Multiscale analysis of the Boron Dilution sequence in the NuScale reactor using TRACE and SUBCHANFLOW

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Abstract

This work has been performed within the framework of the 'High-Performance Advanced Methods and Experimental Investigations for the Safety Evaluation of Generic Small Modular Reactors' McSAFER H2020 European project. In this project, the main aim is to advance the research in safety analysis methods for water cooled Small Modular Reactors (SMRs). Hence, several light-water-cooled SMR concepts were selected along with the corresponding accident scenarios to be modelled to evaluate the potential application of advanced modeling tools based on the coupling of different codes and their benefits and drawbacks. In this study, the simulation of the postulated boron dilution scenario caused by a malfunction of the Chemical and Volume Control System (CVCS) in the NuScale reactor is performed using different modeling tools and approaches, specifically, a TRACE-1D, a TRACE-3D and a TRACE-3D/SUBCHANFLOW (SCF) models have been developed to perform this analysis. Two cases have been carried out in this work: the first case is based on the calculation of the boron concentration evolution within the Reactor Coolant System (RCS) of NuScale to compute the time of lossing of the shutdown margin (SDM), and secondly, a best estimate calculation is performed to evaluate the plant response considering the reactivity feedbacks and the performance of the safeguards and safety systems of NuScale according to the public literature. These investigations have demonstrated that the modeling approaches and tools can capture the physics of the problem providing less conservative results than those obtained using the analytical methods present in the DCA report of NuScale. In addition, , it was shown that the NuScale design is robust against the consequences of the boron dilution transient; no relevant asymmetrical effects are expected to take place in the NuScale design due to the singular arrangement of the Helically Coiled Steam Generators (HCSGs) around the riser.

Keywords: Small Modular Reactors (SMRs); NuScale; McSAFER project; Boron dilution; Coupled codes.

Highlights

- NuScale thermal hydraulic models are built for TRACE and TRACE/SUBCHANFLOW codes.
- Boron dilution accidental scenario is simulated showing a robust behavior of the plant.
- In NuScale, asymmetrical effects within the primary side are almost negletable.
- TRACE and SUBCHANFLOW codes capture the physics involved in the selected sequence.

Abbreviations

SMR: Small Modular Reactor; **RPV**: Reactor Pressure Vessel; **HCSG**: Helically Coiled Steam Generator; **CVCS**; Chemical and Volume Control System; **SCF**: SUBCHANFLOW; **RCS**: Reactor Coolant System; **NPM**: NuScale Power Module; **DHRS**: Decay Heat Removal System; **SDM**: Shutdown Margin; **PWR**: Pressurized Water Reactor; **LOCA**: Loss Of Coolant Accident; **DWS**: Demineralized Water System; **BAS**: Boron Addition System; **LWR**: Light Water Reactor; **BWR**: Boiling Water Reactor; **VVER**: Water-Water Energy Reactor; **MTR**: Material Testing Reactor ; **PRZ**: Pressurizer; **BOC:** Beginning of cycle.

1. Introduction

With the advent of the advanced designs features implemented in Small Modular Reactors (SMRs) such the integrated Reactor Pressure Vessel (RPV), the application of one dimensional thermal hydraulic tools to the safety analyses should be assessed carefully. Appropriate models and correlations should be carefully selected for each case considering the appearance of multi-dimensional phenomena associated with the new geometries or design concepts. Moreover, the small cores, the peculiarities of both helical coiled and compact plate-type steam generators, and the passive safety systems used for long-term removal of the residual heat, which are considerably different from the ones of large nuclear reactors of Gen II, may require a careful selection of the numerical tools for safety evaluations.

In September 2020, the 'High-Performance Advanced Methods and Experimental Investigations for the Safety Evaluation of Generic Small Modular Reactors' McSAFER H2020 European project was launched pursuing the aim of advancing the research in safety calculations related to SMRs (Sanchez-Espinoza et al., 2021) ("MCSAFER: Improving safety analysis methodologies and moving from traditional to high-fidelity safety analysis tools for small modular reactors," n.d.). To do so, a very ambitious research program based on experimental investigations at European facilities along with the application of different multi-physics/-scale modeling approaches and advanced tools was defined to perform the analyses of the postulated scenarios for several SMR designs, such as NuScale or SMART, for more details, see (Sanchez-Espinoza et al., 2021).

NuScale is one of the most disruptive and mature designs close to deployment in USA and Europe. It is characterized by an inherently enhanced level of safety thanks to its improved design relying on natural circulation and the implementation of fully passive safety systems to assure the core coolability under all circumstances. NuScale got as first SMR-design the approval of the Nuclear Regulatory Commission; the begin of the first operation of a NuScale plant in the world is envisaged before 2030 (International Atomic Energy Agency, 2022).

In this work, the boron dilution sequence initiated by the malfunction of the Chemical and Volume Control System (CVCS) is simulated (Kliem et al., 2020) with different modeling approaches based on the system thermal hydraulic code TRACE and the subchannel code SUBCHANFLOW (SCF).

Following the computational approaches considered in this study are: a) sytem thermal hydraulic code TRACE to describe the NuScale plant behavior under accidents using the point kinetics model using 1D and 3D modeling approaches and, b) multiscale coupled code TRACE/SCF based on a 3D representation of the RPV by TRACE except the core, which is represented by the quasi-3D subchannel code SCF.

This paper is organized in eight sections. Section 2 describes the main principles of functioning of a generic NuScale Power Module (NPM), where special attention is paid to the Decay Heat Removal System (DHRS). In section 3, the main features associated with the physics involved in the boron dilution sequence in NuScale are discussed. Then, a brief description of the main capabilities of the thermal hydraulic tools used in this work is given in section 4. Afterwards, the models specifically developed to perform the boron dilution calculations are presented in Section 5 along with the main hypotheses considered in the thermal hydraulic models. Sections 6 and 7 discuss the results of the two analyses performed in this study, the calculation of the time to lose the shutdown margin (SDM) considering the power as constant, and the 'best-estimate' case relying on the point kinetics model to compute the power evolution during the transient. Finally, the conclusions drawn from this study are presented in Section 8.

2. NuScale Power Module description.

This section is divided into two subsections: firstly, the general functioning and the main design features of the NPM are described. Then, the DHRS performance under transient conditions is presented.

2.1. Fundamentals.

The NPM primary side is characterized by the integral PWR design of the RPV, where the whole RCS circuit including the steam generator tubes is allocated, eliminating Large/Medium Break Loss-Of-Coolant-Accidents (LOCAs) by design. Hence, only small leaks of coolant could occur due to the rupture of the piping of the CVCS or by the penetrations made in the RPV for the instrumentation rods and the control rod drive mechanisms. As can be seen in Figure 1, the RPV consists of three sections with different diameters for the lower, transition, and upper sections respectively. Furthermore, a small core loaded with

37 17x17 fuel assemblies generates a rated thermal power of 160 MW (NuScale Company LLC, 2020a), which is removed during the normal operation just by the natural circulation-driven mass flow rate inside the RCS (NuScale Company LLC, 2020b).

The mass flow rate within the RCS loop is established passively; it is driven by density gradients along the flowpath between the heat source (core) and the heat sink (HCSGs regions). More precisely, the water inventory within the core is heated and flows upwards through the central hot riser. At the top of the riser, the flow is redirected radially to the annular region, where the HCSG tubes are located. The primary coolant leaving the HCSG lower part, is colder and denser since the feedwater inside the helical couled tubes was heated up. Consequently, the denser coolant flows downwards through the downcomer by gravity. Finally, the flow is again redirected from the downcomer to the central region in the lower plenum reaching the core inlet plane and closing the RCS loop (NuScale Company LLC, 2020b).

Regarding the secondary side, two dedicated feedwater and steam lines can be found in each NPM with the corresponding isolation valves (two for each steam line and two for each feedwater line) (NuScale Company LLC, 2020c). Therefore, the heat generated in the core is removed by two HCSGs (with two trains each) with a total number of 1380 tubes. However, in NuScale the HCSGs are not mounted as independent units as it is commonly found in other SMR designs. In this case, the helical tube bank is arranged within the annular region between the inner surface of the RPV and the outer surface of the upper riser tube in such a way that they are symmetrically intertwined surrounding the riser itself, see Figure 1. Because of it, non-symmetrical behavior of the RPV is almost eliminated (NuScale Company LLC, 2020d).

Given the important role of the CVCS in this study, it should be noted that this system oversees, among other tasks, the management of the chemistry of the primary coolant and, more in particular, controlling the boron concentration in the core region using the makeup and letdown lines (NuScale Company LLC, 2020e). For this purpose, the injection and discharge points of the CVCS makeup and letdown lines are located above the core to ensure that the core is covered with liquid water in case of penetration failure (NuScale Company LLC, 2020e). More specifically, the CVCS injection point can be found at the height around the upper riser inlet and, the discharge point is located at the same height but in the annular region between the RPV and the riser itself. In addition, CVCS counts on two redundant variable-speed makeup pumps (positive displacement kind), which provide the RCS with the coolant with the proper boron concentration achieved by the adequate contributions to the makeup flow of the Demineralized Water System (DWS) and the Boron Addition System (BAS) lines, (NuScale Company LLC, 2020e). In that sense, the main source of unborated water in the NPM is the DWS system, which makes the boron dilution sequence more challenging to the safety systems. However, although CVCS is not a safety system, two automatic, fail closed and safety-related redundant DWS isolation valves are implemented in the CVCS so that the consequences of the inadvertent injection of unborated water in the RCS can be successfully avoided or stopped (NuScale Company LLC, 2020e).

