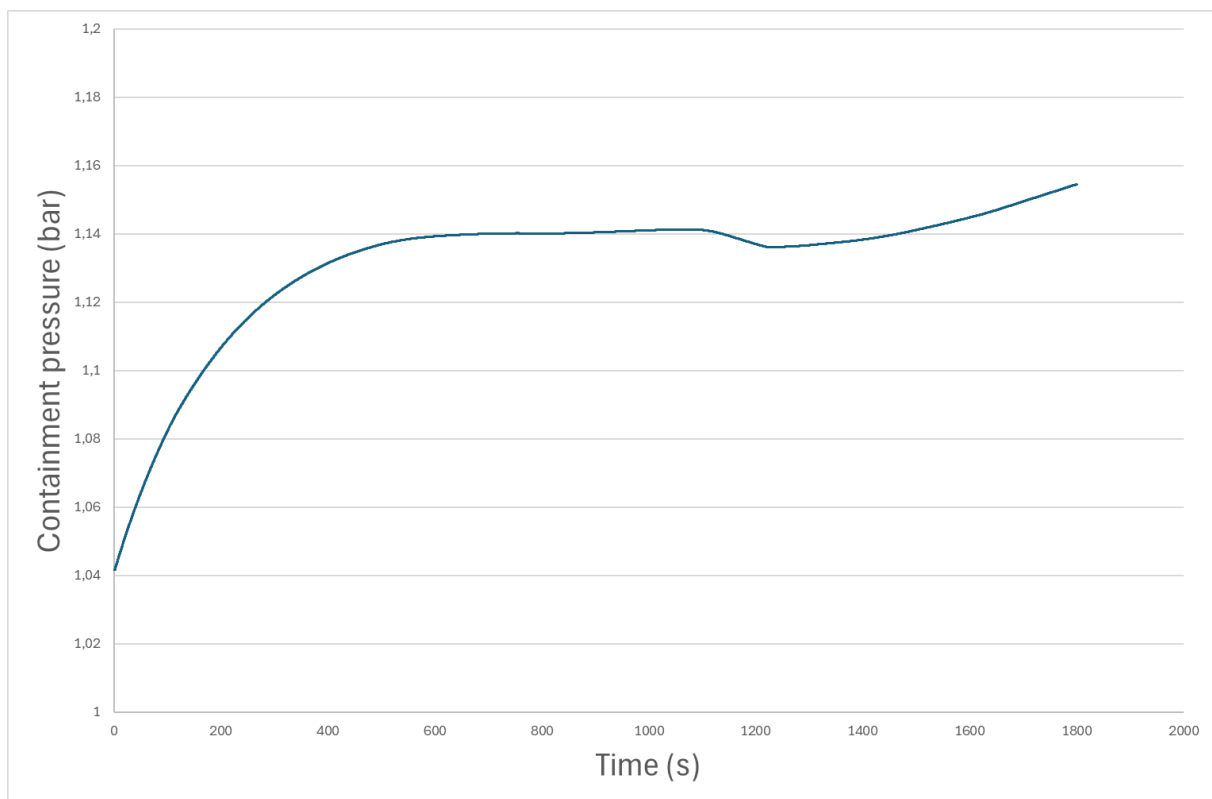


Figure 3-21: Evolution of the containment pressure for transient n°2

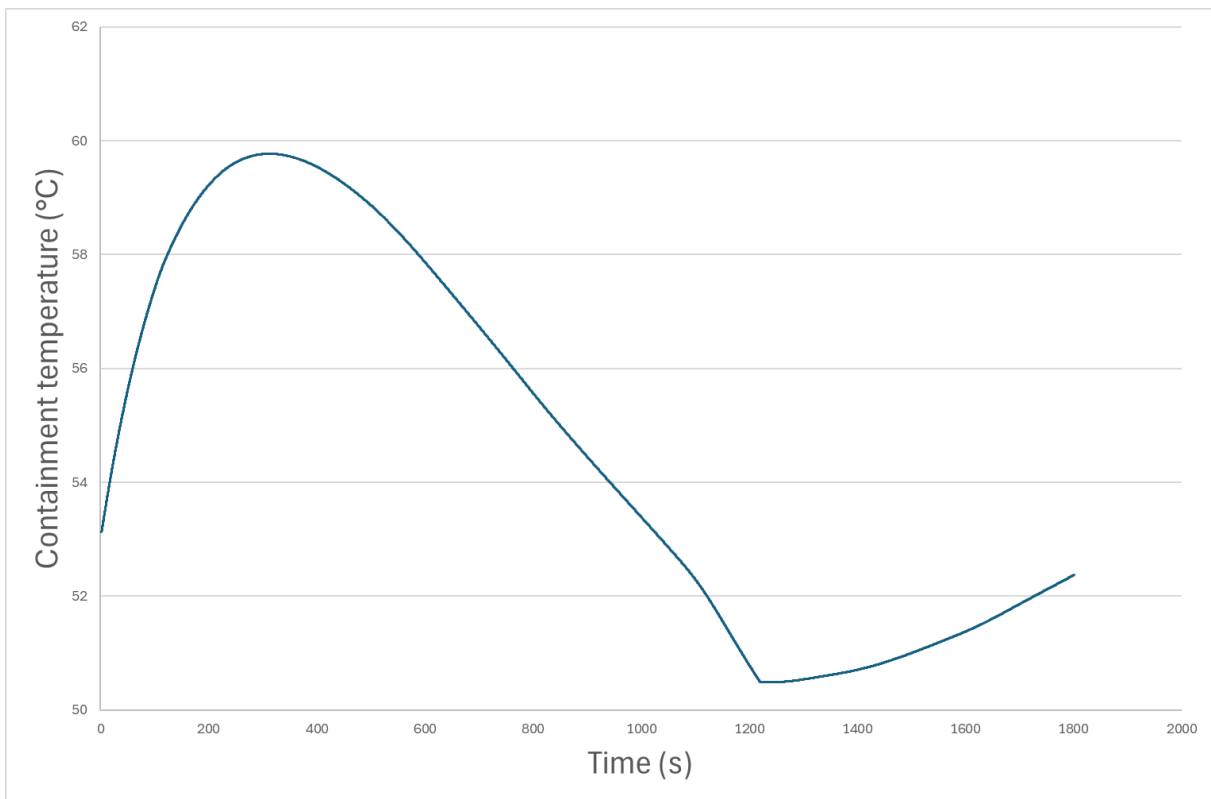


Figure 3-22: Evolution of the containment temperature for transient n°2

