sCO2-4-NPP: Innovative sCO2-Based Heat Removal Technology for an Increased Level of Safety of Nuclear Power Plants

Deliverable 2.1

Definition of initial and boundary conditions for an SBO accident

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DEC	Websites, patents filing, press & media actions, videos, etc.	
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1 List of Acronyms

Abbreviation / Acronym	Description / meaning
AC	Alternating Current
AFW	Auxiliary Feed Water
AMM	Accident Management Measure
ATHLET	Analysis of THermalhydraulics of LEaks and Transients (system code of GRS)
BDBA	Beyond Design Basis Accident
CFD	Computational Fluid Dynamics
COCOSYS	Containment Code System (system code of GRS)
CV	Control Volume
DBA	Design Basis Accident
ECC	Emergency Core Cooling
GCSM	General Control Simulation Module
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit gGbmH (German TSO)
HA	Hydro Accumulator
HCO	Heat Conduction Object
HeRo	Heat Removal
HPI(S)	High Pressure Injection (System)
LOCA	Loss Of Coolant Accident
LOOP	Loss Of Offsite Power
LPI(S)	Low Pressure Injection (System)
MCP	Main Coolant Pump
NPP	Nuclear Power Plant
PSD	Primary Side Depressurization
PWR	Pressurized Water Reactor
RPV	Reactor Pressure Vessel
SBO	Station Black Out
SG	Steam Generator
TSO	Technical Safety Organisation

2 Executive Summary

One of the main challenges of the sCO2-4-NPP project is to integrate the Heat Removal system (developed in sCO2-HeRO project) into various thermal-hydraulic codes and to simulate its behaviour in case of an accidental situation on the nuclear power plant (NPP).

The objective of the work presented in this report is therefore to provide initial and boundary conditions for an SBO accident in each code used in the sCO2-4-NPP project (CATHARE, ATHLET, ATHLET-MODELICA).

In a first time, each partner presented how a Station Blackout accident was scripted in each of the codes (as well as in the Konvoi simulator that will be used in the project).

This work shows that despite a common definition, because the codes used are different as well as the types of power plants (EPR, KONVOI, VVER), the scenarios have some differences. This observation is also due to the fact that the accident scenarios modelled in the codes are also based on the operators' standards to manage these situations.

Another difference in the works presented in the report is the fact that the codes take into account the same types of operating parameters (pressures, temperatures...) but these parameters can be expressed in different units. One of the major points of attention for the further work of the project will therefore be to ensure that each partner makes the effort to convert their simulations results if necessary so that the entire consortium can easily take ownership of these results.

3 Introduction

One of the main challenges of the sCO2-4-NPP project is to integrate the Heat Removal system (developed in sCO2-HeRO project) into various thermal-hydraulic codes and to simulate its behaviour in case of an accidental situation on the nuclear power plant (NPP).

The fact that several codes are involved requires for the consortium a perfect communication and understanding of the parameters that will be used and produced results in the codes in order to be able to carry out a quality benchmark and to guarantee the quality of the results on the behaviour of the Hero System. These parameters cannot be defined if the initial conditions of the accident are not shared and validated by the consortium.

Laying the foundations for the initial conditions of the accident scenario and the parameters referring to them are the objectives of the present deliverable.

As a first step, we will resume the scenario that will be applied for the reference accident situation, with its consequences on the nuclear power plant. Then, we will describe the parameters that will be implemented in the different codes to finally identify the points of attention (similarities, differences, ...) that the consortium will have to manage in the future project.

4 Reference accident scenario

4.1 Description of the accident

The sCO2-4-NPP consortium agreed to work on an SBO (Station BlackOut) accident at a nuclear power plant.

The definition of SBO situation by different international authorities is “the complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). Station blackout does not include the loss of available AC power to buses fed by station batteries through inverters or by alternate AC sources as defined in this section, nor does it assume a concurrent single failure or design basis accident“ (NRC, US, 2019).

The Loss of Offsite Power (LOOP) is also considered to be within the design basis for all plants and is managed through a range of redundant and diverse means.

Prior to the Fukushima Daiichi Accident, the international trend was evolving to consider the station blackout (SBO) as a design basis event. The Fukushima Daiichi accident, which started as a SBO event but had consequences that went well beyond the coping capability of the units, made it a major consideration for commercial nuclear plants to develop a strategy for SBO as a part of Design Extension Condition (DEC). (Duchac, 2015)

For the NPP operators, the SBO management depends on several aspects, such as:

- State of the plant before the SBO event (Full power operation, Shutdown state for refuelling…)
- Capacity to restore the AC power (Quick restoration, restoration within the coping time, out of coping time restoration)

In the sCO2-4-NPP, we consider that the SBO event cannot be recovered from within the coping time. So, we are in an extended SBO event and the objective of the HeRo system is to prevent the core melt scenario. An extended SBO is a SBO event where AC power source is connected within the coping time but is required to continue operating for extended duration (IAEA, 2015).

4.2 Description of the scenario of a SBO for a EPR PWR

4.2.1 GENERAL OPERATING PRINCIPLES

In normal operation, the thermal power from the core is transmitted to the turbines, driving the alternator in steam water produced by the steam generators.

The steam generators are supplied with water (from the condenser at the outlet of the turbines) by the feedwater flow control system.

In normal operation, in certain reactor states (shutdown…), the conditions for pressure and temperature no longer allow for efficient steam generators. Under these conditions, the residual core power is removed by the core cooling circuit of the primary (Safety Injection System-Residual Heat Removal System), itself cooled by the component cooling system which is cooled by Essential Service Water System.

Under certain incidental or accidental operating conditions, the supply to the steam generators is carried out by the Auxiliary Feedwater System, a dedicated backup circuit, instead of the Feedwater Flow Control System. The emergency feed water for the steam generators is then drawn from the Auxiliary Feedwater System storage tanks, which can be replenished from tanks; this feed water is injected into the steam generators by means of the Auxiliary Feedwater System motorpumps.

The Chemical and Volume Control System, directly connected to the primary loop, allows the volumetric and chemical control of the latter. It is associated with the Reactor Boron and Water Makeup System providing the water and boron make-up functions. Under certain incidental or accidental operating conditions, borication of the primary circuit can be carried out by the use of pumps (safety borication system) fed by the borated water storage of the tanks.

In addition to its shutdown cooling function, the Safety Injection system allows borated water to be injected into the primary circuit, drawn from the “In containment refueling water system tank” (IRWST), thus ensuring control of core reactivity and water inventory in the primary circuit.

The Ultimate Ventilation System is an ultimate means of mitigation designed to limit containment pressure and provide heat removal from the containment and IRWST under accident operating conditions when the systems are inoperative or in the event of a severe accident.

4.2.2 LOSS OF POWER SUPPLIES AND COOLING SYSTEMS FOR REACTORS

The following paragraph presents the scenario resulting from an SBO accident on an EPR plant (data set to be used in CATHARE).

The Total Loss of External Power Supplies is an operating condition of the design basis studied under the safety standard. It assumes the loss of the main external power grid, the failure of house load operation (power generation by the plant's Turbo Generator) and the loss of the external auxiliary grid.

The total loss of power supplies to a single unit on the site is a condition of complementary operation of the safety baseline. It results from the loss of external power supplies associated with the impossibility of recharging the control board rescued by the two emergency generators.

In the plant design basis, the situation of loss of external power supplies and loss of the four main diesel generators is an operating condition of the design basis safety area. The duration considered before the electrical sources are returned is 24 hours.

The electrical source used in this situation is the two diversified ultimate back-up diesel generators of the main diesel generators. The diversification of the diesel generators is based on different ranges of equipment, different voltages, a diversified fuel oil supply and different fuel oil tanks. The choice of diesel generator takes into account the feedback from the fleet. The EPR therefore benefits, from the design stage, from redundant, diversified and robust power sources for power loss scenarios.

In this situation, the control clusters are automatically inserted into the core.

Due to the initiator, the primary pumps stop. The residual power is evacuated by thermosiphon (natural circulation provided for in the design).

The cooling of the thermal barrier of the primary circuit pumps is lost. In addition, the charge pumps are lost, which in normal operation provide injection at the seals of the primary circuit pumps. The sealwater system of the primary pumps is automatically put in place.

On the secondary side, the steam generators are no longer supplied by the feedwater flow control system due to the initiating event. The residual power is discharged via the atmospheric steam dump valves.

The two back-up diesel generators are started manually from the control room. One emergency diesel generator may not start because it is undergoing preventive maintenance.

Following the start of the emergency diesel generators, the 2-hour and 12-hour batteries of the nuclear island are charged.

The measures taken make it possible to provide the following functions during the period under consideration (24 h) for the voltage to be restored:

- ensure the integrity of the primary circuit and avoid a breach induced by the simultaneous loss of the thermal barrier and injection at the primary circuit pump seals,
- ensure the operability and control of the components essential to the control of this operating condition,
- maintain certain information in the control room that is essential for this operating condition,
- provide lighting for the control room,
- ensure, in closed states, the evacuation of residual power
- provide, in open states, a means of compensating for the vaporization rate and the discharge of residual power.

4.2.3 LOSS OF THE ULTIMATE COOLING SYSTEM

The total loss of the cold source means that the feedwater plant, the essential service water system, the component cooling system, the residual heat removal system, the primary pumps (loss of bearing, motor, thermal barrier cooling), the safety injection system are unusable.

The loss of the essential service water system induces a progressive heating of the component cooling system. Eventually, the component cooling system no longer provides adequate cooling of the auxiliaries.

In this situation, the reactor coolant pump set automatically shut down by loss of the thermal barrier or loss of injection at the joints. An automatic reactor shutdown is then initiated: the control clusters are automatically inserted into the core. The residual power is evacuated by thermosiphon (natural circulation provided for in the design). The sealwater system of the primary pumps is automatically put in place.