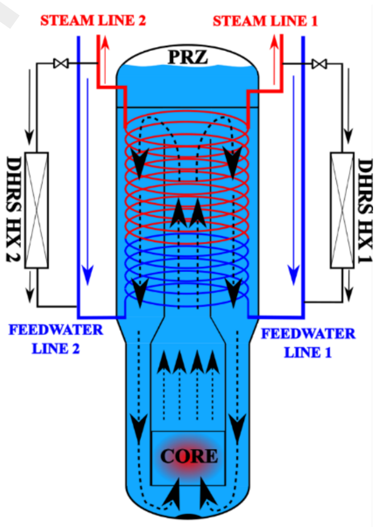

Figure 1: RPV scheme of NuScale including DHRS piping

2.2. DHRS system description.

The DHRS system is a completely passive safety system designed to remove the decay heat generated within the core under certain conditions after a reactor trip. In particular, DHRS actuation is required when the core cooling is not achievable by the normal performance of the secondary side under non-LOCA conditions (NuScale Company LLC, 2020d).

DHRS system is an isolation condenser, which consists of the correspondent piping connected to the steam and feedwater lines along with a heat exchanger with 80 vertical tubes and two collectors attached to the outer surface of the CNV of the NPM and submerged in the reactor pool. It is worth to mention that two DHRS trains are mounted at each NPM and connected independently to each steam line, see Figure 1. Therefore, each secondary side loop has a dedicated DHRS train, which is connected to the correspondent steam line before the MSIVs at the top, and to the feedwater line after the FWIVs at the bottom. In addition, a DHRS actuation valve is located immediately after the connection with the steam line (NuScale Company LLC, 2020d).

From the physical point of view, the mass flow rate in the DHRS is driven by the density gradient in the DHRS loop and the difference in height between the bottom of the DHRS condensers and the bottom of the HCSGs. Hence, when the isolation of the NPM is demanded, the MSIVs and FWIVs are eventually closed, and the DHRS actuation valves open creating the two DHRS natural circulation loops. The saturated steam from the HCSGs outlet is directly flowing to the DHRS piping and, consequently, to the DHRS condensers. When the steam reaches the DHRS submerged pipes and the condensers, it is condensed because of the cooling effect of the reactor pool. For that reason, it becomes denser flowing downwards to the feedwater lines by gravity reaching eventually the HCSGs inlet region again.

The safety signals in NuScale demanding the DHRS actuation are summarized in Table 1.

Table I: DHRS actuation safety signals (NuScale Company LLC, 2020f)

Safety signal	Limit	System Automated Function
High Pressurizer Pressure [MPa]	13.790	Demands the DHRS actuation valves. Demands the actuation of: Primary and secondary MSIVs Primary and secondary MSIBVs FWIVs and feedwater regulating valves
High Narrow Range RCS Hot Temperature [K]	594.261	
High Main Steam Pressure [MPa]	5.516	
Low AC Voltage to Battery Chargers [-]	-	

3. Description of the Boron dilution sequence in the NuScale Power Module

The boron dilution sequence in NuScale can be caused by the malfunction of the CVCS or by the failure of the human action of the operator associated with the control of the boron concentration within the core. The main source of unborated water within an NPM is identified as the DWS (NuScale Company LLC, 2020g).

Consequently, it is highly important to take into consideration that the boron dilution sequence begins with the injection of unborated water within the RCS due to a malfunction of the CVCS. Therefore, a boron dilution front is formed in the CVCS makeup injection point and then, that dilution front begins to go through different RCS regions over and over following the flow path explained in Section 2. When the dilution front reaches the core region, an insertion of positive reactivity takes place due to the reduction of boron concentration. For that reason, the shutdown margin is significantly reduced. Hence, the time to lose that margin needs to be evaluated. As a result of the core boron concentration reduction, the core power is increased and the reactor is eventually tripped by any SCRAM signal, which, in turn, also demands the closure of the DWS's isolation valves finishing the unborated water injection within the RCS. Finally, it should be noted that the decay heat generated in the core can be successfully removed by the actuation of the DHRS system and the reactor pool becomes the ultimate heat sink maintaining the plant conditions under control in the long-term.

4. Modeling tools description

In this section, a brief description of the simulation tools applied in this study is performed making special emphasis on the features of the coupled TRACE/SCF tool.

4.1. The system thermal hydraulic code TRACE

The TRAC/RELAP Advanced Computational Engine (TRACE) code (U.S. NRC, n.d.) is developed by the U.S. Nuclear Regulatory Commission (NRC) as the reference code for analyzing the steady-state and transient behaviour of Light Water Reactors (LWR) of Generation 2 and 3 (PWR, BWR, VVER). The TRACE code has been designed to perform best-estimate analyses of loss-of-coolant accidents, operational transients, and other accident scenarios for light water reactors. It is based on a component approach to model the reactor systems in which the capability to model thermal hydraulic phenomena in both one-dimensional (1D) and three-dimensional (3D) domains is implemented including a set of heat transfer correlations for the wall/coolant heat transfer and interphase heat transfer covering the whole boiling curve (horizontal and vertical flow regimes) i.e. the pre- and –post CHF heat transfer. It solves a system of six conservation equations for mass, momentum and energy for a two-fluids in one or three dimensions. Dedicated models describes the transport of boron in the liquid phase. It also includes correlations to describe the heat transfer in helical coiled steam generators (U.S. NRC, n.d.).

Although the execution time of TRACE simulations is highly case dependent, it can be generally optimized thanks to the usage of the first-order upwind for solving the spatial difference and the stability-enhancing two-step (SETS) numerical method for the time integration (SETS allows the material Courant limit to be exceeded allowing the optimization of the time-step size in slow transients). However, this numerical method cannot be applied when a high-order spatial numerical method is selected. It is the case when e.g. boron dilution sequences are analyzed. Instead a semi-implicit method that imposes the Courant limit is used leading to an increased computational time.

4.2. The subchannel code SUBCHANFLOW

SUBCHANFLOW (SCF) is a fast running and flexible subchannel simulation tool. It can handle rectangular, hexagonal, and plate fuel bundles as well as whole core geometries built from these. The code is based on the legacy COBRA code family (Coolant Boiling in Rod Arrays) (Basile et al., 2010; Rowe, 1973; Wheeler et al., 1976) . Modeling is concentrated on the main physical phenomena occurring in sub-channel flow for steady state and transient conditions in consideration of fast and stable execution. In a basic option steady state and transient problems are solved with an iteration-based COBRA like fully implicit solver, so the flow is restricted to upward direction with lateral exchange. In addition, a solver is available which allows low flow rates, downward flow and buoyancy driven flow. It is based on the semi-implicit SOLA method (Hirt et al., 1975). Coolant properties and state functions are implemented for water using the IAPWS-97 formulation. In addition, property functions for liquid metals (sodium and lead) and gases (helium, air) are available. Two phase flow (boiling) is implemented for water and sodium. The heat transport from the nuclear heated fuel rods to the fluid is based on a cylindrical one- dimensional radial heat conduction model including the gap conductance between the fuel pellets and the cladding. As boundary conditions, the total flow rate or a channel-dependent flow rate can be selected. It is possible to distribute the flow automatically to the parallel channels depending on the friction at the bundle inlet. In addition, a pure top-bottom pressure difference boundary can be used. Fluid temperature at the inlet and pressure at the outlet always have to be prescribed as boundary conditions. An iterative steady-state numerical procedure is available to determine the power at which critical heat flux conditions appear during the simulation. In SUBCHANFLOW, profit is taken from the many valuable empirical correlations for pressure drop, heat transfer coefficients, void generation, etc., collected over the last decades. Consequently, it does not follow the general trend to describe two-phase flow by simulating the processes on a microscale basis e.g., separate conservation equations for liquid droplets, films or vapor bubbles. In SUBCHANFLOW, a three-equation two-phase flow model that is a mixture equation for mass, momentum, and energy balance is implemented. The constitutive relations are expressed as mixture equations for wall friction and wall heat flux as well as a two-phase flow slip velocity correlation. Boron transport can be tracked in a manner conserving strong concentration gradients. The main calculation routines are allowed to be controlled by a series of C programming language functions interfaces. They permit to use SCF as an external recompiled library, to exchange all the data necessary for coupled neutron physics calculations. The input is oriented to a text-based user interface using comprehensive keywords and simple tables. All data are given in SI units apart from temperatures which are in °C to be converted in Kelvin inside the code. Extensive validation work is performed using data of relevant experiments for PWR, BWR, and MTRs (Almachi and Sanchez Espinoza, 2022; Imke and Sanchez, 2012).

4.3. TRACE/SCF coupled tool

The TRACE/SCF multiscale tool developed at KIT has been applied to the analysis of the boron dilution sequence in the NuScale reactor by the UPM research group. It is based on the ICoCo (Interface for Code Coupling) and the SALOME platform (Zhang et al., 2021). This is a server-client system where TRACE and SCF are in charge of performing the corresponding calculations, ICoCo is responsible for inter-code data exchange, and SALOME is the supervisor which synchronizes the calculation of both codes.