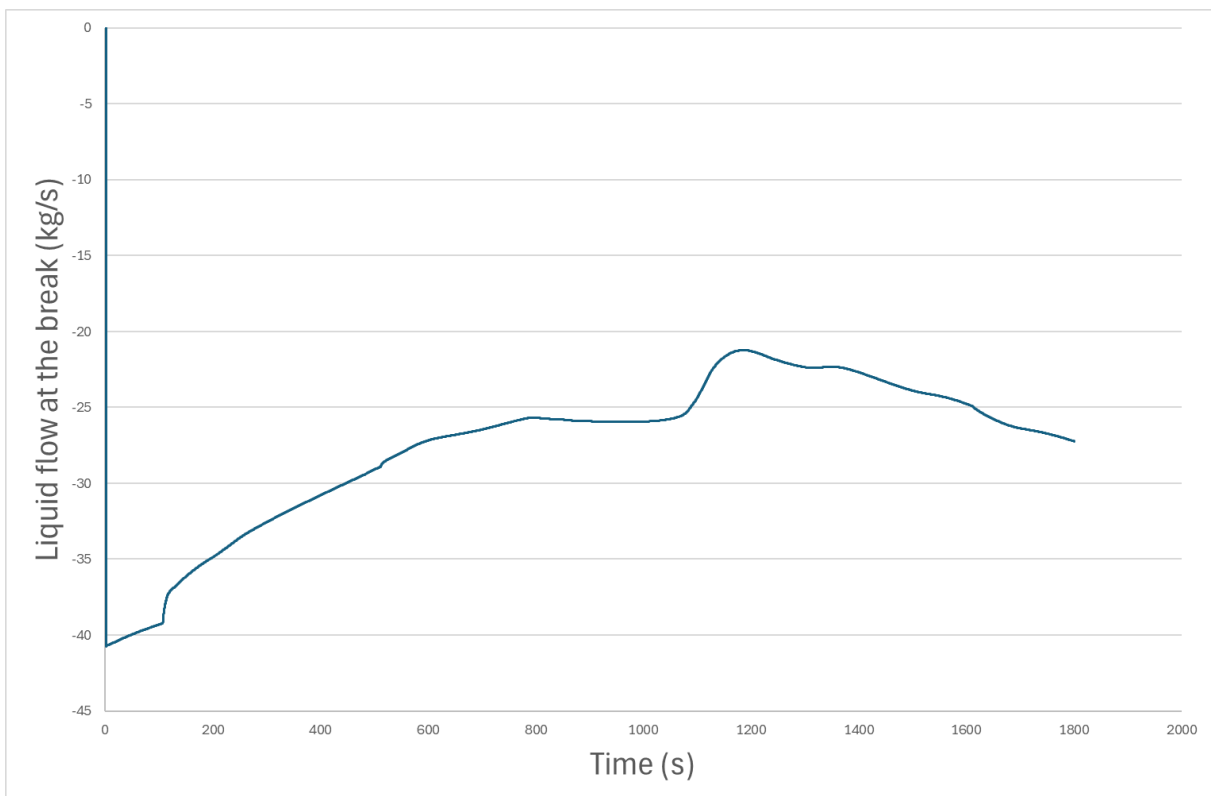


Figure 3-23: Evolution of the liquid flow at the break for transient n°2

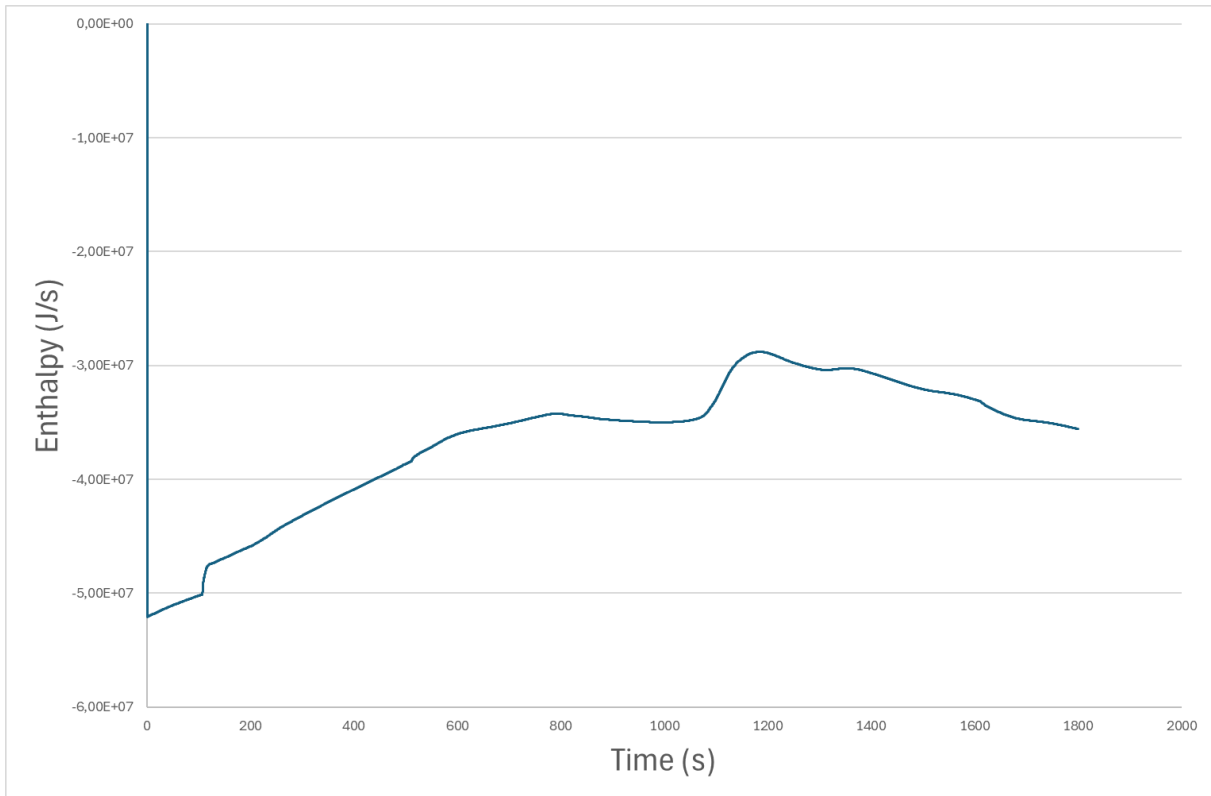


Figure 3-24: Evolution of the enthalpy flow at the break for transient n°2

