TRACE System Code Benchmark of Station Blackout of NuScale-like Reactor and Comparison with NuScale Simulator

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ABSTRACT

Keywords: NuScale Power Module (NPM), Station-blackout (SBO), TRACE

1. INTRODUCTION

Advanced nuclear reactor designs around the world are currently being developed to address the requirement for carbon-free energy. Of these designs, Small Modular Reactors (SMRs) are gaining traction as they offer some supposed advantages compared to the traditional commercial light water reactors (LWRs) including more passive safety features, better scalability, and applications for flexible energy generation through hydrogen generation, industrial process heat, and more [1]. In a recent world climate action summit of the 28th Conference of Parties (COP28), more than 20 countries have launched a declaration to triple nuclear energy capacity by 2050 recognizing the key role nuclear energy plays in achieving carbon neutrality [2]. While traditional large-scale LWR will continue to help in achieving this goal, the deployment of advanced nuclear technologies will be vital in contributing efforts to decarbonize. A leading vendor for advanced reactor designs, NuScale Power, made an important milestone with regards to advanced reactor designs being the first company to have their SMR design, the NuScale Power Module (NPM), licensed by the U.S. Nuclear Regulatory Commission (NRC) in 2023. Plans to build a NuScale reactor were made between the Utah Associated Municipal Power Systems (UAMPS) with commercial operation expected in 2026, but the project was recently cancelled in November of 2023 as project delays and costs raised concerns [3]. Although the cancellation of the NuScale project poses some setbacks, NuScale has paved the way for advanced reactor design and licensing and is continuing efforts to bring their technology to market in other countries namely Romania, Canada and Jordan [4].

System thermal hydraulic codes have been the primary tool of modeling and simulation of reactor designs since their conception and implementation. Much of the development work, including validation studies, have been done based on existing reactor designs for pressurized water reactors (PWRs) and boiling water reactors (BWRs). Advanced reactor technologies have significant design changes and phenomenological differences when compared to these existing technologies in which system codes were designed to model. For example, many designs have incorporated passive safety systems which operate on physical principles of natural circulation for heat removable capabilities such as the decay heat removal system (DHRS) in the NPM. The use of these passive safety systems seems compelling as they don't require any active systems such as pumps or operator intervention to operate, but some researchers have raised concerns on the reliability of passive safety systems [5]. Different steam generator or heat exchanger designs are also being considered for use in advanced reactors. In the case of the NPM, a helical coil steam generator (HCSG) is implemented in the system to extract heat produced from the reactor. Opposite to existing u-tube steam generator designs, the primary fluid passes over the helical tube bundles while the feedwater passes through the internal tubes to generate steam. Some experimental facilities have been built to assess the performance of novel HCSG designs including the Modular Test Loop (MOTEL) at LUT and the SIET helical coil steam generator tube facility [6,7]. Much like the passive safety systems designs, more research is needed to fully understand the phenomena induced by the HCSG design as vibrational effects and instabilities may still raise concerns [8,9]. From the perspective of system codes, more effort is clearly needed to assess and validate these novel advanced reactor technologies as the phenomenology largely differs from the typical light-water reactor designs where the original application and validation of these system codes are derived. As an example, some studies involving system code validation for natural circulation systems have shown that nodalization of the primary components may have large impacts on the accuracy of the model in predicting the primary variables of interest such as pressure, flow rates, and temperatures [10,11]. System codes will continue to evolve their modeling capabilities as additional experimental data and subsequent correlations become available, but caution should be taken when considering the application of these codes in this capacity.

NuScale has developed a full plant simulator with GSE solutions which is a representation of the control room inside of a NuScale plant. The NuScale plant simulator uses a modified version of the RELAP system thermal hydraulic code to model plant dynamics and can simulate many events from normal operating procedures to accident scenarios such as a Station Blackout (SBO). In 2021, a partnership between NuScale, Texas A&M, and the U.S. Department of Energy (DOE), a NuScale plant simulator was installed under the Center for Advanced Small Modular and Micro Reactors (CASMR) at Texas A&M. The simulator serves both as an educational tool for students and for researchers to understand and analyze how an advanced reactor responds to different operational or accident scenarios.

In an effort to perform a cross-code benchmark study, TRACE models of the NuScale Power Module (NPM) are currently being developed at Texas A&M, based on a previous model developed by Universidad Politécnica de Madrid (UPM). The NPM model is currently being developed using the TRACE thermal hydraulic code developed by the U.S. Nuclear Regulatory Commision (NRC) [#]. In general, benchmark studies have shown that discrepancies in results can occur not only from a code-to-code basis but are also dependent on the modeling approach taken by code users. Examples of this are found in the benchmark studies such as the IEA-R1 research reactor [##] where the benchmark study showed discrepancies between system code results even when using the same computational code to perform the transient analysis. For this reason, important contributions can be made in defining modeling approaches and assessing code predictions within the scope of modeling advanced reactors with complex phenomena such as the NPM. Similarly, there is some merit in developing additional code models of the NPM to assess the overall performance in steady and transient conditions to quantify and assess differences between the expected system behavior and whether these discrepancies between the code predictions can be attributed to code methodology or modelling approaches. The purpose of this study is to therefore perform a code-to-code benchmark study of the NPM by developing system code models using design data publicly available in the NuScale Design Certification Application (DCA) and observe differences in plant characteristics for steady and transient conditions compared to the NuScale plant simulator. The observed similarities or differences between the NuScale simulator results and the TRACE models may provide a deeper insight into the TRACE code performance and modeling techniques required in simulations of advanced reactors. Additionally, the results of this study may also provide information useful in the design considerations of the NPM.