4.3.1. ICoCo Interface

ICoCo is a standard Application Protocol Interface (API) for code coupling, which was developed by CEA within the framework of the European NURISP project to handle various code coupling tasks. It is an very flexible, modular object-oriented generic interface (framework) for code coupling. It does not contain any calculation code. Therefore, ICoCo provides only the corresponding functions and methods (setDataFile, initialize, computeTimeStep, solveTimeStep, getOutputMEDField, setInputMEDFields…) to the involved codes along with the description of the activities performed by them. For example, some methods allow to insert various input and output ports to the coupled codes making the interaction between the selected codes very flexible and convenient. On the other hand, ICoCo relies on the MEDCoupling library (also implemented in the SALOME platform) to perform the mesh and field mapping between codes in coupled calculations for the data exchange between them. For further information see (Zhang, 2020).

4.3.2. SALOME platform

The SALOME platform is open-source software for numerical pre- and post-processing activities, but it can be also used as the environment to perform code coupling and relevant simulations. The platform was initiated by CEA, EDF, and OpenCascade. It has been employed in a wide range of engineering fields (CEA/DES et al., n.d.). SALOME allows performing efficient code coupling using a very powerful mesh interpolation tool included in the built-in module MED, known as the MEDCoupling libraries (CEA/DES et al., n.d.). The SALOME built-in module YACS is in charge of the development as well as the execution of the coupling codes.

The development of the TRACE/SCF coupling code in SALOME can be summarized in the following steps (Zhang et al., 2021):

1. Develop ICoCo for TRACE and implement them together into SALOME.
2. Develop ICoCo for SCF and implement them together into SALOME.
3. Develop the interpolation tools to handle the field mapping between different meshes of TRACE and SCF. Implement the interpolation tools into SALOME.
4. Construct the scheme for coupling simulation in SALOME for the selected transient.

In this work, the domain decomposition approach is applied, and the data exchange is only performed at the core inlet and the core outlet (two planes). The fields to be exchanged between the codes are listed below:

TRACE → SCF:

1. Core Inlet:
   - mass-flow rate,
   - coolant temperature
   - boron concentration.
2. Core Outlet:
   - pressure.

SCF → TRACE:

1. Core Inlet:
   - pressure.
2. Core Outlet:
   - mass-flow rate,
   - coolant temperature,
   - boron concentration.

5. Description of the modeling approaches

In this section, the main features of the thermal hydraulic models of the NuScale SMR built to perform the calculation of the boron dilution sequence are briefly explained. Regarding the modeling approaches applied in this study, three different modeling constraints were imposed on each step of the study as follows:

1. The full-plant TRACE model (RPV, steam lines, feedwater lines, DHRS, etc.) is based on only 1D components referred to as the TRACE-1D model.
2. In the full-plant TRACE model, the RPV and core is based on 3D model (VESSEL components); hereinafter it is named as TRACE-3D model.

3. In the full-plant TRACE model under point 2, the TRACE core model is replaced by the SCF model; it is referred to as the TRACE-3D/SCF model.

It is worth to be noted that the development of the different models listed above is based on public data sources, see (NuScale Company LLC, 2020h, 2020a, 2020d, 2020i, 2020e, 2020c, 2020b, 2020f, 2019a, 2019b, 2018, 2016), or some data have been estimated by expert judgment. In the models, the RCS is divided into different volumes and regions according to the NuScale DCA, see (NuScale Company LLC, 2020a, 2020d), as listed below:

1. The core region is divided into Lower/Upper Core Plates, Fuel Assemblies and Bypass volumes.
2. Hot leg region is formed by the Lower Riser, the Transition Riser, and the Upper Riser volumes.
3. The HCSG primary side region is formed by the outer volume of the HCSG tubes.
4. The cold leg consists of Upper DC, Transition DC, Lower DC and Lower Plenum volumes.
5. Pressurizer (PRZ) region.

The key-features implemented in the three different models are briefly described below:

- The modeling of the RCS flow path has been achieved using a 1D PIPE component for each of the regions identified previously for the TRACE-1D model in the first step of this study, see Figure 2. However, for the second and the third cases of this study, the same nodalization scheme of the RCS (apart from the core region) is used. Two 3D cylindrical VESSEL components have been implemented in the models considering 2 radial rings (one for modeling the central region and the other one for the annular region between the inner RPV surface and the outer surface of the central region (riser and core barrel)) and 4 azimuthal sectors, see Figure 3. Finally, to do the modeling of the region where the HCSGs are located, the dedicated kind of PIPE and VESSEL, 'Tube Bank Crossflow', implemented in TRACE is selected respectively,(U.S. NRC, n.d.).
- To do the modeling of the core region, the active core height with a total length of 2 m is divided into 20 axial levels for the TRACE-1D model. However, in addition to including 20 axial levels, a fuel assembly to hydraulic channel ratio of 1:1 is applied in the 3D Cartesian VESSEL and SCF core models. As it was commented in section 2, the NuScale core is formed by 37 fuel assemblies. Therefore, a very detailed model of the active core region with a total of 740 hydraulic cells for the TRACE-3D and TRACE-3D/SCF models has been built. Additionally, the bottom/top nozzle axial levels of the fuel assemblies along with the lower/upper core plates have been included in the models. Finally, the bypass flow is modelled separately using PIPE components (one for the TRACE-1D model and four for the TRACE-3D and TRACE-3D/SCF models).
- Another relevant region to be modelled with care is the one in which the HCSG tube bundles are allocated. In that sense, the reader should keep in mind that the secondary side is modelled in the same manner for the three commented modeling approaches. As commented a few lines above, the primary side is modelled using the dedicated option of PIPE and VESSEL in TRACE ('Tube bank crossflow') switching on the adequate heat transfer and friction correlations to this kind of special geometry. Similarly, the dedicated kind of PIPE, 'Curved pipe', in TRACE in combination with a fine nodalization of the HCSGs tubes with 39 hydraulic cells has been implemented in the models to do the modeling of the HCSGs secondary side representing the inner flow through the HCSGs tubes, as it is recommended in (Mascari et al., 2023). Finally, for the four different HCSG tube bundles to be explicitly modelled, four PIPE components were implemented in the models.
- The modeling of the PRZ region is also performed using the dedicated kind of 1D PIPE component, 'Pressurizer', in TRACE allowing to consider the actuation of the PRZ heaters and PZR spray to control the RCS pressure along with its trip in a very simple manner.
- The CVCS makeup and letdown lines have been implemented in the models as boundary conditions (mass-flow rate, injection temperature and boron concentration are controlled), as can be seen in Figure 2, Figure 3, and Figure 4.
- Regarding the secondary side and the safety systems implemented in the models, it should be noted that the secondary side is modelled up to the turbine stop valve along with the DHRS actuation valves, DHRS piping, DHRS condensers and the reactor pool using 1D components.
- The instrumentation and control signals of the plant, paying special attention to the SCRAM signals logic along with the control rods worth, have been implemented in the thermal hydraulic models to simulate the plant behaviour during the selected transient.

- Finally, for the transport of boron within the RCS to be properly modelled, the semi-implicit method has been implemented in the TRACE models for doing the time integration along with the high-order numerical technique, Van Leer with flux limiters, for the spatial differences allowing to reduce the amount of numerical diffusion. On the SCF side, an explicit solver for the time integration is used in combination with a second-order technique for the spatial differences.

The main hypotheses assumed to perform the steady-state and the correspondent transient calculations are in line with the ones described in (NuScale Company LLC, 2020g) as listed below:

- The reactor is initially working at Hot Full Power (100%) Beginning-of-Cycle (BOC) condition. This condition determines the axial power profile included in the models, see Figure 5. The Decay Heat is increased up to a value of 120%.
- Normal AC power is assumed to be available.
- The plant control systems and safety features perform as designed.
- No human actions are credited to mitigate the effects of the transient.
- The regulating CRA bank is not credited to mitigate the reactivity insertion associated with a boron dilution of the RCS.
- Nominal values are assumed for the RCS except for the mass flow rate (535.24 kg/s).
- The letdown flow rate is equal to the makeup flow rate (3.15 kg/s).
- The initial Boron concentration is fixed at 1600 ppm.
- Reactivity feedbacks are not credited in the steady state calculation. However, in the best-estimate calculation of the boron dilution sequence, the following values (conservative) for the reactivity coefficients have been selected, see (NuScale Company LLC, 2020g):
    - Coolant temperature reactivity coefficient: 0.00 pcm/K
    - Fuel temperature (Doppler) reactivity coefficient: -2.52 pcm/K
    - Boron reactivity coefficient: -10.00 pcm/ppm
- A deviation of around ± 5 % from the nominal value (33.535) of the feedwater mass flow is implemented in each Feed Water (FW) line to evaluate the HCSG performance under asymmetrical conditions.
    - FW line 1 mass-flow rate: 35.22 kg/s
    - FW line 2 mass-flow rate: 31.85 kg/s
- The FW injection temperature is fixed on 421.875 K.

