M&C 2023 - The International Conference on Mathematics and Computational Methods Applied to Nuclear Science and Engineering · Niagara Falls, Ontario, Canada · August 13 – 17, 2023

Coupled Analyses of Steam Line Break for Integral PWR

J-P. Ikonen

VTT Technical Research Centre of Finland Ltd
Kivimiehentie 3, 02150 Espoo

jussi-pekka.ikonen@vtt.fi

ABSTRACT

This study investigates NuScale small modular reactor (SMR) during a steam line break scenario. The scenario is simulated with the thermal hydraulic system code and the neutronics code. The development of the coupling between the codes is done in this work. The main goal is to perform safety analysis, providing corresponding data for code validation, and study the feasibility of the coupling.

The computational tools which are used to perform the numerical calculations are part of VTT's modular calculation system Kraken. Kraken uses modules, which in this study are the nodal neutronics code Ants and the thermal hydraulic system code TRACE. This study is part of the McSAFER project funded by the Euratom research and training programme.

The coupling between TRACE and Ants was achieved, and the steam line break event was simulated successfully demonstrating the feasibility and advantages of Kraken. This work also includes the development of coupling the computational fluid dynamics program OpenFOAM with Kraken and simulating stationary state. However, this part is not in the scope of this report.

The NuScale SMR behavior complied with the limitations set by the acceptance criteria, but additional studies with other computational tools and experimental facilities are needed to validate the results. Oscillatory behaviour was found in the secondary circuit warranting further research on the novel helical-coil steam generators the NuScale design incorporates.

KEYWORDS: nuclear thermal hydraulics, nodal neutronics, multi-scale and multi-physics simulation

1. INTRODUCTION

Nuclear technology is deemed to be hard to change and slow-moving, but small modular reactors (SMRs) are new emerging nuclear power plant types that are attempting to change the view of the nuclear industry. The novel SMR designs try to ease manufacturing and assembly as well as decrease capital costs by simplifying, modularizing, and integrating nuclear power plant systems and components. This paper studies the NuScale SMR, which is a small, 60 MWe, light-water cooled integral pressurized water reactor. [1]

The novel SMR designs differentiate quite a lot from conventional nuclear power plants (NPPs): some of the designs have implemented novel safety systems relying on natural circulation, and the actual geometry and the power output of the SMR reactors are much smaller. This raises the question of whether the current safety analysis methods can demonstrate the reliability and safety of the SMRs with enough accuracy. The thermal hydraulic system codes rely heavily on correlations and simple 1-D or even 0-D modelling.

In this paper, the NuScale SMR is behaviour is investigated during a steam line break transient. The transient is simulated with a thermal hydraulic system code and a neutronics code. The codes are coupled with the VTT-developed Kraken. This work is part of the McSAFER project funded by the Euratom research and training programme.

2. NUSCALE DESIGN

The NuScale power plant is made up of one to twelve NuScale Power Modules (NPMs). Each module has its own dedicated passive safety systems, and they each generate 60 MWe of electrical power and operate independently of the other modules. The NPM is a small, compact, pressurized water reactor with integral helical coil steam generators (SGs). Modules are manufactured in a factory and use a passive approach to safety systems. The NuScale design does not include any primary circulation pumps; instead, all operational states rely on natural circulation. [2]

Steam generators, Reactor Coolant System (RCS) injection and discharge, pressurizer spray, reactor safety valves, Emergency Core Cooling System (ECCS), and Decay Heat Removal System (DHRS) are all integrated within the reactor pressure vessel (RPV). The RPV, control rod drive system, and associated components are housed in a high-strength steel containment vessel (CNV) that is partially submerged in the reactor pool. The ECCS is a critical safety system that provides core cooling during and after accidents such as loss of coolant accidents (LOCA). The integral design eliminates the possibility of large break LOCAs. During non-LOCA events and when feedwater is unavailable, the DHRS provides secondary side reactor cooling. [1]

The NuScale fuel assembly design is the typical 17x17 PWR design commonly used in the nuclear industry. However, the height of a NuScale fuel assembly is half the nominal height of a standard fuel assembly design. The reactor core consists of 37 fuel assemblies and 16 control rod assemblies (CRAs). The assembly is composed of 264 fuel rods that are supported by five spacer grids, 24 guide tubes, and a top and bottom nozzle. The fuel is uranium dioxide (UO2) with the enrichment limited to 4.95 percent. [3]

The NuScale design incorporates novel safety systems that differ significantly to the conventional NPP design. In conventional NPPs, most of the safety systems are active, consisting of pumps and valves which are powered by external power.

