TANDEM

Research and Innovation Action (RIA)

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or he European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

Start date : 2022-09-01 Duration : 36 Months

Summary report on safety case studies for a SMR with cogeneration

Authors : Natalia RODIONOV (IRSN), Sebastian Buchholz, Thorsten Hollands (GRS), Calogera Lombardo, Massimiliano Polidori (ENEA), Artur Hashymov, Vadim Semiros, Stanislav Sholomitsky (ENERGORISK)

TANDEM - Contract Number: 101059479

Project officer: Angelgiorgio IORIZZO

Document title	Summary report on safety case studies for a SMR with cogeneration
Author(s)	Natalia RODIONOV, Sebastian Buchholz, Thorsten Hollands (GRS), Calogera Lombardo, Massimiliano Polidori (ENEA), Artur Hashymov, Vadim Semiros, Stanislav Sholomitsky (ENERGORISK)
Number of pages	57
Document type	Deliverable
Work Package	WP4
Document number	D4.5
Issued by	IRSN
Date of completion	2025-06-27 18:15:52
Dissemination level	Public

Summary

Summary report on safety case studies for a SMR with cogeneration based on final results of the safety cases. They will be discussed in a dedicated workshop between WP participants. Recommendations and further needs for the implementation of the SMR in the hydrid system as a cogeneration system from the safety point will be drawn. The capabilities of the assessment methods and tools (both system codes and coupling to Modelica) will be evaluated, guidance for their application will be formulated, and needs for improvement and further development will be identified.

Approval

Date	By
2025-06-27 18:18:30	Natalia RODIONOV (IRSN)
2025-06-27 18:28:36	Claire VAGLIO-GAUDARD (CEA)

D4.5 – Summary report on safety case studies for a SMR with cogeneration

WP4 - Task 4.3

June 27st, 2025 [M34]

Natalia Rodionov (ASNR), Sebastian Buchholz, Thorsten Hollands (GRS), Calogera Lombardo, Massimiliano Polidori (ENEA), Artur Hashymov, Vadim Semiros, Stanislav Sholomitsky (ENERGORISK)

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

History

Date	Version	Submitted by	Reviewed by	Comments
27/06/2025	V1	Natalia Rodionov	Sebastien Israel	

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

Table of Contents

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

List of Figures

List of Tables

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

Abbreviations and Acronyms

Acronym	Description
AOO	Anticipated Operational Occurrences
API	Application Programming Interface
ASNR	Autorité de Sûreté Nucléaire et de Radioprotection
BDBA	Beyond Design Basis Accident
BoP	Balance of Plant
CIRTEN	Consorzio Interuniversitario per la Ricerca Tecnologica Nucleare
CHP	Combined Heat and Power
CSA	Cross Section Area
CSG	Compact Steam Generator
DBA	Design Basis Accident
DBC	Design Basis Condition
DEC	Design Extension Condition
DEC-B	Design Extension Condition with core degradation
DiD	Defense in Depth
ECS	Évaluations Complémentaires de Sûreté
E-SMR	European Small Modular Reactor

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

FMU	Functional Mock-up Unit
GCSM	General Control Simulation Module within ATHLET
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit gGmbH
HES	Hybrid Energy System
HP	High Pressure
IAEA	International Atomic Energy Agency
IE	Initiating Event
IP	Intermediate Pressure
IRSN	Institut de radioprotection et de sûreté nucléaire
LOOP	Loss Of Offsite Power
LP	Low Pressure
LWR	Light Water Reactor
NSSS	Nuclear Steam Supply System
PI	Proportional-Integral
PID	Proportional-Integral-Derivative
PWR	Pressurized Water Reactor
SBO	Station Black-Out
SG	Steam Generator

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

SMR	Small Modular Reactor
SS	Steady State
SSC	Systems Structures Components
TANDEM	Small Modular ReacTor for a European sAfe aNd Decarbonized Energy Mix
TDV	Time-Dependent Volume
UNIPI	Università di Pisa
WENRA	Western European Nuclear Regulators Association
WP	Work Package

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

Executive Summary

This report presents the outcomes of an extensive investigation into the safety and energy efficiency challenges associated with integrating Small Modular Reactors (SMRs) into hybrid cogeneration systems. With the growing need for flexible and resilient energy solutions, the study focused on assessing the impact of energy scenarios on SMR safety, defining key safety criteria, and evaluating operational constraints within these integrated systems.

One of the central aspects of this research was the analysis of SMRs' adaptation to dynamic energy demands and their response to transient and abnormal operations within connected systems. Task 4.1, led by IRSN[1], identified the potential impacts on safety margins by surveying best practices from Europe. A methodology was developed to assess critical physical parameters and safety margins, incorporating the existing EU Network Codes governing reactor production flexibility.

Building upon these insights, Task 4.2, coordinated by GRS, established a set of relevant safety cases to evaluate cogenerating SMR resilience against the derived safety criteria. These cases included anticipated operational occurrences (AOO) and design basis accidents (DBA), addressing transmission grid disturbances, transient interactions with cogeneration loads, and specific failure scenarios. Four AOO-type and one DBA-type scenarios were specified, ensuring a comprehensive assessment of boundary conditions, major events, and assumptions.

Task 4.3, also coordinated by GRS, carried out in-depth simulation studies using thermal-hydraulic codes ATHLET and CATHARE, alongside their coupling with MODELICA-based models. These simulations enabled a realistic analysis of transient responses within both the nuclear (E-SMR) and cogeneration subsystems. The results provided valuable insights into the robustness of hybrid SMR systems, informing the development of refined safety strategies and operational recommendations. These studies were documented in two technical reports—one dedicated to AOO-type events (D4.3) and another to DBA-type events (D4.4). Additionally, the DBA case was also simulated in standalone mode of ATHLET and CATHARE to get insights of the impact of the coupling on the scenario evolution.

The summary of WP4 activities, presented in Deliverable D4.5, the present report, contributes to the broader effort of refining assessment methodologies, enhancing safety evaluations, and

[1] As of January 1, 2025, the French Institute for Radiological Protection and Nuclear Safety (IRSN) has been reorganized and officially renamed the Nuclear Safety and Radiation Protection Authority (ASNR). This change reflects a structural reform aimed at consolidating nuclear oversight functions under a single authority responsible for both safety and radiation protection

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

supporting the safe integration of SMRs in hybrid cogeneration settings. Through collaboration with leading institutions such as EDF, CEA, ENEA, ENERGORISK, GRS, and UNIPI - in collaboration with WP2 -, this study has reinforced confidence in modern nuclear technologies while advancing energy efficiency and resource optimization.

The results obtained through TANDEM WP4 underscore the importance of advanced modeling techniques in assessing the dynamic behavior of E-SMRs under various transient conditions. The coupling of MODELICA with ATHLET or CATHARE has proven valuable in simulating the operational stability of the reactor under diverse scenarios, providing detailed insights into the interplay between system components.

The study's findings support the premise that cogeneration can serve as a stabilizing factor in specific transient scenarios, particularly for mitigating load rejection events. By enabling heat extraction from the secondary circuit, cogeneration reduces peak thermal stress and enhances the adaptability of the system, promoting a safer operational environment. However, the effectiveness of cogeneration in ensuring stable reactor performance is highly dependent on the nature of the transient event, as demonstrated in cases of external power loss (DBA) where its role is minimal.

Ultimately, the tools developed within the TANDEM project offer a robust framework for future investigations into the safety and flexibility of E-SMRs. By refining these modeling approaches and extending the scope of analysis, further improvements in reactor design, control system optimization, and energy integration strategies can be achieved, paving the way for enhanced nuclear cogeneration applications.

Keywords

SMRs, E-SMR, Hybrid energy Systems, Safety Methodology, Flexibility, Cogeneration, Europe, AC^2/ATHLET, CATHARE, MODELICA

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

1 Introduction

The European TANDEM project plays a key role in advancing the safe integration of Small Modular Reactors (SMRs) into Hybrid Energy Systems (HES). As these nuclear technologies become increasingly relevant in the energy transition, their interaction with other components of hybrid systems introduces unique operational challenges and potential risks. Recognizing these complexities, TANDEM has focused on developing and implementing a robust safety methodology aimed at identifying and assessing these risks, ensuring that SMRs can operate safely and efficiently within diversified energy environments. The TANDEM project consists of six Work Packages (WP). The present report specifically addresses to WP4— "Safety Analysis of SMRs Integrated into Hybrid Systems". The study assesses SMR adaptability to dynamic energy demands, highlighting safety concerns occurring from integration with cogeneration systems.

The International Atomic Energy Agency (IAEA) series No NP-T-1.17 Guidance on Nuclear Energy Cogeneration [6], highlights two major concerns: first, the risk of radioactive material infiltration within cogeneration facilities, and second, the occurrence of reactor transients triggered by abrupt fluctuations in heat or electricity demand. These variations, which can disrupt the reactor's secondary circuit, require rigorous analysis to define failure scenarios—whether they stem from anticipated operational occurrences or design-basis accidents—to ensure the system's continued safe operation.

In this framework, the European TANDEM project makes a significant contribution by exploring the integration of SMRs in hybrid production environments. This report is part of a comprehensive analysis of the safety and energy efficiency challenges associated with integrating Small Modular Reactors (SMRs) into hybrid cogeneration systems. In the context of an accelerating energy transition, these technologies promise to combine flexibility and performance, adapting nuclear production to the evolving and diverse demands of the market.

An innovative modeling approach has been developed, coupling thermal-hydraulic simulation codes for the reactor with a MODELICA model dedicated to non-nuclear industrial processes. Thanks to this integration, it is possible to simulate the effects induced by rapid variations in thermal or electrical load, allowing for identification of safety margins and optimization of the dynamic behaviour of facilities.

The studies conducted in collaboration with partners such as EDF, CEA, ENEA, ENERGORISK, GRS, and UNIPI has relied on ATHLET and CATHARE models to assess the behavior of SMRs under operational transients and design-basis accident scenarios. The results provide fresh comprehension into the resilience of these hybrid systems, reinforcing confidence in modern

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

nuclear technologies while highlighting the potential of cogeneration to enhance overall energy efficiency and resource utilization.

Beyond a purely technical approach, this report aims to establish a solid foundation for fostering an informed and constructive debate on the development pathways for sustainable energy solutions. By combining scientific rigor with industrial pragmatism, this study contextualizes the future of SMRs in cogeneration mode, emphasizing their strategic role in optimizing energy systems in response to contemporary challenges.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

2 Methodology

2.1 General issues

The status of European research on the safety analysis of small modular reactors (SMRs) in terms of operational flexibility and cogeneration was analyzed in Deliverable D4.1, [1], to examine the need for defining a specific safety methodology which was issued in Deliverable D4.2, [2]. The recommended methodology was based on the IAEA SSG2 guidance to ensure the compliance with relevant nuclear safety regulations and standards (IAEA, WENRA, national regulatory bodies...). Its objective was to identify these specific risks and assess them. In particular, the integration of SMRs into future low-carbon and smart grids, where some non-dispatchable generators are prioritized over dispatchable generators, will inevitably lead to specific interactions with the grid and constraints for grid reliability (e.g., ensuring that generation always equals demand). Then a crucial step was to establish an exhaustive list of postulated initiating events. Energy network balancing through SMR load following capabilities, energy storage systems or cogeneration flexibility, has been discussed.

In addition, the safety of small modular reactors should be improved by reducing the factors that can cause these risks. Such a goal should be addressed in the new operational management of the integration of SMRs in HES. For this reason, DEC were not considered. Therefore, discussions were limited to identifying constraints that could affect the maneuverability of the SMR. The general issues taken into consideration were as follows, [2]:

- Determining appropriate separation distances between the reactor building and the industrial plant helps to minimize mutual risks.
- There are two possible configurations for coupling an SMR with an industrial process in a cogeneration context. Either the coupling can be characterised as weak if the interface is managed via the BoPs, which provide a certain buffer, or the coupling is strong if it interacts directly with the nuclear island.
- An hazard does not necessarily lead to a failure and therefore there is the need to consider an initiating event (IE).
- All reactor states should be considered (power operation, shutdown, maintenance, etc.).
- In the same way, as design provisions have been put in place for the turboalternators (located tangentially to the nuclear island), the same kind of consideration must be given to the industrial cogeneration process systems.
- Stopping potential projectiles without any special provisions other than the strength of the building materials.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

- It should be noted that the modifications planned as part of the post-Fukushima stress-tests would make it possible to fulfil some of these safety functions - provided that these modifications withstand the thermal and overpressure effects of an explosion.
- Taking domino effects into account is necessary.
- Local storage (in a form to be defined) as a solution to avoid transients should be considered.

Assessing the safety of a Small Modular Reactor (SMR) integrated in a hybrid energy system requires a comprehensive methodology that considers various aspects of nuclear safety. The proposed methodology to assess the safety of such a system is based on the IAEA guide SSG2-rev 1 and can be split in several phasis:

Phase 1: Firstly, perform two separate safety analyses (one for the NPP and one for the industrial plant)

Phase 2: Detail precisely the design and operational characteristics of the SMR

Phase 3: Identify sources of risk and sensitive system components

Phase 4: Define a DBC list

Phase 5: Perform safety analysis

The main challenge has been to identify IEs and hazards that depend on both HES configurations and SMR design. The methodology proposed in Deliverable D4.2, [2], was tested on the key IEs identified in that document using MODELICA, ATHLET and CATHARE through specific safety case studies [3],[4], allowing validation of existing modelling tools despite the lack of operational experience with SMR reactors.

2.2 Identification of Suitable AOOs and DBA

The list of AOO and DBA was established applying the methodology described in D4.2, [2].

The primary objective in identifying relevant AOO and DBA is to ensure that cogeneration within the HES plays a significant role. A key focus was assessing on :

- The impact of cogeneration on SMRs.
- The influence of grid interactions on both cogeneration and SMRs.

The presence of cogeneration has a significant influence on the evolution of the safety case, either by enhancing its robustness or introducing additional challenges. In the context of DBA scenarios, conventional PWR accidents demand an immediate disconnection of the turboalternator from the electrical grid. This separation ensures proper accident management and allows independent evaluation of the event without necessitating integration into the HES. The role of cogeneration in this framework requires careful assessment to determine its overall impact on operational safety and system stability.

Based on the results of Work Packages 1, 2 and 3, a set of safety cases relevant to assessing the safety of cogeneration SMRs according to the criteria defined in Task 4.1 was established with the support of all participants. The cases included postulated initiating events mainly of the AOOs or DBAs type, examining in particular:

- Transmission network disturbances based on the European network code;
- Interactions with cogeneration loads;
- Transients in the cogeneration section and ramp-up requirements linked to network management.

To conduct the assessment, four AOO scenarios and one DBA scenario were specified, incorporating both initial and boundary conditions for nuclear and Combined Heat and Power (CHP) plants. The modeling approach was structured as follows:

AOO Modeling:

ATHLET/MODELICA coupling:

- Case 1: Sudden decrease in heat demand (drop from 50 % to 3 % in 10 s).
- Case 2: Sudden decrease in heat demand (drop from 100 % to 10 % in 10 s).
- Case 3: Sudden increase in heat demand.

CATHARE3/MODELICA coupling:

- Case 4: Thermal load rejection (evaluated using the quasi-static BoP (Balance of Plant) model)

DBA Modeling:

CATHARE3/MODELICA coupling:

- Unsuccessful electrical load rejection leading to Loss of Offsite Power transient (>2h).

Standalone simulations of Loss of Offsite Power transient using:

- CATHARE3.
- ATHLET.

All assumptions and modeling details are documented in deliverables D4.3, [3] and D4.4, [4]. Several of these elements are reiterated in the following chapters to enhance understanding. Annexes 3 and 4 provide details for other calculations.

2.3 Coupling tools

The TANDEM project examines the safety impact of SMR hybridization, specifically the effects of connecting heat users to the BoP, which may influence key safety parameters through initiating events or system interactions. To model this dynamic interaction, a full hybrid system representation is necessary, with MODELICA chosen for modular simulation.