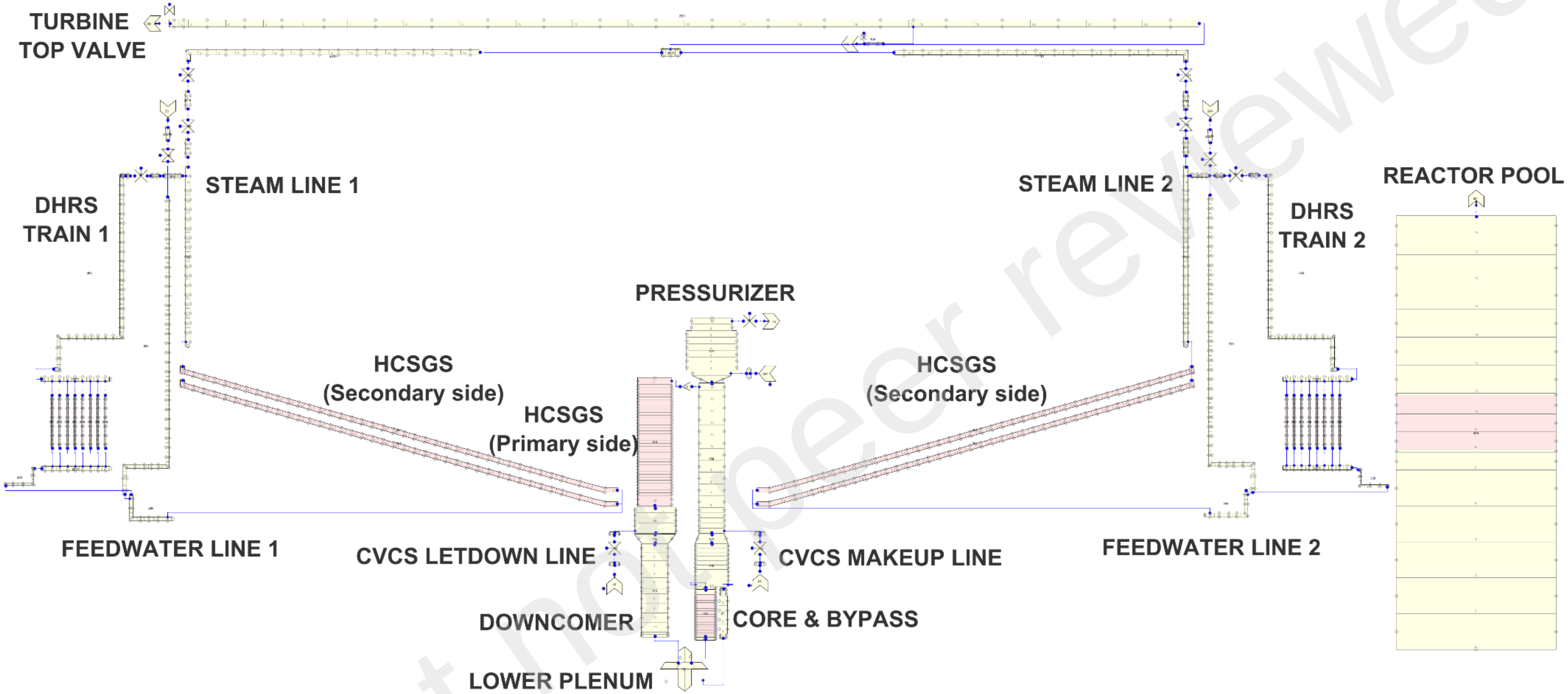

Figure 2: Nodalization scheme of the full-plant TRACE-1D model

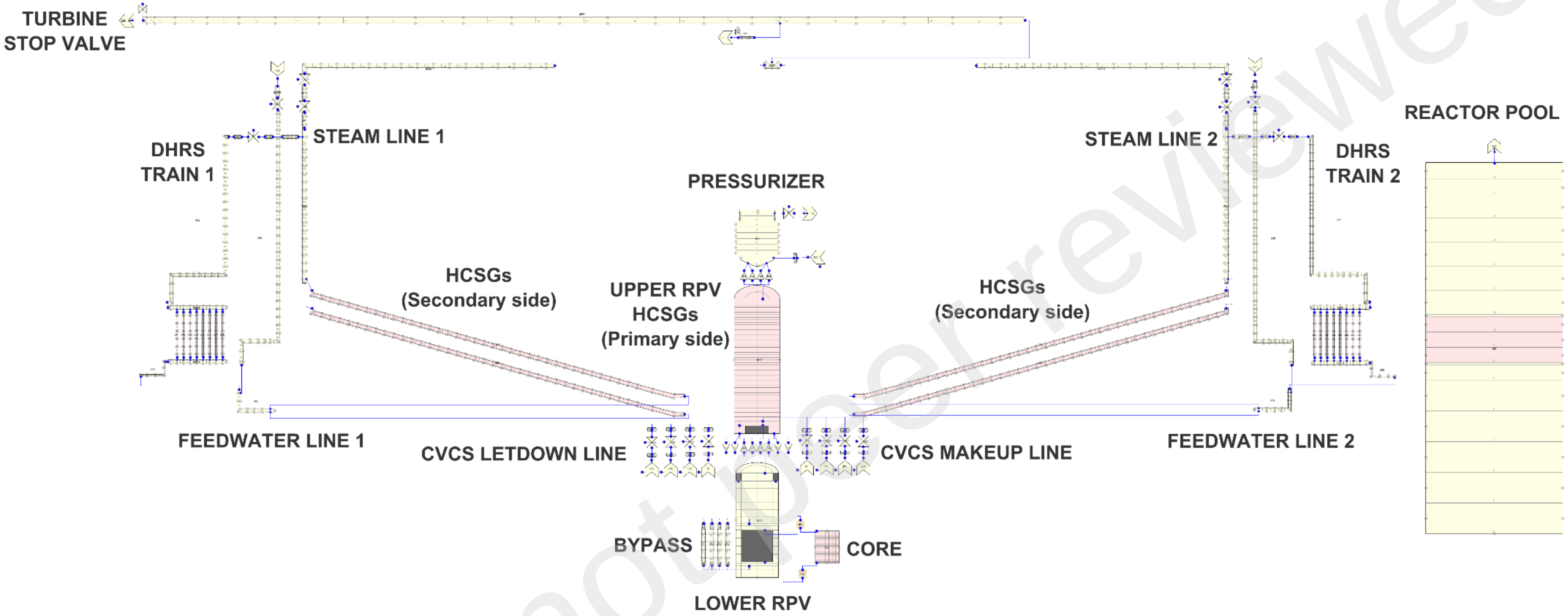

Figure 3: Nodalization scheme of the full-plant TRACE-3D model

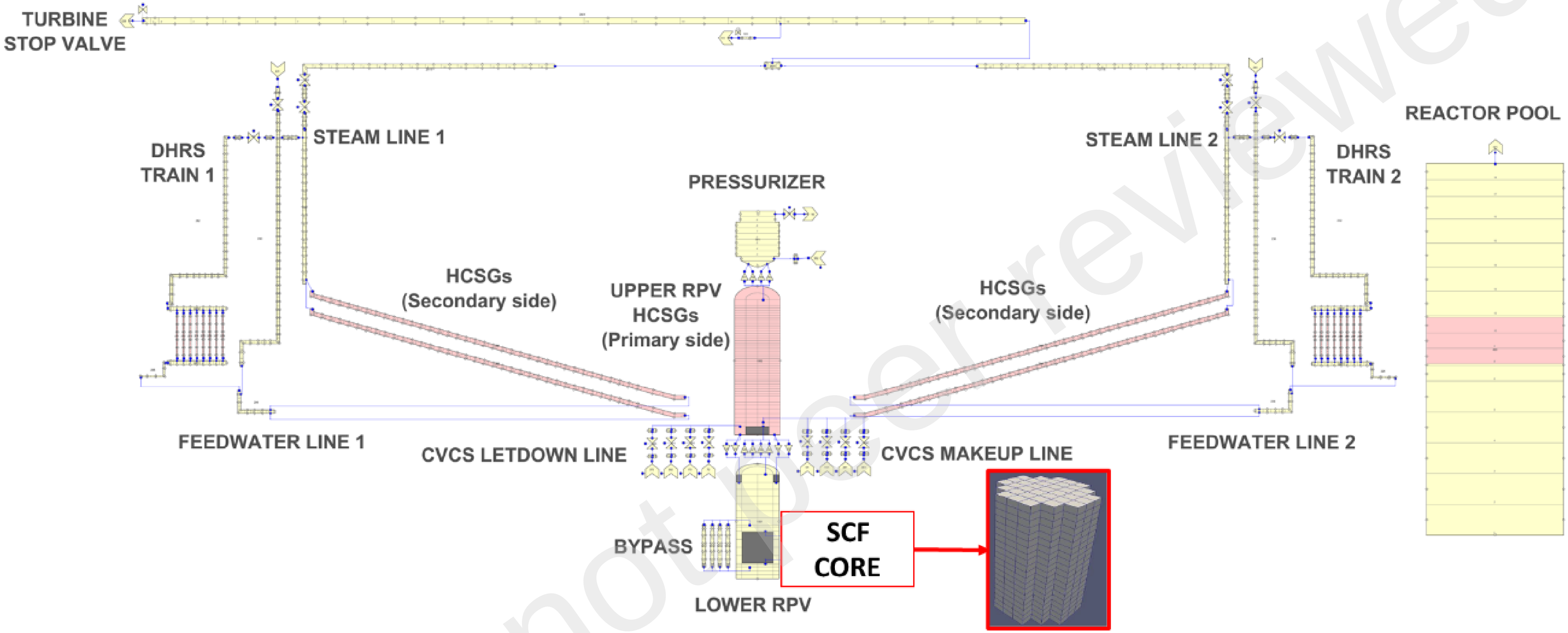

Figure 4: Nodalization scheme of the full-plant TRACE-3D/SCF model

5.1. Steady-State calculation

A steady-state calculation has been performed using the described models for NuScale obtaining a very good agreement with the reference data included in the DCA of NuScale (NuScale Company LLC, 2020g), see Table II. The 3D distributions of the main parameters of the core (power, pressure, temperature and mass-flow rate) obtained using the TRACE-3D/SCF model during the steady-state calculation are presented in Figure 6 and Figure 7. More in detail, Figure 8 and Figure 9 present the radial distributions channel-by-channel for the core mass-flow rate and for the core outlet temperature obtained using the TRACE-3D and TRACE-3D/SCF models.