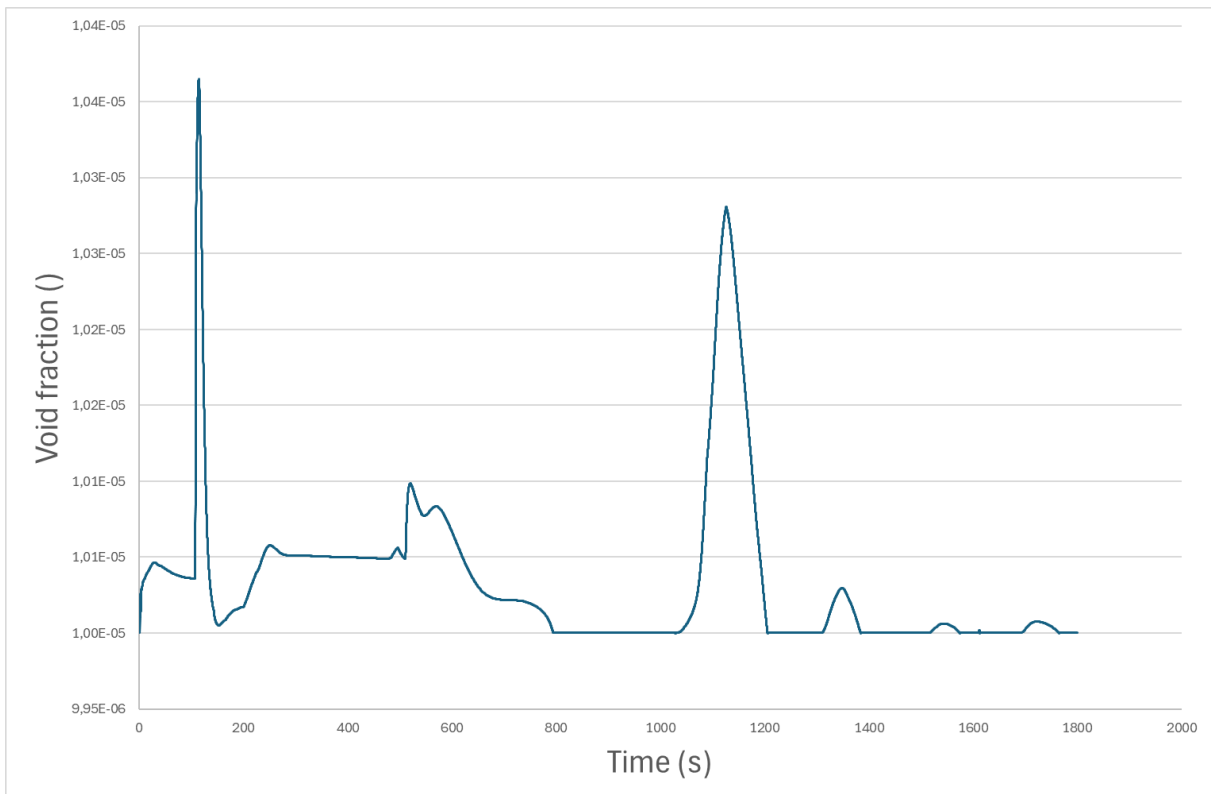


Figure 3-25: Evolution of the void fraction at the break for transient n°2

3.3. Transient n°3: small break on pressurizer head

The break is open at 1 second. Its size is 1,74 inches and located on the pressurizer head.

The evolutions of the main thermal hydraulic parameters are presented below. It must be noted that containment spray system is not actuated, the containment pressure staying under 1,5 bar.

The simulation is realized for 30 minutes.