Moreover, safety codes such as ATHLET and CATHARE3 focus on primary system behavior. A coupling between these safety codes and MODELICA models must be established to ensure comprehensive analysis. As it was noted previously at Chapter 2.2, three types of modeling were developed in the framework of task 4.3 to realize the scenarios chosen for the modeling and to carry out the benchmarks for certain cases. To perform safety studies on the E-SMR power plant, separate models were developed and subsequently integrated:

- An NSSS (Nuclear Steam Supply System) model created using the CATHARE3 or ATHLET codes,
- Two BoP models built with MODELICA:
    - Quasi-static one (developed by the CEA with the ThermoSysPro library)
    - Dynamic one (developed by PoliMi with the Thermopower library)

All models were equipped with Proportional-Integral-Derivative (PID) controllers to emulate the behavior of the plant's automatic control systems.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

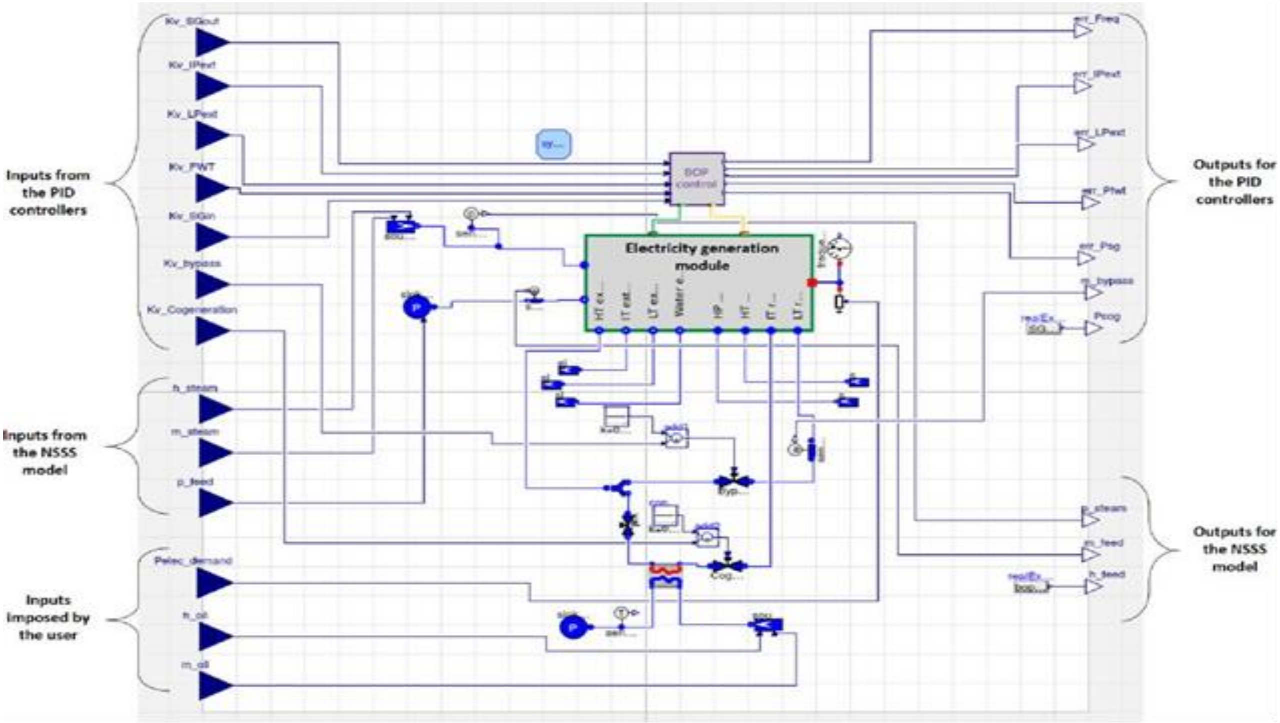

Figure 1: Dynamic model (Thermopower library) of BoP, [4]

3 AOOs simulations, [3]

3.1 ATHLET/MODELICA simulations

3.1.1 Overview about ATHLET model of the E-SMR

Some modifications of the ATHLET model have been made with respect to the deliverable D2.7 [10]. In this report, the coupling of ATHLET and MODELICA is presented including first simulations verifying its applicability using two MODELICA sub models, the NSSS control and the BoP model.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

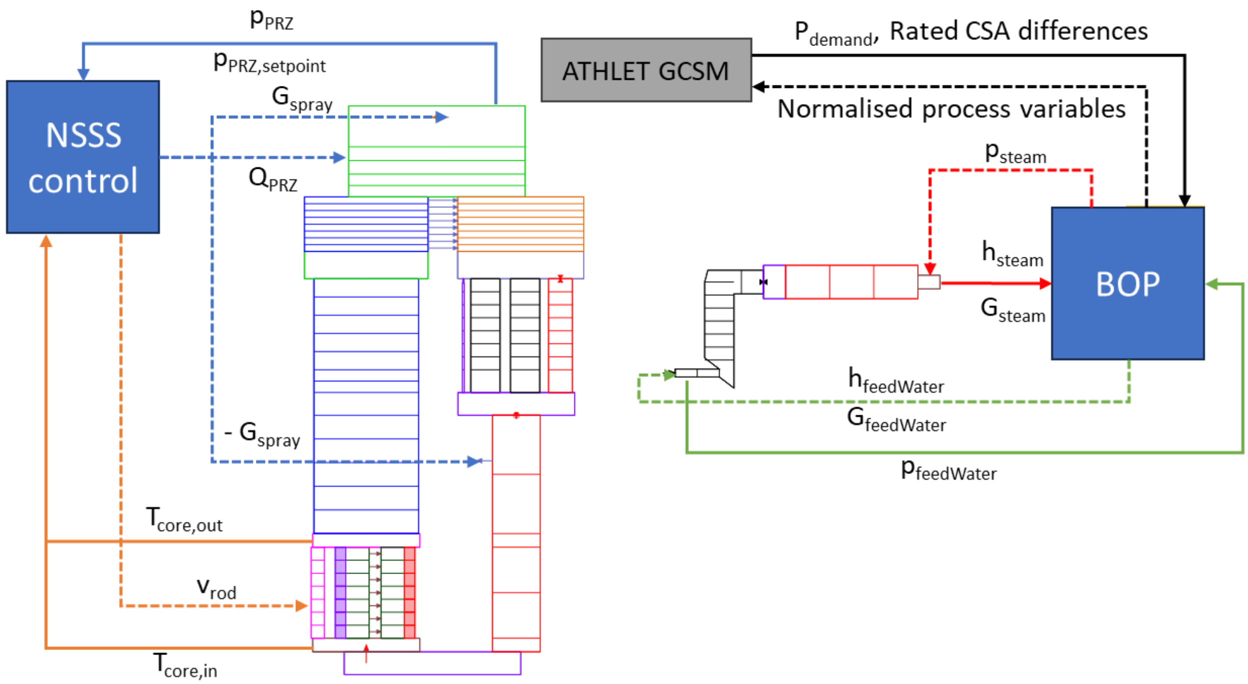

Figure 2: Coupling interfaces used for coupled simulations dashed lines: MODELICA → ATHLET, solid lines: ATHLET → MODELICA, [3]

The coupled model in Figure 2 illustrates variable exchange between ATHLET and the MODELICA submodels for NSSS control and the BoP . While ATHLET GCSM explicitly shows only key process parameters (e.g., demanded power, CSA differences, error signals), it serves as the main conduit for all data exchange. Colored arrows indicate parameters exchanged with time-dependent volumes and fills. Additionally, core outlet/inlet temperatures and control rod velocity are transferred between ATHLET and NSSS. The spray mass flow rate enters the pressurizer via a fill and is removed from the downcomer below the main coolant pump, as marked by a negative sign. The detailed description of this coupled simulation is provided by the deliverable D4.3, [3].

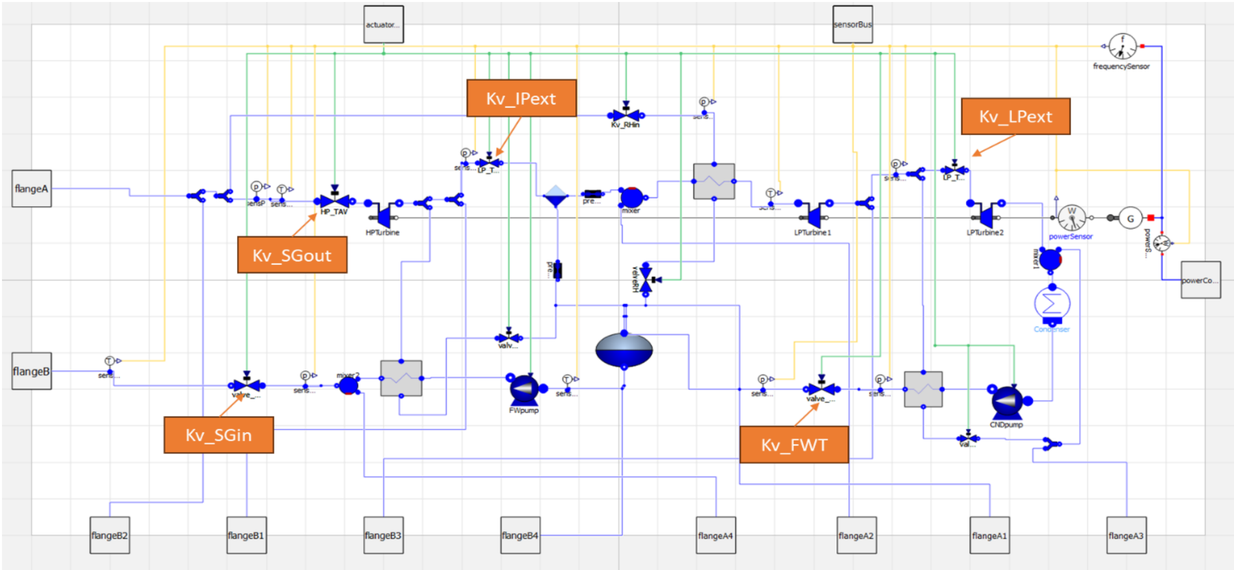

Figure 3: BoP model developed by POLIMI, [3]

3.1.2 Investigated AOOs

The work was shared between partners GRS and UNIPI. UNIPI employed a quasi-static BoP model, meaning that its output variables responded instantaneously to imposed boundary conditions. In contrast, GRS utilized a dynamic BoP model that accounted for component dynamics within the system. GRS conducted simulations to analyze the effects of sudden fluctuations in energy demand from heat users.

Two different cases are simulated concerning the decrease of power demand by the heat users. First, the sudden drop of heat demand from 50 % to 3 % in 10 s is considered. In the second case, the drop is even larger - from 100 % to 10 % (also in 10 s). The cases are listed in Table 1 (1 and 2). The drop is simulated by closing the cogeneration valve to 3 % or 10 % respectively including a reduction of the oil mass flow rate to the same values.

The third case is the sudden increase in thermal load by opening the cogeneration valve to 100 % in 10 s.

Table 1: Investigated power demand from the heat users – cases, [3]

No	Case	Notes
1	Full core power distributed to 25 MW_{th} cogeneration, rest (164.8 MW_{el}) electrical power	Drop from 50 % to 3 % in 10 s

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

2	Full core power distributed to 46 MW_{th} cogeneration, rest (153.0 MW_{el}) electrical power	Drop from 100 % to 10 % in 10 s
3	Full core power distributed to 25 MW_{th} cogeneration, rest (164.8 MW_{el}) electrical power	Increase from 50 % to 100 % in 10 s

3.1.3 ATHLET/MODELICA simulation results

In all cases the initial conditions cover a full power E-SMR (540 MW_{th}). The secondary steam mass flow rate is distributed in case 1 and 3 to provide 25 MW_{th} cogeneration heat and 164.8 MW_{el} electrical power. In case 2 the cogeneration valve is opened to 100 % leading to a cogeneration power of approximately 46 MW_{th}. The rest is used to generate electrical power of 153 MW_{el}. In the simulation, at t = 0 the cogeneration valve is either closed to a certain value or fully opened in 10 s. All cases show that the SMR can handle the AOO initiated from the external load variation and lead to stable reactor conditions. The detailed description of simulation results is provided by the deliverable D4.3, [3].

3.2 CATHARE/MODELICA simulations

3.2.1 Overview about E-SMR CATHARE 3 Model Implementations

The general layout of the CATHARE 3 model of the E-SMR was not modified with respect to the one illustrated in the reference deliverable D2.6 [9], hence it is not described in this document. A few implementations introduced regarding the reactivity control systems are described in the D 4.3 [3].

3.2.2 Thermal load rejection assessed with the quasi-static BoP model

Using the developed coupling procedure between CATHARE 3 and MODELICA, a thermal load rejection scenario was assessed using the BoP model developed with the ThermoSysPro MODELICA library. The adopted BoP model is described in detail in the reference project deliverable D2.3 [7].

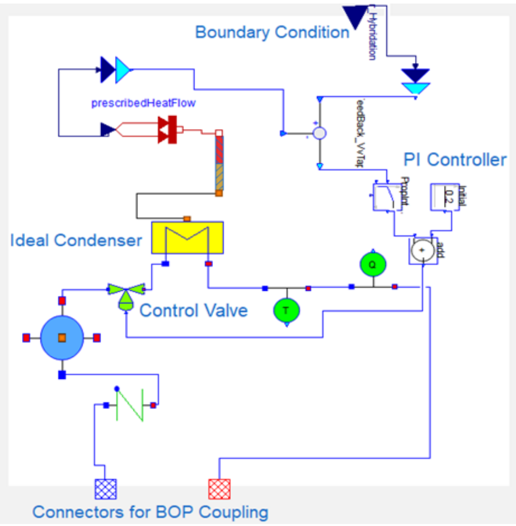

Figure 4: Cogeneration section of the ThermoSysPro BoP, [3]

The addressed thermal load rejection event is simulated as follows:

- Starting from a stationary fully-electric power generation mode of the coupled system, the thermal power demand imposed on the cogeneration section is brought to -86.5 MW, corresponding to a heat-to-power ratio equal to 50 %, through a ramp lasting 600 s as shown in Figure 5. After 1,000 s of cogeneration, a sudden disruption of the thermal power is simulated through a ramp lasting 20 s considering e.g. a sudden failure of the heat user. In this regard, it must be mentioned that the thermal load rejection was simulated with a steep ramp and not with a step, since the adoption of a vertical drop of the thermal load would lead to an abnormal disruption of the co-simulation.
- Using a ramp instead of a step to simulate the thermal load rejection is somehow coherent with what may happen in real conditions. In fact, in a real case scenario, the inertia of the entire system (BoP + Coupling Circuit) would likely make the thermal demand seen by the BoP to follow a downward excursion, which is anyway different from a step, even in response to a sudden and complete loss of the thermal user.

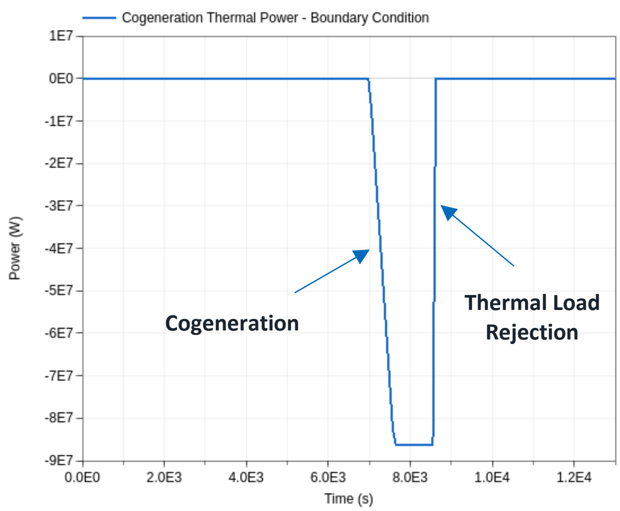

Figure 5: Adopted boundary condition for the thermal load rejection analysis, [3]

3.2.3 CATHARE/MODELICA simulation results

The simulations results are detailed in the deliverable D4.3, [3]. The key points from them are as follows:

- The coupling between CATHARE 3 and MODELICA, managed by Dymola 24x as the co-simulation supervisor and using a quasi-static Balance of Plant (BOP) model, is proved robust. It effectively handled sudden changes in boundary conditions, allowing the system to return to normal full-electric operation after a thermal load rejection event.
- Simulations showed that the reactor's control systems—including reactivity feedback and control rods—maintained safe operation without requiring a SCRAM. When steam and heat flows were shifted between electricity generation and cogeneration, the impact on the primary circuit remained minimal. During AOOs, the core power increased only slightly—by about 0.8%—and temperature and pressure changes in the primary coolant loop were mild.
- Based on the simplified BOP layout used, the results indicate that redirecting rejected cogeneration heat to the turbine allows the system to handle load changes of up to 15% of thermal power without activating safety systems or significantly affecting primary side parameters.

3.3 AOO Analysis conclusions

The analyses conducted within task 4.3 of TANDEM WP4 demonstrate that the coupled code system integrating MODELICA with the thermal-hydraulic system codes ATHLET or CATHARE is applicable for modeling various AOOs. The study of thermal load rejection transients highlights that the E-SMR and its control systems are capable of efficiently compensating for load loss on the secondary side, ensuring the reactor remains within safe operating conditions without requiring a SCRAM. The core and moderator reactivity feedbacks, combined with the regulation of core power through control rods, play a significant role in maintaining stability.

When steam and heat flows are redistributed between power generation and cogeneration, the impact on the reactor's primary side is minimal. However, a sudden increase in cogeneration heat demand at full power can cause the core power to exceed nominal levels. While this might be acceptable depending on the reactor design, a power limitation system—absent in this study—would typically be included to manage such situations.

The study also highlighted the complexity and time-consuming nature of coupling the codes and tuning parameters, especially in ATHLET, to ensure stable and realistic simulations.

4 Simulations of DBA transients, [4]

4.1 CATHARE/MODELICA simulations

4.1.1 CHP plant modeling

For simulation of DBA transients, the E-SMR power plant layout chosen to conduct the safety studies on cogeneration is presented in Figure 6.

The nominal core power is 540 MWth, and the maximum gross electric power output is 177 MWe. The main reactor parameters in nominal operation are reported in Table 2, [4].

A hybrid system consisting of the E-SMR thermal-hydraulic system code model and a cogeneration model in MODELICA was used to simulate the sudden increase or decrease in thermal load due to fast electricity or heat demand changes.

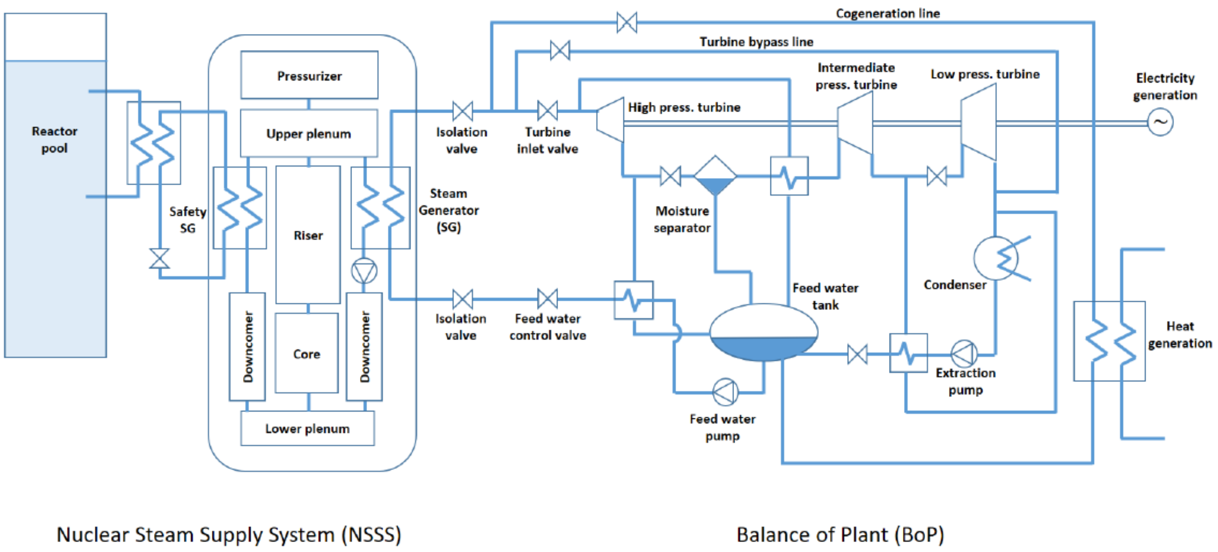

Figure 6: Power plant layout, [4]

The Nuclear Steam Supply System (NSSS) comprises six Steam Generators (SGs) that transfer core thermal power from the primary to the secondary circuit, with coolant flow regulated by six pumps. Additionally, two safety SGs operate without pumps, using natural circulation to remove decay heat under abnormal conditions (DBC-2, 3, and 4). For simplicity, Figure 6 represents one operational SG instead of six and one safety SG instead of two.