2.1 Decay Heat Removal System

The DHRS can be seen in Figure 1. In the NuScale design, each steam generator loop is accompanied with its own redundant decay heat removal equipment train, and each train can remove 100% of the decay heat load and keep the primary system cooled. When the DHRS receives an actuation signal, DHRS actuation valves open, while main steam isolation valves (MSIVs) and feedwater isolation valves (FWIVs) close. As a result, steam is directed from the SGs into the DHRS piping and DHRS passive condensers. The steam condenses in the passive condensers, which are mounted on the outside of the CNV. The reactor pool manages the heat generated in the passive condensers. The system is a closed loop, and the passive condensers are at a higher elevation than the SGs. This arrangement induces natural circulation driven by density differences between the condensate and the steam produced in the SGs. [3]

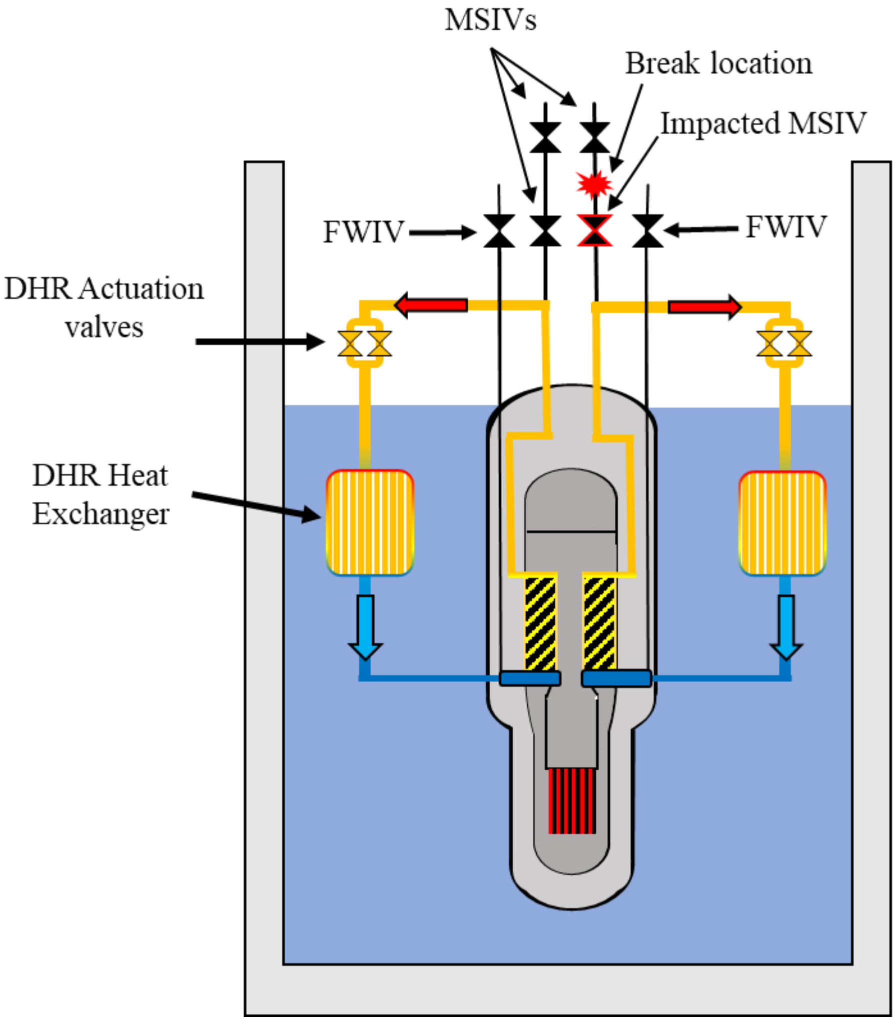


Figure 1. Decay Heat Removal System

2.2 Steam Line Break sequence

The accident event selected to perform in this study is a double-ended guillotine break (DEGB) located at one of the steam lines between the two MSIVs. The modelled break is located in steam line 2. The break flow area in DEGB is 200 %. These kinds of studies are very important to prove the reliability of novel systems, like the DHRS and the helical-coil steam generator