2. METHODOLOGY

2.1. Description of the NuScale Power Module (NPM)

A typical NuScale plant is designed to house 12 NPM units operating at 160 MW_{th} with a total output of around 600 MW_e for the whole plant. The NPM is an integral type of PWR where the primary side flow is driven through natural circulation of the primary coolant as opposed to traditional commercial reactors which rely on active systems, such as pumps, to force flow of the primary coolant. Each NPM is equipped with two secondary side loops which pass to their respective steam generators which are rated for a load of 80 MW each, equivalent to half of the total reactor power. The NPM features a unique steam generator design known as a helical coil steam generator (HCSG) where tube bundles are helically wound along the upper region of the downcomer to increase heat transfer area and conserve space within the primary system. Two separate decay heat removal systems (DHRS) are also connected to the steam line and feedwater piping to cool the reactor under accident conditions but are not in operation under normal conditions. Another unique feature of the NPM is that their containment is a steel vessel which surrounds the entire reactor pressure vessel and then that containment vessel is submerged in a large reactor pool which acts as the ultimate heat sink for all 12 units. In the event of an accident, the containment serves as its own natural circulation system for the primary coolant to cool the core by venting steam generated through valves at the top of the pressurizer where heat is then rejected to the reactor pool to condense the steam and recirculated back to the primary system through valves located in the downcomer region. Doing this ensures that the primary coolant inventory is never lost during an accident and the water level will always remain above the active fuel region, assuming the reactor pool is able to remove the necessary amount of decay heat. Figure ## below shows a schematic of the NPM.

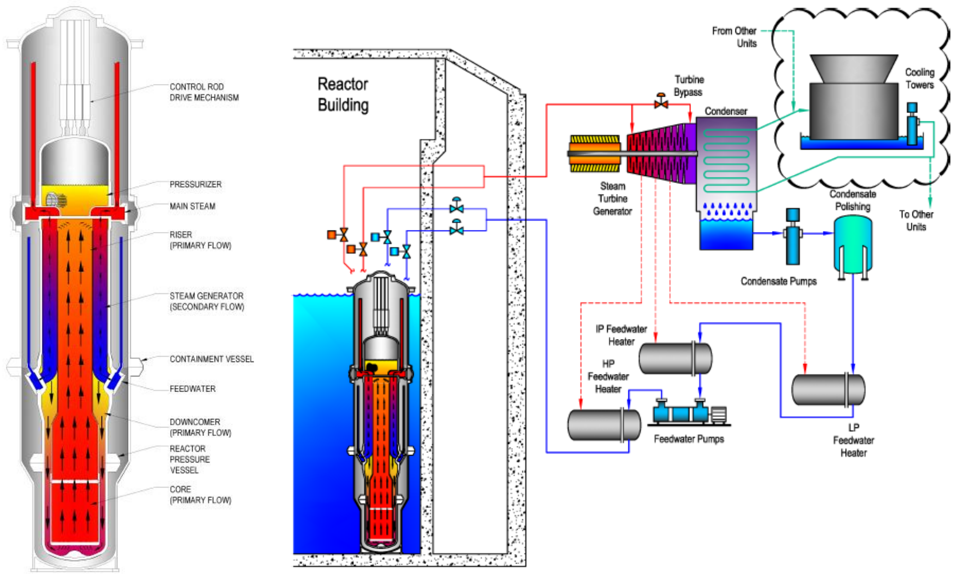

Fig. ###. Schematic of a NuScale Power Module [#]

NuScale is currently seeking to uprate the power in their NPM to 77 MW_e (250 MW_{th}) which would equate to a total energy production of 924 MW_e for a 12 unit plant [#]. The additional uprate in power would improve the economics of the NuScale plant, but considerations should be taken into account when considering the additional heat removal required from the passive safety systems. NuScale has also announced a collaboration with Lightbridge Corporation for use of their advanced fuel design in their NPM [#]. A grant was awarded through the Nuclear Energy University Program (NEUP) for Texas A&M and the Lightbridge Corporation to fund experimental programs to assess the capabilities of the Lightbridge fuel design [#]. The Lightbridge fuel design features metallic fuel which will allow the NPM to operate at higher burnup while also having desirable thermal qualities similar to accident tolerant fuels (ATF) to make for safer reactor operation. For the purpose of this study, only the design characteristics for their licensed 160 MW_{th} reactor are considered.

2.2. Description of TRACE Model

As mentioned previously, TRACE is a system-level thermal hydraulic code developed by the U.S. NRC. TRACE solves the standard two-fluid, two-phase field equations of mass, momentum, and energy in a finite volume approach with additional closure correlations to describe additional phenomena. The TRACE models were developed through the Symbolic Nuclear Analysis Package (SNAP) [##] suite using TRACE version v5.0P7 [##]. Governing equations solved by the TRACE system code are shown below in Equations 1 through 6.