The BoP follows a standard power conversion cycle. Steam at 300°C and 45 bar passes through High-Pressure (HP), Intermediate-Pressure (IP), and Low-Pressure (LP) turbines before being condensed, pressurized, and preheated for recirculation to the SGs.

In cogeneration mode, some steam is diverted to a heat exchanger, transferring heat to thermal oil (200°C–250°C) for industrial applications (D2.5, [8]).

The BoP also includes a bypass line, normally closed, which can discharge steam directly into the condenser if secondary circuit pressure rises too quickly.

Table 2: Main reactor parameters in nominal operation, [4]

Parametre		Value
Primary circuit	Core power	540 MWth
	Core inlet temperature	300 °C
	Core outlet temperature	324.5 °C
	Primary mass flow rate	3700 kg/s
	Pressurizer pressure	150 bar

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

Secondary circuit	Steam generator inlet temperature	160 °C
	Steam generator inlet pressure	49 bar
	Steam generator outlet temperature	300 °C
	Steam generator outlet pressure	45 bar
	Mass flow rate (x6 steam generators)	40 x 6 kg/s
	IP turbine inlet pressure	7.55 bar
	LP turbine inlet pressure	0.8 bar
	Condenser pressure	0.07 bar
	Feed water tank pressure	7.15 bar
	Turbine rotational speed	1500 rpm

4.1.2 Simulation of selected DBA scenarios (load rejection leading to a loss of offsite scenario)

<u>Load rejection transient</u>

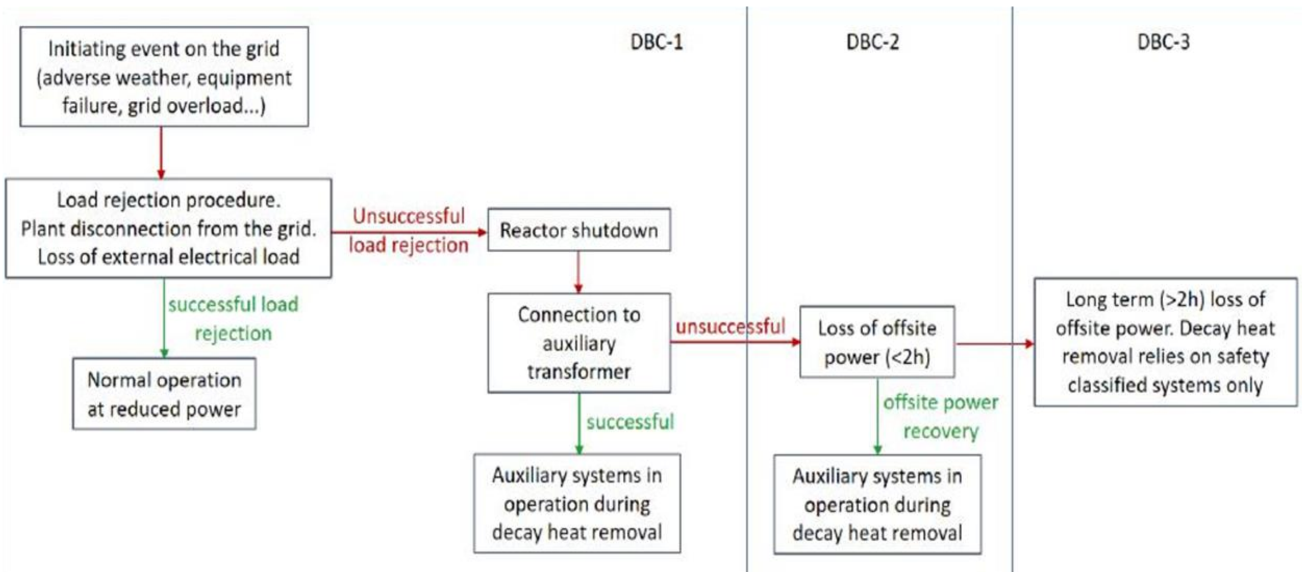

Figure 7: Event three leading to a loss of offsite power scenario, [4]

As it is shown at the Figure 7, a load rejection transient may be successful or lead to a LOOP.

A grid disturbance, like storm damage, can disrupt frequency and voltage, threatening a nuclear plant's auxiliary systems. To prevent shutdown, the plant initiates a load rejection, disconnecting from the grid and redirecting power to key systems. The generator and pumps accelerate, increasing coolant flow and core power.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

The turbine control system adjusts valves to stabilize temperatures, while pressure buildup triggers steam release into the condenser. Control rods are inserted to regulate core power. If successful, the plant stabilizes at 30% capacity until grid restoration. Failure risks reactor shutdown and reliance on auxiliary power, with a possible loss of offsite power.

This scenario examined whether cogeneration helps or worsens plant stability during such events.

Unsuccessful load rejection leading to a LOOP

The long-term loss of offsite power (>2 hrs) was simulated for both fully electric and cogeneration setups to assess their impact on accident severity. Both scenarios began with load rejection at full core power. After 60 seconds, a safety threshold is exceeded, triggering reactor shutdown. With the auxiliary grid unavailable for hours, the plant loses external energy, stopping the primary pumps.

Isolation valves on the steam and feed water lines close, disconnecting the NSSS from the BoP, while intermediate circuit valves open to transfer decay heat to the reactor pool. Since the BoP setup has no influence after isolation, any differences between setups occur only in the first minute. No significant discrepancies were observed beyond that, confirming accident severity remains unchanged.

The results of simulations are documented in the D4.4, [4].

A further simulation of a load rejection transient leading in a LOOP has been performed by GRS using the ATHLET/MODELICA coupling also used for the AOO simulations mentioned in the section 3.1. Also here, the BoP part of the model is switched off during the simulation, since it does not influence the results after SCRAM and secondary isolation. The results show that the coupled code system ATHLET/MODELICA can be reliably used to simulate such transients, ANNEX 4. In this process, the Balance of Plant (BOP) model is only required during the initial phase of the simulation—before secondary side isolation—and can be switched off thereafter.

However, the current ATHLET/MODELICA coupling does not support full deactivation of a Functional Mock-up Unit (FMU). Instead, in this case, the BoP model is deactivated by assigning nominal boundary conditions to MODELICA. The downside of this approach is that, despite being unnecessary after the initial phase, the BoP model continues to run, which may lead to inefficient resource usage.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

4.2 Standalone modeling and results

Additional DBA simulations focused on loss of offsite power (LOOP) were conducted separately by ENEA and ENERGORISK to compare ATHLET and CATHARE models, see Figure 8.

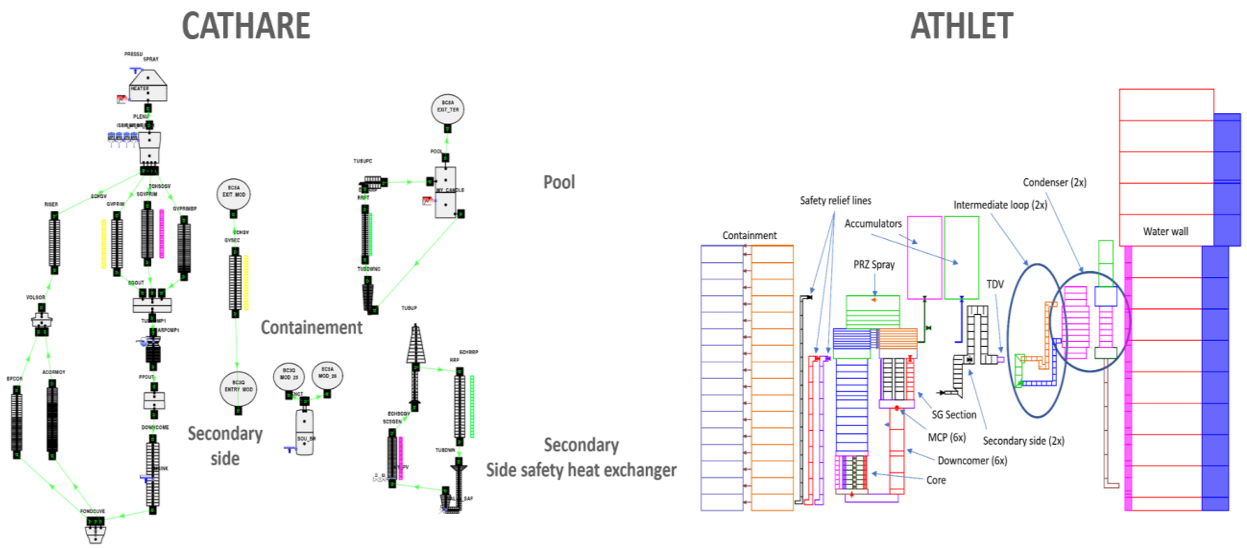

Figure 8: Safety codes models used for standalone SS conditions, [ANNEX 1, ANNEX 2]

This section presents the results of comparing the calculations of the IE "Loss of external grid" performed on the calculation models of the E-SMR reactor plant developed for the ATHLET and CATHARE codes. For comparison, the main representative parameters that characterize the operation of safety systems and the thermal inertia of the reactor plant are selected:

- Core inlet mass flow rate
- Pressurizer pressure
- Core outlet temperature
- Mass flow rate from relief valve in the pressurizer
- Mass flow rate discharged by accumulators in the primary side
- Core inlet temperature

Table 3: Steady state condition parameters

Parameter	E-SMR dataset	CATHARE SS	ATHLET SS
Nominal thermal power [MW]	540	540	540

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

Primary pressure [bar]	150 (Tsat= 342)	152	152
Pressurizer level [m]	1	0.999	0.95
Secondary pressure [bar]	45	45	45.7
Secondary coolant flowrate [kg/s]	40	40	249
Primary coolant flow rate [kg/s]	3700	3722	3702
Core inlet temperature [°C]	300	299	301
Core outlet temperature [°C]	324.5	323.2	326

Comparison of CATHARE 3/ATHLET standalone LOOP simulations, ANNEX 3

The results of calculations performed using computational models of the SMR reactor developed for the ATHLET and CATHARE codes show a general coincidence of the trend of the main parameters and the magnitude of temperature and pressure stabilization by the end of the calculation. The magnitude of the disturbances introduced by the operation of safety systems is also comparable (for example, a drop in the temperature of the primary circuit coolant when the heating system is activated). At the same time, it should be noted that there is a significant difference in the rate of decrease in the temperature of the primary circuit, which, with a comparable level of residual energy release and the values of stabilization parameters, can be explained by different approaches to modeling steam generators. The details of each simulation are presented in ANNEX 1 and ANNEX 2.

4.3 DBA analysis conclusions

The analysis conducted in Report D4.4 focuses on the simulation of relevant transient scenarios for the safety assessment of cogenerating Small Modular Reactors (SMRs). It was postulated that the DBA most impacted by cogeneration is the loss of the external electrical load leading to a long-term (>2 hrs) loss of offsite power. Cogeneration can reduce the risk of failing a load rejection procedure and incurring in a loss of offsite power scenario, which is a design basis transient that can also evolve into a DBA if it lasts for several hours; the actual impact of cogeneration in preventing the loss of offsite power depends on the initiating event that causes the load rejection. However, it does have a stabilizing effect on electrical load, potentially reducing the risks associated with external power loss.

Regarding Design Basis Accidents, the study finds that cogeneration has a limited effect, and generally not a negative one. The only scenario where its presence could be negatively influential is a small steam piping failure, where the additional thermal extraction might slightly increase the risk of rupture. However, once the reactor is shut down and isolated, cogeneration ceases to play a role in cooling processes, which are managed entirely by the reactor’s dedicated safety systems.

One of the most relevant aspects of D4.4 is its detailed assessment of prolonged external power loss. While cogeneration might help stabilize the electrical load under normal operating conditions, its role becomes negligible once power is lost entirely, as safety protocols ensure the isolation of the Nuclear Steam Supply System from the Balance of Plant, preventing interaction between the systems.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

5 Conclusion

5.1 Studies Conducted in the frame of WP4 of Tandem project

Based on methodology defined in D4.2, different aspects of SMR operation under transient and accident conditions were explored. While D4.3 focuses on modeling thermal load rejection transients, D4.4 extends the analysis by investigating the implications of cogeneration on safety events. Both studies contribute valuable insights into the stability, resilience, and flexibility of SMRs with cogeneration capabilities.

To conduct these studies, sophisticated modeling approaches were employed, including the use of MODELICA coupled with the thermal-hydraulic codes ATHLET and CATHARE. Additionally, a detailed NSSS model developed using CATHARE3 was integrated with BoP model built with the ThermoPower library in MODELICA. PID controllers were designed to regulate key physical parameters of both NSSS and BoP, ensuring realistic simulations of automatic system responses during load rejection and external power loss events.

In addition, standalone calculations using ATHLET and CATHARE models for the SMR reactor show a general consistency in parameter trends and stabilization of temperature and pressure. Disturbances from safety system operations, such as coolant temperature drops when heating is activated, are also comparable. However, a notable discrepancy exists in the rate of primary circuit temperature decrease, likely due to differences in steam generator modeling despite similar residual energy release and stabilization values.

5.2 Key Results Obtained

Cogeneration and Load Rejection Transients

- Cogeneration reduces the severity of load rejection transients by stabilizing heat extraction, limiting thermal surges in primary and secondary circuits.
- The presence of cogeneration lowers the electrical load, which reduces the stress on reactor components during sudden load rejection scenarios.
- However, in cases where thermal and electrical loads are lost simultaneously, the beneficial effects of cogeneration are diminished.

Impact of Cogeneration on External Power Loss

- Once external power is lost entirely, the NSSS is isolated, preventing cogeneration from playing a role in mitigating accident severity.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

- The response of the reactor remains unchanged in cogeneration versus pure electricity generation modes, as post-trip cooling is entirely managed by the reactor's safety systems.

Cogeneration and DBAs

- Cogeneration does not significantly alter DBA severity but could contribute to prevention strategies by stabilizing power demand.
- A small steam piping failure is the only scenario where cogeneration may slightly increase rupture risks, though safety protocols ensure proper mitigation.

5.3 General Conclusions and Future Research Directions

The findings from AOO and DBA selected transients demonstrate that cogeneration offers advantages for managing certain transients, particularly load rejection scenarios, but does not have a significant impact on more severe accidents such as external power loss. The modeling techniques developed within the TANDEM project have proven valuable in assessing E-SMR behavior under different operating conditions, providing insights that can inform reactor design, control system optimization, and operational safety improvements.

Looking forward, future research should explore:

- **Enhanced reactor power limitation systems**: Investigating control mechanisms to automatically limit core power when unexpected load increases occur in cogeneration mode.
- **Expanded accident modeling**: Extending studies to analyze more complex interactions between cogeneration systems and severe accident conditions.
- **Optimization of dynamic response control**: Refining PID controller configurations to improve automatic system stability and response times.
- **Integration with energy distribution networks**: Studying how cogeneration-enhanced SMRs can interact with smart grids to optimize energy delivery and improve resilience.
- **Extension of SMR designs**: Beside E-SMR other (generic) concepts which might be of interest for Europe should be investigated for the integration into a HES and safety evaluation.

Overall, the results of TANDEM WP4 confirm the viability of cogeneration in SMR designs and highlight promising directions for continued development in nuclear safety and operational efficiency.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

5.4 Policy briefs

After the WP4 safety assessment and based on the results of D4.3 and D4.4, WP4 members developed the following policy briefs.

Key Challenges

- In many instances, SMRs will be located next to industrial clusters meaning therefore licensing will need to take into account requirements stemming from the other industries in order to ensure compatibility between the nuclear and industrial regulatory frameworks and avoid duplication/overlaps
- Safety requirements may need to be updated to consider potential impacts from surrounding sites.
- SMRs are likely to be sited close to populated areas raising potential issues in terms of public acceptance.

Policy Recommendations

- Ensure that the nuclear safety regulators have the resources they need (e.g skills and workforce) to support their involvement in project development early on (pre-licensing).
- Establish a collaborative framework between nuclear safety regulators and regulators covering other industries.
- Ensure involvement of downstream users in the licensing process at an early stage (pre-licencing).
- Ensure that nuclear safety regulators provide clear and understandable information to the public early on in order to answer any concerns.
- Update the licensing framework to take into account the co-generation possibilities linked to the integration of SMRs within hybrid energy systems.

Bibliography

[1] Pucciarelli A., et al, (2023). Status report on safety analysis in Europe from the operational flexibility and cogeneration viewpoint. TANDEM technical report D4.1.

[2] Miss J., et al, (2023). Identification of potentially impacted safety margins and methodology for safety analysis of a SMR integrated in a hybrid system. TANDEM technical report D4.2.

[3] Buchholz S., et al, (2025). Report of operational transient safety case Studies for a SMR with cogeneration. TANDEM technical report D4.3.

[4] Olita P., et al, (2025). Report of operational Design Basis Accident case studies for a SMR with cogeneration. TANDEM technical report D4.4.

[5] International Atomic Energy Agency (IAEA): Deterministic Safety Analysis for Nuclear Power Plants, Specific Safety Guide. IAEA Safety Standards Series, SSG-2 (Rev. 1), ISBN 978-92-0-102119-9, IAEA, 2019.

[6] IAEA, Guidance on Nuclear Cogeneration, IAEA Nuclear Energy Series No. NP-T-1.17, 2019.