On the secondary loop, since the condenser is unavailable due to the initiating event, the residual power is discharged through the atmospheric steam dump valves. Subsequently, the auxiliary feedwater system pumps start to supply water to the steam generators.

The total loss of the heat sink implies the loss of cooling functions for the core and the reactor building through the natural heat sink.

Based on the feedback from the other PWR units, the EPR has a dedicated system, the Fire Water Production System designed to enable the Auxiliary feedwater system to be replenished for several days. This dedicated system is composed of 2 concrete tanks, one of which has a capacity of 2600 m3.

4.3 Description of the scenario of a SBO for a Konvoi/pre-Konvoi PWR

The following chapter describes the event propagation after SBO for a pre-Konvoi plant, aimed to be used as reference for the demonstration of HeRo system at its full scope simulator. Alongside the typical evolvement of an unmitigated scenario, some remarks are included about possible countermeasures prepared by training. Some aspects for the use of the HeRo system are described as well.

Following the definition of the SBO as the loss of all AC sources that are not backed from batteries on site, the plant scenario can be described along some basic safety parameters:

- Subcriticality
- Core cooling, with the sub-parameters
    - primary coolant content,
    - pressure limiting and
    - heat removal (where secondary side water resources are taken into account)
- Containing integrity.

From operator’s view, many parameter subsets are defined, when it comes to the handling of the event, but the list should be kept short to get a quick overview. All actions necessary to mitigate the event and to cope with faulty signals or un-appropriate safety locks are described in procedures the operators are trained about. So, the focus can remain on thermal hydraulics and physics.

Subcriticality

The loss of power causes SCRAM from several criteria, first of all the loss of the main coolant pump (MCP) function. The reactivity binding from control rods and xenon guarantees criticality margins fulfilled for hot shutdown conditions. Without a proper heat sink, lacking abundant coolant resources, a rapid cooldown is improbable, so recriticality is out of concern. Depressurisation of secondary side will be done later only to limit temperatures in primary circuit and to add some options for injection of low pressures sources for secondary side coolant.

Later concerning an operating HeRo system, a cooldown without proper boron injection could lead to recriticality, when xenon poisoning had been gone. The point of recriticality depends from the initial conditions, mainly burnup.

Core cooling

With subcriticality fulfilled, only decay heat has to be managed. Initially, after rundown of MCP, single phase natural circulation will provide transport of heat from the core to the steam generators. Pressuriser (PRZ) will remain mainly on former operating pressure, without electrical heating but without spraying too. Hence surface losses of about 5 K/h will determine the saturation point. Therefore, for the first critical hours, with secondary side coolant and battery power still available (this can be held for the „coping time“) the pressure will be kept providing subcooled conditions, unless the reactor outlet temperature will rise to saturation point. There should be mentioned that the first criteria for secondary side depressurisation (for the simulator reference plant: hot leg temperature above 330 °C, another for PRZ level increase over 9.5 m) are set for still subcooled hot leg condition.

Comparable to the loss of offsite power (LOOP), the atmosphere is the heat sink. Main-steam relief valves will blow out the steam from the steam generators (SG). The set-point for blowout is different among the plants – there is some battery powered control for the reference plant, to keep the pressure at 74 bar ("partial cool down"), after hitting 82 bar once, following the loss of the main heat sink. Other Konvoi/pre-Konvoi plants rely on their safety valve setpoint of about 87 bar. All these valves are working with battery backed power for their solenoid pilot valves.

However, the mass of secondary side coolant becomes the crucial parameter now. Going into the accident from full power, decay heat will boil-off the inventory of the SG within about 50 min down to less than 2 m (as an averaged simulator experience), when the heat transfer from primary to secondary side will be affected remarkably. Each SG is assumed to start with about 60 t of coolant mass for the reference plant, with feedwater pumps and auxiliary feedwater pumps lost at the initialisation of the event, unable to fill up after SCRAM.

Differences in the secondary coolant mass available between Konvoi/ pre-Konvoi PWR may result from void content and the existence of a pre-heater chamber. Other differences may result from the assumption of the power before the event. In case of in house operation, power is about 35%, with a lower void content, hence with more mass, and with less decay heat after SCRAM. If the scenario starts from Hot Standby, as in Fukushima Daiichi about 50 min after shutdown, SG would be filled up even more.

During boil off, the average primary coolant temperature will be kept about 10 K above the secondary side temperature, so it is about 300 to 310 °C, depending on saturation point at secondary side. But with reduced heat exchange area, after level having dropped below about 2 m, the average temperature will be increased slowly. Hot leg temperature is about 10 K above average temperature, so after some time the saturation point will be reached at the outlet of the hot channel, according to the slightly reduced pressure from the PRZ. Now, the reactor vessel head will collect a steam bubble, defining the pressure on primary side. Level in PRZ will increase. At a certain point of time, the surface of the vessel head bubble would reach the hot leg, so steam will flow below top of the pipe to the SG and to the surge line to the PRZ.

The primary pressure would now exceed the setpoint of the PRZ relief valve (about 166 bar a), at a higher pressure the two safety valves (about 169 bar a/ 175 bar a at the valve, spring loaded). Continuous heat transfer to the pressurizer relief tank would lead to heatup and burst of the rupture disk, so pressure in containment would increase.

These signs for insufficient heat removal (increase of hot leg temperature above 330 °C, increase of level in PRZ, pressure increase above setpoint of relief valve, indication for leak out of the PRZ relief tank) trigger the operators' actions for secondary side blowdown. Only one condition has to be fulfilled for action, and first criteria are fulfilled with the coolant in the hot leg still subcooled liquid.

With the remaining cooling capabilities of the secondary side, a rapid cooldown with recriticality is out of concern. Even with a HeRo system starting now, the problem could be handled only by reducing the pressure and temperature on secondary side remarkably, producing overheated steam to be cooled first and condensed afterwards in the CHX. It could be beyond the latest point of time to engage with HeRo. Typically, this is about 1 hour after beginning of the scenario.

In the classical approach, with the remaining battery power, the secondary side will be relieved (secondary side depressurisation), temporarily recovering heat removal and reducing pressure and temperature on primary side (from some subcooling). Some coolant, up to 400 m³, may come from the feedwater tank and feedwater system, which acts as an accumulator. Ultimately, some low pressure feeding of the secondary side has to be established quickly, to provide heat removal to the atmosphere. This could be done e.g. with mobile fire protection equipment, using the filtered water from mobile tanks.

Even with some injection, the relief valves on secondary side have to be kept open. Getting short of batteries (long term SBO, refer to Fukushima Daiichi unit 3 at the morning of March 13, later unit 2 at the evening of March 14), remaining depressurised becomes a problem. Hence, a situation could evolve, where even a half filled secondary side after loss of battery power for relief valves would heat up again, parallel to the primary circuit. With HeRo, this could become a second chance to start the system, using the coolant injected already and recharging the batteries. For that, the coolant mass inside the SG affected has to be conserved, not to get lost slowly to drains or auxiliary steam components (mostly, for the resources available, only a single SG will attempted to be refilled).

Without proper heat removal, e.g. after an unsuccessful attempt for secondary side depressurisation, temperature and pressure increase on primary side will lead to a slow boil off from the primary circuit to the pressurizer relief tank and into the containment.

During heatup and boil-down of the primary circuit, there is a limited timeframe of about 1.5 hours to recover secondary side heat removal, before “MIN3” will be reached. MIN 3 represents the “wetted level” below mid loop. If that fails, uncovering the top of the core will follow soon. Now, with increasing temperatures in hot leg beyond 400 °C, the last attempt is to depressurize the primary circuit (battery power for control has to be available, but is assumed to be so after only 2.5 hours after loss of AC) to finally let inject the 8 accumulators into the hot legs and cold legs. By this the boron concentration for cold subcritical conditions will be reached. The coolant injected gives about 3 hours of coping time to recover heat removal via secondary side or to establish some sources for electrical power.

Otherwise, high pressure core melt is unavoidable after about 3 hours from the beginning of the event. With hydrogen production from metal-water burning, heat transfer to secondary side will be blocked from non-condensable gases finally, consequently rendering any HeRo system useless.

Containing activities

As long as the core could be kept covered with liquid water, only a small amount of activity from contaminated coolant will become released to the containment. Most activities will become washed out from steam bubbles passing the pressurizer and subsequently the water in the pressurizer relief tank. Therefore, in all successful HeRo scenarios, the activity will remain contained, even after coolant release from the pressurizer relief tank.

If the stress from secondary side depressurisation would lead to a rupture of a u-tube, the affected steam generator would have to be isolated, to avoid activity release to the environment. Unfortunately, this would not prevent the SG from heating and filling up (solid water) to the setpoint of its safety or relief valves. With pressure in primary circuit kept high, there will be a continuous loss of primary coolant to the SG, and subsequently to the atmosphere. Therefore, the combination of DBA with SBO is not discussed willingly, because it will lead to core melt and early activity release near surface, with high doses and contamination on site and nearby the plant.

With a HeRo loop available at the affected SG, the increasing coolant mass on secondary side could be used to cool down primary circuit, with primary pressure kept slightly above secondary side, with two-phase conditions on both sides. This would reduce the leakage flow remarkably, providing more time for countermeasures, e.g. restoration of electrical power or heat removal with another steam generator.

4.4 Description of the scenario of a SBO for a VVER1000 PWR

The SBO results in a long-term failure of the AC power supply of all appliances without the possibility of its recovery. The SBO is modeled as a loss of working and reserve power supplies (ZPRZEN) with blocked start-up of all emergency power supplies (system and non-system diesel generators).