$$\frac{\partial[(1-\alpha)\rho_l]}{\partial t} + \nabla \cdot \left[(1-\alpha)\rho_l \vec{V_l}\right] = -\Gamma \tag{1}$$

$$\frac{\partial(\alpha\rho_g)}{\partial t} + \nabla \cdot \left[\alpha\rho_g \vec{V_g}\right] = \Gamma \tag{2}$$

$$\frac{\partial\left[(1-\alpha)\rho_l\vec{V_l}\right]}{\partial t}+\nabla\cdot\left[(1-\alpha)\rho_l\overrightarrow{V_l}\overrightarrow{V_l}\right]+(1-\alpha)\nabla\mathrm{P} = \vec{f_\iota}+\overrightarrow{f_{wl}}+(1-\alpha)\rho_l\vec{g}-\Gamma\vec{V_\iota} \quad (3)$$

$$\frac{\partial\left[\alpha\rho_g\vec{V_g}\right]}{\partial t}+\nabla\cdot\left[\alpha\rho_g\overrightarrow{V_g}\overrightarrow{V_g}\right]+\alpha\nabla\mathrm{P}=\overrightarrow{-f_\iota}+\overrightarrow{f_{wg}}+\alpha\rho_g\vec{g}+\Gamma\vec{V_\iota} \quad (4)$$

$$\frac{\partial\left[(1-\alpha)\rho_l(e_l+V_l^2/2)\right]}{\partial t}+\nabla\cdot\left[(1-\alpha)\rho_l\left(e_l+\frac{P}{\rho_l}+\frac{V_l^2}{2}\right)\overrightarrow{V_l}\right] = q_{il}+q_{wl}+q_{wsat}+q_{dl}+(1-\alpha)\rho_l\vec{g}\cdot\overrightarrow{V_l}-\Gamma h_l'+\left(\vec{f_\iota}+\overrightarrow{f_{wl}}\right)\cdot\overrightarrow{V_l} \quad (5)$$

$$\frac{\partial\left[\alpha\rho_g(e_g+V_g^2/2)\right]}{\partial t}+\nabla\cdot\left[\alpha\rho_g\left(e_g+\frac{P}{\rho_g}+\frac{V_g^2}{2}\right)\overrightarrow{V_g}\right] = q_{ig}+q_{wg}+q_{dg}+\alpha\rho_g\vec{g}\cdot\overrightarrow{V_g}+\Gamma h_v'+\left(\overrightarrow{-f_\iota}+\overrightarrow{f_{wg}}\right)\cdot\overrightarrow{V_g} \quad (6)$$

Where….

Two models in TRACE were developed to model the NPM system where the models are differentiated through modeling of the primary system components as 1D or 3D components, respectively. Components within the models are developed based on data found in the publicly available literature; mainly in NuScale's design certification application (DCA). Any additional parameters which were not included in the DCA are estimated or derived based on engineering judgement. All component geometries and parameters were held consistent for both the 1D and 3D models where applicable. Figures ## and ## below show the nodalization diagram used for the 1D and 3D TRACE models.

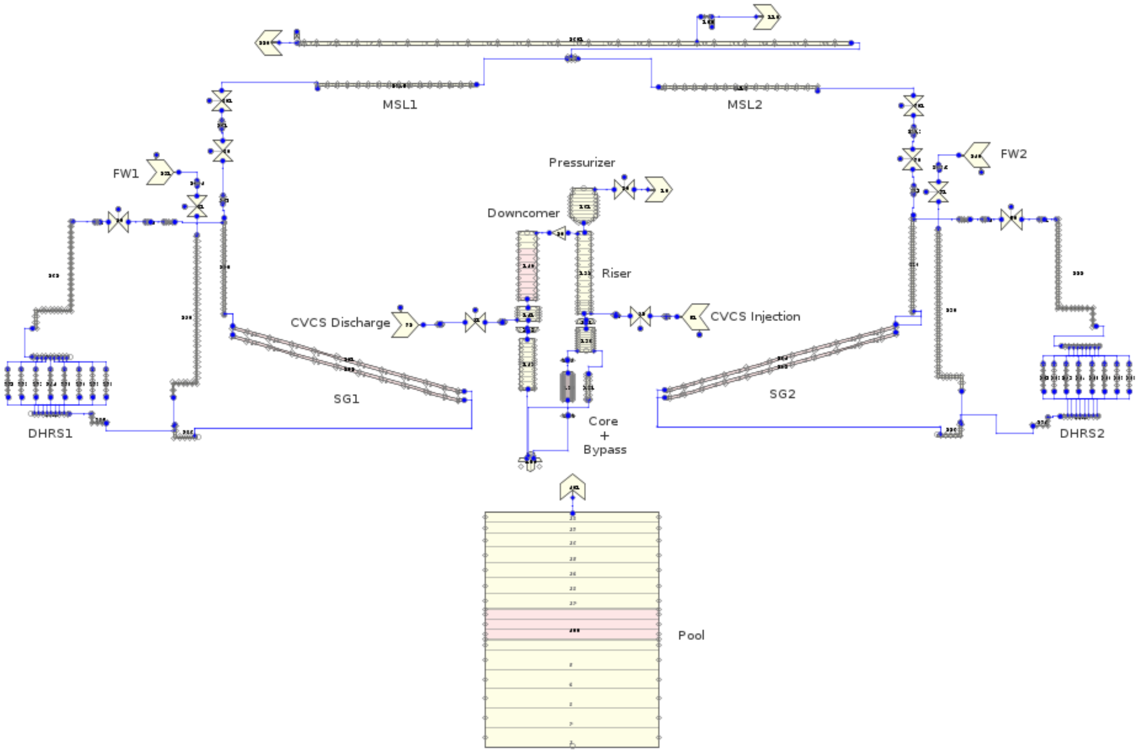

Fig. ##. Nodalization diagram of the 1D TRACE model.