Table II: Comparison of results of the steady-state calculations

Plant parameter (SI Units)	Reference	TRACE-1D (Error, %)	TRACE-3D (Error, %)	TRACE-3D/SCF (Error, %)
RCS Pressure [MPa]	12.755	127.68 (0.10)	127.72 (0.13)	12.754 (-0.00)
RCS Inventory (No PRZ) [kg]	-	40009.51	39920.26	39856.82
Avg. RCS Mass-flow rate [kg/s]	535.24	533.81 (-0.27)	532.38 (-0.55)	535.35 (0.02)
RCS Mass-flow rate S1 [kg/s]	-	-	133.05	133.79
RCS Mass-flow rate S2 [kg/s]	-	-	133.14	133.89
RCS Mass-flow rate S3 [kg/s]	-	-	133.05	133.78
RCS Mass-flow rate S4 [kg/s]	-	-	133.14	133.89
Avg. Core Mass-flow rate [kg/s]	496.17	495.59 (-0.12)	496.34 (-0.03)	497.27 (0.22)
Core Mass-flow rate S1 [kg/s]	-	-	123.81	123.95
Core Mass-flow rate S2 [kg/s]	-	-	124.36	124.69
Core Mass-flow rate S3 [kg/s]	-	-	123.81	123.94
Core Mass-flow rate S4 [kg/s]	-	-	124.36	124.69
Avg. Core Inlet Temperature [K]	531.48	531.13 (-0.06)	531.12 (-0.07)	532.72 (0.23)
Core Inlet Temperature S1 [K]	-	-	530.99	532.61
Core Inlet Temperature S2 [K]	-	-	531.24	532.84
Core Inlet Temperature S3 [K]	-	-	530.99	532.60
Core Inlet Temperature S4 [K]	-	-	531.24	532.84
Core Inlet [B] [ppm]	1600.00	1600.00 (0.00)	1600.00 (0.00)	1600.00 (0.00)
PZR level [%]	60.00	59.998 (-0.00)	59.995 (-0.01)	60.03 (0.05)
Avg. Steam Temperature [K]	580.04	585.50 (0.94)	585.01 (0.86)	582.37 (0.40)
Avg. SG Outlet Pressure [MPa]	3.447	3.464 (0.48)	3.451 (0.12)	3.447 (0.00)

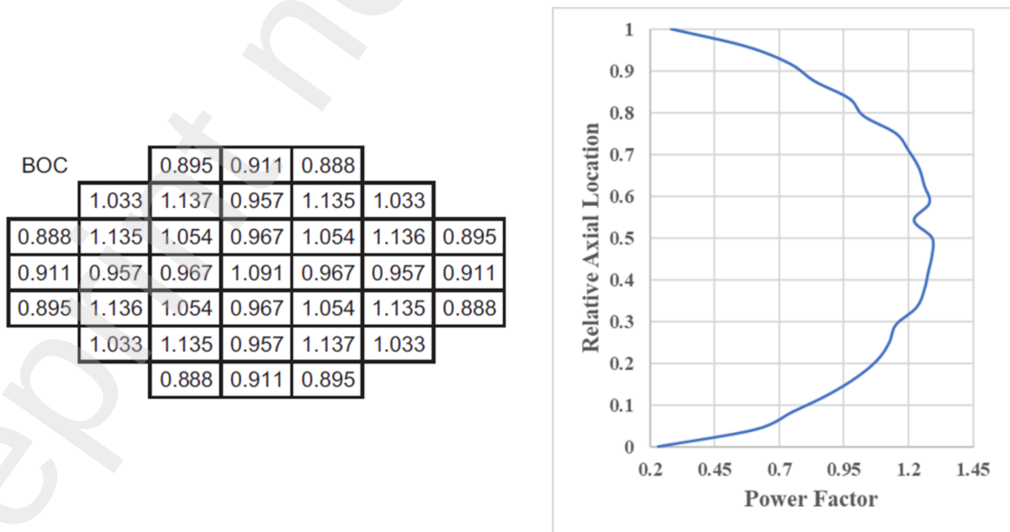

Figure 5: BOC Radial (left) and Axial(right) power profile implemented in the core models (NuScale Company LLC, 2020a)

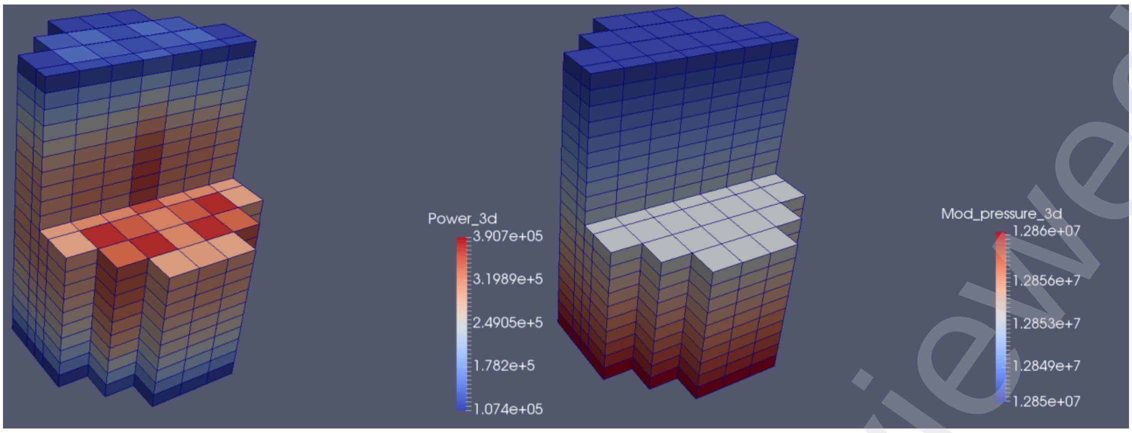

Figure 6: Core power (left) and pressure (right) in the steady state calculation (TRACE-3D/SCF)

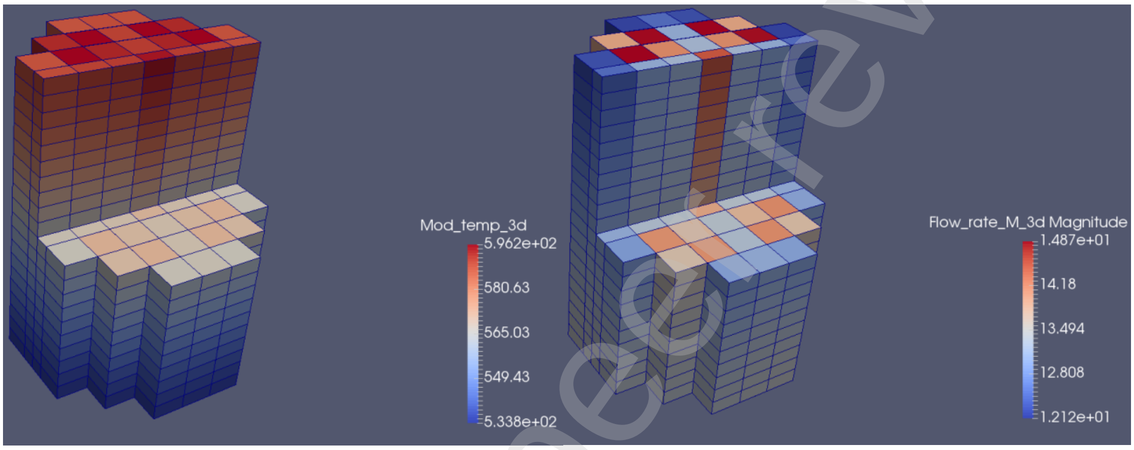

Figure 7: Core temperature (left) and mass-flow rate (right) in the steady state calculation (TRACE-3D/SCF)

		13.31	13.27	13.24		
	13.58	13.56	13.42	13.50	13.53	
13.30	13.56	13.56	13.48	13.51	13.51	13.25
13.27	13.42	13.48	13.54	13.48	13.42	13.27
13.25	13.50	13.51	13.48	13.56	13.56	13.30
	13.53	13.50	13.42	13.56	13.58	
		13.24	13.27	13.31		

		12.24	12.44	12.13		
	13.95	14.89	13.00	14.86	13.95	
12.13	14.83	14.16	13.14	14.14	14.84	12.21
12.43	12.98	13.13	14.50	13.13	12.98	12.43
12.21	14.84	14.14	13.14	14.16	14.83	12.13
	13.95	14.86	13.00	14.89	13.95	
		12.13	12.44	12.24		

Figure 8: Radial distribution of the core mass-flow rate (TRACE-3D (left) and TRACE-3D/SCF (right))

		587.02	587.58	586.77		
	592.18	595.29	589.12	595.20	592.16	
586.78	595.22	592.84	589.52	592.82	595.23	587.01
587.58	589.12	589.52	593.98	589.52	589.12	587.59
587.01	595.23	592.82	589.51	592.84	595.22	586.79
	592.16	595.20	589.12	595.29	592.19	
		586.77	587.58	587.03		

		588.71	589.26	588.49		
	593.30	596.22	590.75	596.18	593.33	
588.17	595.96	593.91	591.08	593.91	596.04	588.43
588.95	590.55	591.02	594.97	591.02	590.55	588.95
588.43	596.04	593.91	591.08	593.91	595.96	588.17
	593.33	596.18	590.75	596.22	593.30	
		588.49	589.26	588.71		

Figure 9: Radial distribution of the core outlet temperature (TRACE-3D (left) and TRACE-3D/SCF (right))

6. Boron dilution transient calculation with constant power

Regarding the boron dilution transient, the acceptance criterion is based on the fact that the available time for the operator to perform the human action required to terminate the boron dilution transient should be greater than 15 minutes under normal operation conditions (NRC, 2007). It means that the time to lose the shutdown margin (SDM) should be greater than 15 minutes. According to the results of the analysis performed in the DCA report of NuScale, that time is 30.5 minutes (NuScale Company LLC, 2020g). It can be highlighted that the analysis of the time to loss of the SDM can be performed by analytical methods or utilizing numerical simulations introducing boron dilution rate in the CVCS interface with the RCS (through the CVCS make-up flow rate) and maintaining the power as constant so that the boron concentration evolution within the RCS can be evaluated up to reaching the value corresponding to the loss of SDM.