The most limiting factor in SBO (heat removal from IO, heat removal from fuel pool, loss of cooling of I&C rooms, discharging time of accumulators) is the time during which the unit is able to withstand without fuel damage in the core. In the case of the SBO during the unit operation in the hot state, the heat from the primary side can be only by steam into the atmosphere via PSA or PVPG. Due to the loss of SG power supply after the SBO, the water supply on the secondary sides of the SB then decreases, accompanied by a gradual detection of the heat exchange tubes in the SG and a reduction in the effective heat transfer surface from the primary side. This situation would lead to the loss of secondary heat dissipation without the intervention of the operating personnel. Since SG is unable to dissipate any residual heat produced in the core, the temperature of the primary coolant has increased accompanied by an increase in the pressurizer levels and primary pressure up to the pressurizer relief valve or pressurizer safety valve opening value. Operational personnel follow SBO's procedures following SBO. When the total levels in all SGs fall below 140 cm (a level ensuring sufficient heat dissipation from the SG), the critical safety function is violated, and the operating personnel follow the relevant procedures. According to the procedure, the operative personnel first temporarily refill the selected SG with feed water by means of gravity tank and after exhaustion of the tank, ensure that the selected SG is replenished with a mobile fire pump. These activities will significantly delay the need for transition to emergency heat dissipation using the primary bleed&feed method and extend the time window for the emergency power supply of pumps from SBO DG.

5 Description of CATHARE parameters

5.1 Cathare description

The CATHARE code is a French thermal-hydraulics system code developed since 1979 and extensively validated in collaboration between CEA, EDF, IRSN and FRAMATOME (F. Barre, 1990). It was first devoted to best estimate calculations of thermal-hydraulics transients in nuclear reactors. CATHARE-2 is the current industrial version of the CATHARE code. It is internationally used for nuclear power plant safety analysis and licensing (International Atomic Energy Agency, 2002) and in plant simulators (Miettinen, 2008). CATHARE-3 is the new development version of the code (P. Emonot, 2011), in continuity with the CATHARE-2 code. The CATHARE code is based on a two-phase six-equation model including additional equations for non-condensable gases and radio-chemical components transport.

CATHARE-3 is the next generation of multi-concept thermal hydraulics system code, and will address safety, design and simulation topics for both LWR and Generation IV reactors. Concerning physical models, the main enhancement of CATHARE-3 consists in the use of multi-field models eventually coupled with interfacial area equation and two-phase turbulent model.

Improving the modeling of three dimensional flows is another important topic. The three dimensional module while keeping today's numerical schemes has been redesigned so as to be able to deal with non conforming structured meshing in adapted coordinate systems. CATHARE-3 architecture will be a completely renewed object-oriented one in order to achieve better coupling ability with both other thermal hydraulic scales and other disciplines like neutronics and fuel. The new architecture will also enable the use of modern software development tools.

This new version of the code allows us to use library like REFPROP (NIST Reference Fluid Thermodynamic and Transport Properties Database) in order to work with a new gas (like supercritical CO2 in our case) and its equation of state.

5.2 Plant model

The CATHARE data set includes a 0D-1D modeling of the primary and secondary circuits. The tertiary circuit, which is made up of a sCO2-based heat removal system, will be developed and included in this data set.

The primary circuit modeling mainly contains:

- The reactor vessel including the core;
- The four primary loops including the pipes and the primary coolant pumps;
- The four steam generator tube bundles (flow inside the tubes);
- The pressurizer;
- Safety systems such as accumulators.

The secondary circuit modeling mainly contains:

- The four steam generator risers (flow outside the tubes);
- The four main steam pipes;
- The turbine, which is modeled by a pressure boundary condition.

The primary and secondary circuits are connected in the data set by heat transfer occurring through the steam generator tubes.

In the data set, tee junctions are defined and connected on the main steam pipes to collect steam produced in the steam generators. In the developed concept, this steam is supposed to be sent in heat exchangers connected to the tertiary circuit in order to evacuate energy by condensing. Then the resulting liquid phase is sent inside the steam generator to boil again.

The main interests of modeling the three circuits coupling are to take into account the inertial thermal phenomena during the transient and to allow to define a pre-sizing of the heat exchangers in the tertiary circuit.

The following figures show an example of nodalization of the vessel and the primary loop with CATHARE.

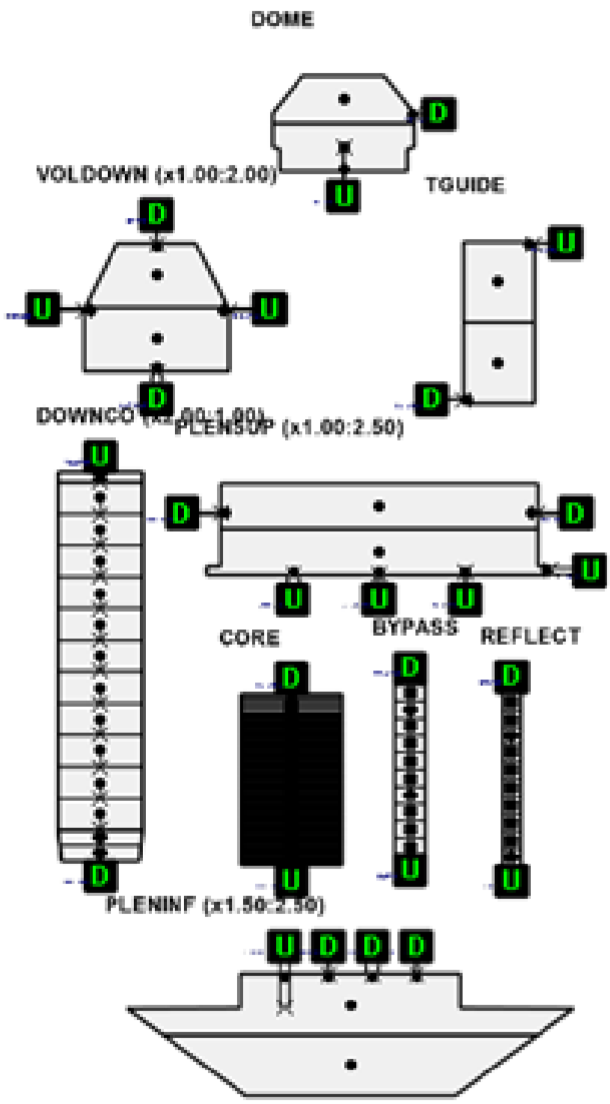

Figure 1: Reactor Pressure Vessel Nodalization Scheme (Gurgacz, 2015)

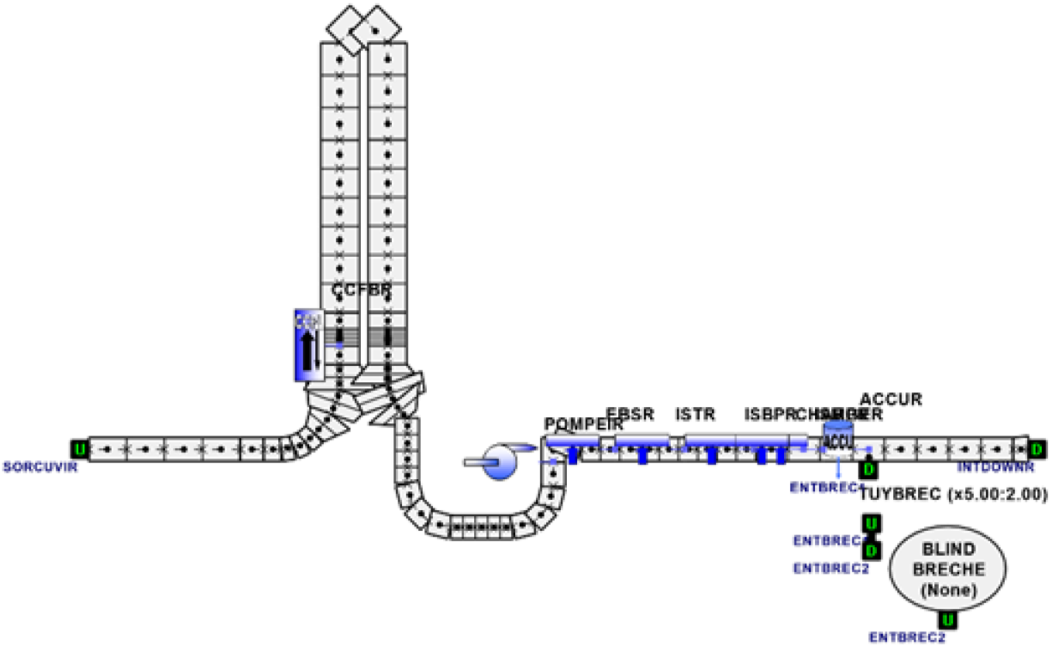

Figure 2: Primary loop Nodalization Scheme (Gurgacz, 2015)

5.3 Plant parameters in normal operation conditions (steady state)

Table 1: Plant parameters in CATHARE

Parameter	Unit	Plant reference	CATHARE
reactor thermal power	MWth	4300	4386
steam generator power	MW	n/a	
pressure in upper plenum of RPV	MPa	15.70	15.70
water level in pressurizer		/	48.9% Maximum Level
coolant temperature at RPV inlet	°C	295.6	310.7 (mean)
coolant temperature at RPV outlet	°C	331.6	
total coolant mass flow rate	Kg/s	22 235	21 931
pressure in steam generators	MPa	/	9.2
water level in SG		56	64.75%
feedwater mass flow rate	kg/s	7551.4	7551
live steam mass flow rate	kg/s	2552.4	2555

5.4 Station blackout scenario

The SBO event especially causes:

- The primary pump shutdown and the reactor shutdown;

- The secondary pump (feedwater of the steam generator) shutdown;
- The loss of the RCV circuit and the pressurizer heater;
- The loss of the safety motor pump for the steam generator.

The relevant set points of the reactor protection system are:

- To durably evacuate the residual power of the core;
- To cool down the primary circuit in order to reach a certain level of pressure and temperature;
- To control the reactivity in the primary circuit in order to prevent a critical reaction in the core.

For the first two points, the power evacuation is performed by the steam generators, then it would be performed by the sCO2-based heat removal system.