As a result of the steam line break (SLB), steam is discharged through the rupture, lowering the pressure in the secondary circuit. As the pressure drops, the steam saturation temperature decreases inducing more boiling in the steam generators. This increases heat transfer to the secondary circuit causing overcooling in the primary circuit. The overcooling effect decreases core inlet temperature and may cause asymmetric cooling of the core. Additionally, a decrease in RCS temperature causes positive reactivity to be inserted within the core due to the negative moderator temperature coefficient. Reactor power is expected to increase until the reactor trip system (RTS) actuates.

The RTS is expected to actuate due to either a low main steam pressure signal or a high reactor power signal. Though, given the size of the rupture, rapid depressurization of the secondary side is expected and thus, the low main steam pressure signal is triggered first. The DHRS is the safety system expected to manage core cooling during the SLB event. The DHRS is the safety system expected to mitigate the effects of the transient. DHRS is expected to actuate due to the high main steam pressure signal.

See Table I below for the assumptions made in this work. No decay heat is modelled during the transient and one of the control rod assemblies is assumed to get stuck in the core sector where the overcooling effect is the strongest. Conservative assumptions are made regarding the RCS flow rate, average temperature, and SG pressure. Additionally, one of the MSIVs located before the rupture fails to close when the secondary side isolation is actuated. As a result, the affected SG train becomes non-isolable and experiences a complete blowdown. During the break sequence, the feedwater flow is allowed to increase by 0.181 kg/s for every 6890 Pa decrease in SG pressure. In conclusion, these assumptions increase the gravity of the overcooling effect, creating more challenges to the NuScale design.

Table I. Assumption made for the SLB simulation.

Parameter	Nominal Value	DEGB-SL
Core Power (MWt)	160	100%
Rupture Location	-	SL between the primary and secondary MSIVs
PZR Pressure (MPa)	12.755	12.755
Pressurizer Level (%)	60	60
RCS Flow rate (kg/s)	544.29	535.24
RCS Avg. Temp. (K)	558.15	563.71
SG Pressure (MPa)	3.447	3.689
Core exposure		BOC (0.0 MWd/tHM)
Decay Heat		0

3. COMPUTATIONAL METHODS

The SLB scenario is simulated with the thermal hydraulic system code TRACE and the nodal neutronics code Ants. The coupling is made with VTT's modular calculation system Kraken. Kraken is a Serpent-based computational framework that has been developed at VTT since 2017 for coupled core physics problems [4]. Cerberus works as the application programming interface for the Kraken framework, enabling communication between the computational cools. Cerberus is a Python package that provides basic multi-physics driver features such as coupled solution and time stepping controls, field and variable transfers between the computational tools, and convergence checks [5]. Data transfer between the coupled computational tools is executed through sockets, and then the data is transformed into accessible classes by Cerberus.

3.1 The Thermal Hydraulic System Code TRACE

The SLB scenario is simulated with the thermal hydraulic system code TRACE coupled to the 3D nodal neutronics code Ants. The TRAC/RELAP Advanced Computational Engine (TRACE) is a nuclear thermal hydraulic system code developed by the United States Nuclear Regulatory Commission [6]. TRACE version 5 patch 6 is used in this work. The NuScale TRACE model has been developed by Universidad

Politécnica de Madrid (UPM). As seen in Figure 2, the TRACE model includes the primary circuit and also, the secondary circuit up to the turbine stop valve.