[7] G. Simonini., et al, (2024) MODELICA models description for the 'TANDEM' library, TANDEM technical report D2.3.

[8] Amezcua, et al, (2024). Hybrid System simulator description.TANDEM technical report D2.5.

[9] Lombardo C., et al,(2024). CATHARE SMR model description. TANDEM technical report D2.6.

[10] Buchholz S., et al, (2024), ATHLET SMR model description, D2.7 TANDEM technical report.

ANNEX 1. LOSS of OFFSITE POWER ATHLET calculation (ENERGORISK)

Table 4: Steady state parameters at the beginning of the transient

Parameter	E-SMR dataset	CATHARE SS	ATHLET SS (E-risk results)
Nominal thermal power [MW]	540	540	540
Primary pressure [bar]	150 (Tsat= 342)	152	152
Pressurizer level [m]	1	0,999	0.95
Secondary pressure [bar]	45	45	45.7
Secondary coolant flowrate [kg/s]	40	40	249
Primary coolant flow rate [kg/s]	3700	3722	3702
Core inlet temperature [°C]	300	299	301
Core outlet temperature [°C]	324.5	323,2	326

Table 5: Event chronology

EVENT	MAAP	TRACE	RELAP5	AC2	APROS	ATHLET
Start of Transient	0	0	0	0	0	0.0
FW pump trip	0	0	0	0	0	0.0
Primary pump trip	0	0	0	0	0	0.0
Primary pressure 172.3 bar	4.0	4.3	0		4.57	4.15
RPV safety valve opens	4.1	4.4			4.89	4.1 (start)
Stop FW flow	4.0	5.0	0		4.89	5.4
Containment pressure 1.5 bar	Not reached	7.1				Not reached

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

SCRAM	4.0	8.1	0		4.89	0.0
S-CSG valve opens	4.1	8.1			4.89	3.8
Accumulator injection starts	11476	189711	4100	113840	99102	116508
Accumulators empty				115340	100171	
Water-wall temperature = 98°C or Saturation	Not reached	610000	-	29800		
End of calculation	72000	610000	6500	300000	259200	140000

Table 6: Figures

#	Identificator	Description
Figure 9	N_core	Core Power, W
Figure 10	P_prz	Pressurizer Pressure, Pa
Figure 11	L_prz	Pressurizer Collaps Level
Figure 12	mf_Core	Core inlet Mass flow rate, kg/s
Figure 13	T_inlet T_outlet	Core inlet temperature, C Core inlet temperature, C
Figure 14	mf_Accu1 mf_Accu2	Accumulator-1 Mass flow rate, kg/s Accumulator-2 Mass flow rate, kg/s
Figure 15	L_Accu1 L_Accu2	Accumulator-1 Level, m Accumulator-2 Level, m
Figure 16	W_mass_Accu1 W_mass_Accu2	Accumulator-1 Mass of water, kg Accumulator-2 Mass of water, kg
Figure 17	P_Cont	Containment (bottom) Pressure, Pa
Figure 18	T_WW_max	Water Wall (MAX) Temperature, C

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

Figure 19	P_W-Wall	Water Wall (top) Pressure, Pa
Figure 20	P_second	SG1 secondary Pressure, Pa
Figure 21	mf_SG1_2k mf_SG2_2k	SG1 secondary Mass flow rate, kg/s SG2 secondary Mass flow rate, kg/s

The results of calculating the main parameters of the reactor plant are shown in Figure 9- Figure Figure 21 to demonstrate the response to the safety systems and the thermophysical properties of the plant. The initiating event is the loss of an external electrical load, which triggers the reactor's emergency protection and reduces power to the level of residual energy, shutting down the main coolant pumps and feed pumps of the steam generator. Termination of forced circulation of the coolant through the core leads to an increase in the pressure and temperature of the primary circuit coolant, as a result of which the pressurizer safety valve is activated to limit further parameter growth. After the main coolant pumps are stopped, the valves of the passive residual energy discharge system are opened and stable circulation of the coolant through the emergency cooling heat exchangers is established. The natural circulation of the coolant and heat sink from the primary circuit to the Water Wall leads to a gradual decrease in the parameters of the primary circuit, as a result of which the conditions for the operation of the cooling system and the supply of the primary circuit with water from accumulators are achieved, which leads to a sharp decrease in the pressure of the primary circuit. By the time the calculation is completed, the parameters of the reactor plant are stabilized as a result of achieving an equilibrium of the heat sink and the level of residual energy.