For the third point, no boron injection can be performed in the case of a SBO before the safety injection by the accumulators. As a consequence, the cooling of the primary circuit has to be adapted depending on the sCO2 system performance and the range of residual power which needs to be evacuated.

5.5 Sequence and timing of major events during the accident

The SBO event and the battery exhaustion lead to a total loss of electrical power. It causes an overflowing of the steam generators due to total opening of safety valves. The safety turbo-pumps in contact with water are lost and the water inventory in the secondary side decreases. To restore this inventory, the turbo alternator compressor has to be started up in order to supply a safety motor pump with power.

The transient is simulated with the CATHARE code in order to determine the time before making the steam generators unavailable. This situation occurs when steam generators are empty.

The main thermal-hydraulic parameters to follow during the transient are:

- The primary and secondary pressure
- The primary temperature
- The steam generator level
- The secondary mass flow rate in the steam generator
- The total liquid mass inside the steam generator

6 Simulation of a German KONVOI Reactor with ATHLET

Within WP 2 USTUTT will apply the German system code ATHLET for the simulations of a station blackout scenario in a German KONVOI reactor. The plant model (input data set for ATHLET) of the generic KONVOI reactor is initially based on the PWR example data set (GRS, 2012) provided by GRS together with ATHLET and was elaborated by GRS, IKE and HZDR within the framework of the German WASA-BOSS project (Tusheva, et al., 2015), (Buck, Pohlner, & Trometer, Oktober 2017), (Jobst, Kliem, Kozmenkov, & Wilhelm, 2017).

The plant being modelled is a generic one, i.e. it is not identical with a specific NPP in Germany. The parameters have mostly been taken from publically accessible literature, e.g. (GKN, 2011), (Ziegler & Allelein, 2013), (Kleinedler, 2002).

6.1 System code ATHLET

The ATHLET code system (Lerchl, Austregesilo, Schöffel, von der Cron, & Weyermann, March 2016) is developed by the German technical safety organisation GRS. The thermal-hydraulic system code ATHLET is designed to simulate the thermal-hydraulics of light water reactors, including DBAs and BDBAs. The highly modular code contains different optional and non-optional modules, so that the user can optimize the simulation tool for its particular purpose.

ATHLET contains simulation models for thermo-fluid-dynamics, heat conduction and transfer, neutronics and reactor controls. It is possible to couple the ATHLET code system to the containment code COCOSYS in order to simulate a severe accident in feedback with the containment behaviour. For the simulation of 2D- or 3D-thermo fluid dynamics, interfaces to couple common CFD / CFD codes are available.

6.1.1 Thermo-fluid-dynamics (TFD)

To simulate the thermo-fluid-dynamic behaviour of the coolant, the user has to specify the whole piping system with all its geometrical and fluid dynamic properties. Therefore, different kinds of objects are available, basically pipes and branches, but special objects like a steam water separator as well. Each object can be subdivided into several cells, the so-called control volumes (CVs), in order to refine the simulation's resolution. All objects have to be coupled via junctions, in order to provide valid flow paths. In principle, mass and energy conservation is based on the CVs, while the momentum balance is based on the junctions.

The user can chose between the classic 5-equation model (with drift flux approach) and a two-fluid model with separate momentum balance equations for liquid and vapour. The two-fluid model (or six equation model) treats the liquid and vapour phases of the coolant independently, so that for both phases the equations for mass, energy and momentum have to be solved. The ATHLET-CD code system performs spatial integrations on the basis of a finite volume approach, which leads to a set of first order differential equations, stemming from 2 mass and energy conservation equations per CV and one or two momentum conservation equations per junction. For the integration, a change in the geometry of flow channels and structures is neglected, as well as the dissipation energy and potential energy distribution to the energy balance equations.

Non-condensable gases can be simulated as well, e.g. hydrogen, nitrogen, oxygen, or an arbitrary user defined gas. They are basically treated as ideal gases and are assumed to have the same temperature and velocity as the vapour. For the coolant not only light water is available, but also heavy water, carbon dioxide, helium and some liquid metals. In addition, a boron tracking model can be applied to simulate the transport of boric acid, which is dissolved in the liquid phase of the coolant only, through the system.

6.1.2 Heat conduction and heat transfer (HECU)

The simulation of the heat conduction in structures, heat exchangers, fuel rods, electrical heaters and spheres (pebble bed) is performed by the basic module HECU. It permits the user to assign heat conduction objects (HCOs) to all thermal-fluid-dynamic objects of a given network. The HECU module simulates one-dimensional heat conduction in solid materials with up to three material zones separated by a gap. A HECU object can be coupled to a thermos-fluid-dynamic CV on one or both sides, in order to allow heat transfer between two CVs or between a CV and the environment. The basic equation to be solved by the HECU module is the conservation of energy inside of the heat conduction objects (HCOs). The basic assumptions made for the heat conduction theory are that the solid material is homogeneous and isotropic and has no dependencies on pressure. Phase changes of the material are neglected as well.

The specific equations for the heat flow into and out of the layer depend on the geometry of the layer, which can be cylindrical, flat or spherical. HECU also handles the heat transfer between the fluid and the solid structures. In principle, the ATHLET-CD simulation tool divides the heat transfer into several regimes (and sub-regimes), e.g. single phase heat transfer, nucleate and film boiling, condensation, etc., for which different correlations for the calculation of heat transfer coefficients are applied. Details can be found in the ATHLET Manual (Lerchl, Austregesilo, Schöffel, von der Cron, & Weyermann, March 2016).

6.1.3 General simulation control (GCSM)

To simulate the balance-of-plant systems, the general control simulation module (GCSM) is used. It is a block-oriented language, which describes the control and protection systems of the simulated nuclear power plant. [24] Every GCSM control element is restricted to a maximum of four input values, and gives only one output value.

The GCSM provides interface data to the simulation of the thermal dynamic plant behaviour by means of process signals like temperature and pressure values and control signals or hardware actions such as valve positions. There is a total of 41 process signals and 26 different hardware actions available, which are described in detail in (Lerchl, Austregesilo, Schöffel, von der Cron, & Weyermann, March 2016).

6.2 Plant model in ATHLET

6.2.1 General description, thermal-hydraulics and BOP models

The plant model used for the ATHLET simulations includes all components, systems and regulations of a generic KONVOI reactor, which are needed to simulate the accident progression in scenarios with SBO or LOCA accident sequences. The thermal-hydraulic system of the primary and secondary circuits is represented by a two-loops model, see Figure 3. The pressurizer (PRZ) is connected to the single loop (which could optionally also include a leak); the triple loop represents all other loops. The primary and secondary side relief and safety valves, relevant for performing the AMMs are modelled as well. The so-called CDR1D model, which calculates critical flow through nozzles, is applied to simulate the mass flow through the PRZ and steam generator (SG) valves.

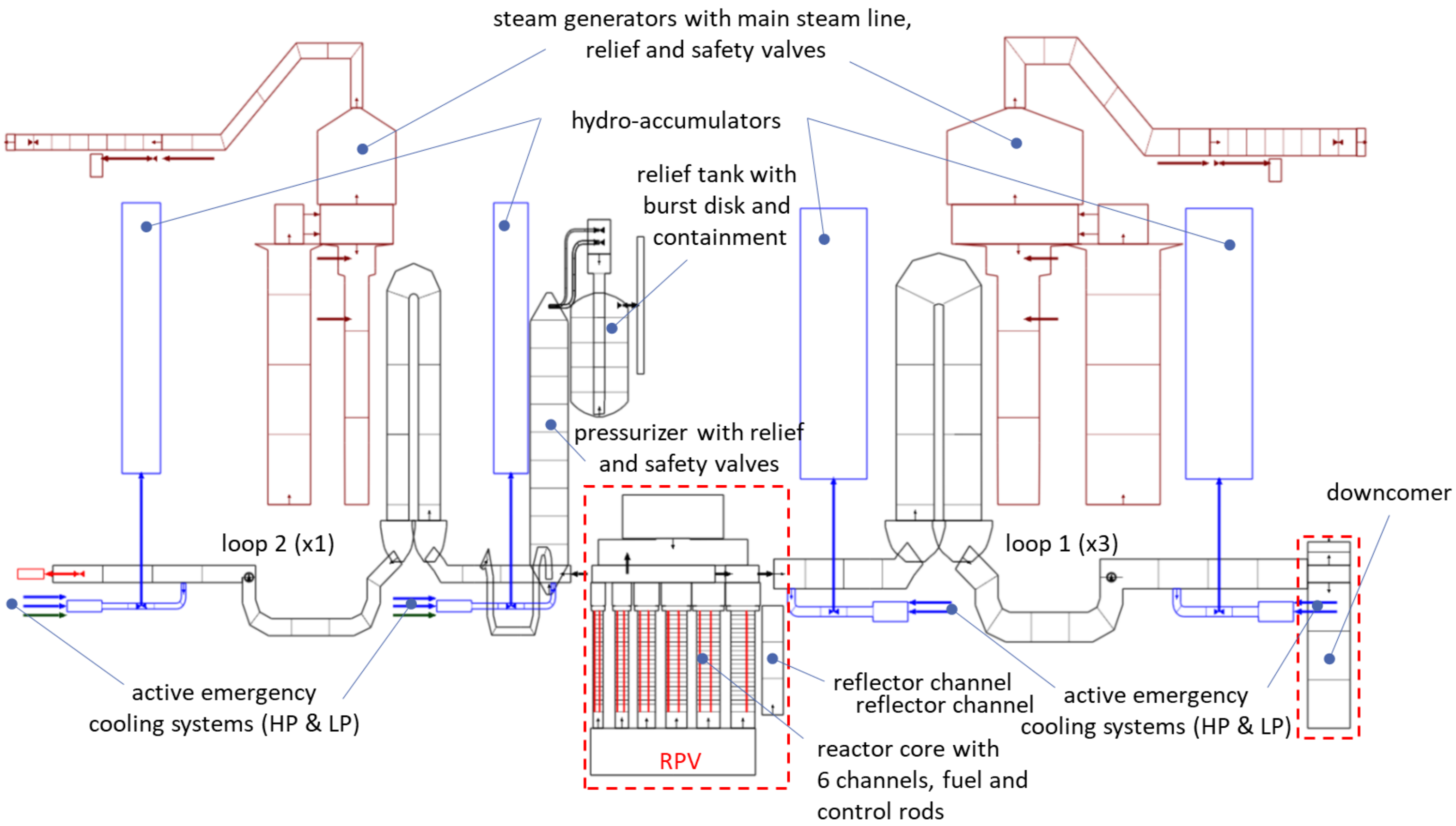

Figure 3: Nodalization scheme for the KONVOI PWR model (two loop representation).