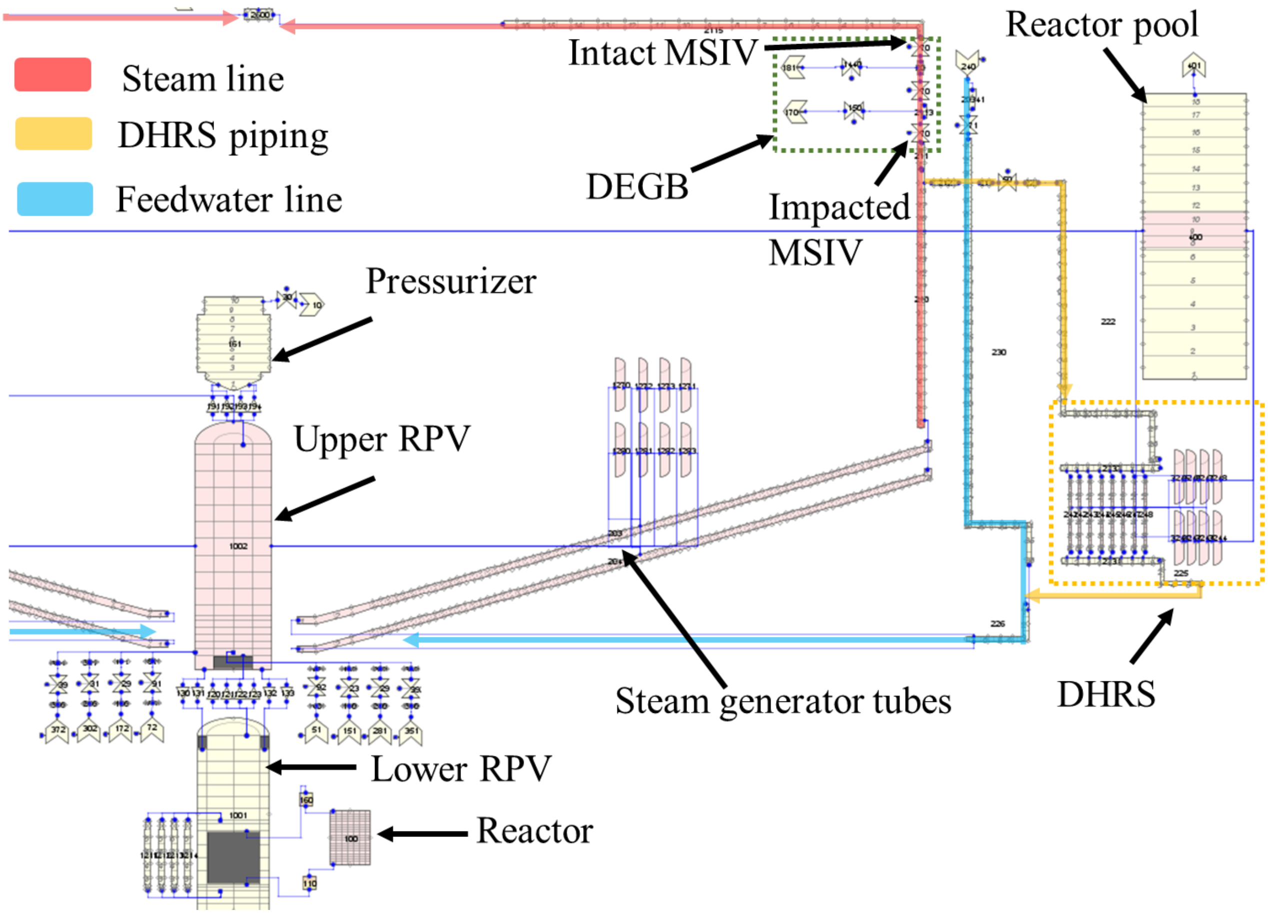


Figure 2. TRACE model of the RCS and one-half of the secondary circuit including the DEGB.

The pressurizer is modelled with a 1D component, but the rest of the thermal hydraulics in the primary circuit are modelled in 3D with two separate vessel components which comprise the rest of the RPV. The secondary side components are modelled in 1D from the SGs to steam and feedwater lines. Each steam line includes one DHRS train that is connected to the reactor pool. The DEGB is modelled in one of the two steam lines, but otherwise, the steam lines are identical. Boundary conditions in the secondary circuit are modelled with FILL and BREAK components, no pumps, turbines, or control systems are included in the secondary circuit. TRACE calculates fuel temperature and coolant properties which are transferred to the Ants model through Kraken.

3.2 Ants

The neutronics are modelled in 3D with nodal diffusion code Ants developed by VTT [7]. The Ants model includes the whole reactor core along with reflector and spacer grids. The reactor is modelled at full reactor power at beginning of cycle condition, but no decay heat is considered during the simulations. The model uses four-energy-group constants generated with the Serpent code. Ants calculates reactor power, which TRACE uses to solve the heat transfer within the fuel rods and out from the cladding surface.