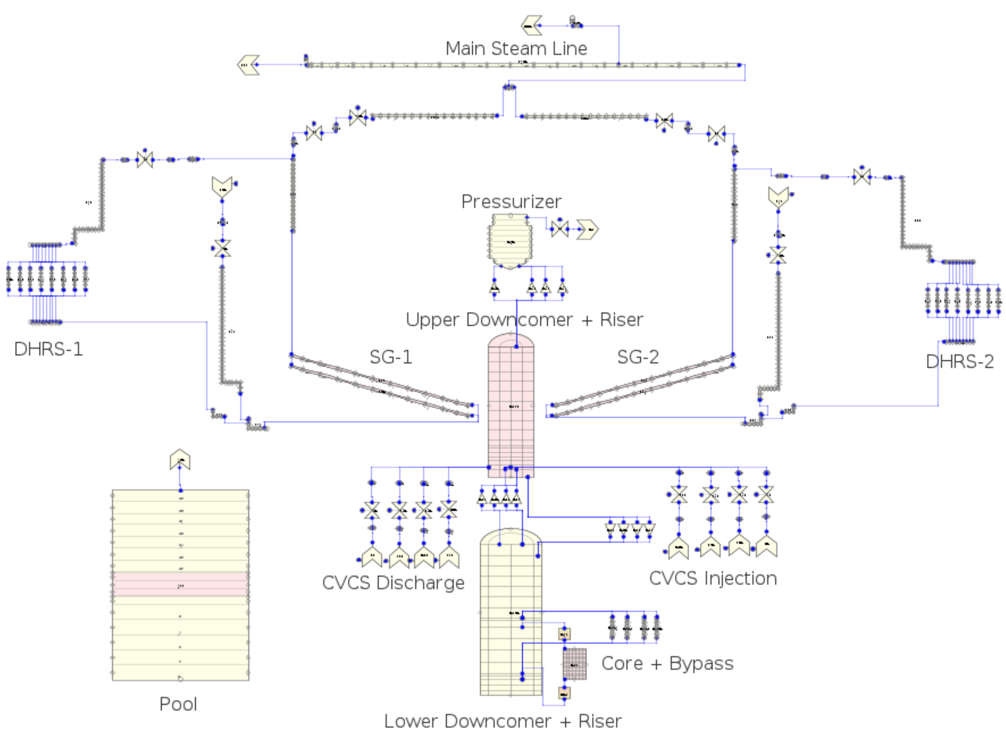

Fig. ###. Nodalization diagram of the 3D TRACE model.

As shown in Fig. ### above, the 1D TRACE model uses several 1D pipe components to model the primary system. The primary system is divided into 6 main components including the downcomer, lower plenum, core channel, bypass, riser, and pressurizer regions. Chemical Volume and Control System (CVCS) charging and letdown lines are attached in the lower riser and lower downcomer regions, respectively, to regulate the amount of boron within the primary system. The core channel is modeled as a single pipe coupled to a single heat structure in which power is regulated using the implemented point reactor kinetics within the TRACE code. The pressurizer component is modeled with a valve located at the top of the pressurizer to provide a pressure boundary condition for the primary system which, after reaching stable conditions, is closed and the pressure in the primary system is then regulated to the desired value using control logic via heat input to the pressurizer.

The 3D TRACE model uses separate 3D VESSEL components in TRACE to model the primary system as opposed to 1D pipes. Two cylindrical VESSEL components are used to model the main portion of the primary system and are divided into 18 axial cells, 2 radial cells, and 4 azimuthal cells. The inner ring of the VESSEL components is representative of the lower plenum and riser regions, and the outer ring is used to model the downcomer region of the primary system. An additional VESSEL component is also used to model the core and is coupled to the lower VESSEL component through junction connections. The core VESSEL component is nodalized into 20 cells in the axial or z direction, and then 7 cells in both the x and y directions to allow for flexible power profile shapes in both the axial and radial directions.

Modeling of the secondary side is kept consistent between both the 1D and 3D TRACE models. One of the most important parts in the secondary side modelling involves the HCSG as the HCSG design is a fairly new design to be considered for nuclear technologies as opposed to standard u-tube steam generators which have many years of research and design efforts for use in nuclear applications. Many experimental studies have been conducted to develop correlations for pressure drop and heat transfer in both the shell and tube side of the HCSG. [LIST REFERENCES] For this study, built in correlations within TRACE were used to model the HCSG which required some additional user inputs based on the helical geometry. Figure ## below shows an example of the HCSG design used in the NPM.

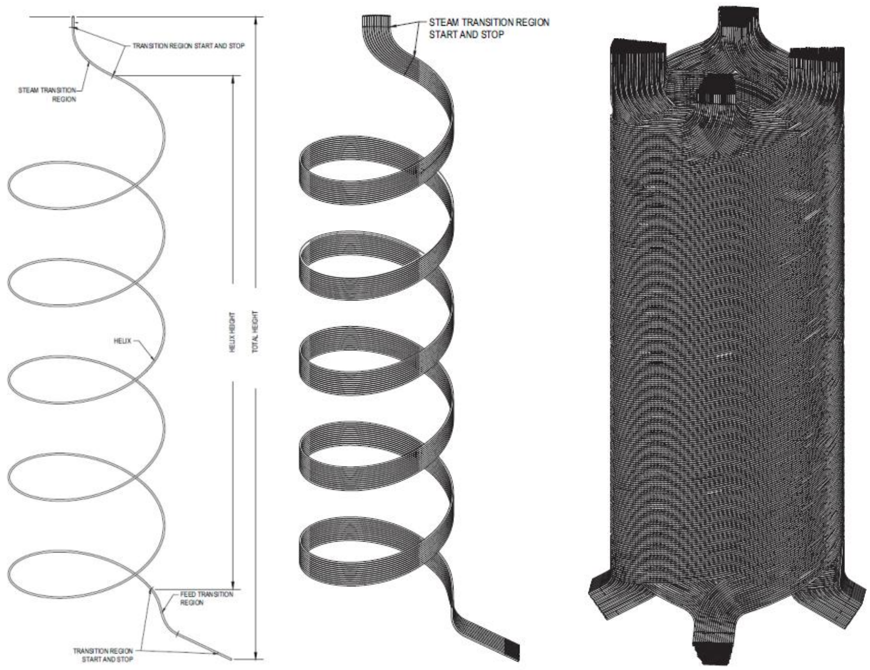

Figure #. Schematic of the helical tube bundle used in the NPM. [#]