In the DCA report of NuScale, (NuScale Company LLC, 2020g), the time to lose the shutdown margin analysis under a boron dilution scenario is evaluated employing two simplified models: the Perfect mixing and the Boron Dilution Slug models:

- The perfect mixing model assumes that the unborated water injected in the RCS by the CVCS is mixed instantaneously with the whole RCS volume in a homogenous way resulting in a continuous and decreasingly monotonic variation in the Boron concentration of the RCS. The formula used can be found below:

$$\frac{dC}{dt} = \frac{Q_{in}(kg/s)}{M_{RCS}} \cdot C(t) \quad (1)$$

where,

C = boron concentration [ppm]

Q_{in} = CVCS makeup mass-flow rate [kg/s]

M_{RCS} = Current RCS mass without considering the PRZ

Integrating (1):

$$C(t) = C(0) \cdot e^{-\frac{Q_{in}(kg)}{M_{RCS}}t} \quad (2)$$

- On the other hand, the boron dilution front or slug flow model is based on the movement of the dilution front along the RCS. Therefore, the Boron concentration variation within the core is a function of the transit time, the number of times that the dilution front had passed through the core and the dilution front Boron concentration itself. In addition, the slug flow model is a discrete one and the formulas used are presented below:

$$C(N) = C(0) \cdot \left[\frac{W_{NC}}{W_D + W_{NC}}\right]^N \quad (3)$$

$$t = \frac{M_{RCSI} + (N-1) \cdot M_{RCS}}{W_D + W_{NC}} \quad (4)$$

where,

C(N) = the N-the front boron concentration passing through the core region [ppm]

C(0) = initial boron concentration (1600 ppm)

t = time [s]

N = number of times that the boron dilution front has passed through the core region [-]

W_{NC} = RCS mass-flow rate (535.24 kg/s)

W_D = CVCS mass-flow rate (3.155 kg/s)

M_{RCSI} = mass between the CVCS injection point and the core inlet level [kg]

M_{RCS} = total RCS inventory without considering the PZR volume [kg]

In this study, a comparison between the results computed using the simplified analytical models presented in (NuScale Company LLC, 2020g) explained above and the results obtained from the simulations with the thermal hydraulic models presented in section 5 is made. To do so, it is important to remember that only the transport of the boron concentration through the RCS should be computed in the simulations without any change in the reactor power or plant conditions since the Figure of Merit of the analysis is the time to reach a specific core boron concentration. By doing so, it will be possible to compare the time, at which the boron concentration causing the loss of the SDM is reached, obtained in the simulations with the results presented in the DCA of NuScale.

As can be seen in Table II, the values of the RCS inventory computed in the steady-state calculation using the three modeling tools are very similar. In that sense, the TRACE-1D model data will be used to obtain the time to lose the SDM using the explained analytical methods. To do so, it should be noted that according to the results presented in the DCA of NuScale, the time to lose the SDM is 30.5 minutes (NuScale Company LLC, 2020g). Therefore, introducing that value in (2) and considering that the value of the mass between the CVCS injection and the core inlet calculated using the TRACE-1D model is 33566.52 kg along with the parameters presented in section 5 and Table II, a value of 1384.99 ppm is obtained for the boron concentration causing the loss of the SDM the TRACE-1D model. A summarizing table with the results obtained using the commented analytical methods and the simulations using the thermal hydraulic models described is included, see Table III.

Finally, it should be noted that a very good agreement has been achieved among the results obtained using the different cited methods and the reference value (30.5 min) for the time to lose the SDM. Therefore, it can be concluded that the obtained results are within a reasonable range in comparison to the reference value and that the developed thermal hydraulic models of the RCS for the different cases are simulating properly the physics of the problem, see Figure 10, despite the appearance of numerical diffusion at the very endings of the simulations, see Figure 11.

Last but not least, it has been also demonstrated that the acceptance criterion associated with the time to loss the SDM is met no matter which calculation approach is applied and that the analytical methods provide more conservative values to this analysis than the best estimate models, see Table III, as it was expected.

Table III: Comparison of the time to loss of the SDM computed by the analytical methods (Perfect mixing and boron dilution slug models) and the numerical simulations.

Plant parameter	Time to lose the SDM [min]
Perfect mixing model	30.50
Boron dilution front or slug model	30.76
TRACE-1D	30.96
TRACE-3D	31.08
TRACE-3D/SCF	30.98

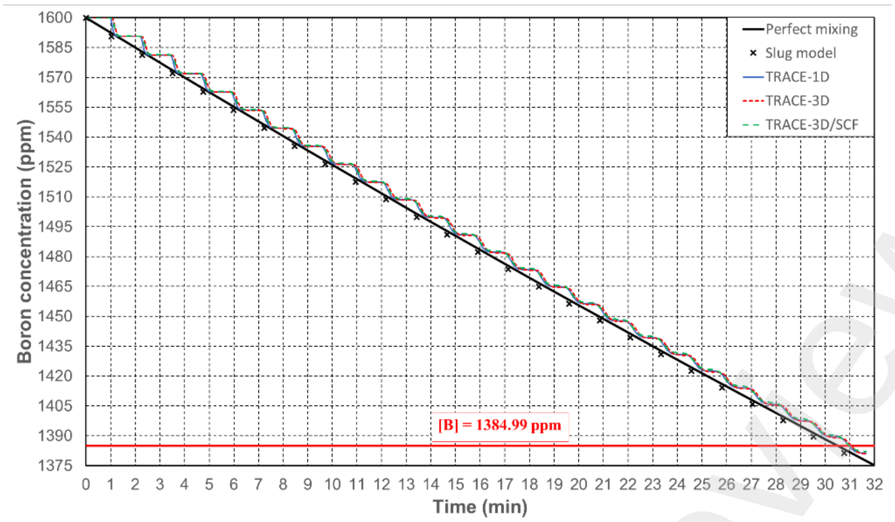

Figure 10: Boron concentration evolution in the constant power analysis

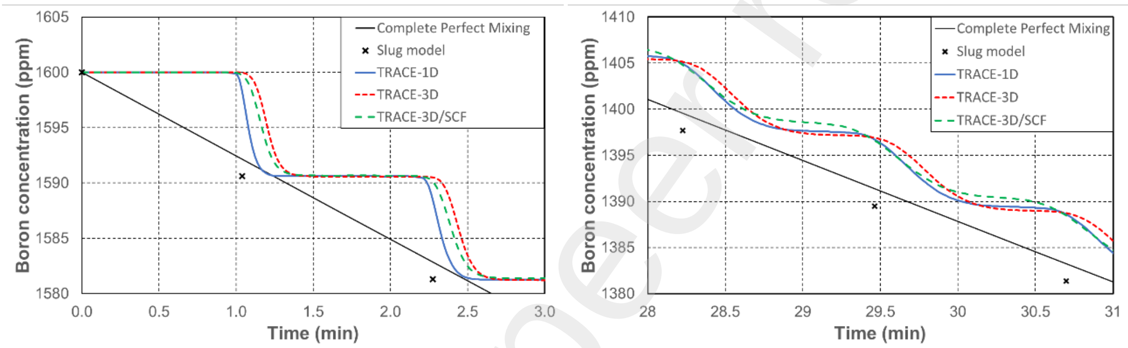

Figure 11: Boron concentration detail at the beginning (left) and the ending (right) of the transient

7. Boron dilution transient analysis with point kinetics

In this section, the plant response to the boron dilution sequence is simulated and the results obtained using the modeling approaches presented previously are analyzed. It is must be noted that the core power evolution predicted by the Point kinetics model of TRACE-3D is transferred to the SCF model so that the SCRAM actuation can be considered in the boron dilution simulations. Therefore, in the TRACE-3D/SCF case, the boron dilution front transport is computed but without any boron reactivity feedback since the core power is input through a time-dependent table. Now, a brief description of the events occurring during the simulation of the accidental sequence is included below:

The injection of unborated water starts at the transient begin due to the CVCS malfunction. Firstly, the boron dilution front is formed immediately after the CVCS injection point and begins to travel following the RCS flow path up to reaching the core inlet region, see Figure 12. As can be seen in Figure 13 and Figure 14, when the boron dilution front passes through the core region, the reduction in the boron concentration causes a positive insertion of reactivity and, consequently, an important increase in the core power. For that reason, the pressure in the primary circuit begins to rise and the 'High PZR Pressure' setpoint is eventually reached, see Figure 15, and the associated SCRAM signal is consequently triggered. After a delay of 2 seconds, the control rod insertion starts and the reactor emergency shutdown takes place and the turbine is also tripped, as it is depicted in Figure 14 by a very sharp reduction of the core power. The commented SCRAM signal activation also demands the CVCS isolation so that the injection of unborated water can be stopped, the NPM isolation from the secondary side by the closure of the MSIVs and FWIVs, and the DHRS actuation.