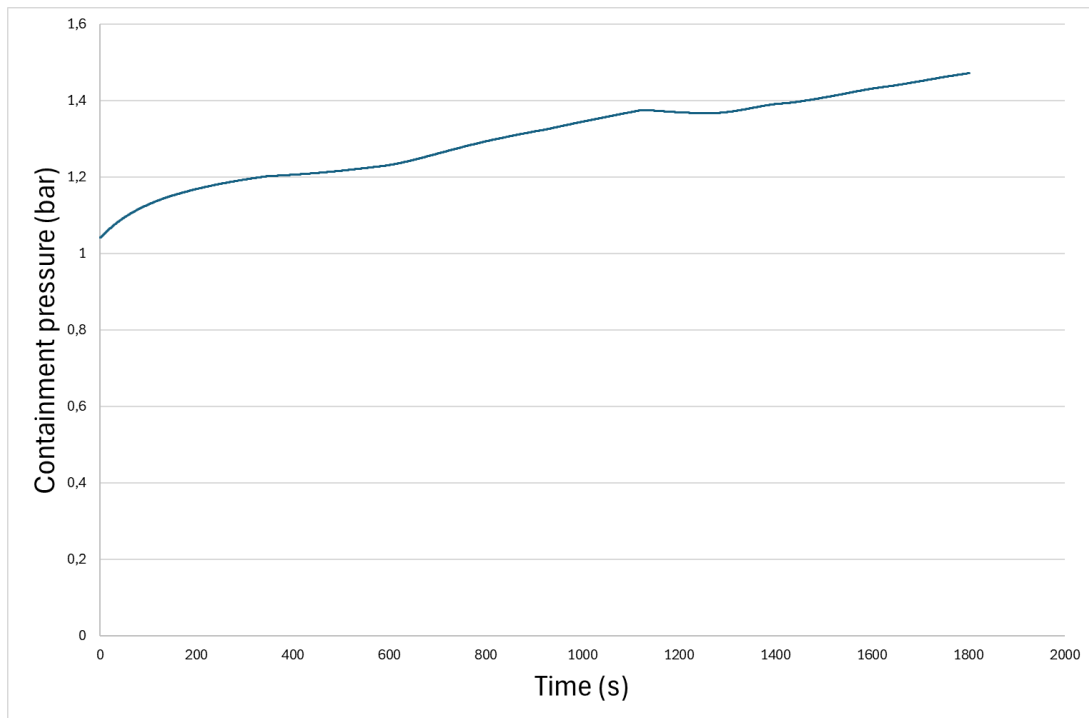


Figure 3-26: Evolution of the containment pressure for transient n°3

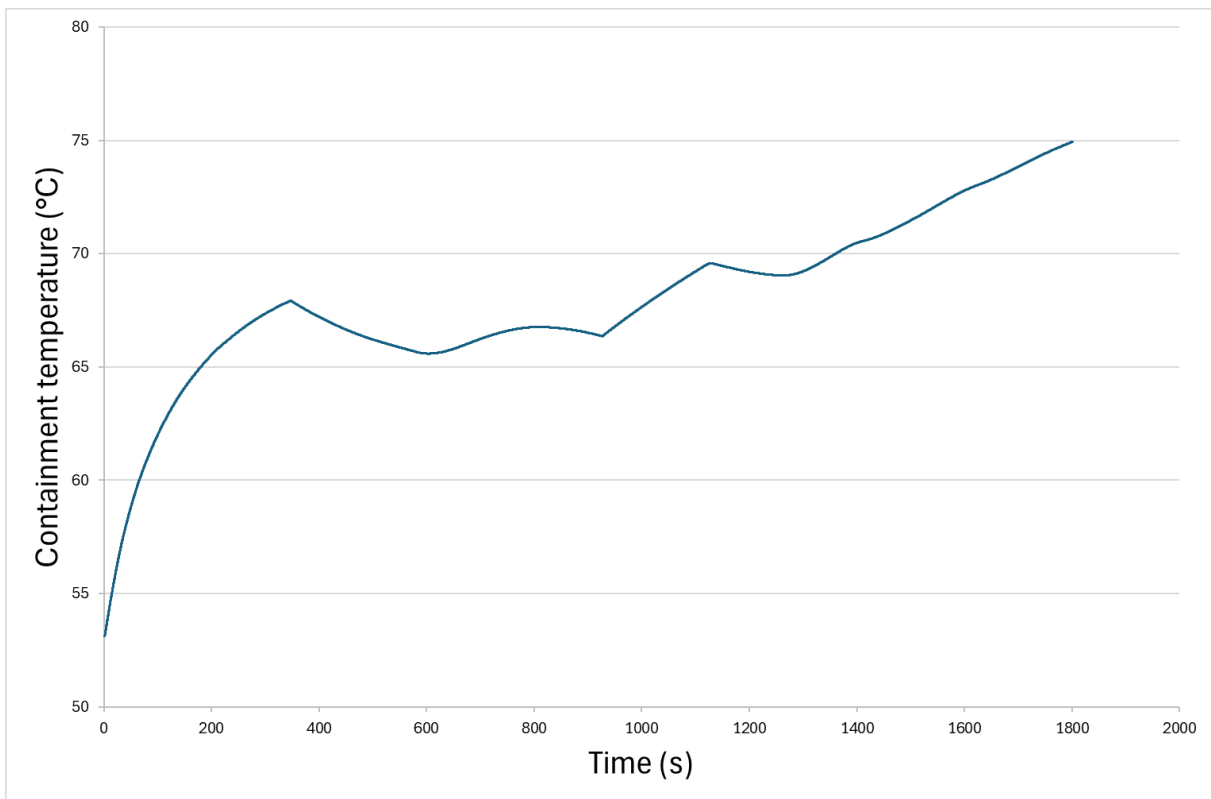