3.4 Coupling scheme

The coupling scheme used in this work is created by coupling the neutronics code Ants with the system code TRACE. Figure 3 illustrates the transfer of data fields between the codes in the developed coupling. In the Trace-Ants coupling, the whole nuclear power plant including the control system is modelled with TRACE. Both of the models include the reactor core; however, TRACE is used for modelling the thermal hydraulics and Ants for modelling the core neutronics.

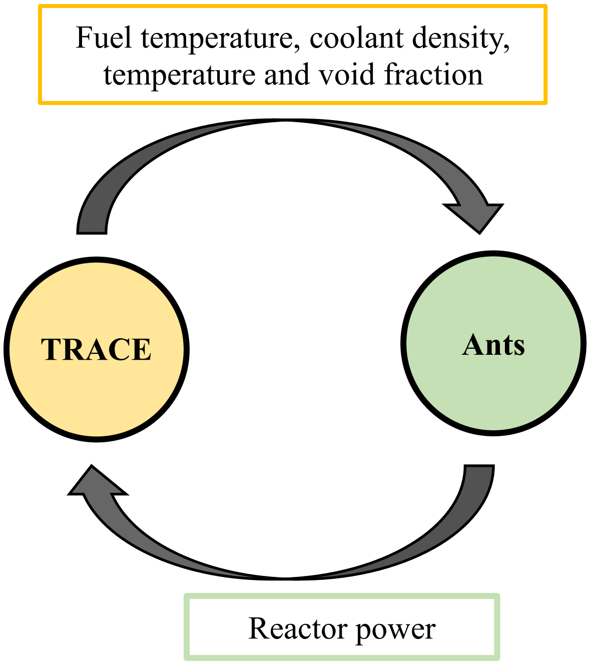


Figure 3. The coupling scheme used in this work.

4. RESULTS

At present, no validated experimental data from experimental facilities is available for comparison. The only data available is from NuScale's FSAR and from other projects conducted in the McSAFER project. See Table II below for steady-state results calculated in this work compared against the FSAR reference values and the results calculated by other McSAFER participants.

Helmholtz-Zentrum Dresden Rossendorf (HZDR) NuScale model was developed with the system code ATHLET and the nodal diffusion code DYN3D. The UPM NuScale model was developed with TRACE and the nodal diffusion code PARCS. The only differences with UPM's TRACE model and the model used in this work are the break model and model of the RPV. UPM modelled the RPV with 1D components and the VALVE components used in the break model are composed of 2 cells instead of 0 cells.

Table II. Comparison of steady-state results.

Parameter	**FSAR**	**VTT (error%)**	**HZDR (error%)**	**UPM (error%)**
Reactor Power [MW]	160.00	160.00 (0.00)	160.00 (0.00)	159.96 (-0.03)
RCS Pressure [MPa]	12.76	12.75 (0.00)	13.09 (2.61)	12.86 (0.85)
Core inlet temperature [K]	-	538.66	527.62	534.95
Core outlet temperature [K]	-	597.12	588.32	595.38
Core Avg. Temp. [K]	563.71	567.27 (0.63)	557.97 (-1.01)	565.17 (0.26)
Core mass flow [kg/s]	496.17	490.81 (-1.09)	494.21 (-0.39)	484.18 (-2.42)
Pressurizer level [%]	60.00	59.16 (-1.42)	58.35 (-2.75)	60.08 (0.13)
SG1 Secondary outlet temperature [K]	-	591.23	584.31	581.25
SG2 Secondary outlet temperature [K]	-	591.92	584.31	581.02
SG1 outlet pressure [MPa]	3.69	3.73 (-2.34)	3.43 (-7.12)	3.685 (-0.10)
SG2 outlet pressure [MPa]	3.69	3.73 (-2.35)	3.42 (-7.18)	3.69 (0.00)
Flow rate MSL 1 [kg/s]	33.53	33.81 (0.84)	33.53 (0.00)	33.75 (0.66)
Flow rate MSL 2 [kg/s]	33.53	34.02 (1.44)	33.53 (0.00)	34.24 (2.12)
DHRS Inlet Temperature [K]	-	345.54	514.10	345.85

The steam line break scenario was simulated with coupled TRACE-Ants calculation scheme. The sequence of events computed by each McSAFER participant can be seen in Table III. Figure 4 presents the main and secondary system parameters during the transient. The simulated SLB event propagated as described in NuScale's FSAR. The overcooling effect caused by the SLB was modest and no rapid power excursion was experienced during the transient as seen in Figure 4 (a). Failure of primary MSIV located in the impacted steam line was assumed which resulted in rapid depressurization of the impacted steam line as seen in Figure 4 (b).