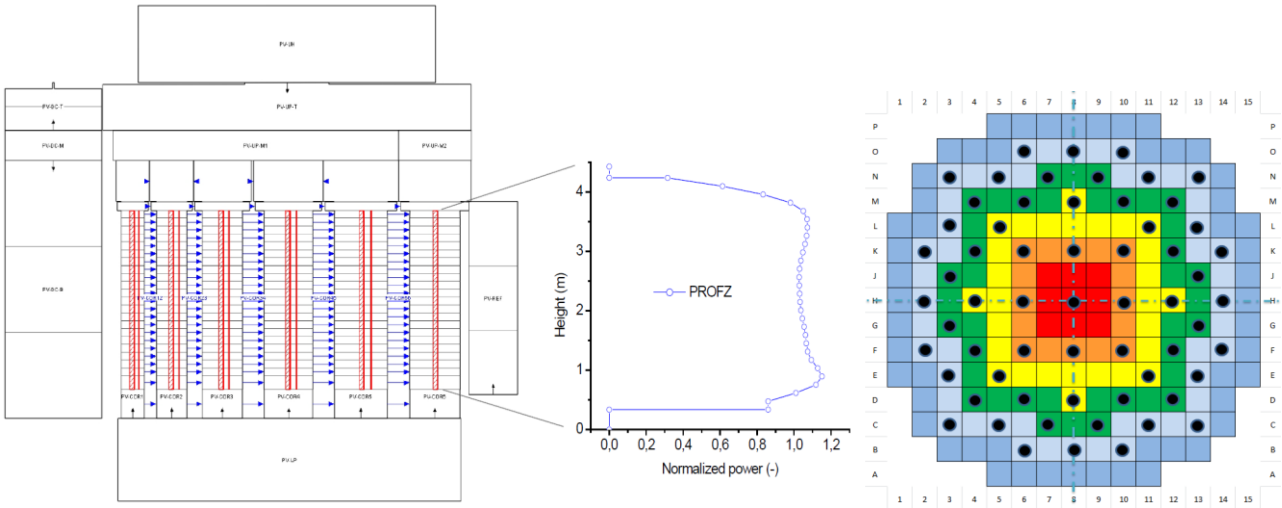

Figure 4: Nodalisation scheme for the RPV of the KONVOI reactor with parallel channels and axial power distribution (left) as well as partitioning of the core in 6 rings (right).

In the reactor pressure vessel (RPV) the downcomer, lower plenum, core, upper plenum and upper head are modelled. For the reactor core a six channel representation is used (see **Error! Reference source not found.**), the parallel channels are interconnected by cross connection objects. The core channels No 1-5 are connected to a common volume in the upper plenum (PV-UP-M1). The core channel No 6 is connected to a separate volume (PV-UP-M2). In this way an ECC water injection from the hot legs via the upper plenum to the core channel No 6 can be modelled ("break through channel"). The core model consists of 57900 fuel rods with an active length of 3.9 m and 1464 control rods. For all fuel rods an axial power profile with the peak in the lower part of the core is applied. The core bypass is modelled by a separate channel.

The pressurizer model also includes a model for the blow down tank. The blow down tank is connected to the containment (modelled as volume with prescribed time dependent pressure and temperature). The nodalization scheme for the PRZ is depicted in **Error! Reference source not found.**. The control logic for the pressurizer relief and safety valves has two main functions: to keep the primary pressure within the set-points of the valves' operation (pressure limitation) or full opening of the valves for PSD.

For the blow down tank the condenser model of ATHLET has been applied. The PRZ relief and safety valves are connected to a common volume. The steam released during the valves' operation is condensed in the water stored in the tank. During steam condensation both water level and pressure in the tank will rise. The burst membrane is also modelled as a valve. When the pressure in the tank exceeds 15 bar, then the membrane breaks and the steam or two-phase flow from the PRZ valves will be released to the time dependent volume which models the containment atmosphere. For PSD all three PRZ valves will be fully opened. The criteria for starting PSD are the signals RPV-level < MIN3 or Tcore-exit > 400 °C (for more detailed information on the application of AMMs see e.g. ((Roth-Seefrid, Feigel, & Moser, 1994), (GRS, Oktober 2001)). Both signals have been modelled in the GCSM part and can be selected by the user. With PSD initiation the pressure limitation is stopped and the depressurization will start.

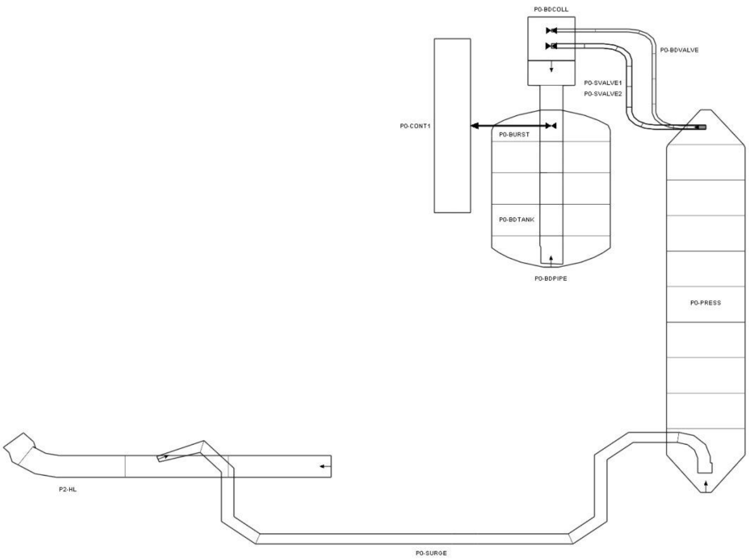

Figure 5: Nodalization scheme for the pressurizer with surge line and blow down tank.

On the secondary side, the steam generators and the main steam lines up to the position of the blow-off and safety valves are modelled (see Figure 3). To be consistent with the modelling on the primary side, the secondary systems are also divided into a single and a triple loop. The thermo-fluid-dynamic systems of the primary and secondary side are connected through heat conducting objects which model the heat transfer in the steam generator U-tubes. The secondary side of the steam generators is divided into the downcomer, the riser, the separator and the steam dome. In the downcomer, feed water flows downwards, is diverted in the lower part of the steam generator and then flows upwards again in the riser. The resulting steam flows through the separator to the dome and then to the main steam line in the direction of the turbine. Steam and water are separated in the separator, the water part flows back into the downcomer and mixes with the feed water.

The turbine is modelled in a simplified manner as a time dependent volume (TDV) in combination with a valve. The TDV defines a pressure-enthalpy boundary condition representing a thermodynamic state near the turbine inlet. Feedwater and emergency feedwater supply are modelled by FILL objects (prescribed mass flow and enthalpy boundary conditions), which can be activated or deactivated by the user depending on the assumed accident scenario

The secondary cool-down procedures (100 K/h cool-down in case of scenarios where a leak is detected or partly cool-down to 75 bar) and the auxiliary feedwater (AFW) supply are modelled by FILL objects and a GCSM control. These can be optionally be activated/deactivated depending on the chosen scenario. For SBO scenarios it is assumed that the secondary pressure regulation is battery powered, see Section4.3. It starts when the secondary pressure exceeds the set-point of 86 bar[1]. Then the secondary pressure will be reduced and kept at a constant value of 75 bar. If available, the AFW supply starts when the SG level drops below 5 m. Then the AFW pumps will feed the SG and, after reaching the nominal level, the regulation will keep the level at this value. The GCSM signals for the cool-down procedure and the AFW availability, to be applied for a specific simulation, have to be selected by the user.

The present plant model also considers the ECC systems – high and low pressure injection systems (HPIS and LPIS) for active, and the HAs for passive safety injection. In a German PWR each of the four independent ECCS

1 Note that the set points of the safety and relief valves or for the pressure regulation vary slightly between the different KONVOI plants, thus also between this section and section 4.3.

has two HAs, one HPIS pump, one LPIS pump, one diesel generator and one flooding pool. The availability of all ECC systems can be varied depending on the investigated scenario.

The reactor protection signals for the reactor SCRAM, turbine trip, stop of MCPs, activation of the safety systems and initiation of countermeasures like secondary cool-down and PSD are also modelled by GCSM signals.

6.3 Plant parameters in normal operation conditions (steady state)

In general, stationary plant conditions should be found before executing a transient ATHLET calculation. This means that the energy, mass and momentum balances must be fulfilled in all parts of the system, and that the time derivatives of the ATHLET solution variables disappear as far as possible (Lerchl, Austregesilo, Schöffel, von der Cron, & Weyermann, March 2016). In doing so, this state should correspond to the given data of the real plant, such as reactor power, steam generator power, temperatures and mass flows. For the model of the generic KONVOI, the adjustment of the steady-state plant state takes place in two steps: 1. steady-state calculation (ATHLET Steady State Calculation SSC) with a simplified, algebraic 4-equation system, 2. subsequent zero transient with 6-equation system.

Table 2 shows the steady state conditions according to ATHLET calculation for the generic KONVOI reactor in comparison with plant parameters. The plant parameters have mostly been taken from publically accessible literature, e.g. (GKN, 2011), (Ziegler & Allelein, 2013), (Kleinedler, 2002).