Figure 3-27: Evolution of containment temperature for transient n°3

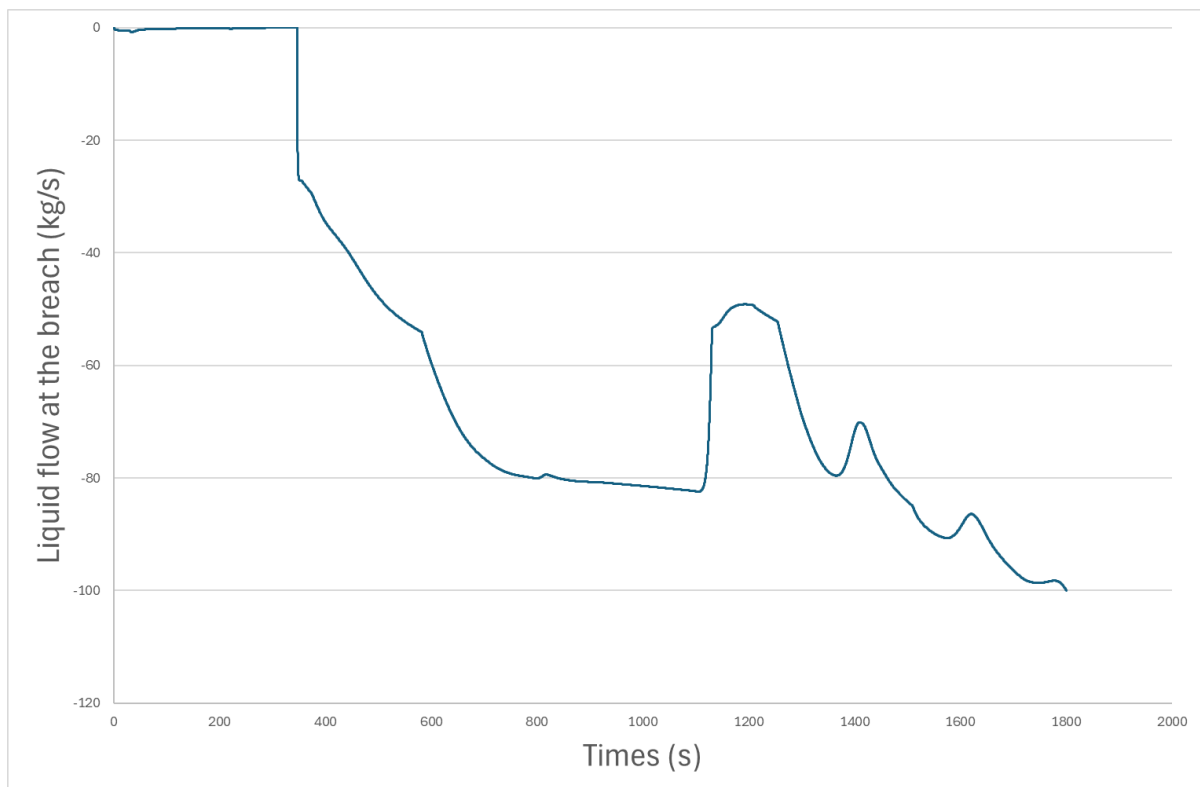


Figure 3-28: Evolution of the liquid flow at the break for transient n°3