In TRACE, heat transfer and pressure drop of the HCSG is handled separately based on either shell or tube side flow. For shell side flow, pressure drop and heat transfer are both governed by the Zukauskas model [#] for inline and staggered tube bundles. The resulting correlations for shell side heat transfer of the tube bundle is shown below.

$$Nu = C \cdot Re^{B} \cdot Pr^{D} \left(\frac{Pr}{\mathrm{Pr}_W}\right)^{0.25}$$

Reynolds Number	Inline			Staggered		
	B	C	D	B	C	D
$1 \leq Re < 100$	0.4	0.9	0.36	0.4	1.04	0.36
$100 \leq Re < 1000$	0.5	0.52	0.36	0.5	0.71	0.36
$1000 \leq Re < 2 \cdot 10^5$	0.63	0.27	0.36	0.6	$0.35 \cdot \left(\frac{\left(\frac{P}{D}\right)_T}{\left(\frac{P}{D}\right)_L}\right)^{0.2}$	0.36
$2 \cdot 10^5 \leq Re < 2 \cdot 10^6$	0.8	0.033	0.4	0.8	$0.031 \cdot \left(\frac{\left(\frac{P}{D}\right)_T}{\left(\frac{P}{D}\right)_L}\right)^{0.2}$	0.36

Where Nu is the Nusselt number, Re is the Reynolds number, Pr is the fluid Prandtl number, Pr_W is the Prandtl number of the associated tube wall, and B, C, and D are associated coefficients based on tube bundle geometry and Reynolds number as shown in Table #. The correlation for the pressure drop across the shell side of the tube bundle used in TRACE is shown below in Equation ##.

$$Eu = \frac{2 \cdot \Delta p}{\rho \cdot V^2}$$

Where Eu is the Euler number, Δp is the pressure drop, ρ is the fluid density, and V is the fluid velocity at the minimum flow area of the tube bank. In this way, tabular functions based on tube bundle geometry and Reynolds number can be used to evaluate the effective pressure drop across the tube bundle. Correlations for tube side heat transfer phenomena implemented into the TRACE code are based on correlations developed for laminar and turbulent flow inside helical tubes. Heat transfer correlations for fully developed laminar flow inside helical tubes developed by Manlapaz and Churchill [#] implemented in TRACE are shown below in Equations # through #.

$$Nu = \left[\left(4.364 + \frac{4.636}{x_3}\right)^3 + 1.816 \cdot \left(\frac{De}{x_4}\right)^{\frac{3}{4}}\right]^{1/3}$$

$$De = Re \cdot \left(\frac{d}{2R}\right)^{1/2}$$

$$x_3 = \left(1 + \frac{1342}{De^2 \cdot Pr}\right)^2$$

$$x_4 = 1 + \frac{1.15}{Pr}$$

Where De is the Dean number, d is the tube diameter, R is the coil radius, Re is Reynolds number, and Pr is Prandtl number. Fully developed turbulent flow heat transfer correlations used in TRACE were developed by Schmidt [#] and Pratt [#] and are shown below in Equations # and #.

$$\frac{Nu_c}{Nu_s} = 1 + 3.6 \cdot \left[1 - \frac{r}{R}\right] \cdot \left(\frac{r}{R}\right)^{0.8} ; 20000 < Re < 150000$$

$$\frac{Nu_c}{Nu_s} = 1 + 3.4 \cdot \frac{r}{R}; 1500 < Re < 20000$$

Where Nu_c is the Nusselt number for the curved duct, Nu_s is the Nusselt number for a straight corresponding straight duct, r is the tube radius, and R is the coil radius. Pressure drop correlations for the internal helical tube flow implemented in the TRACE code were based on correlations developed by Srinivasan et al. [#] and Ito [#]. Corresponding friction factors for fully developed laminar flow inside of a helical pipe are shown below in Equations # through #, and Equations # and # represent the friction factor for fully developed turbulent flow within a helical pipe.

$$\frac{f_c}{f_s} = \begin{cases} 1, & De < 30 \\ 0.419 \cdot De^{0.275}, & 30 < De < 300 \\ 0.1125 \cdot De^{0.5}, & De > 300 \end{cases}$$

$$f_c = \begin{cases} 0.00725 \cdot \left(\frac{R}{r}\right)^{-0.5} + 0.076 \cdot [Re]^{-0.25}, & Re \cdot \left(\frac{R}{r}\right)^{-2} < 300 \\ 0.084 \cdot \left[Re^{-0.2} \cdot \left(\frac{R}{r}\right)^{-0.1}\right], & Re \cdot \left(\frac{R}{r}\right)^{-2} \geq 300 \end{cases}$$

Where f_c is the friction factor of the curved duct, f_s is the friction factor of a corresponding straight duct, and the other variables Re, R, and r are the same as described previously. More information about the correlations used for helical geometries can be found in the TRACE code theory manual [#].

The DHRS is connected to the secondary side through a valve in each of the steam lines and the feedwater lines and coupled to the reactor pool via heat structures. In this way, the DHRS removes decay heat from the primary system through the HCSG and transfers that energy to the reactor pool during the SBO scenario. The reactor pool is modeled as a large pipe where the volume of the pool is adjusted to account for only a single NPM unit as opposed to the NuScale simulator which accounts for the total pool volume and decay heat of all 12 NPM units.