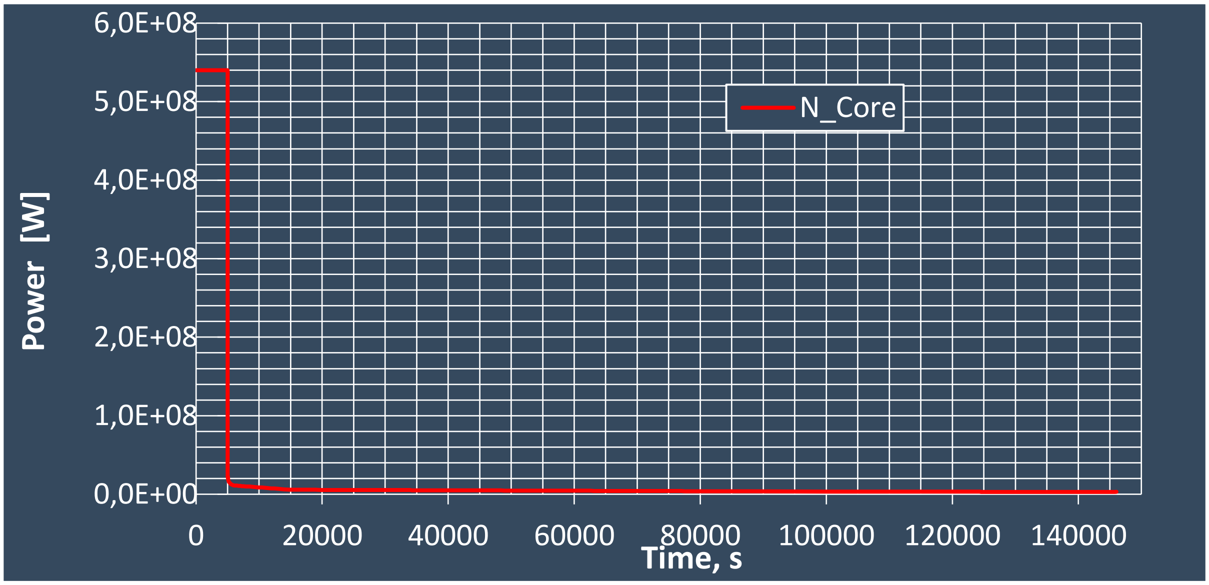

Figure 9: Core power

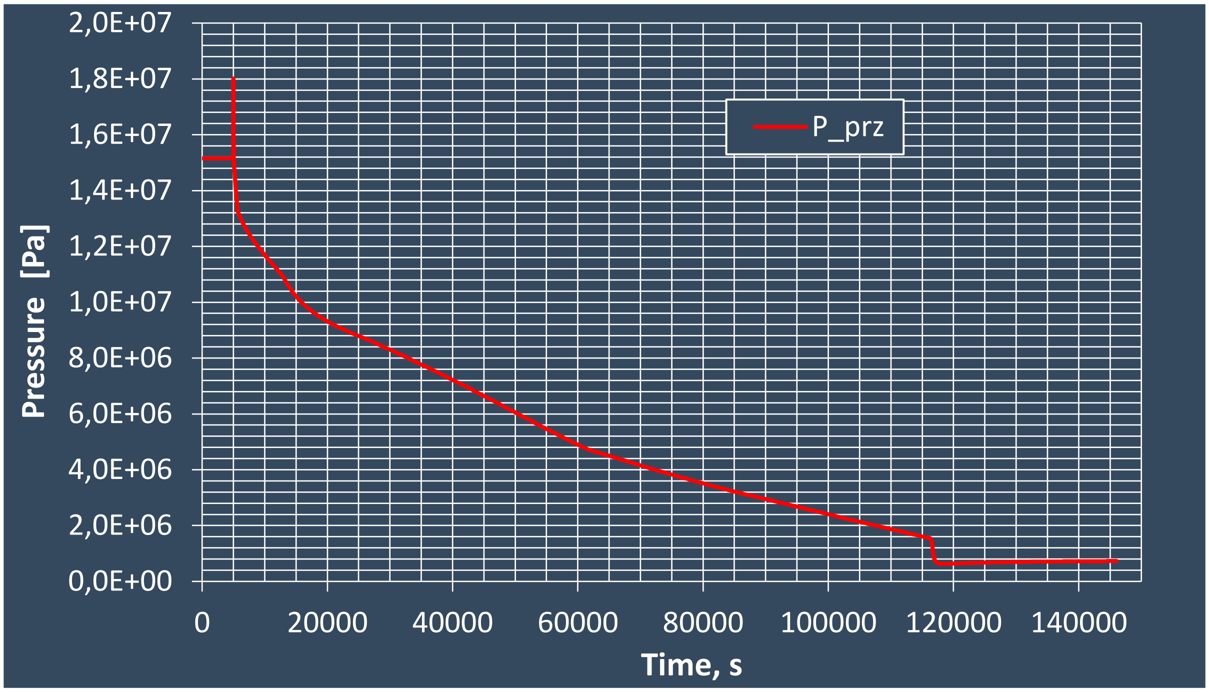

Figure 10: Pressurizer pressure

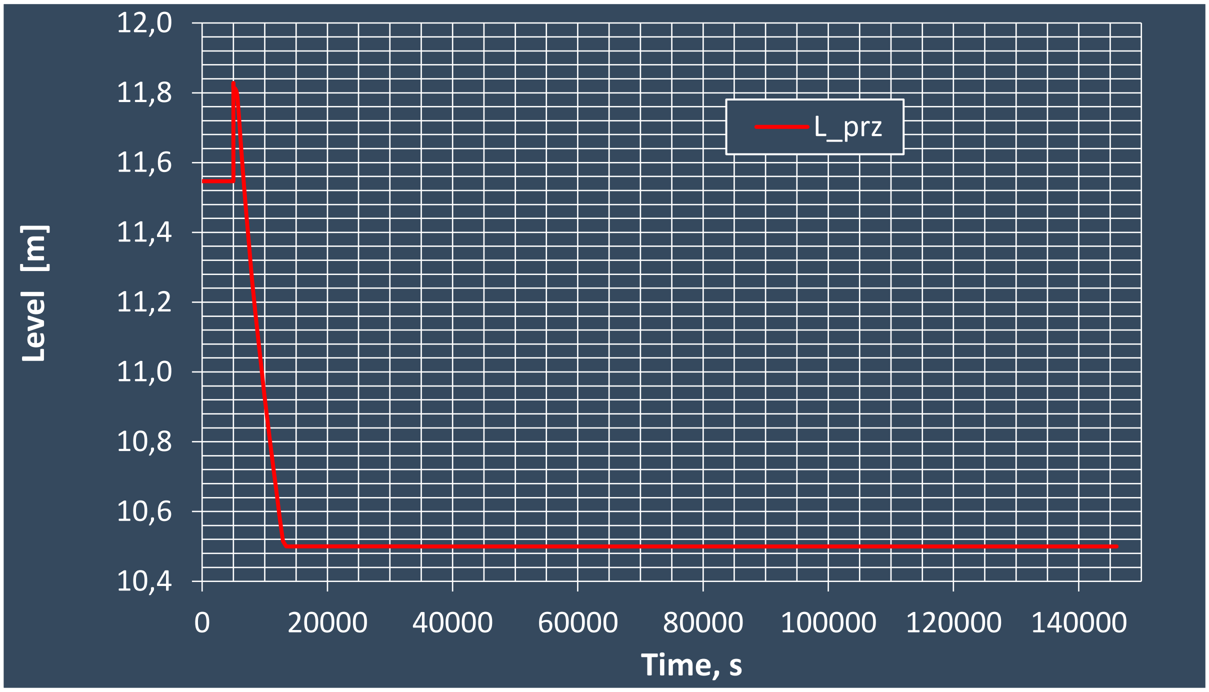

Figure 11: Pressurizer collaps Level

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

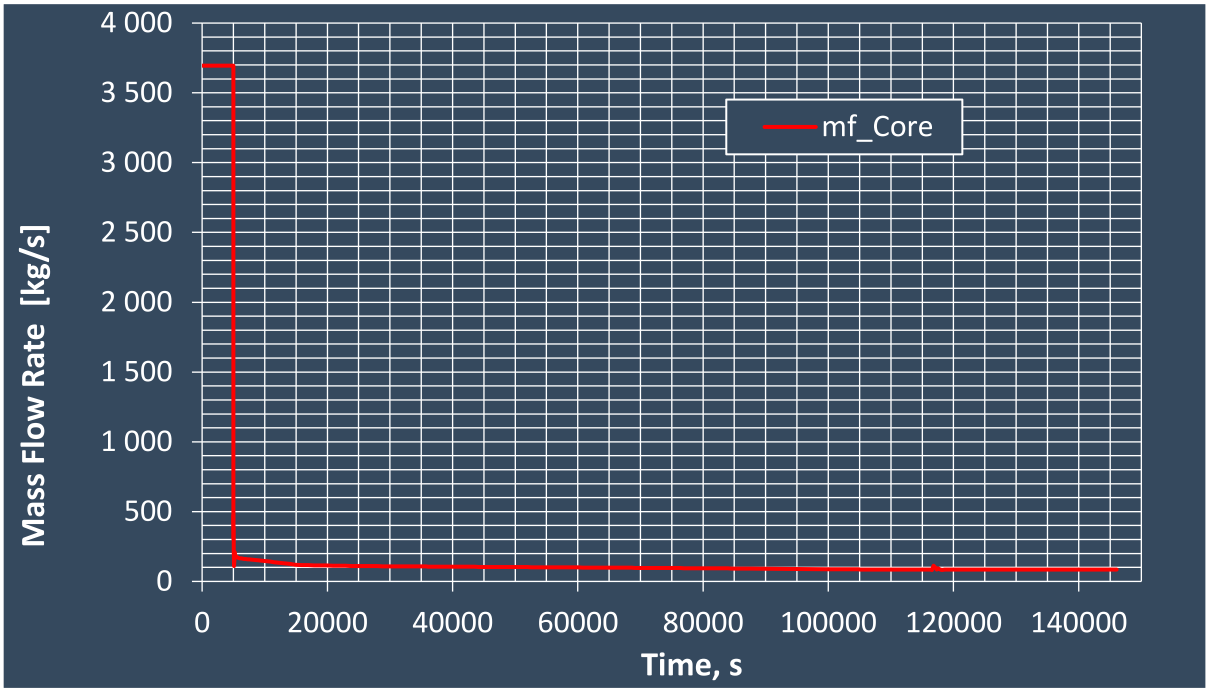

Figure 12: Core inlet mass flow rate

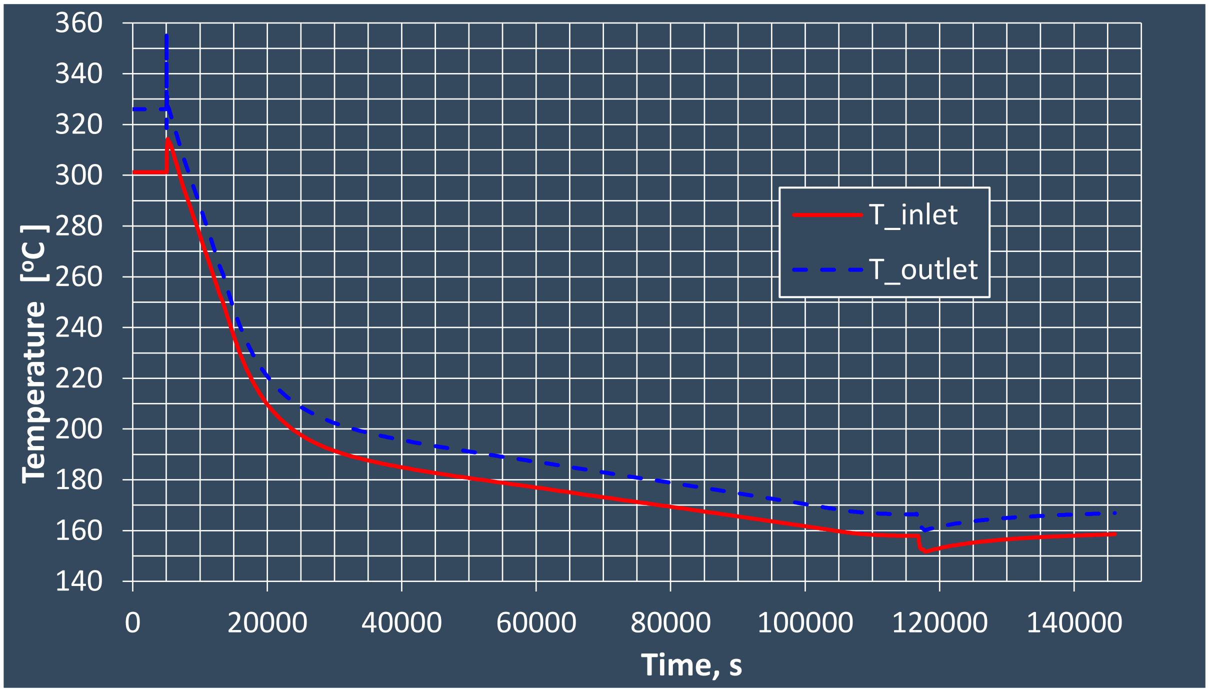

Figure 13: Core inlet & outlet temperature

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

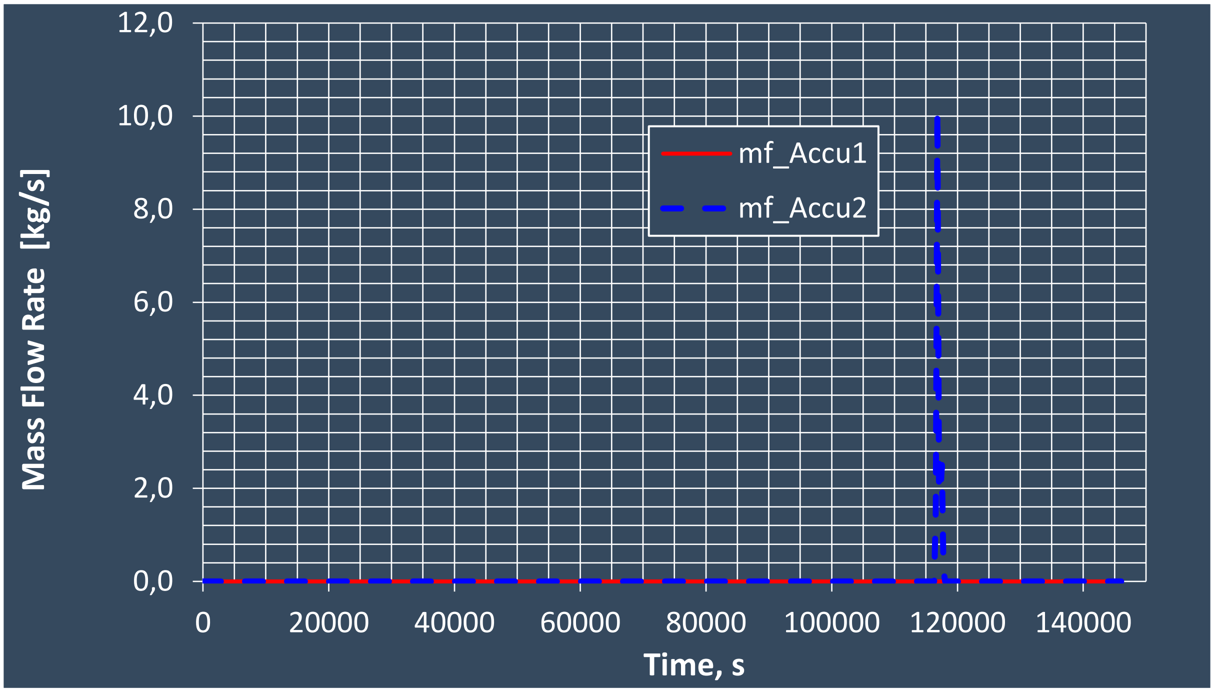

Figure 14: Accumulator-1,2 mass flow rate

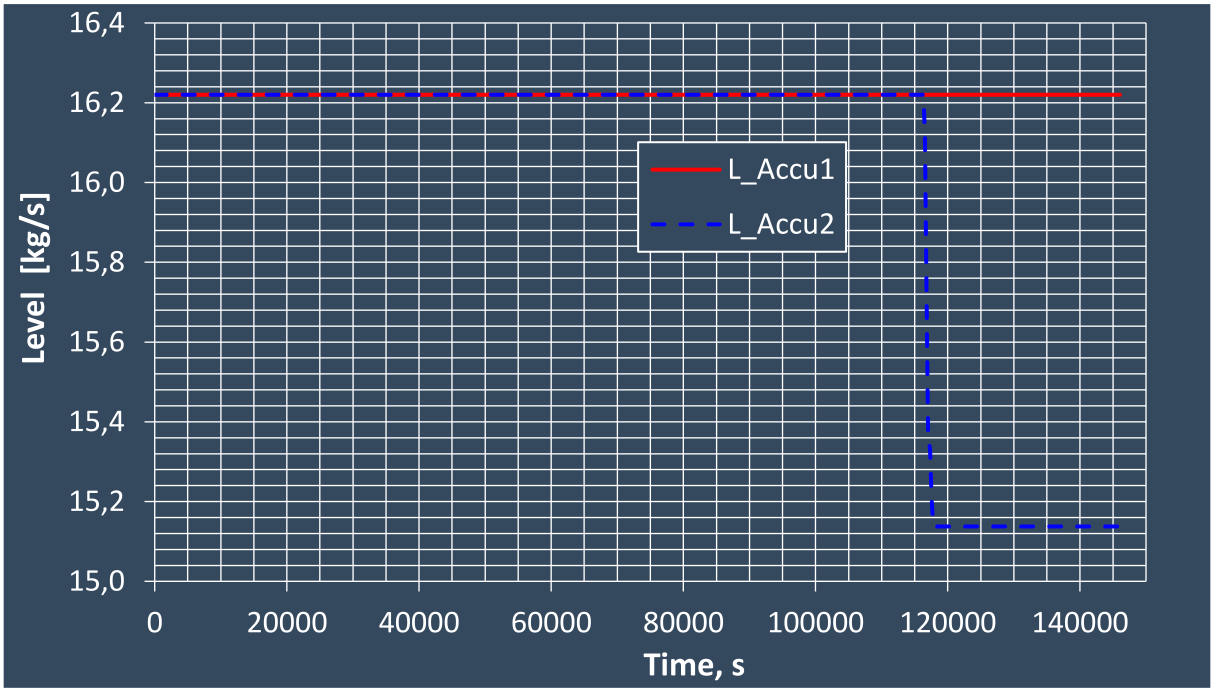

Figure 15: Accumulator-1,2 level

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

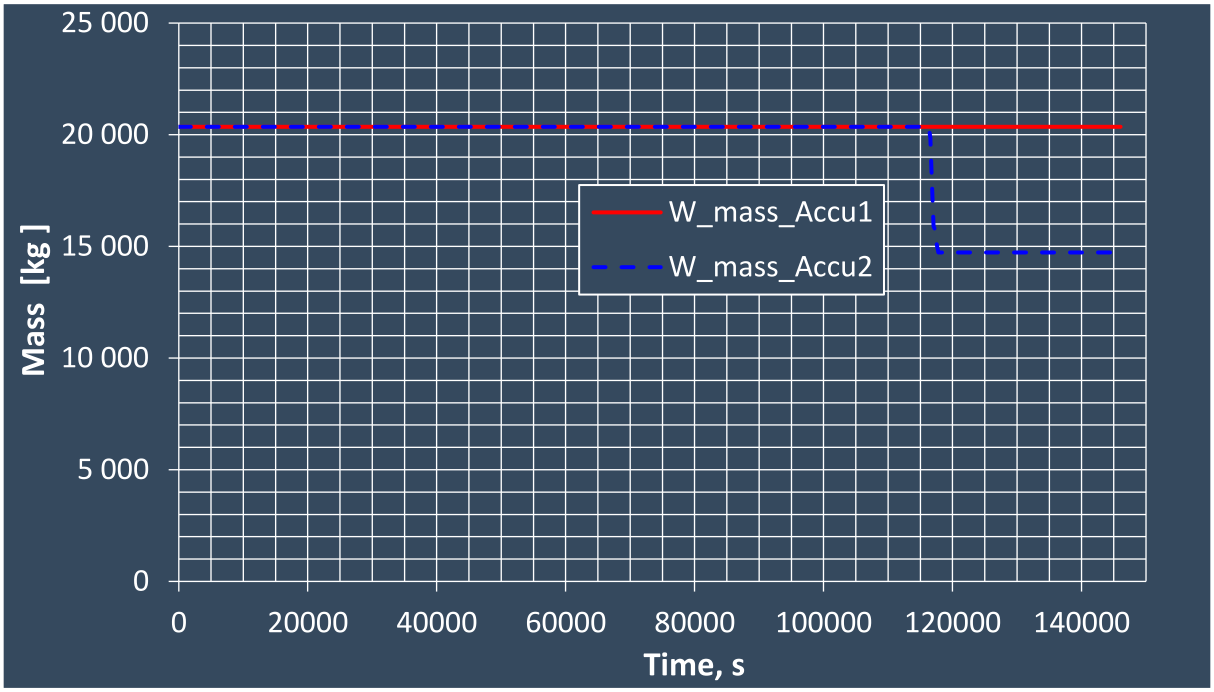

Figure 16: Accumulator-1,2 mass of water

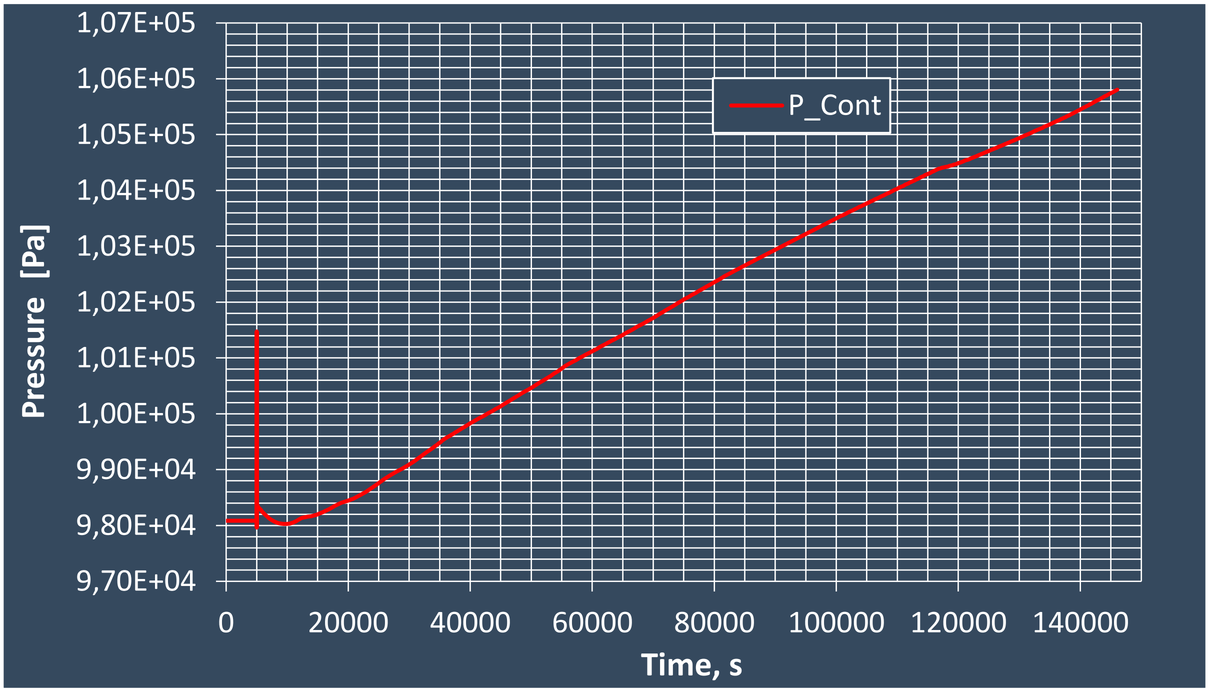

Figure 17: Containment (bottom) Pressure

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

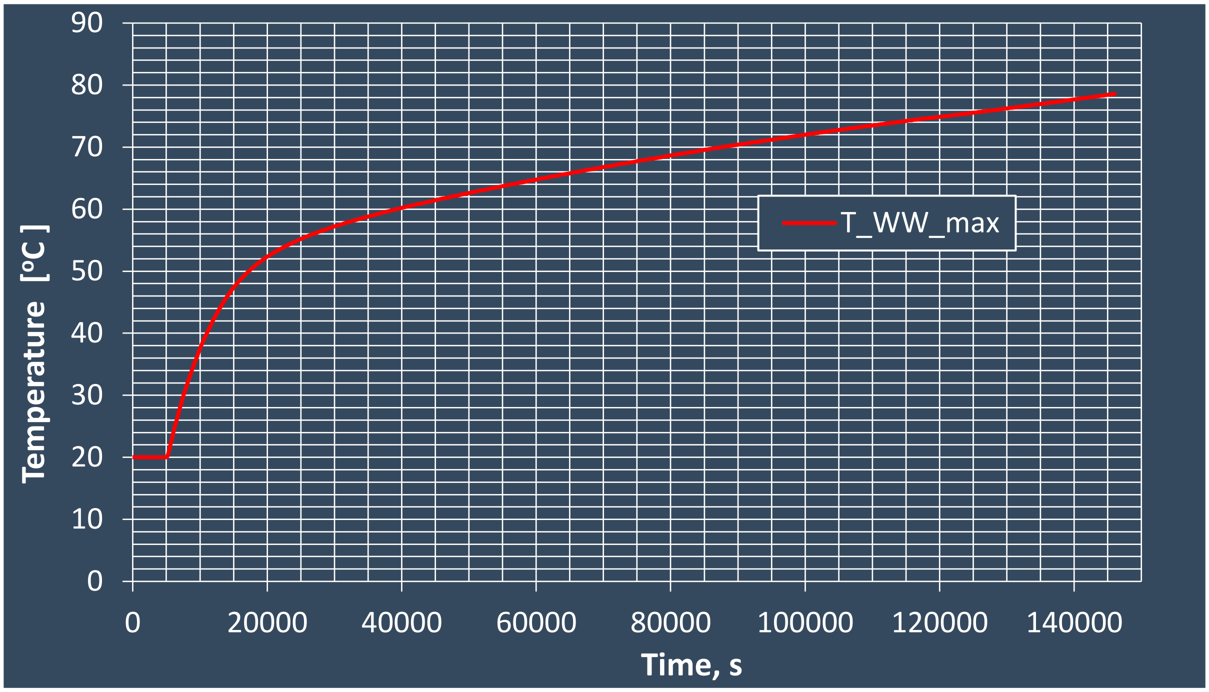

Figure 18: Water Wall (MAX) temperature

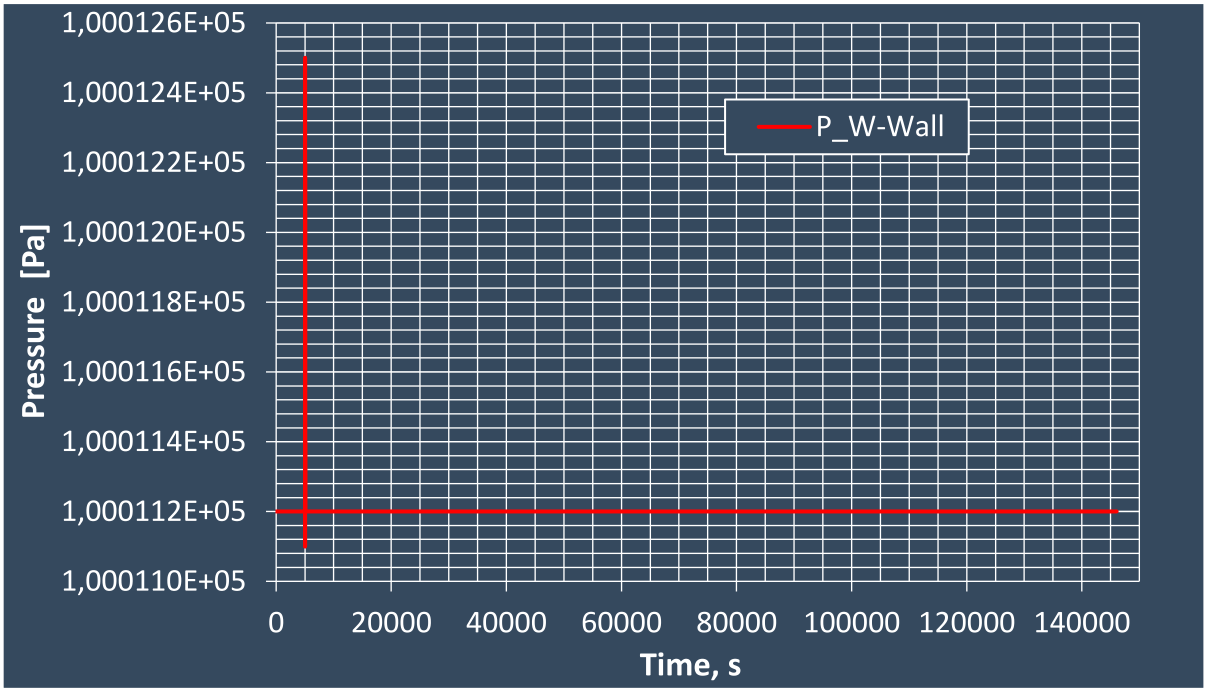

Figure 19: Water Wall (top) Pressure

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

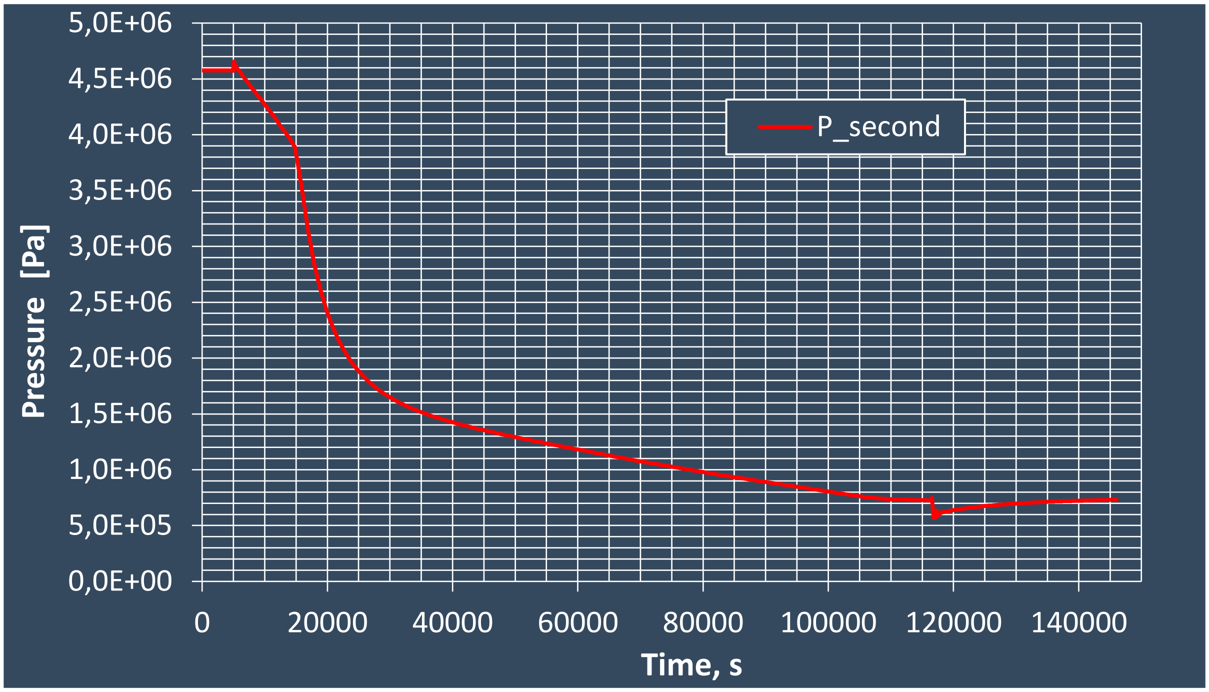

Figure 20: SG1 secondary Pressure

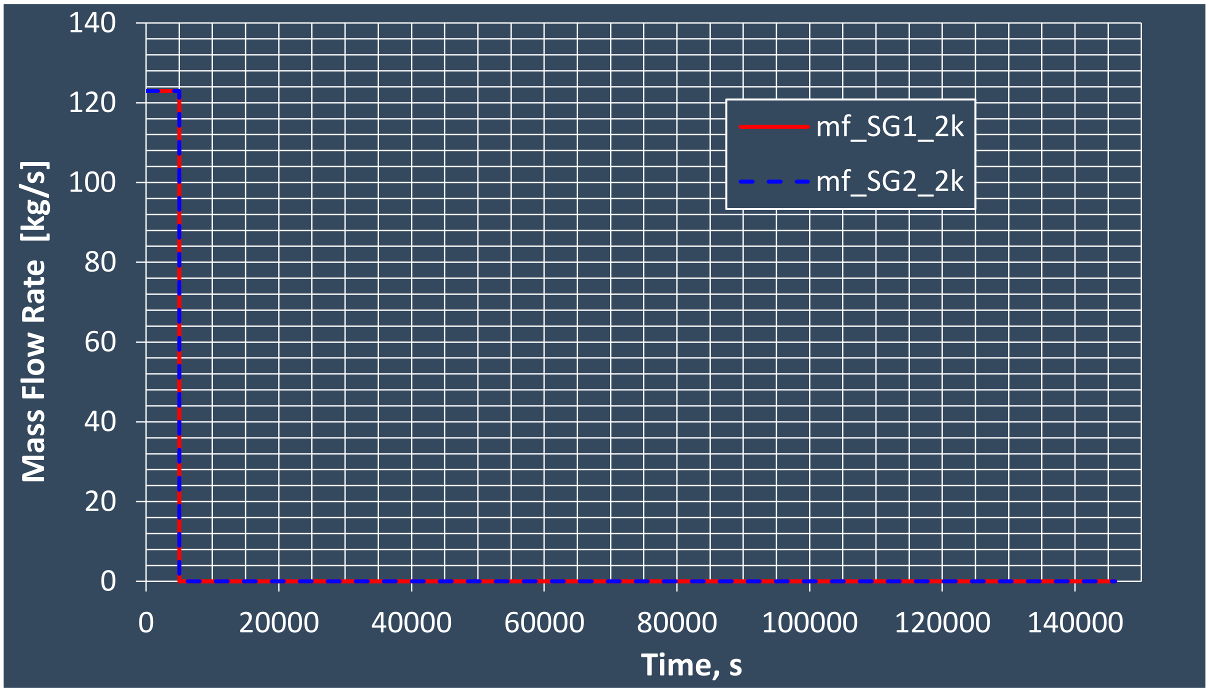

Figure 21: SG1,2 secondary Mass flow rate

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

ANNEX 2. CATHARE3 LOSS of OFFSITE POWER simulation (ENEA)

DESCRIPTION

The E-SMR model has been developed using the thermal-hydraulic system code CATHARE3 [1], within the WP2 of the TANDEM project. All the hydraulic components of the E-SMR are modelled as a set of 1-D and 0-D elements. The core component includes a 0-D reactor physics model (point kinetics) to compute the core power evolution during transients. The details of the CATHARE nodalization are reported into the deliverable D2.6 [2]. For the simulation of the LOOP transient, here reported, a relief valve has been added to the top of the pressurizer, with aim to extract steam from the primary circuit when its pressure becomes larger than 172 bar (open pressure 172 bar – 166 close pressure).

The nodalization of the CATHARE3 model is presented in Figure 2.1.

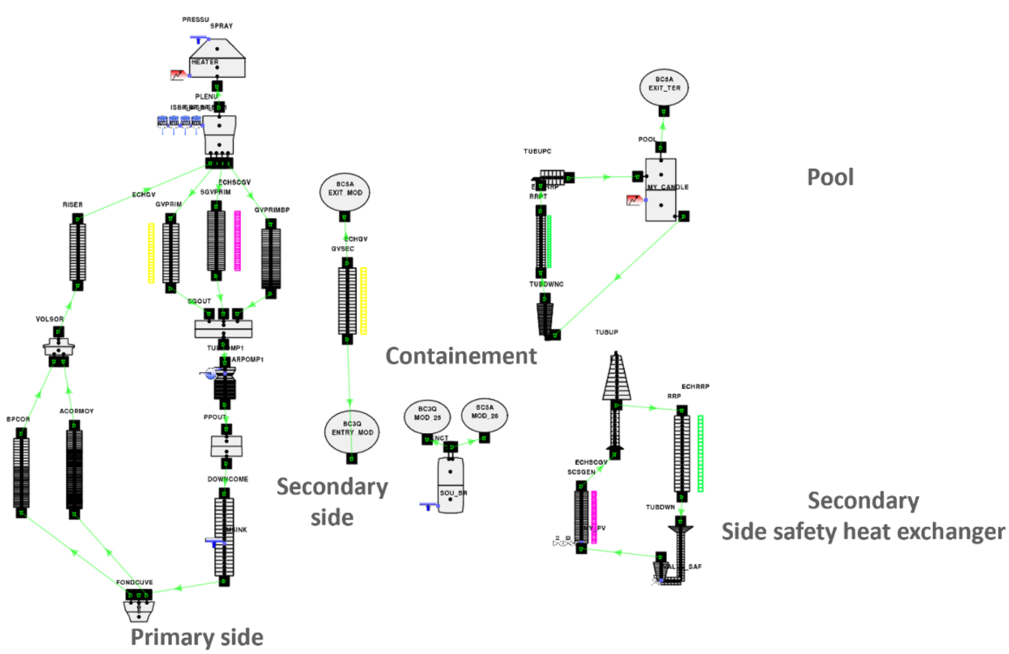

Figure 22*:* CATHARE3 noding of the E-SMR

The Steady State conditions reached in the plant are reported in Table 7: Steady state condition reached in the CATHARE versus E_SMR dataset***.***