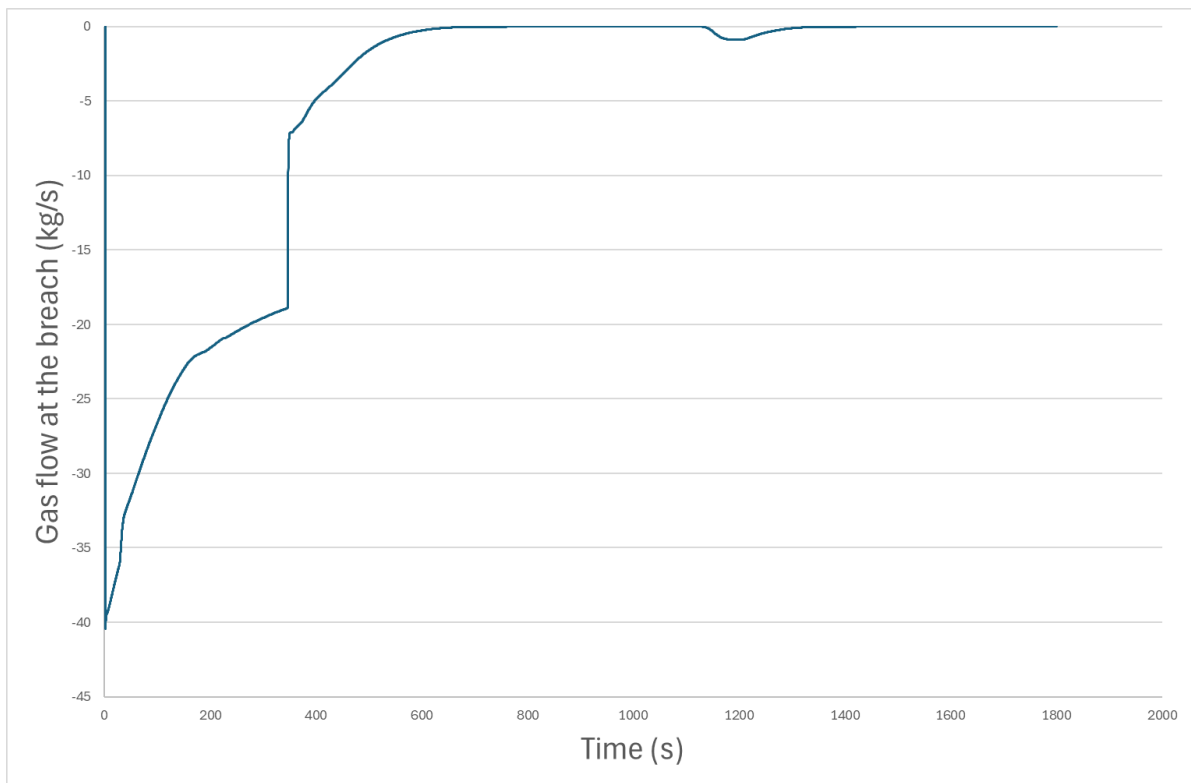


Figure 3-29: Evolution of the gas flow at the breach for transient n°3

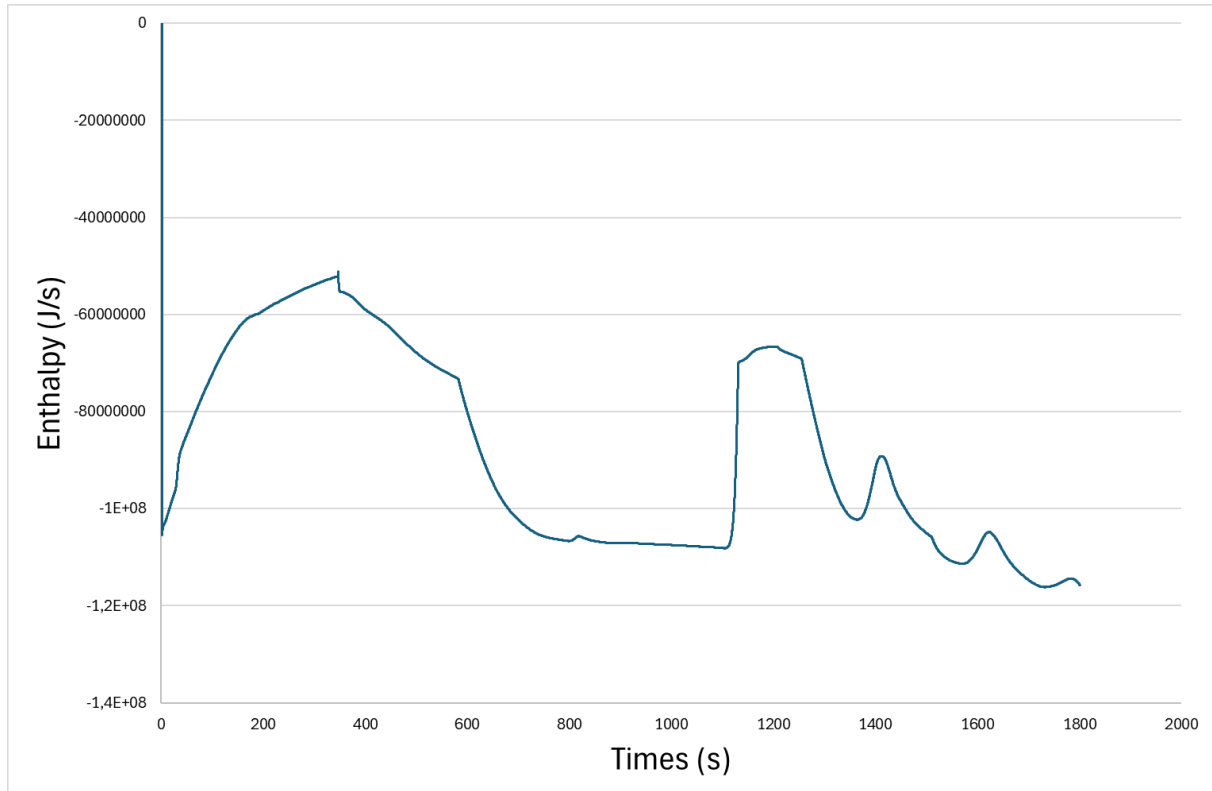