On one hand and as NuScale relies on natural circulation, the reactor trip causes a mass-flow rate transient with a strong reduction of the core and RCS mass-flow rate in which very low values can be reached, see Figure 16. At this point, it should be noted the mass-flow rate presents certain oscillations rapidly dumped in a similar behaviour to the one described in the manometer problem by other authors such as in (Hoon

Kim et al., 2014). Additionally, the reactor trip along with the DHRS actuation allows to control and eventually to reduce the RCS pressure, see Figure 15. Finally, it should be noted that the PZR level follows the same trend as the RCS pressure, as can be seen in Figure 17.

On the other hand, the turbine trip and the rapid secondary side isolation, achieved 5 seconds after the reactor trip, the secondary pressure sharply increases, see Figure 18. However, the opening of the DHRS actuation valves creates a new flow path establishing an approximately steady mass-flow rate driven by natural circulation through the DHRS piping in the long term, as shown in Figure 19. In that sense, the secondary pressure and temperature begin to decrease due to the DHRS performance, see Figure 18 and Figure 20, and the reactor pool becomes the new ultimate heat sink of the system. Finally, the temperature, the mass-flow rate and the boron concentration at the core inlet region are depicted in Figure 21 and Figure 22, respectively, to demonstrate that no relevant asymmetrical consequences appear in the transient simulation results.

Finally, to put all the above in a nutshell, a summarizing table of the computed sequence of events during the boron dilution calculations using the previously described modeling approaches, see section 5, is included below:

Table IV: Sequence of events in the simulation of the boron dilution transient

Event	TRACE-1D	TRACE-3D	TRACE-3D/SCF
Boron Dilution begins	0	0	0
SCRAM set point	96.0	106.1	108.2
SCRAM (signal + 2 s delay): *Control rods begin to move* **Secondary side isolation:** *MSIVs/FWIVs begin to close* **DHRS actuation:** *DHRS actuation valves begin to open*	98.0	108.1	110.2
Max. pressure peak in primary side	101.7	111.4	110.0
MSIVs and FWIVs fully closed	103.0	113.1	115.2
CVCS isolation valves fully closed	105.0	115.1	117.2
Max. pressure peak in secondary side	122.0	131.7	132.5
DHRS valves fully open	128.0	138.1	140.2