Table 7: Steady state condition reached in the CATHARE versus E_SMR dataset

Parameter	E-SMR dataset	Cathare SS
Nominal thermal power [MW]	540	540
Primary pressure [bar]	150 (Tsat= 342)	152
Pressurizer level [m]	1	0.999
Secondary pressure [bar]	45	45
Secondary coolant flowrate [kg/s]	240	240
Primary coolant flow rate [kg/s]	3700*	3722
Core inlet temperature [°C]	300	299
Core outlet temperature [°C]	324.5	323.2

*The value of the primary coolant flow rate equal to 3700 kg is the sum of the flow rate in the core (3550 kg/s) and the by pass flowrate (150 kg/s).

The list of events used to simulate the LOOP is reported in Table 8: List of events to simulate LOOP transient.

Table 8: List of events to simulate LOOP transient.

EVENT	CATHARE
Start of Transient	0
FW pump trip	0
Primary pump trip	0
RPV safety valve opens	

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

Containment pressure 1.5 bar	Not reached
SCRAM	0.0
Accumulator Pressure	15 bar

The transient start with the:

1. Scram of the reactor (stop of the core power and consequently introduction of the decay heat power)
2. Stop of pump
3. Stop of the feedwater in the secondary side of steam generators
4. Start of the emergency steam generators
5. Automatically opening of the relief valve in the PRZ

The results of the transient are reported in the Figures below.

The primary flowrate versus time is reported in Figure 23, at time 0 when is imposed the stop of pump the flow rate in the primary side decreases rapidly from the nominal value (3550 kg/s) up to a value close to 60 kg/s.

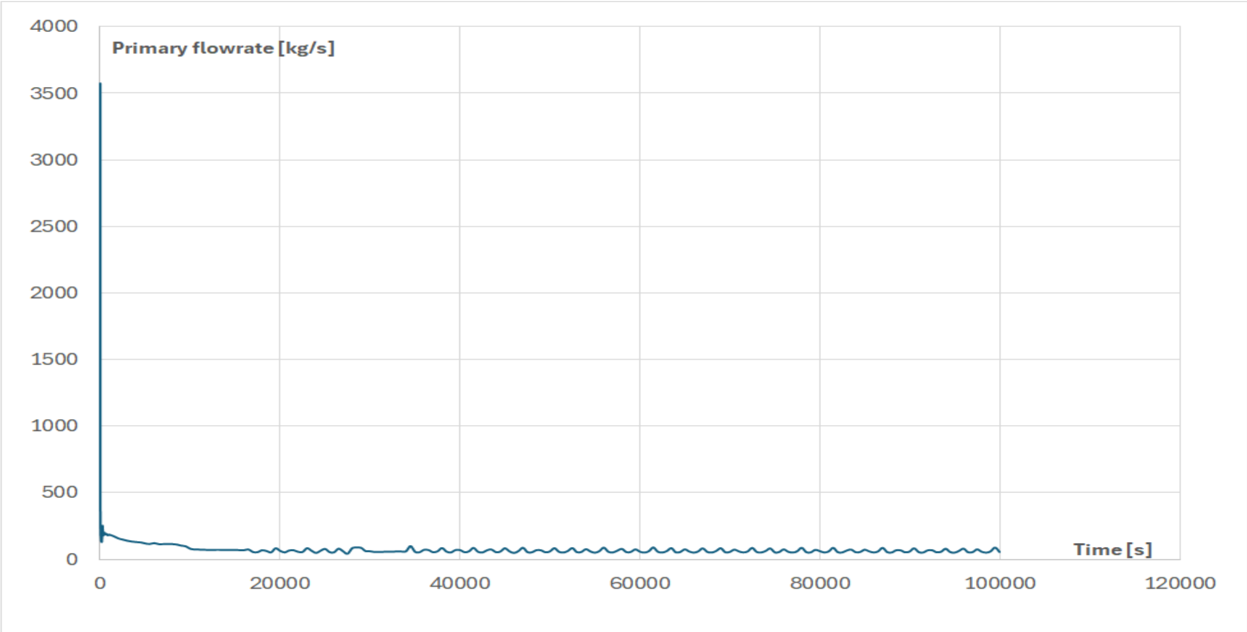

Figure 23: Primary flowrate versus time

As you can see in the Figure 24the pressure in the pressurizer start to increase immediately after the scram of the reactor, when the pressure in the PRZ reaches the 172 bar the relief valve is opened and the two phase flow is discharged and the relief valve is closed when the pressure in the PRZ is below the 166 bar.

At time equal to 28000 s, when the pressure in the PRZ reaches the value of 15 bar, the 4 accumulators intervene and discharge in the primary side a flowrate of cold water (the intervention pressure of the 4 accumulators was set at 15 bar and they are filled with water at 25 °C).

The total flowrate discharged by the accumulators in the primary side is reported in Figure 24.