2.3. Station Blackout (SBO) Scenario

A Station Blackout (SBO) scenario is a standard design basis accident scenario required by the U.S. NRC to model when licensing reactor designs [#]. Dangers of an unresolved SBO scenario were apparent in the Fukashima Diachi accident [##] and has gathered more attention in recent years. A SBO event generally assumes that the plant has lost all onsite and offsite AC power to the plant where battery operated DC power is still available. Commercial reactors are equipped with backup diesel generators to power safety systems in the event of a SBO, but if the integrity of diesel generators are lost, much like in the Fukashima Diachi accident, core damage can occur in a matter of hours. Much research has been conducted to analyze and assess reactors coping mechanisms in extended blackout scenarios. The Nuclear Energy Institute (NEI) published a report in 2012 on diverse and flexible coping strategies (FLEX) which recommends the installment of on-site and off-site portable equipment and other strategies to handle SBO scenarios [#]. The Electric Power Research Institute (EPRI) has also worked to develop a set of guidelines known as the Severe Accident Management Guidelines (SAMG) which detail strategies and subsequent actions to take in the event of a severe accident to mitigate core damage and radioactivity release [#]. One of the primary concerns of a SBO event with the loss of backup diesel generators is that the primary system pressure has no means of being lowered with loss of the reactor heat sink.

Because of this, operators are not able to inject water through the high pressure coolant injection (HPCI) lines to cool the reactor. For this reason, the SAMG guidelines highlight depressurization of the reactor coolant system as a high priority even with the loss of inventory that would occur.

One study from KAERI modeled SBO accident progression with failure of diesel generators using the MAAP severe accident code for different reactor designs to assess the phenomena and core damage process over the course of the transient [#]. The authors modeled a typical 1000 MW PWR where they assumed the only operating safety system available were the auxiliary feed water (AFW) turbine driven pumps (TDP) which had a total battery life of 4 hours. After the battery life of the AFW TDPs were lost, the primary pressure of the system increased until reaching the set point of the pressure relief valves. As the cycling of the pressure relief valve continues, the water in the primary system decreases and eventually the core becomes uncovered and quickly rises to the fuel melting temperature where core damage begins.

The passive design of the NPM allows it to handle a SBO event for extended periods while maintaining core integrity. With no active safety systems, the operators do not need to worry about leakage from the reactor coolant pump seals, and any inventory lost from the pressure relief valves will be kept within the containment which is used to provide long-term core cooling through the ECCS vent and recirculation valves. In effect, either the DHRS or the ECCS recirculation within the containment should be able to maintain core cooling by rejecting heat to the reactor pool which is designed to provide adequate cooling of the 12 units for greater than 72 hours. In effect, the design implications of the NPM highlight the system capabilities of passively handling an event which is considered as a severe accident in traditional light water reactors.

As the focus of this study is based on a cross-code benchmark activity, only the first 8000 seconds (~2 hours) of the SBO event was simulated to compare against data from the NuScale simulator. The time frame for the SBO was chosen based on the NPM ability to establish a quasi-steady state around this time frame due to the passive cooling provided by the DHRS as seen in results from the NuScale simulator. During this time, only the DHRS is used to remove heat from the primary system, and for this reason containment effects are not considered in this study. Assumptions of the SBO event are consistent with similar assumptions made in the NuScale DCA. The assumptions are as follows:

- Reactor initially operating at 100% rated thermal power (160 MW).
- At time zero, an SBO occurs as a result of complete loss of onsite and offsite AC power.
- Turbine and Feedwater pumps trip as a result of loss of AC power.
- Power from the highly reliable DC power system (EDSS) is available.
- No credit is taken for manual operator actions.
- No additional single failures occur.

Under these assumptions, the subsequent SBO event tree was evaluated based on data from the 1D and 3D TRACE models and is shown below in Table #.

	TRACE 1D	TRACE 3D
Event	Time (seconds)	Time (seconds)
Loss of AC power	0	0
High Pressure Signal	20	25
Reactor SCRAM	21	26
DHRS Actuation	35	45

Results from the SBO event tree for the 3D TRACE model show a larger time delay in the control logic for high pressure signals and subsequent DHRS actuation, but are around the same relative time scale. After DHRS actuation, no other action is taken, and the primary system pressure and temperatures continue to decrease over the course of the SBO transient. During the early stages of the SBO transient, the power input into the fluid from the reactor decay heat is the dominant driver of the flow in the primary side. For these simulations, the standard ANS-94 curve was implemented to model the decay heat in the NPM. Since the simulations only model the early stages of the transient, the containment was not used although it would be necessary at later stages in the SBO.

3. RESULTS

3.1. Steady-State Results

Initial steady-state simulations of the 1D and 3D models were done to provide a baseline comparison for values reported by the NuScale simulator and values provided in the NuScale DCA. Steady state was achieved in the simulations after reaching converged solutions of the system parameters including temperatures and pressures in the primary and secondary systems after around 3000 seconds of simulations time, and the simulations were run for a total of 5000 seconds before completing. Results of the steady-state data from both models are summarized below in Table ##.

Table ##. Summary of the steady-state NPM results compared to simulator and DCA data.