Figure 3-30: Evolution of the enthalpy flow at the break for transient n°3

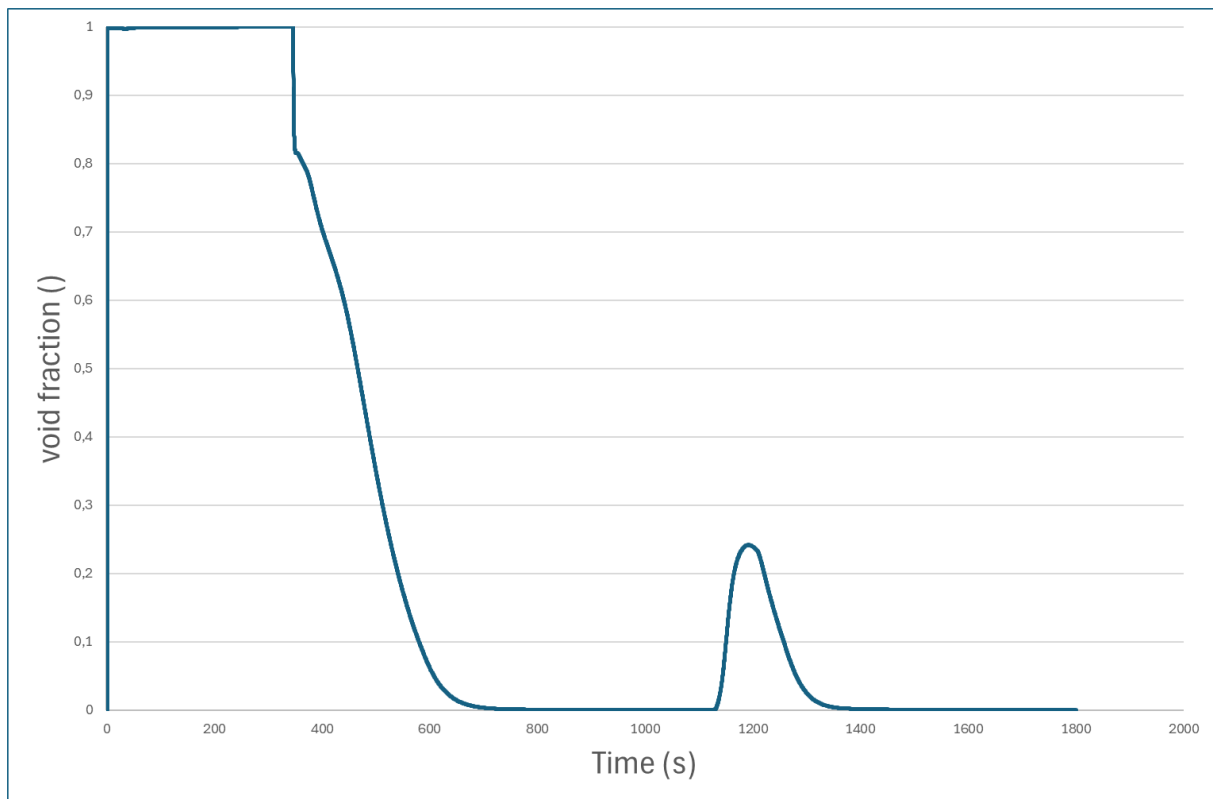


Figure 3-31: Evolution of the void fraction at the break for transient n°3