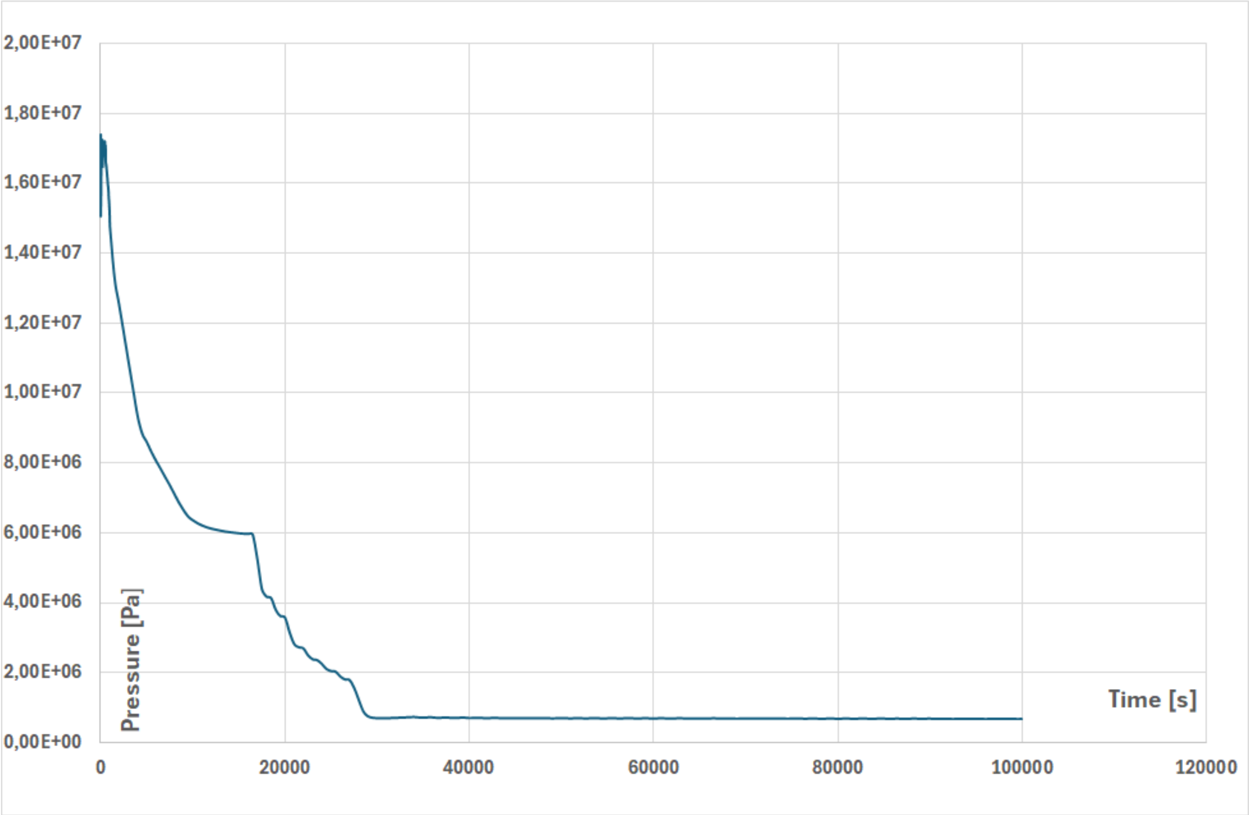

Figure 24: Pressurizer pressure

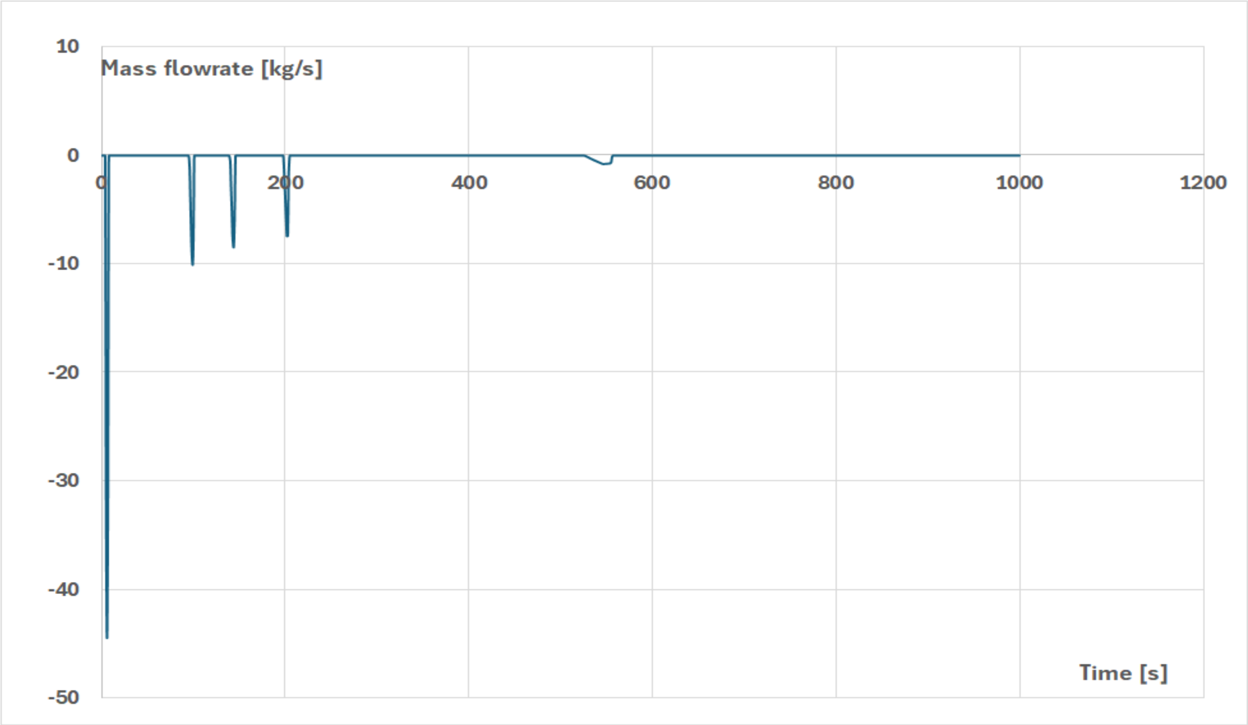

Figure 25: Mass flowrate from relief valve in the pressurizer

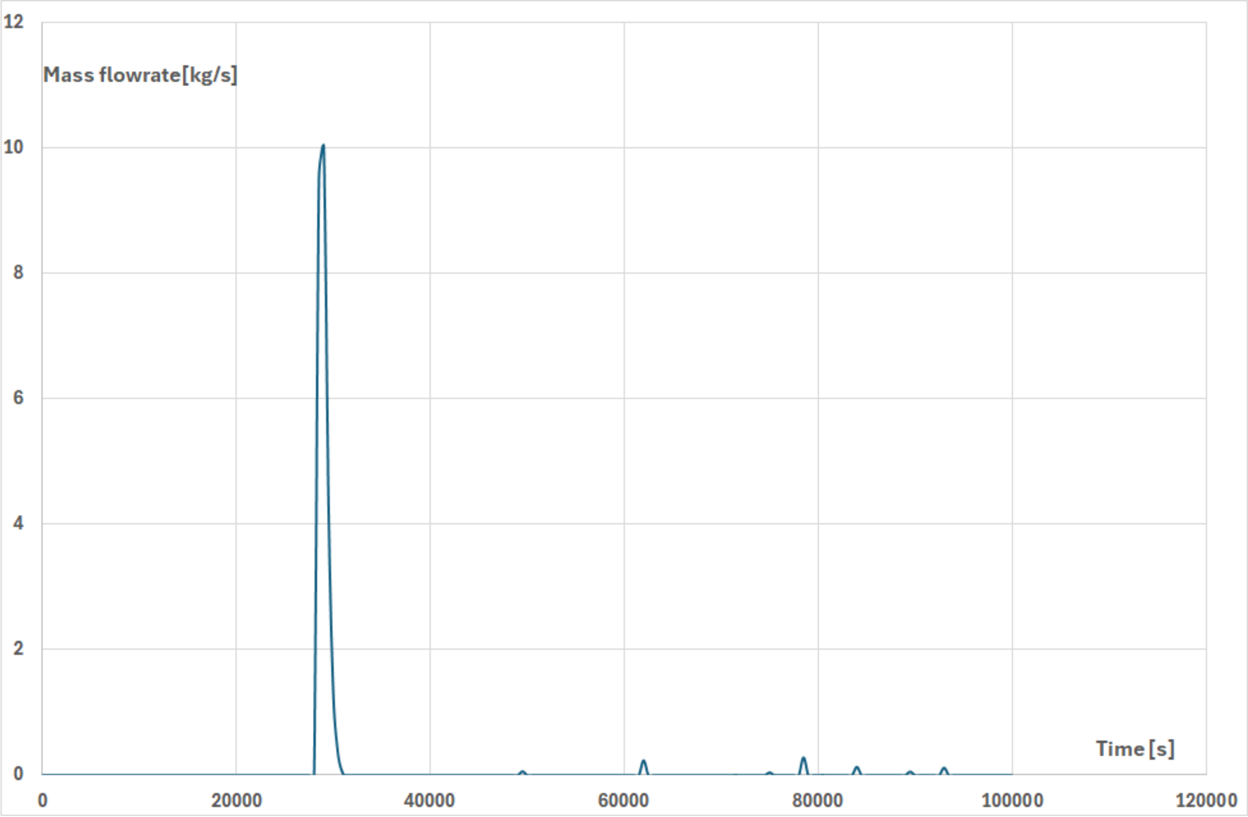

Figure 26: Mass flowrate discharged by accumulators in the primary side

The Figure 27 shows the inlet primary temperature, as you can see the temperature increases in the first part of the transient (when the pressure in the primary side increases) and decreases when the pressure decreases.

In the Figure 28 is reported the outlet primary temperature: as you can see in the first seconds of the simulation the outlet primary temperature reaches a maximum value equal to 353 °C.

This value is greater than the saturation temperature at 151 bar: this evidence needs to be considered in the future to assure the possibility to manage this accident in the E-SMR nuclear power plant safely.

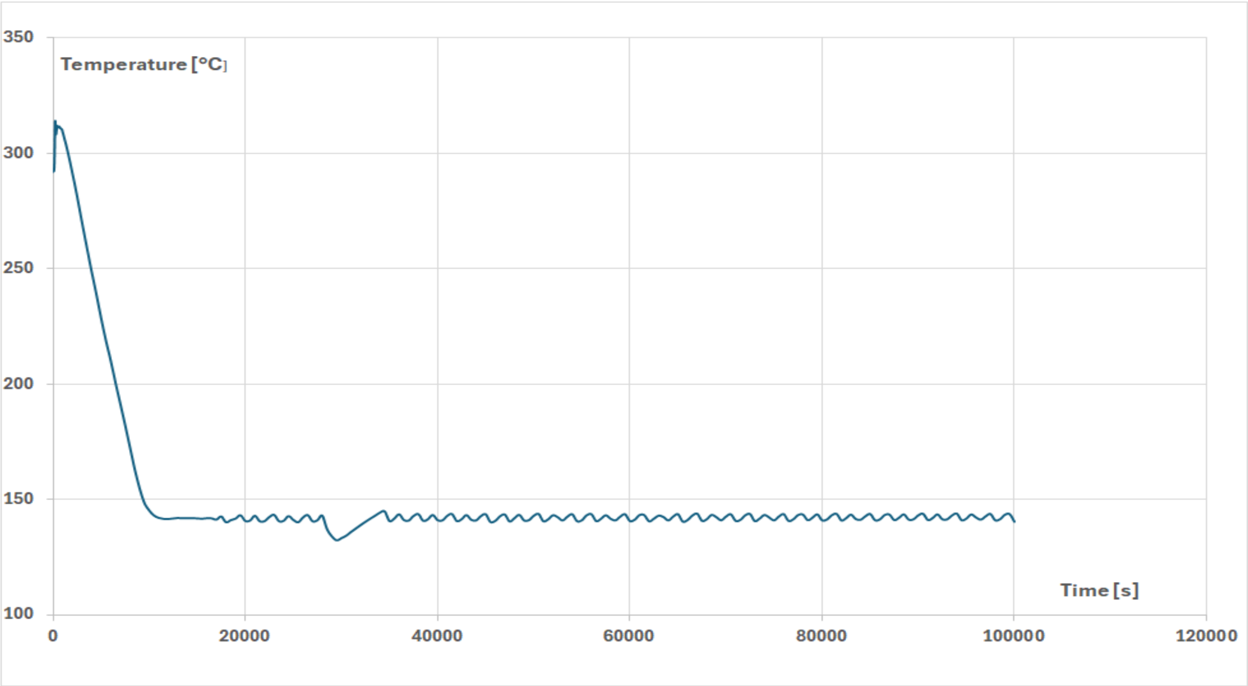

Figure 27: Inlet temperature in the primary side

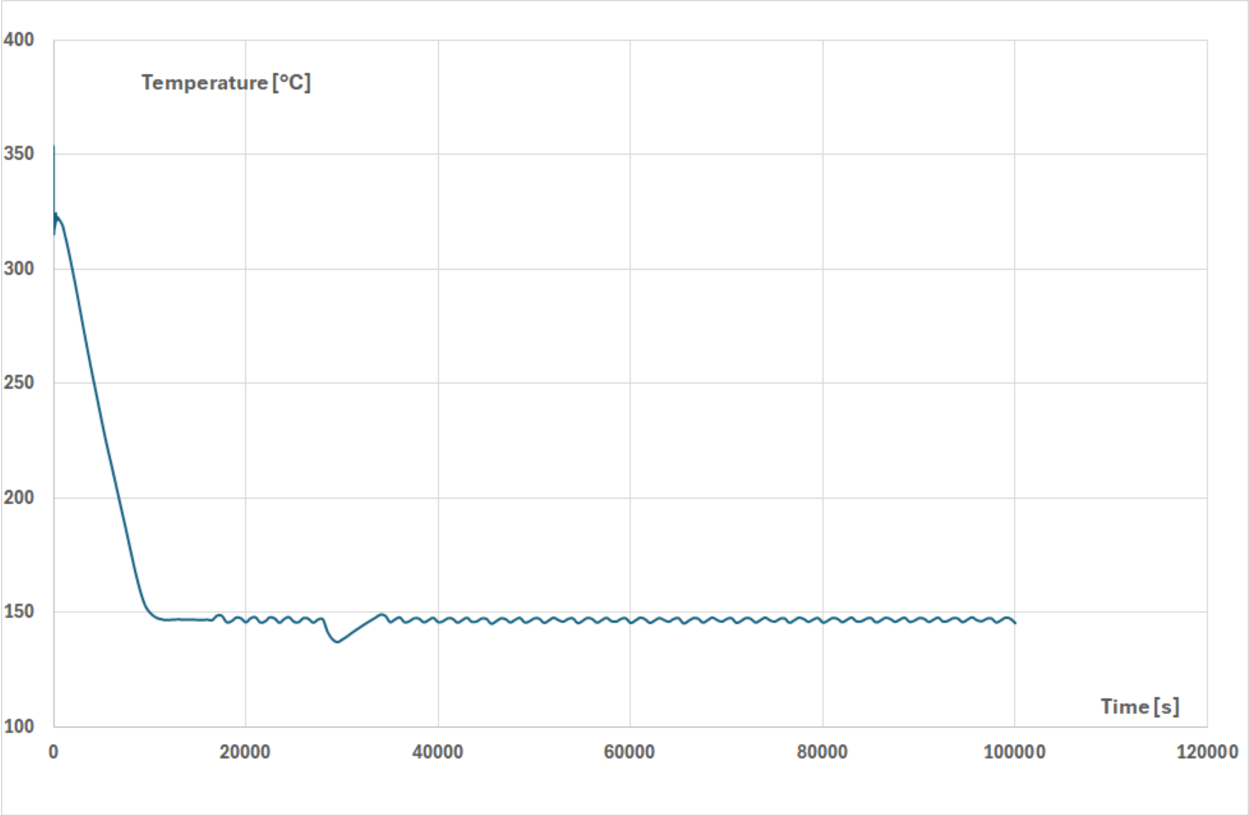

Figure 28: Outlet temperature in the primary side

<u>Conclusions</u>

- The LOOP was simulated using the CATHARE3 code to nodalize a E-SMR nuclear power plant
- The results above presented shows a good capability of the code to describe the behaviour of the plant during the transient
- However, it is important to highlight that during the firsts seconds of the transients when the pressure increase the consequent increase of the outlet temperature in the primary side shows that the its value may be higher than the saturation temperature at pressure equal to 151 bar.
- It is important to note that this needs of further investigation in the future.
- Further investigation could be referred to the need to study the neutronic part of the E-SMR: for example could be interesting to know the exact value of the energy fraction and the decay constants of each decay heat group.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

ANNEX 3. Comparison of ATHLET (ENERGORISK)/CATHARE (ENEA) LOOP calculation

This section presents the results of comparing the calculations of the IE "Loss of external grid" performed on the calculation models of the ESMR reactor plant developed for the **ATHLET** and **CATHARE** codes. For comparison, the main representative parameters that characterize the operation of safety systems and the thermal inertia of the reactor plant are selected:

- Core inlet mass flow rate
- Pressurizer pressure
- Core outlet temperature
- Mass flow rate from relief valve in the pressurizer
- Mass flow rate discharged by accumulators in the primary side
- Core inlet temperature,

Results of comparison are presented in Figure 29- Figure 34.

The calculation results show coincidences of the trend of changes in the main parameters and close values of the final stable state. At the same time, it should be noted that there is a significant difference in the rate of change of parameters, such as the pressure and temperature of the primary circuit coolant, which causes a difference in the time frame for the operation of safety systems. In Figure 31 we can see a significant difference in the rate of drop in the temperature of the primary circuit coolant, which in turn leads to a sharp decrease in pressure. This difference can be explained by different approaches to modeling primary circuit heat exchangers, primarily steam generators.