Table 2: Steady state conditions according to ATHLET calculation for the generic KONVOI reactor in comparison with plant parameters

Parameter	Unit	Plant reference	ATHLET
reactor thermal power	MW	3850	3850
steam generator power	MW	n/a	3869
pressure in upper plenum of RPV	MPa	15.80	15.80
water level in pessurizer	m	7.5	7.5
coolant temperature at RPV inlet	°C	291.7	291.66
coolant temperature at RPV outlet	°C	325.6	325.7
total coolant mass flow rate	kg/s	19760.0	19758.6
main coolant pump speed	1/min	1500.0	1523.4
pump head of MCP	MPa	n/a	0.7139
pressure in steam generators	MPa	6,60	6.526
water level in SG	m	n/a	12.17
feedwater mass flow rate	kg/s	2100	2107.1
live steam mass flow rate	kg/s	2100	2107.1

6.4 Station blackout scenario

The Station blackout scenario will be based on the following assumptions and boundary conditions:

- initial event: total loss of AC power supply (loss of the offsite electric power supply concurrent with a turbine trip and unavailability of the emergency power supply);
- unavailability of all active safety systems and loss of SGs' feedwater supply;
- only passive safety systems and systems powered by batteries are available;
- pressurizer valves and secondary side pressure regulation are available.

The availability of systems and AMM is summarized in Table 3.

Table 3: Assumptions for SBO scenario in generic KONVOI reactor

Primary side depressurization	no
Secondary side cooldown 100K/h	partial
Availability of HPI	no
Availability of LPI	no
Availability of hydro accumulators (HA)	yes
Isolation of cold leg HA	no
Availability of secondary side emergency feedwater	no

6.5 Sequence and timing of major events during the accident

The progression of the accident under the conditions described in Chapter 6.4 is expected to follow the sequence as calculated previously by (Jobst, Kliem, Kozmenkov, & Wilhelm, 2017) using the ATHLET code.

The transient starts at nominal conditions. Following the initiating event, the main coolant pumps and the feed water supply to the steam generators fail. Failure of the main coolant pumps triggers the reactor and the turbine shutdown. The live steam and feed water mass flows drop to zero within a few seconds. After the main coolant pumps have run down, the decay power is discharged on the primary side by natural circulation to the steam generators and from there via the secondary side. With the rapidly decreasing reactor and steam generator power, the steam generator water levels also drop rapidly at the beginning. The signal "secondary side steam pressure > 86 bar" triggers the secondary side partial cool-down to 75 bar.

Due to the secondary-side blow-off control and because of the missing feed water supply, the steam generator levels gradually drop. After the steam generators have dried out (about 45 min after the start of the transient), the steam generator power collapses and the primary circuit pressure rises to the activation pressure of the pressurizer relief valve. A further pressure increase in the primary circuit is limited by periodic valve opening. Due to the rising level in the pressurizer, an increasing amount of two-phase mixture or water flows via the pressurizer relief valve into the containment.

With continued discharge of coolant via the pressuriser relief valve, the level in the reactor pressure vessel begins to decline and steam increasingly forms in the primary circuit. When the level in the reactor pressure vessel drops below the elevation of the hot leg nozzles, steam also flows into the U-tubes of the steam generators. Due to the increasing pressure loss in the U-tubes, the natural circulation breaks down. As a result, no more heat is dissipated to the secondary side. As a result, the primary pressure rises to such an extent that the first pressuriser safety valve also opens. Since the decay heat cannot be dissipated via the pressurizer valves alone, the temperatures on the primary side rise. Further level reduction results in core exposure and subsequent heating. After about 2 hours a core exit temperature of 400 °C is reached and already 10 min later the core exit temperature becomes > 650 °C, which marks the transition to a severe accident (SAMG signal), with strongly increasing cladding tube and fuel temperatures.

The sequence and timing of major events during the accident is summarized in Table 4.

Table 4: Sequence and timing of major events during the SBO accident according to (Jobst, Kliem, Kozmenkov, & Wilhelm, 2017)

Event	Time (s)
Initiating event SBO	1
Turbine shutdown	1
MCP trip	1
MCP speed < 94%	3
SCRAM	4
Initiation of secondary side partial cool-down	14
Complete dryout of SG	2740 – 2800
Rupture disk of blow down tank fails	4122
Water level in RPV below hot leg nozzles	6171
Start of hydrogen production (Zr oxidation)	6695
Core exit temperature > 400 °C	6930
Triggering of SAMG signal **core exit temperature > 650 °C**	7565

7 Simulation of a VVER 1000 Reactor with ATHLET

7.1 System code ATHLET

Refer to the chapter 6.1

7.2 Plant model VVER 1000 in ATHLET

Primary side

Two loops model was used for calculations. First loop represents one real loop and the second loop represents other three real loops. The Pressurizer is connected to the first single loop.

By means of Input Graphics and ATLAS simulator schemes were created. The schemes are presented on fig. 7.1-7.5.

The core is modelled by three parallel channels. Two channels represent 1/3 outer and 2/3 inner fuel part of the core. The core consists of 163 channels. Third channel is a core by-pass. Inner channel contains 34006 average fuel rods, one middle (k=1.456) fuel rod and one hot (k=1.8636) fuel rod. Outer channel contains 16847 average fuel rods and one hot (k=1.8636) fuel rod.

Upper plenum represents the core division - 1/3 and 2/3. Each channel is vertically divided into two volumes. Neighbouring channels are connected due to mixing.

Steam generator tube bundle is modelled by two horizontal layers. Hot and cold collectors are divided by the same way.

Reactor lower plenum is modelled by three volumes. Downcomer is modelled by two channels and geometry respects loops. One channel is connected to one real loop and the second channel is connected to three real loops.