The low main steam pressure signal is triggered at 0.6 seconds after the break when pressure decreases to 2.09 MPa. Then, the reactor trip signal is triggered after a 2 second delay. The low main steam pressure signal also closes MSIVs and FWIVs, isolating the intact steam line. Pressure in the intact steam line starts increasing, reaching 5.52 MPa at 17.54 seconds, which triggers the high mean steam pressure signal. This starts the actuation of the DHRS. After a 2 second delay, the valves to the DHRS start opening and they are fully open after 30 seconds. Natural circulation in the RCS is achieved after the actuation valves are fully open at around 50 seconds, as seen in Figure 4 (c). So far, the calculation results provided by HZDR and UPM are surprisingly similar.

Table III. Sequence of events during the SLB simulation.

Event	**VTT**	**HZDR**	**UPM**
Steam Line Break (s)	0	0	0
Low Main Steam Pressure setpoint (2.0684 MPa) (s)	0.6	0.17	1
SCRAM setpoint	Low Main Steam Pressure		
SCRAM signal generation (2 s delay) (s)	2.6	2.17	3
CRs start to insert (s)	2.6	2.17	3
Turbine valve closure	3.61	0.0	4
MSIVs isolation signal (s)	2.6	2.17	3
FWIV isolation signal (s)	2.6	2.17	3
MSIV of SG1 failure (s)	MSIV of the SG1 fails to close		
MSIV of SG2 closed (s)	7.7		8
FWIV of SG1,2 closed (s)	7.7	7.17	8
High Main Steam Pressure (5.5158 MPa) in MSL2 (Not Affected SG)	17.54	23.2	15
Start of DHRS valves opening	19.54	25.2	21
DHRS valves fully opened	49.54	55.2	51

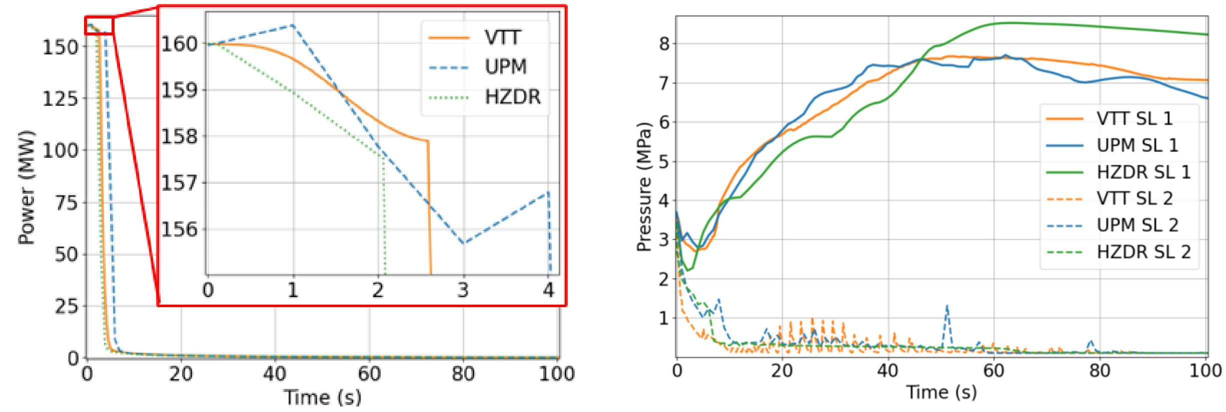
The model of the reactor pressure vessel was divided into four sectors to investigate whether asymmetric reactor cooling appears in the NuScale power plant during the steam line break event. Figure 4 (d) has four graphs: one for each sector. There is only a slight deviation, about 1 K, in the core inlet temperatures calculated at sectors 1 and 3. These sectors depict the impacted side of the reactor, where the DHRS is not functional. Additionally, one of the control rod assemblies was assumed to get stuck during the transient. This rod is located in the first sector. The temperatures calculated at the reactor outlet are very uniform. This indicates that the novel helical-coil steam generators mitigate the effects of the SLB efficiently. The core inlet temperatures calculated in this work decrease more steeply when compared to the other results as can be seen from Figure 4 (e).