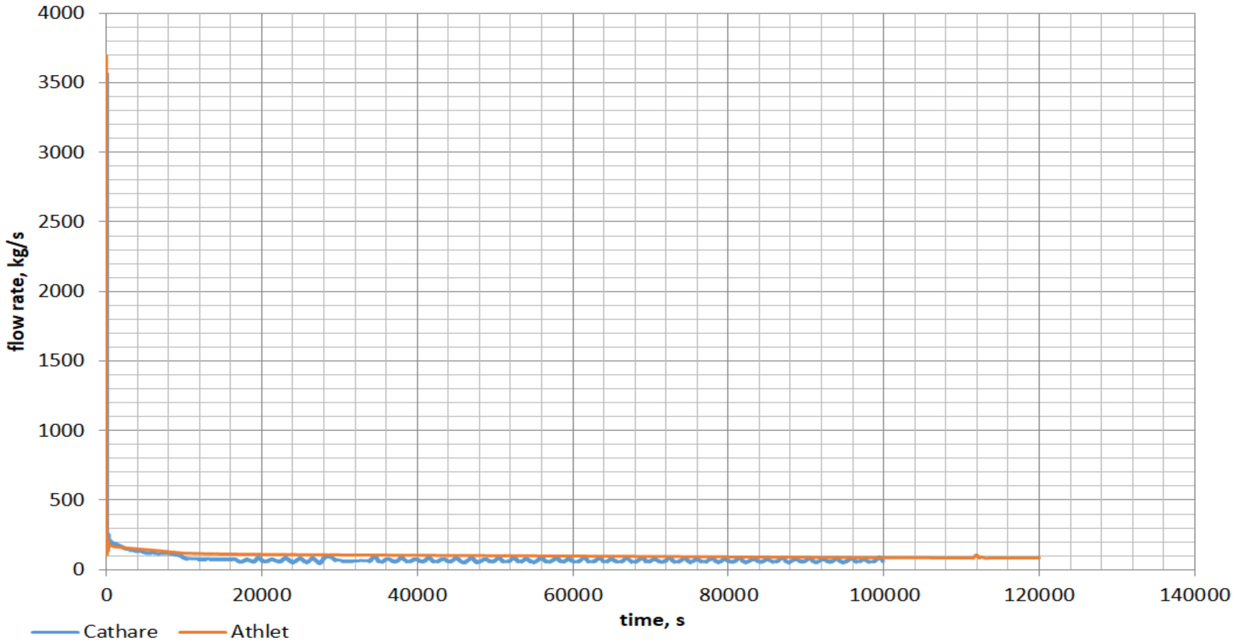

Figure 29: Core inlet mass flow rate

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

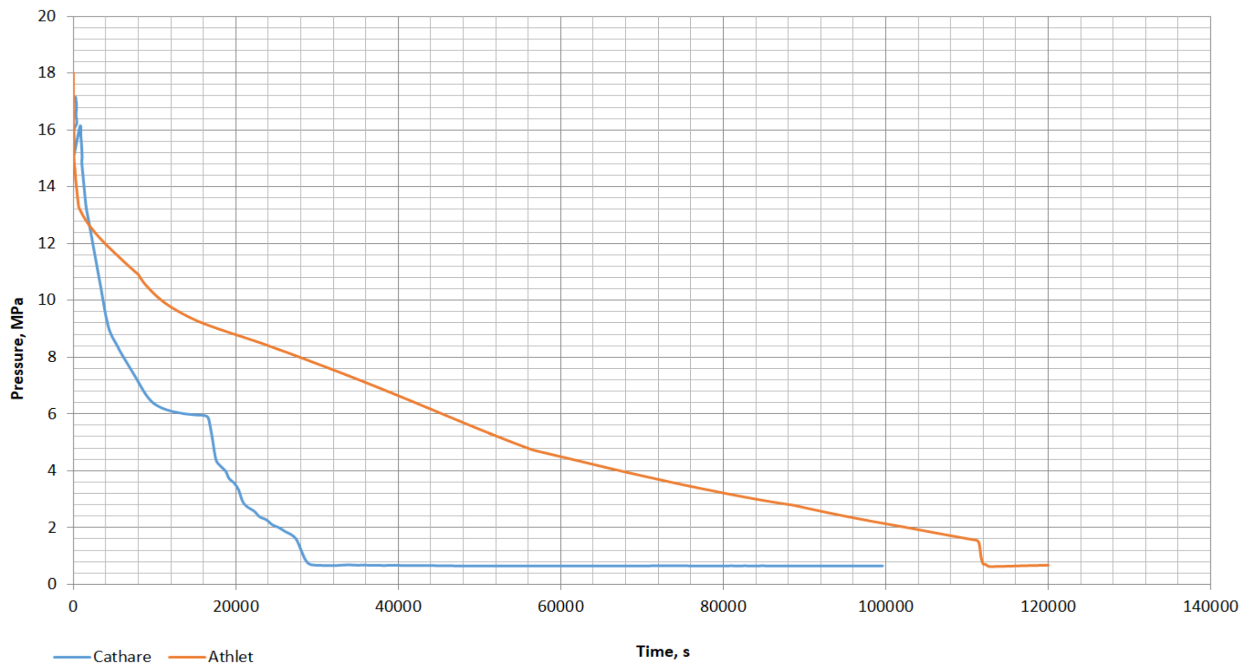

Figure 30: Pressurizer pressure

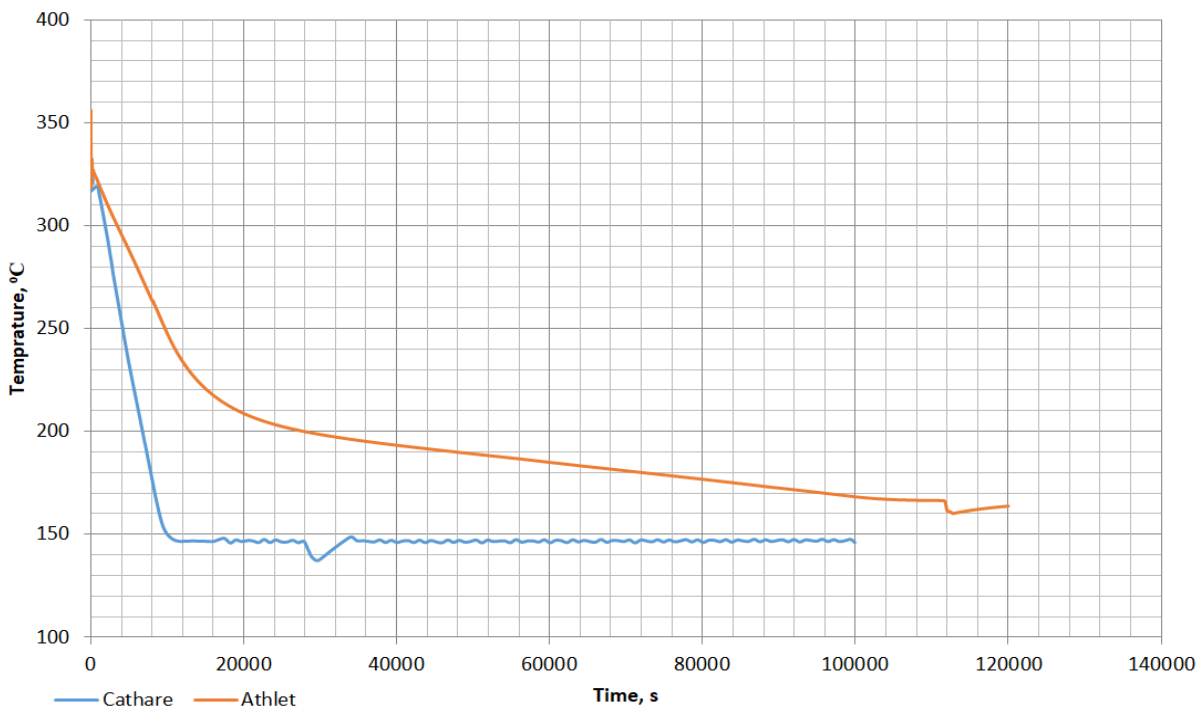

Figure 31: Core outlet temperature

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

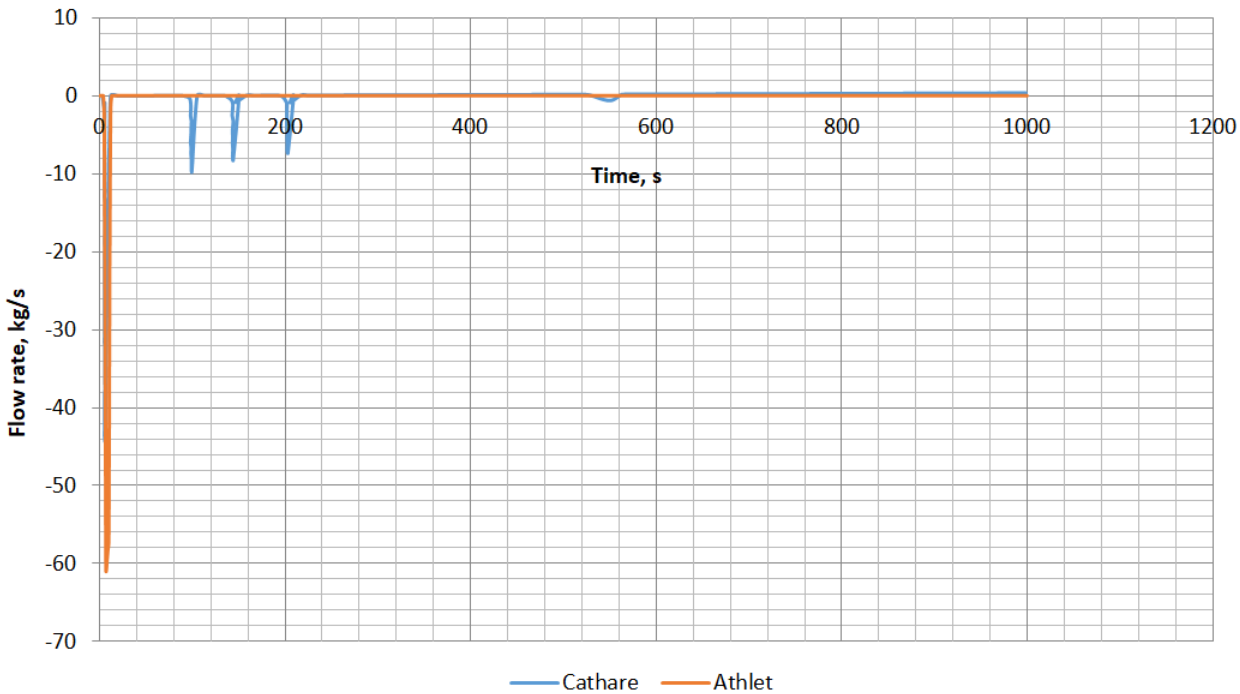

Figure 32: Mass flow rate from relief valve in the pressurizer

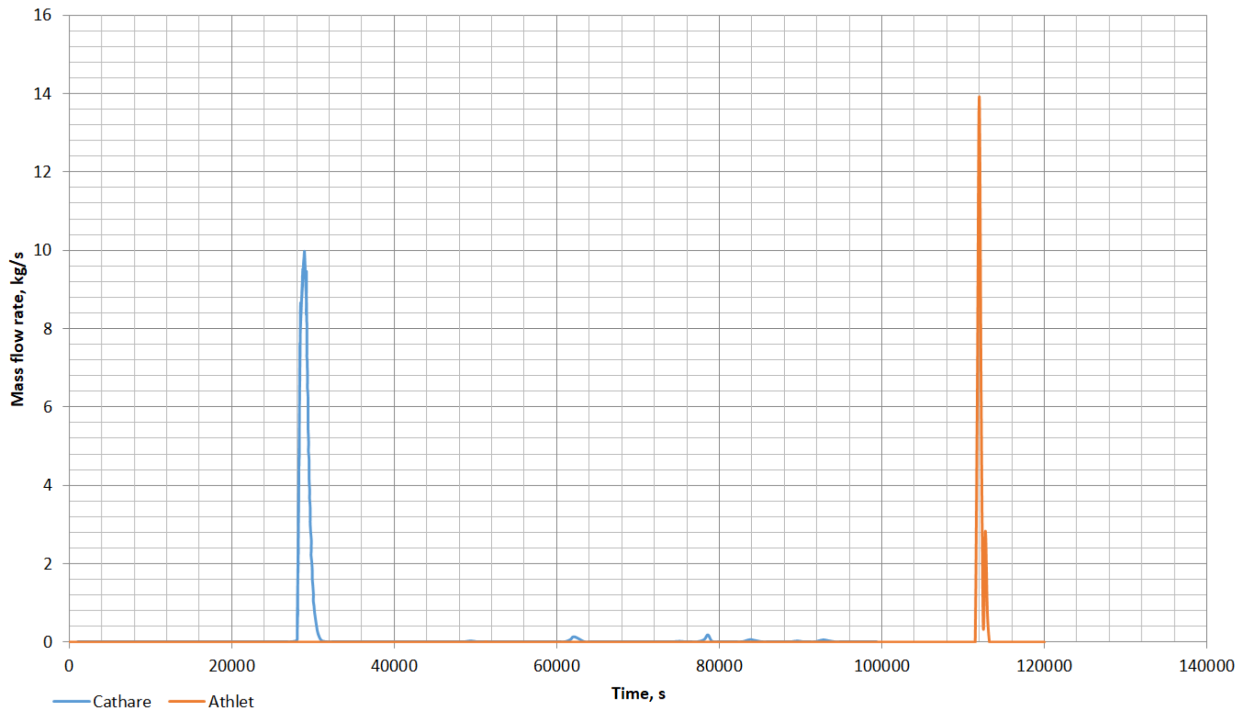

Figure 33: Mass flow rate discharged by accumulators in the primary side

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

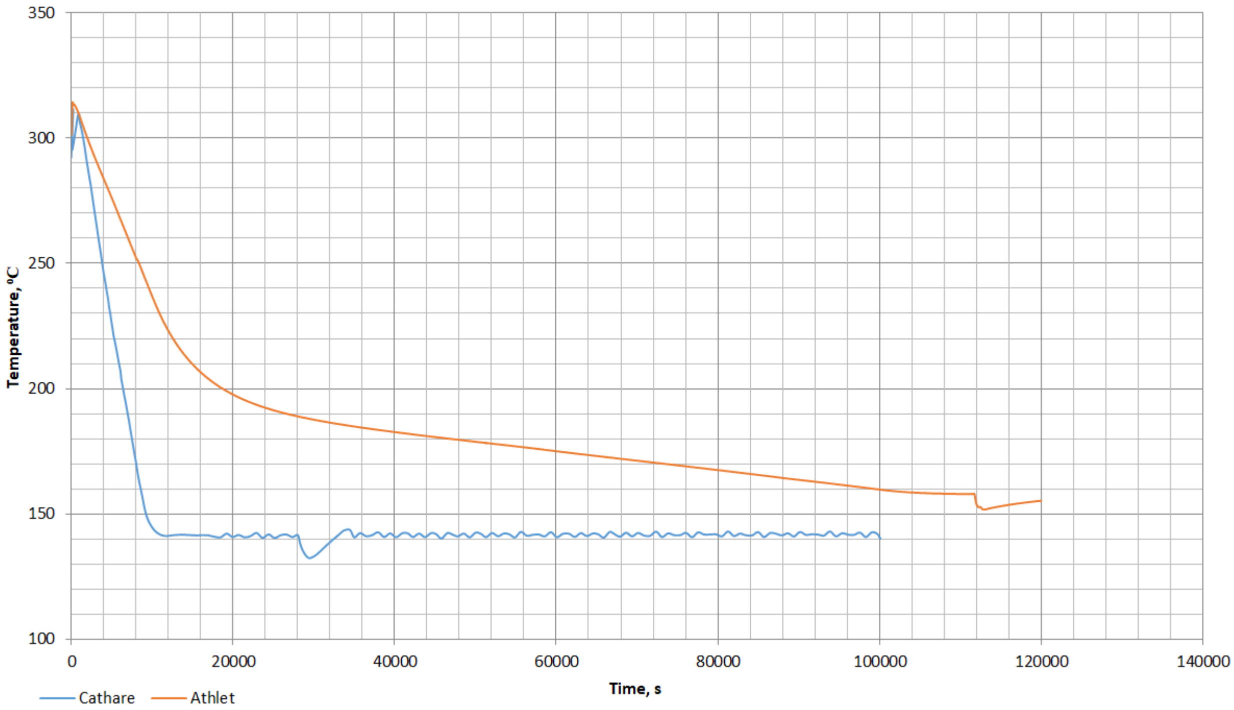

Figure 34: Core inlet temperature

CONCLUSION OF RESULTS COMPARISON

The results of calculations performed using computational models of the SMR reactor developed for the ATHLETE and CATHARE codes show a general coincidence of the trend of the main parameters and the magnitude of temperature and pressure stabilization by the end of the calculation. The magnitude of the disturbances introduced by the operation of safety systems is also comparable (for example, a drop in the temperature of the primary circuit coolant when the heating system is activated). At the same time, it should be noted that there is a significant difference in the rate of decrease in the temperature of the primary circuit, which, with a comparable level of residual energy release and the values of stabilization parameters, can be explained by different approaches to modeling steam generators.

ANNEX 4. LOOP calculation using coupled code system ATHLET/MODELICA (GRS)

A load rejection transient leading to LOOP has been simulated by GRS using the coupled code system ATHLET/MODELICA. The ATHLET input data set, the MODELICA model as well as the coupling itself used for this investigation is the same as described in paragraph 3.1 of this report and in D4.3 [3].

The simulation starts with full power E-SMR which produced steam is fully used for electricity generation. The initiating event is a load rejection from 100 % to 60 % in 20 s. The loss of torque of the generator shaft leads to an increase of the turbine rotation speed, which further leads to SCRAM. To avoid any crash of the MODELICA BoP model, the boundary conditions after SCRAM are set to nominal conditions (full power) for the MODELICA BoP model. However, in the ATHLET model after SCRAM a number of actions are assumed:

- Secondary side isolation (feed water supply set to zero, full close of turbine admission valve)
- Deactivation of primary pressure control (spray/heating)
- Primary pumps set off
- Activation of passive residual heat removal system

The increase of the turbine rotation speed can be seen in Figure 35 top left. The increase directly leads to SCRAM which reduces the core power to the decay heat curve (Figure 35 top right). The primary pumps are set off due to the loss of power which can be seen in the drop of the primary mass flow rate to natural circulation (Figure 35 bottom left). The secondary side is isolated leading to a decrease of steam generator power. The steam generators cannot be used anymore for decay heat removal. Instead, the safety compact steam generators are activated and transfer heat to the intermediate loop (Figure 35 bottom right). The decay heat is transferred ultimately into the water wall.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

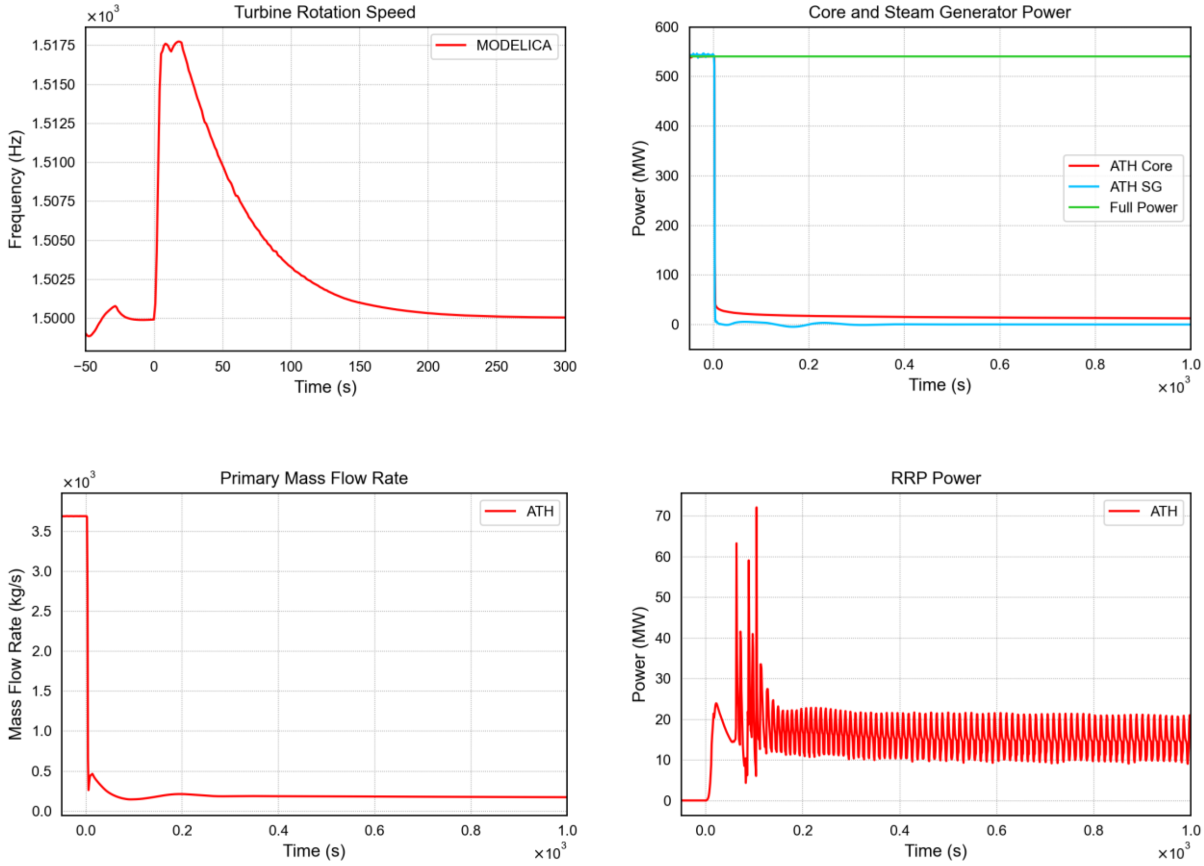

Figure 35: Turbine rotation speed, power and primary mass flow rate during LOOP in the ATHLET/MODELICA simulation

During the first seconds, the core power is still high compared to the possible heat removal. While the steam generators are already isolated and not available anymore, the residual heat removal system still needs some time to be fully activated. The result is a power peak in the primary system (Figure 36 top left) which leads to an actuation of the pressuriser relief valve (Figure 36 top right). The steam is dumped into the small water pool located at the bottom of the containment vessel. However, when the passive heat removal system is fully running it is capable to remove more than the residual heat which can be observed in the decreasing trends of the primary temperature (Figure 36 bottom left) and pressure (Figure 36 bottom right) in the long term.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.

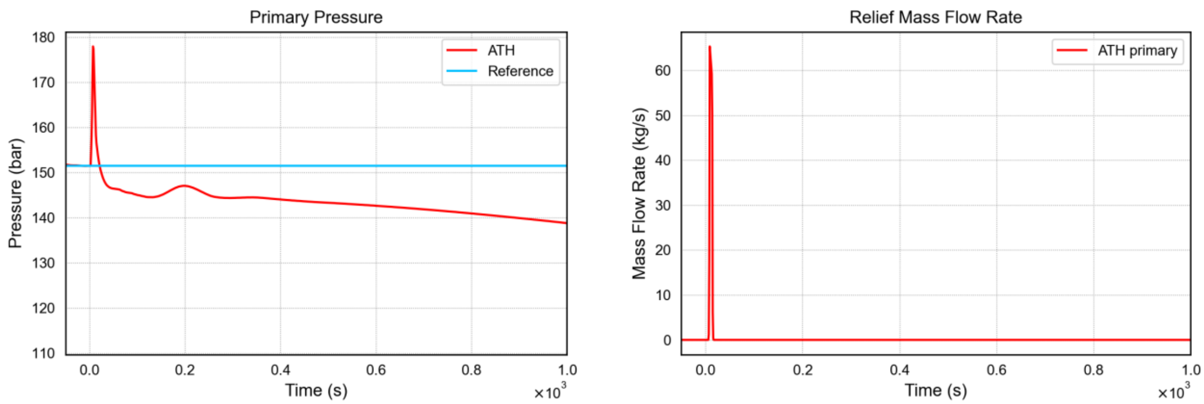

Figure 36: Primary pressure / temperature and pressuriser relief mass flow rate

The results show, that the coupled code system ATHLET/MODELICA can be used to reliably simulate transients like the LOOP case. Since the whole BoP model is not needed anymore, it can be switched off during the transient and is used only in the very first part of the simulation before secondary side isolation. Since the current ATHLET/MODELICA coupling implementation does not provide a real deactivation of an FMU, in the current case the deactivation of the BoP model is performed by provision of nominal boundary conditions to MODELICA. A drawback of this method is that the BoP model is further running although it is not needed anymore.

Funded by the European Union

Funded by the European Union. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Atomic Energy Community ('EC-Euratom'). Neither the European Union nor the granting authority can be held responsible for them.