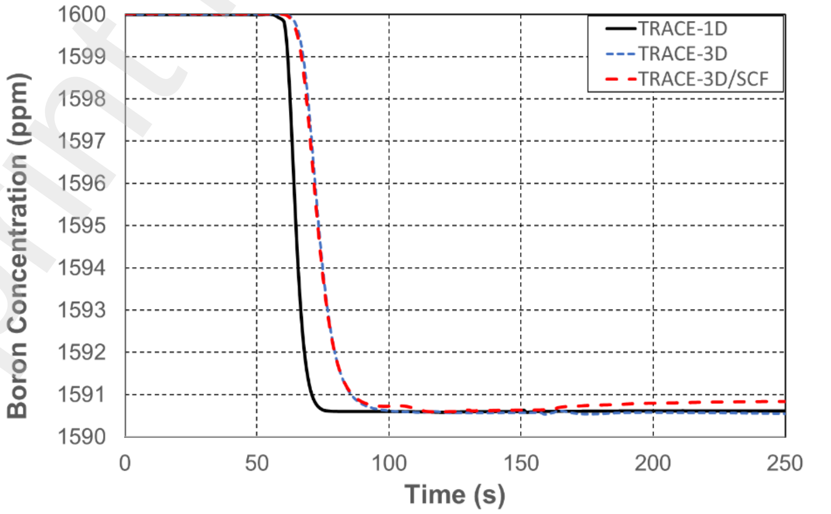

Figure 12: Averaged core inlet boron concentration

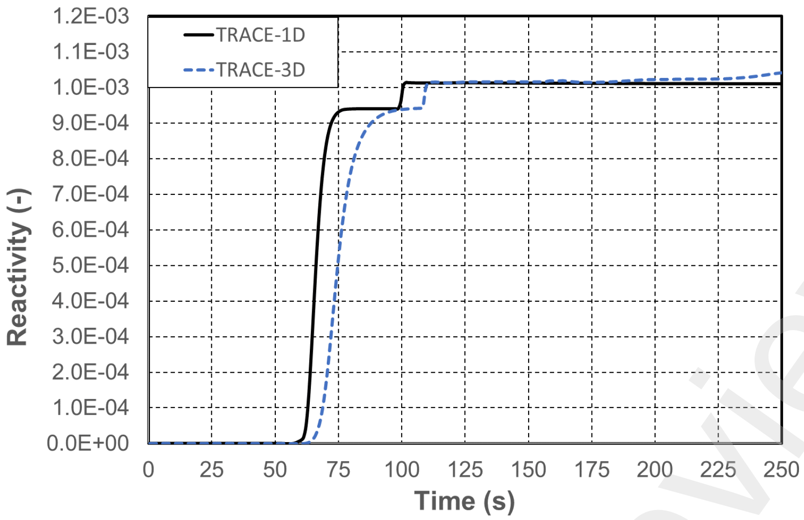

Figure 13: Boron reactivity

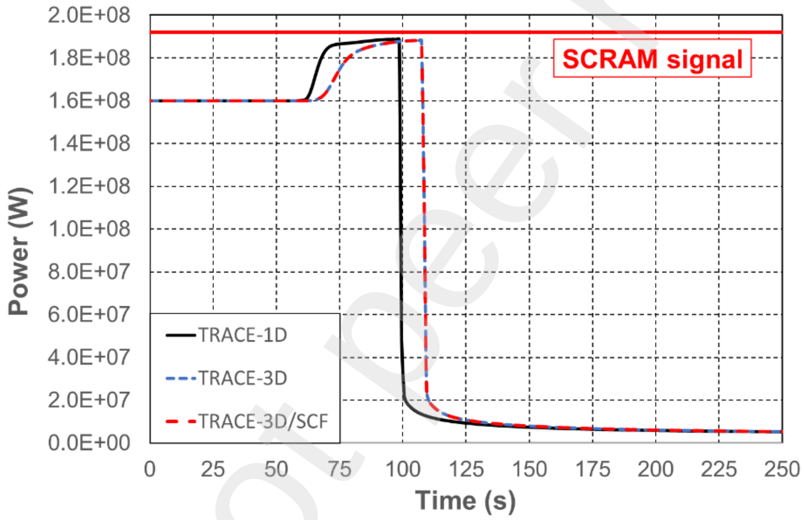

Figure 14: Core power evolution

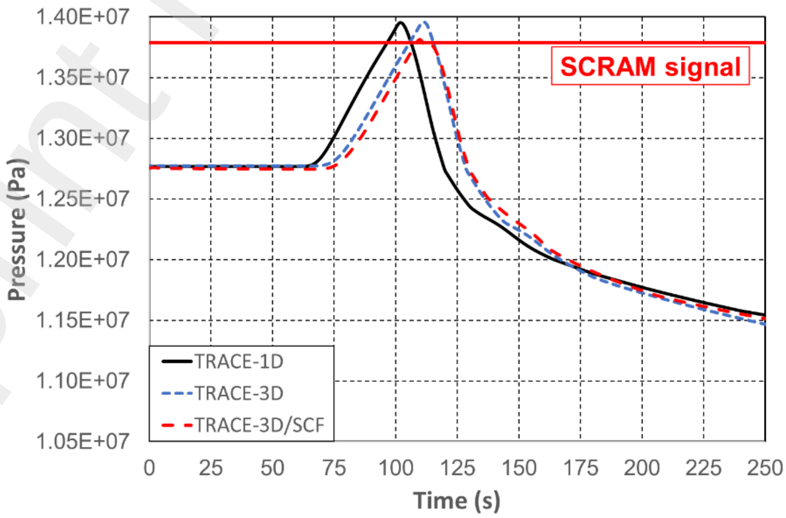

Figure 15: Primary pressure evolution

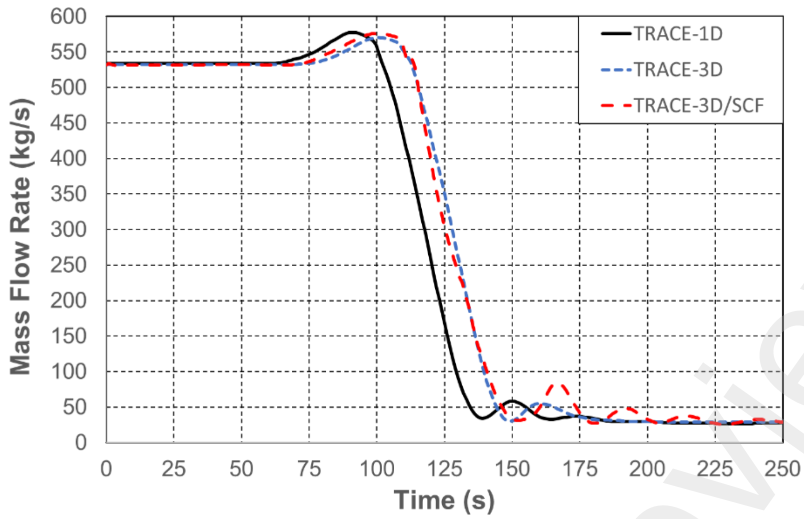

Figure 16: RCS mass-flow rate

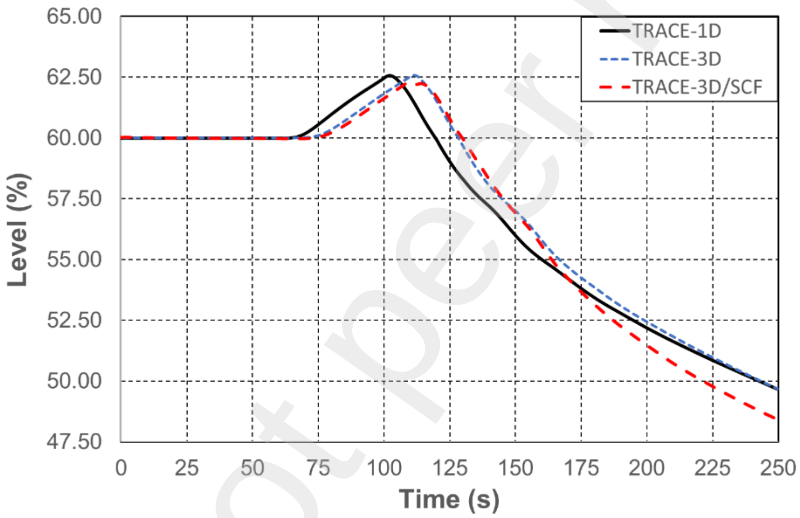

Figure 17: PZR level

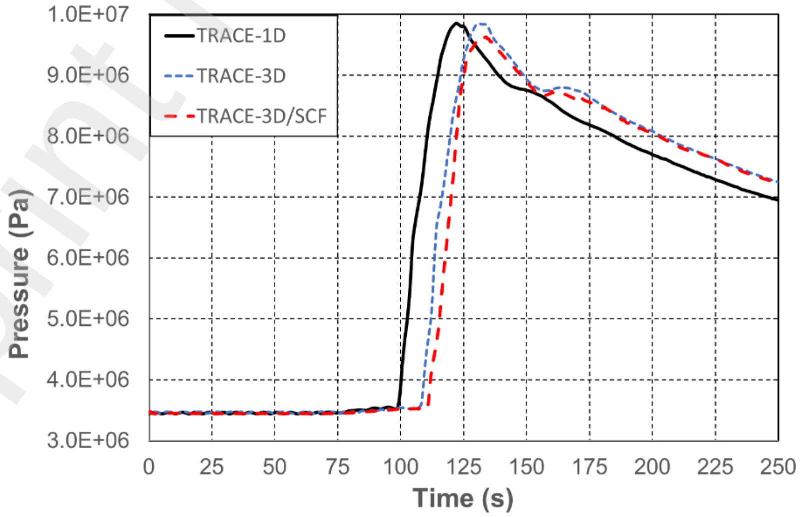

Figure 18: Secondary pressure

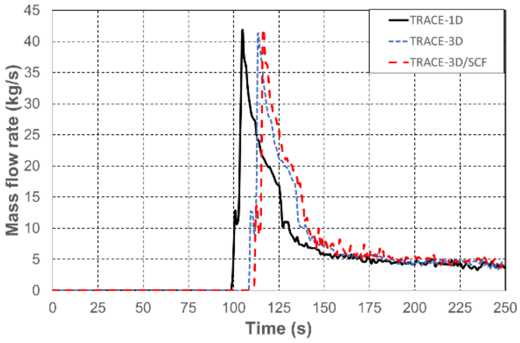

Figure 19: DHRS1(left) and DHRS2(right) mass-flow rate

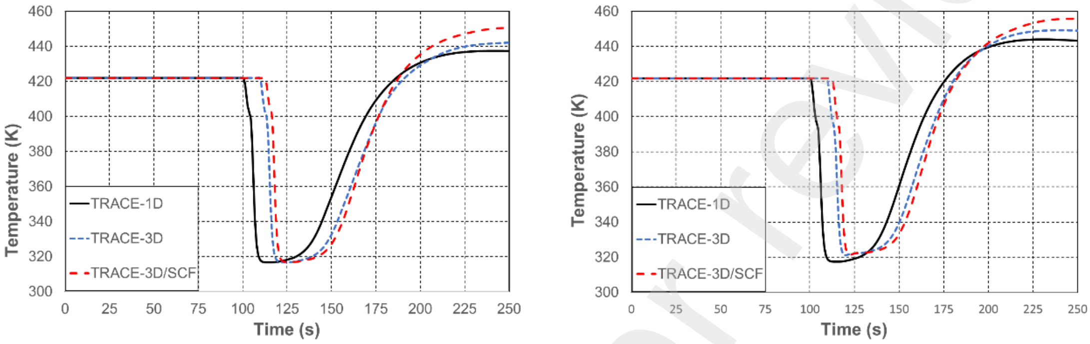

Figure 20: SG1(left) and SG2(right) inlet temperature

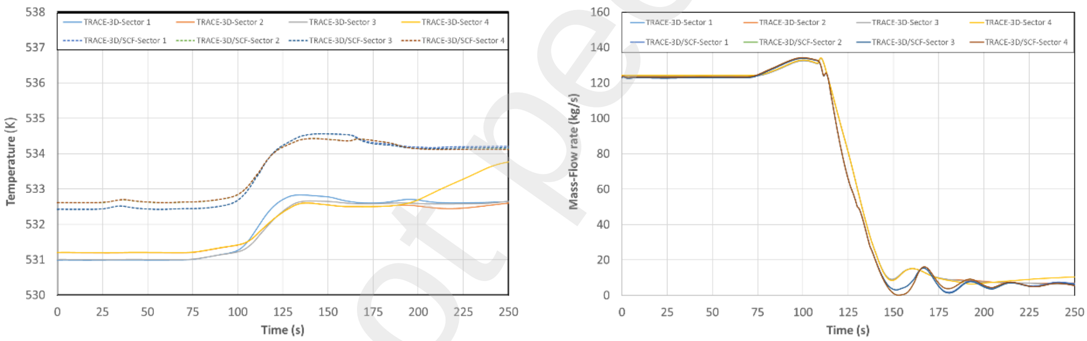

Figure 21: Radial distribution of the temperature (left) and mass-flow rate (right) at the core inlet level during the transient

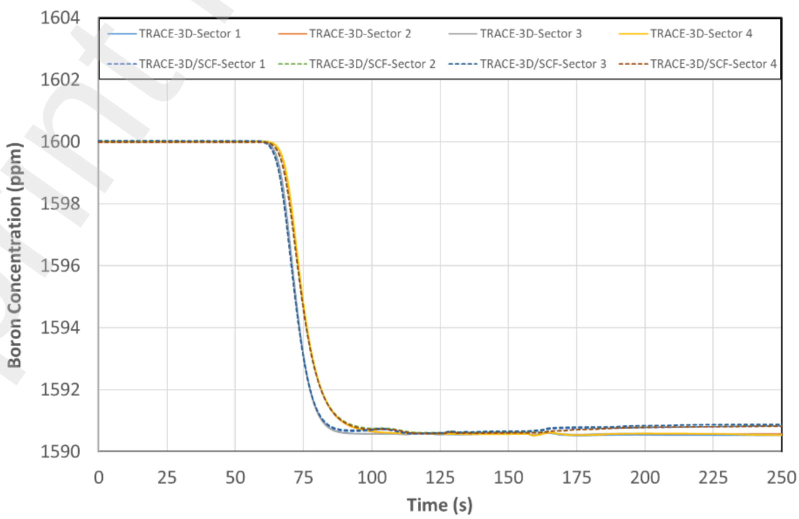

Figure 22: Radial distribution of the core inlet boron concentration during the transient

8. Conclusions

The main conclusions drawn from these investigations are the following:

- The boron dilution sequence has been successfully simulated using different modeling approaches. The usage of high-order numerical techniques for solving the spatial differences allows the physics involved in the transport calculation of the boron concentration through the NuScale RCS to be captured properly.
- In this study, a very good agreement is obtained among the results of the boron dilution sequence in the NPM using TRACE-1D, TRACE-3D and the TRACE-3D/SCF models.
- It has been demonstrated that the acceptance criterion based on the time to lose the SDM has been verified and following the results of this study, the results of the analytical methods used in the DCA report of NuScale are conservative. Even further margin to the 15 minutes for the time to lose the SDM imposed by the acceptance criterion can be obtained with the application of the best estimate modeling approaches presented in this study.
- It has been demonstrated that the TRACE/SCF coupling tool developed at KIT can simulate the boron dilution and the exchange of data between both codes is performed properly.
- Considering the results of both one- and three-dimensional thermal hydraulic analysis of the NuScale behavior under boron dilution accident conditions, it can be stated that the applied tools have predicted only very small differences between the mass-flow and the temperature at the core inlet region, although an asymmetrical mass-flow rate in the feedwater lines was assumed. In addition, the design of the helically coiled heat exchanger in NuScale greatly prevents the development of asymmetrical thermal hydraulic conditions inside the RPV. .
- Finally, the advantages of the use of three-dimensional thermal hydraulic codes to analyze the SMR-plant behavior under accident conditions compared to one-dimensional tools will become more apparent for other SMR-designs where non-symmetical transients cannot be prevented as it is the case in the NuScale design.

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