However,. As can be seen from Figure 4 (f), the mass flow in the impacted steam line is oscillating quite a lot. It is uncertain whether the oscillations describe the physical phenomena in NuScale SMR realistically or whether the oscillations are a result of inadequate modelling with the computational tools. An experimental facility studying NuScale's helical-coil steam generators has noticed similar behaviour, density wave oscillations, at SIET laboratories [8]. Consequently, the perceived. The NuScale SMR behavior complied with the limitations set by the acceptance criteria.

The calculation result of the transient shows that the nuclear power plant behaviour complies with the limitations set by the acceptance criteria. Maximum pressure in the primary circuit and in the steam generators according to the acceptance criteria is 17.37 MPa, and the results were far below it. Maximum pressures calculated are 12.76 MPa in the RPV, 7.64 MPa in the intact steam line, and 3.63 MPa in the impacted steam line. However, the temperature in the intact steam line rose above the design temperature of the secondary steam system by 54 K. The design temperature for both the main steam system and RPV during an SLB scenario is 616 K. The maximum calculation results in the rest of the nuclear power plant were below the design temperature.

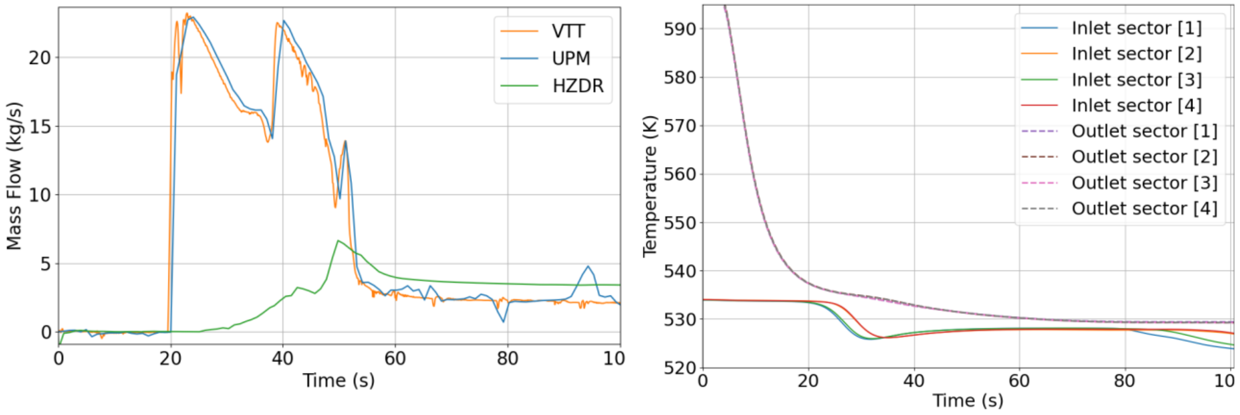
(a) Reactor power

(b) Secondary system pressures



(c) DHRS flow rate

(d) Core temperatures by sector



(e) Core inlet temperatures

(f) Mass flow in steam line 1

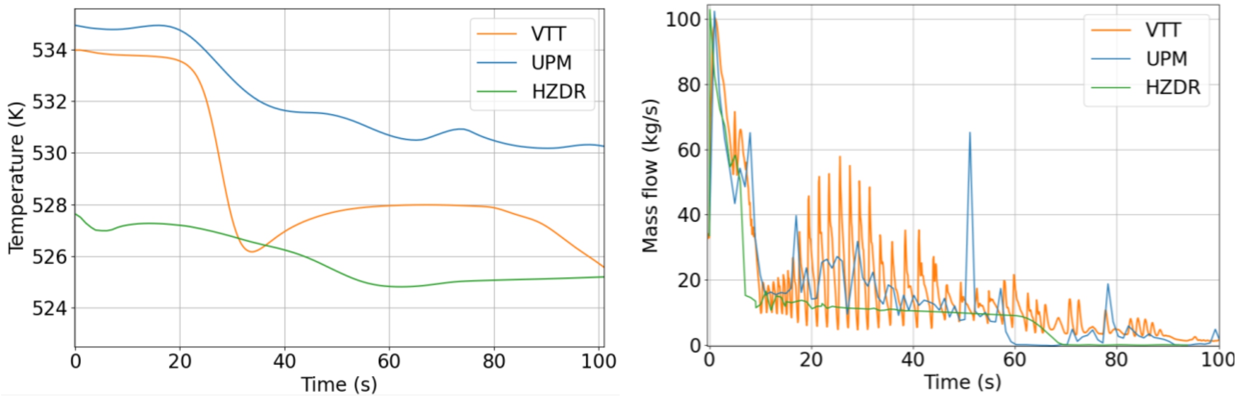


Figure 4. Main primary and secondary system variables calculated during the SLB transient.

5. Conclusions

In this work, the thermal hydraulic system code TRACE was coupled with the neutronics solver Ants. The steady-state results calculated were in fair agreement with the reference values and other calculation results provided by McSAFER participants. The simulated transient event had some differences, but in general, the sequence of events was quite similar. The core inlet temperatures were uniform, indicating that the novel SGs mitigate the effects of the SLB efficiently. The calculation results show that the NuScale SMR complied with the limitations set by the acceptance criteria. However, the temperature in the intact steam line rose above the design temperature.