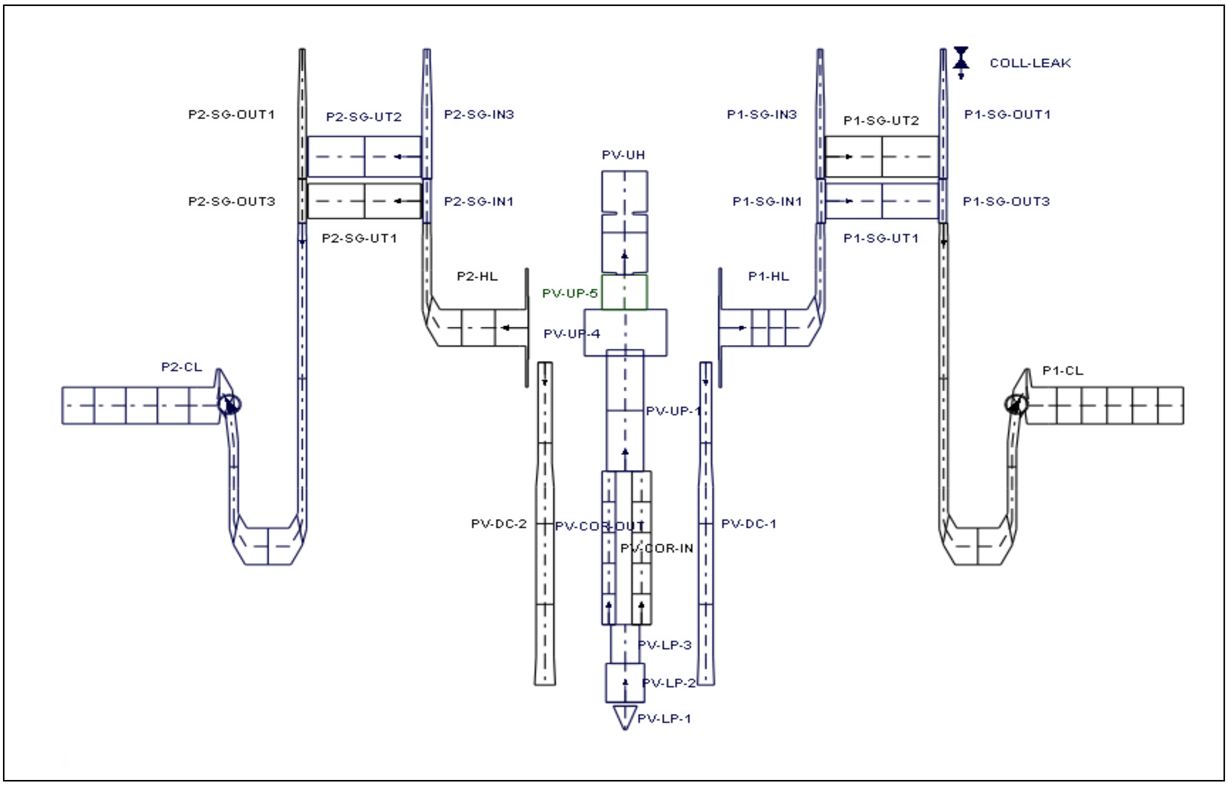

Figure 6: Reactor and primary circuit nodalization

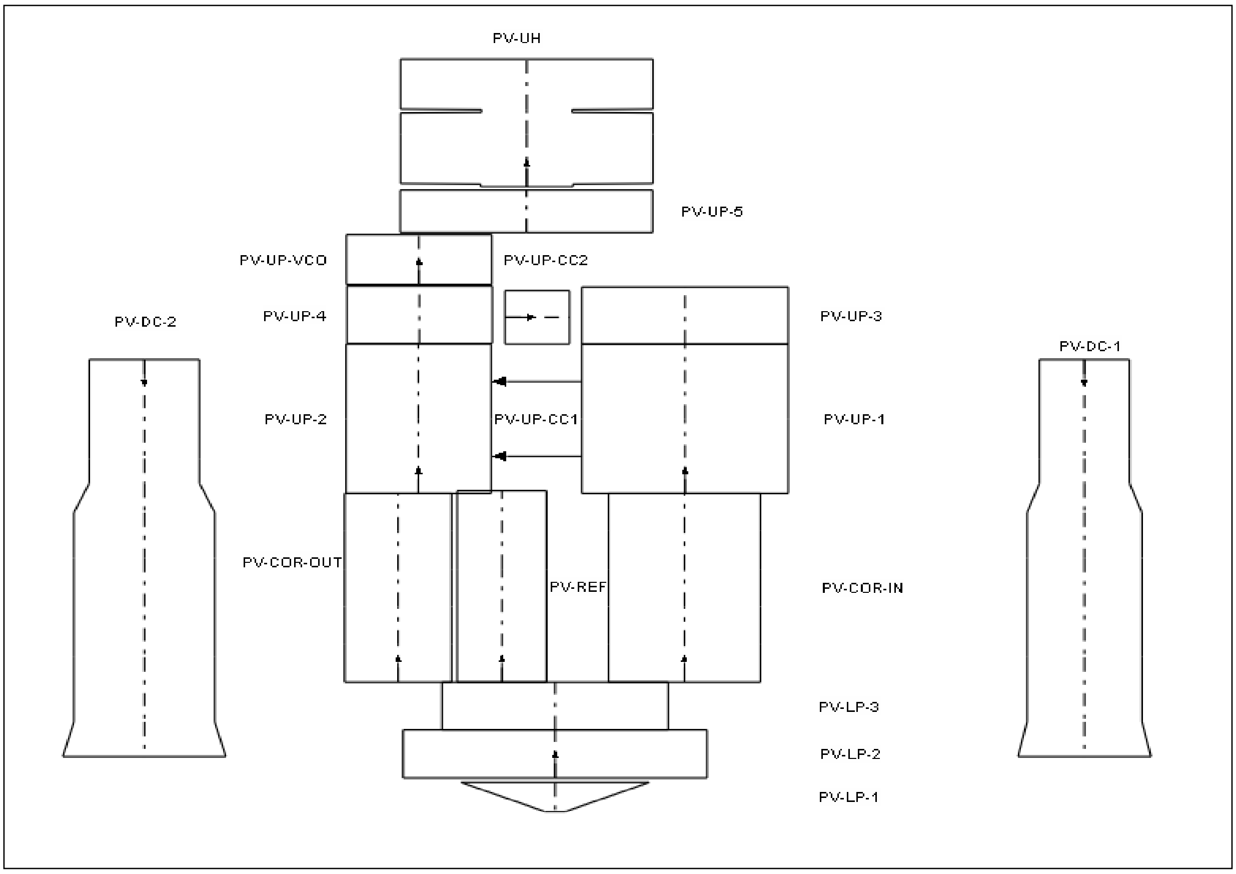

Figure 7: Reactor nodalization

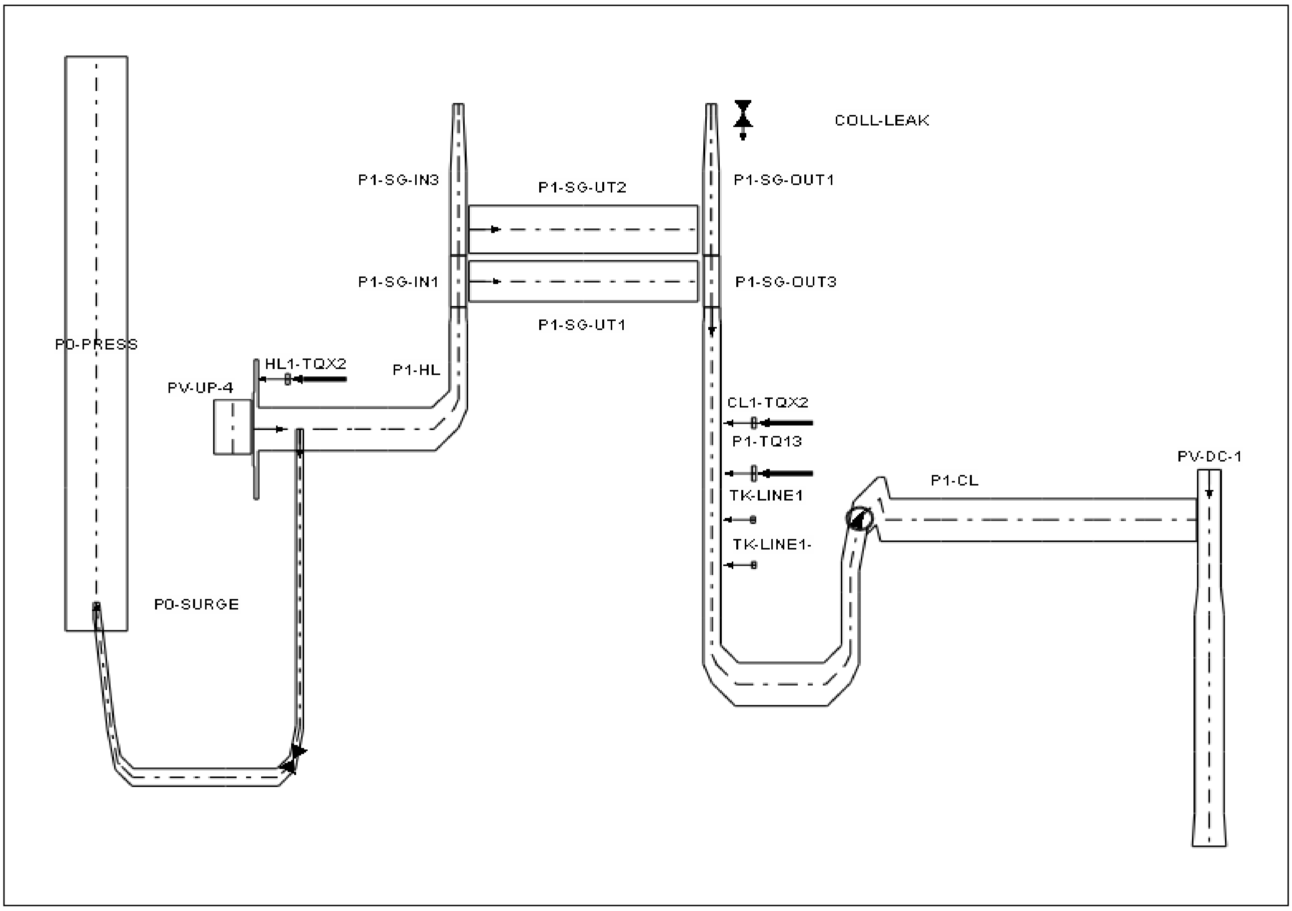

Figure 8: First main loop nodalization

Secondary side

The secondary side nodalization follows the primary side dividing. Steam generators are modelled by one volume which is divided into three layers in the vertical direction. Main steam lines are connected by the main steam header that contains lines with steam dump to the condenser. Steam generator safety valves and steam dump to the atmosphere are connected to the main steam lines.