Parameter	DCA	NuScale Simulator	TRACE-1D	Percent Difference (DCA)	TRACE-3D	Percent Difference (DCA)
System Pressure	12.75 MPa	12.75 MPa	12.75 MPa	N/A	12.75 MPa	N/A
Reactor Power	160 MW_{th}	160 MW_{th}	160 MW_{th}	N/A	160 MW_{th}	N/A
Primary System Flow Rate	587.15 kg/s	594.35 kg/s	584.68 kg/s	-0.42%	587.14 kg/s	-0.002%
Core Outlet Temperature	583.2	582.59	586.3 K	0.53%	580.17 K	-0.52%
Core Inlet Temperature	531.48	532.04	530.5 K	-0.18	529.1 K	-0.45%
Feedwater Flow Rate (per SG)	33.53 kg/s	32.29	32 kg/s	-4.67%	32 kg/s	-4.67%

Steam Generator Inlet Temperature	421.87	422 K	421.87 K	N/A	421.87 K	N/A
Steam Generator Outlet Temperature	580.04	579.8 K	570.9 K	-1.59%	573.5 K	-1.13%
Steam Generator Outlet Pressure	3.45 MPa	3.45 MPa	3.38 MPa	-2.05%	3.46 MPa	0.29%
Pressurizer Level Percentage	N/A	58%	58%	N/A	59%	N/A

As shown in Table ## above, the steady-state results of the1D and 3D models show good agreement between the DCA and simulator data with a maximum percent difference of 4.67% in the feedwater flow rate. The steady-state results between the 1D and 3D TRACE models also show good agreement between each other, as expected. Primary flow rates predicted in the 3D model were slightly higher than the 1D model which also led to lower inlet and outlet core temperatures. It was also found that both models underpredicted the steam superheat in the HCSG but are within an acceptable range compared to the DCA. The primary flow rate given by the NuScale simulator was also found to be larger than expected compared to the DCA. Within the simulator the exact location of the mass flow rate is not given except for stating it is measured in the downcomer, so if the measurement occurs before reaching the CVCS discharge line, the measured flow rate would have some additional flow from the CVCS injection (~3 kg/s). Otherwise, all the data reported in the NuScale simulator shows good agreement with the data from the DCA.

3.2. Station Blackout (SBO) Results

The SBO simulations were run for a total of 8000 seconds after restarting from the initial steady-state simulations, which ran for a total of 5000 seconds. The SBO was initiated at the start of the simulation by tripping the feedwater pumps where shortly after the reactor was scrammed due to the high-pressure signal in the pressurizer. Then, the main steam line valves closed while simultaneously opening the valves to the DHRS where the system was then self-regulated for the remainder of the transient. Results of the system characteristics over the course of the transient are shown below in Fig. ###.

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FREQUENCY ANALYSIS OF 1D

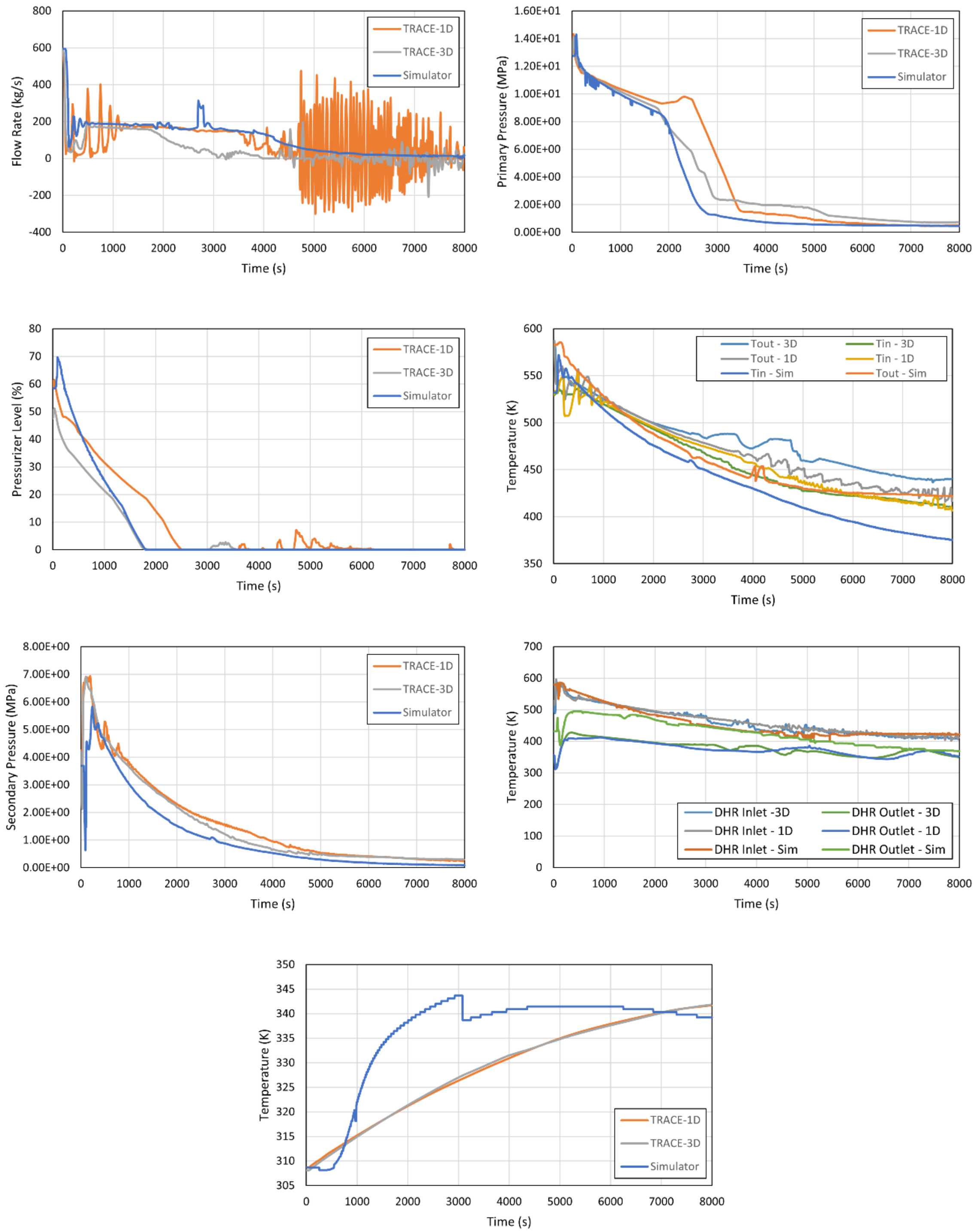

Fig. ##.