However, the calculations carried out in this work revealed oscillatory behaviour in the secondary system. The oscillatory behaviour requires further investigation, and it will be studied further in the McSAFER project with the experimental facility MOTEL, and the thermal hydraulic system code APROS at LUT University. This work is also continuing with a focus on computational fluid dynamics. An OpenFOAM model of the NuScale reactor pressure vessel is added to the TRACE-Ants coupling to simulate the primary circuit in even higher detail.

The issue could be within the computational tools, or the models used, but the novel helical-coil steam generators could be the reason for the oscillations. The novel steam generators are being investigated with an experimental test facility at SIET Laboratories, and similar oscillatory behaviour, density wave oscillations, have been observed in the studies.

ACKNOWLEDGMENTS

McSAFER has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063. The content of this paper reflects only the authors' view and the European Commission is not responsible for any use that may be made of the information it contains.

REFERENCES

1. Sanchez-Espinoza, V.H. et al. "The H2020 McSAFER Project: Main Goals, Technical Work Program, and Status," *Energies*, **14**(19:6348), p. 14. Available at: https://doi.org/10.3390/en14196348. (2021)
2. IAEA "Advances in Small Modular Reactor Technology Developments," A Supplement to: IAEA Advanced Reactors Information System (ARIS) 2020 Edition, pp. 339. (2020)
3. NRC, "Application Documents for the NuScale Design: NuScale Final Safety Analysis Report," *NRC Web*. Available at: https://www.nrc.gov/reactors/new-reactors/smr/nuscale/documents.html. (2022)
4. Valtavirta, V. and Tuominen, R. "A Simple Reactor Core Simulator Based on VTT's Kraken Computational Framework," Mathematics & Computation (M&C) 2021, 50003(American Nuclear Society (ANS)), pp. 304–313. (2021)
5. Leppänen, J. et al. "Current Status and On-Going Development of VTT's Kraken Core Physics Computational Framework," *Energies*, **15**(3), p. 876. Available at: https://doi.org/10.3390/en15030876. (2022)
6. TRACE V5.0 P6 THEORY MANUAL "Field Equations, Solution Methods and Physical Models," Washington, DC: Division of System Analysis Office of Nuclear Regulatory Research. (2020)
7. Sahlberg, V. and Rintala, A. "Development and first results of a new rectangular nodal diffusion solver of Ants," *in Proceedings of the PHYSOR 2018*. Cancun: American Nuclear Society (ANS), pp. 3861–3871. (2018)
8. Reyes, J. "A Semi-Empirical Correlation for the Onset of Density Wave Oscillations in a Helical Coil Steam Generator," in Conference: 19th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-19). Brussels, Belgium, March 6 - 11, 2022, pp. 14. (2022)