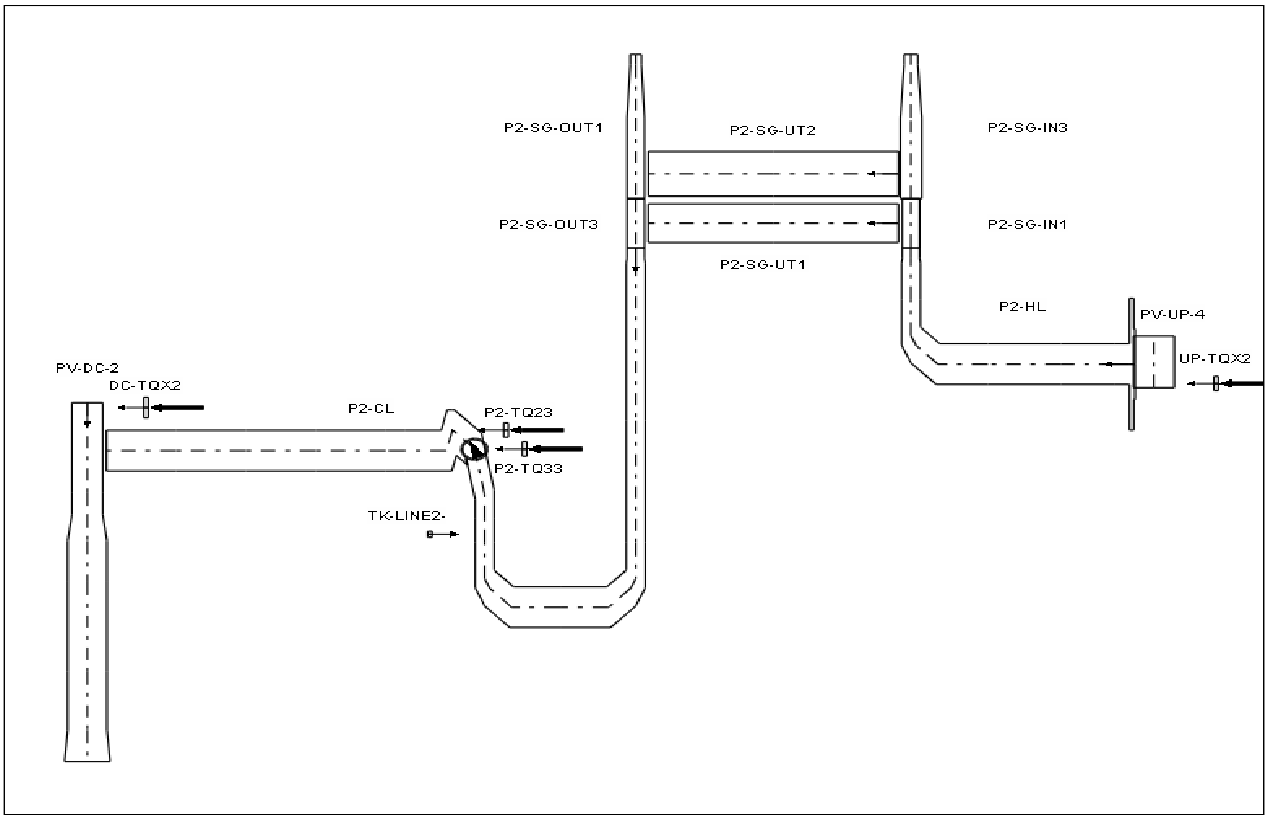

Figure 9: Second main loop nodalization

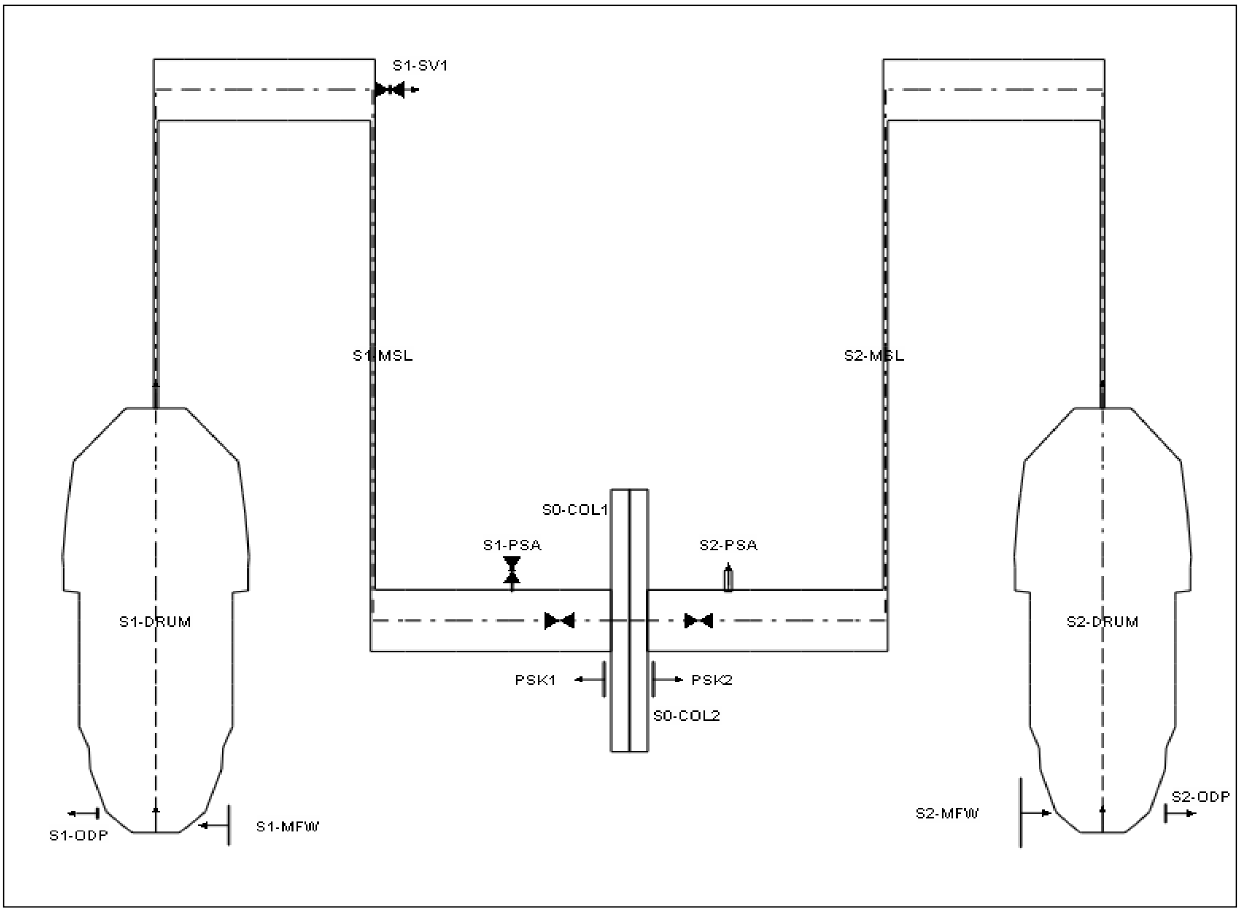

Figure 10: Nodalization of secondary side

7.3 Plant parameters in normal operation conditions (steady state)

Table 5: Primary circuit

Parameter	AHLET calculation	Reference value	Unit
Thermal power	3000	3000	MW_T
Neutron power	2751.6		MW_T
Decay heat	248.4		MW_T
H_3BO_3 concentration at the primary circuit / TK system	0.0		g/kg
Reactor inlet temperature	290.5	288.11	°C
Reactor outlet temperature	322.3		°C
Average reactor temperature	306.4	304.2	°C
Core heat up	31.8	33.3	°C
Reactor mass flow	16780	16711	kg/s
Core by-pass	852		kg/s
Pump speed	1000		1/min
Pressurizer pressure	15.7	15.7	MPa
RCP1 Δp	0.67		MPa
Reactor Δp	0.39		MPa
Core Δp	0.19	0.1808	MPa
Power of pressurizer heaters	270		kW - 1st group
Pressurizer spray	0.2		kg/s
Total pressurizer level (from the bottom)	8.17		m
Primary discharge flow	8.8		kg/s
Primary make-up flow	8.8		kg/s

Table 6: Secondary circuit

Parameter	AHLET calculation	Reference value	Unit
SG1 outlet pressure	6.23	6.28+/-0.2	MPa
SG1 steam temperature	278.4		°C
SG1 steam flow	411.0	408.3	kg/s
SG1 collapsed level	2.13		m
Continual blow-down	2.08	2.08	kg/s

7.4 Station blackout scenario

After SBO, HCC failure associated with LS (c) occurs and at +2.9 s the reactor is shut down by RTS signal. The loss of the external electrical load after switching off the 400 kV circuit breaker leads to a rapid relief of the turbine and consequently to a sharp reduction of the steam withdrawal from all SGs. This initiation event leads to an increase in pressure on the secondary side of the SG, which leads to a reduction in heat transfer from the primary circuit and consequently to an increase in the temperature of the primary coolant, the level in the pressurizer and the pressure in the primary circuit.

SD-C does not open after SBO at all due to low control oil pressure after oil pump failure. The steam from the SG is then discharged through the SD-A after the SBO.

The time delay between turbine unloading and closing of the high-pressure control valves causes the turbine speed to overrun. This overrun is stopped by the rapid closure of the high-pressure control valves and the control flaps. This closure is temporary (in the order of a few seconds), meanwhile, the automatic closure of the fast turbine valve.

After increasing the pressure in the steam collector, the feed water rate to each SG quickly decreased from the original 426 kg / s to about 300 kg / s. The pulse on the fast turbine valve magnet of moving turbine was modeled at +10 s. Subsequent inertia of both TBN then increased this amount to 16.5 t.

Due to the absence of SG feed water and water losses through SD-A, the water supplies on the secondary sides of all SGs are decreasing, accompanied by the gradual detection of heat exchange tubes in the SG and the reduction of the effective heat transfer surface from primary side. After the levels in all PGs fall below 140 cm, the SGs are no able to remove any residual heat produced by the core through the SD-A, and the primary temperature increases by the trend of 39 ° C / hour.

Due to the thermal water expansion, the pressurizer level and pressure increase. At +78 min the pressurizer relief valve is first opened, and the steam is discharged from the pressurizer.

At +100 min, the hot nozzles of the reactor are uncovered, and natural circulation is stopped.

At +121 min, it is not possible to maintain the core outlet temperature at saturation and therefore the fuel coverage temperature will start to rise significantly. At the end of the calculation +127.5 min, the fuel coverage temperature at the top of the fuel assemblies was 672 ° C and increased by a steady trend of about 1.2 ° C / hour. With this trend, a coverage temperature of 1200 ° C would be achieved in about 7.4 min at +134.9 min after the SBO.

7.5 Sequence and timing of major events during the accident

Table 7.3: Main calculation events and operator interventions

Time	Calculation step	Note
0 s	Initial event – SBO	
0 s	Blocked opening of SD-A – from pressure	-
0 s	Loss of all electrical equipment	-
+2.5 s	LS(c) High neutron flux in 3 loops	automatically
+2.9 s	RTS signal Low power in 3/4 MCPs	automatically
+10 s	Turbine and feedwater valve closing	automatically
+12 s	SD-A opening in all SGs, pressure regulation	automatically
+16 min	Initiation of primary temperature increasing (cca 39 °C/h)	-
+16.8 min	SG total water level < 140 cm	-
+73 min	Water level under SG tubes	-
+78 min	Pressurizer relief valve first opening	automatically
+85 min	Steam bubble in the dome	-
+91 min	Pressurizer full	-
+100 min	End of primary natural circuit	-
+101 min	Reaching the saturation temperature at the center in the core	-
+108 min	Reactor water level at the end of the core	Level 5.33 m
+121 min	Rapid fuel cladding temperature increasing (1.2 °C/s)	-
(134.9 min)	Fuel cladding temperature > 1200 °C	-
127.5 min	End of calculation	-

8 Conclusion

The objective of the work presented in this report is therefore to provide initial and boundary conditions for an SBO accident in each code used in the sCO2-4-NPP project (CATHARE, ATHLET, ATHLET-MODELICA).

In a first time, each partner presented how a Station Blackout accident was scripted in each of the codes (as well as in the Konvoi simulator that will be used in the project).

This work shows that despite a common definition, because the codes used are different as well as the types of power plants (EPR, KONVOI, VVER), the scenarios have some differences. This observation is also due to the fact that the accident scenarios modelled in the codes are also based on the operators' standards to manage these situations.

Another difference in the works presented in the report is the fact that the codes take into account the same types of operating parameters (pressures, temperatures...) but these parameters can be expressed in different units. One of the major points of attention for the further work of the project will therefore be to ensure that each partner makes the effort to convert their simulations results if necessary so that the entire consortium can easily take ownership of these results.

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