First, overall trends of the SBO will be discussed and then a more detailed analysis will be done to compare results from each of the models. At the start of the SBO, the feedwater pumps and turbine trips resulting in an increase of the primary pressure from the loss heat removal from the steam generators. After reaching the high-pressure signal setpoint, the reactor is shut off and DHRS flow is established after some time delay. Once flow in the DHRS is established, decay heat from the reactor is removed and the primary system pressure and temperatures begin to decrease.

As shown in Fig. ## above, the primary system flow rate predicted by the TRACE 1D model was found to have a much more oscillatory behavior compared to the simulator and 3D TRACE results. The 1D model was found to predict large oscillations further in the transient (~4500s) even predicting large flow reversal within the primary system but was found to show relatively good agreement to the simulator data until around 3500 seconds into the transient. Results from the 3D TRACE model were also found to predict some oscillatory behavior and flow reversal but on a much smaller scale than compared to the 1D model. The flow prediction from the 3D TRACE model showed good agreement to the simulator data until around 1900 seconds where it was found to underpredict the mass flow in the primary system compared to the 1D mode and simulator results. Time averaged results of the 1D model flow rate showed good agreement compared to time averaged results of the 3D model which underpredict the mass flow compared to the simulator data. A peak in the primary flow rate predicted by the simulator is seen around 2900 seconds into the transient. The cause of this peak is thought to be caused due to the small difference in DHRS temperature, as shown in Fig. ##, which suggests a decrease in the steam generator heat removal and as such the natural circulation flow increases in the primary system to compensate. The large oscillatory behavior may suggest a limitation in the modeling capabilities for natural circulation in the TRACE 1D model, but more investigation is needed here.

Figure ### above shows the results of the primary pressure trends through the SBO transient. The primary pressure predicted for both models show good agreement with the simulator data early in the transient, even predicting the pressure peak of around 14.3 MPa (2078 psi) compared to the simulator value of 14.3 MPa (2074 psi). Around 2000 seconds into the transient, the 1D and 3D models are found to overpredict the pressure in the primary system while the 3D model more closely follows the trend of the simulator. Two reasons for this discrepancy are likely due to differences in the predicted pressurizer levels and temperatures between the models. A rapid decline in pressure is consistent in all the results after the pressurizer level drops below zero. The 3D model predicts the time at which the pressurizer empties around the same time as the simulator as shown in Figure ##. However, the 1D model predicts the pressurizer empties around 2500 seconds where the pressure then rapidly declines and approaches values closer to the simulator results.

The overprediction in pressure is also likely due to the 1D and 3D models predicting higher primary temperatures compared to the simulator values as shown in Fig. ##. Both the 1D and 3D models were found to over predict primary side temperatures compared to the simulator data likely due to the lower predicted mass flow rate in the primary system from the TRACE models. Figure ## above shows that the TRACE 1D and 3D model predictions in the DHRS inlet temperature shows good agreement compared to the simulator data over the course of the transient. The TRACE models predict a lower DHRS outlet temperature compared to the simulator data but show good agreement toward the end of the transient. Differences in the outlet temperature are likely caused from differences in the predicted DHRS flow rate and pool temperature calculations as seen in Fig. ##. Unfortunately, the NuScale simulator does not provide flow rate data for the DHRS, so a sensitivity analysis should be done to assess pressure drops in the secondary loop.

4. CONCLUSIONS

Steady-state simulations and a SBO transient of the NPM were conducted using 1D and 3D TRACE models and compared to data provided in the NuScale DCA and simulator. Results from the steady-state models showed good agreement between the data available in the DCA and simulator with a maximum percent difference of 4.67% in the feedwater flow rate for both 1D and 3D TRACE models.

Analysis of the SBO transient c that the DHRS can provide adequate cooling to remove decay heat from the primary system in the early stages of the SBO. The primary mass flow rate predictions of TRACE models were found to predict lower mass flow rates than the simulator data and they even show oscillatory behavior and reverse flow at some point during the transient. Nonetheless, this oscillatory or reversed flow does not appear to produce any serious hazard to the core integrity in the 1D model. Primary system TRACE pressures were in agreement relative to the simulator in early stages of the transient but were then found to overpredict it compared to the simulator data. It was also found that the trend in primary pressure for all cases has a strong correlation with the pressurizer liquid level where the pressure was found to rapidly decrease once the pressurizer completely emptied.

The large oscillations found in the 1D TRACE model may suggest limitations in the 1D modeling capabilities of natural circulation of the system as the 3D TRACE model predicted a much smaller oscillatory behavior in the flow rate. For this reason, the authors recommend modeling natural circulation systems such as the NPM with multi-dimensional components to better capture the complex flow phenomena occurring in these transient conditions; however, more sensitivity analysis of the oscillatory behavior may need to be conducted. With the limited design data of the DHRS, more work can also be done to better optimize pressure losses and component geometries in the secondary side system.

What could be happening in pool temperature during the transient in simulator???

Make a table for key components and parameters, cpu times, etc….

Mention secondary side modeling and why what is happening in descrapancies

Acknowledge pressure of primary and secondary pressures equalize or don't equalize (should equalize)

CPU TIMES:

3D Model: ~29 hours for full steady to SBO

1D Model: ~3.8 hours for full steady to SBO

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