SAFETY TRANSIENTS IN HELICAL COIL STEAM GENERATORS

EVALUATION OF SAFETY TRANSIENTS IN HELICAL-COIL STEAM GENERATORS WITH RELAP5-3D CODE

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Abstract

Around the world, countries are increasingly considering carbon-free energy generation options as the threat of climate change grows. Small modular reactor designs, promising such carbon-free energy generation, are thriving worldwide with novel and innovative technologies that improve safety as well as economic performance. Canada is also considering small modular reactors (SMRs) as a means of achieving net zero carbon emissions by 2050.

Some of these reactor designs utilize pressurized water for cooling and moderator. Reactors with pressurized water have been subjected to steam generator tube ruptures in the past, and a detailed investigation into the possible consequences of such incidents in SMRs should be conducted.

In this research, a model for one of the newer designs, the NuScale Integrated Small Modular Reactor, was developed with the RELAP5-3D code for assessing safety-related transients. The NuScale Small Modular Reactor incorporates helical coil steam generators within its reactor pressure vessel, which are more efficient in terms of heat transfer than the U-tube steam generators that are widely used in nuclear reactors.

In the first part of the research, a detailed model is created and used to obtain steady state conditions with parameters collected from NuScale's Final Safety Analysis Report (FSAR). The Steam Generator Tube Rupture event is analyzed in the second part of the work. Slight differences in the broken and intact steam generator pressures as well as decay heat removal system flow rates are seen in the comparison of reference values and calculated results. Among the reasons for those differences could be that the correlations used by the RELAP5-3D code for heat transfer coefficient and pressure drop in the helical coil steam generators are different than those of the NuScale proprietary code NRELAP5, with which the analyses have been performed in the FSAR. Also, post-dryout heat transfer at the exit of helical coil steam generators and evaporator sections could cause differences in the outlet conditions of the steam, resulting in

different mass flow rates as well.

The final section of the research simulates a comparable but more severe tube rupture incident without the availability of decay heat removal systems in order to assess the reactor's emergency core cooling system reaction. Passive decay heat removal systems are crucial components for removing heat after reactor shutdown through heat exchangers that are submerged in the reactor pool and connected to steam generators by a closed loop. The containment pressures, the containment wall temperatures, and the peak fuel clad temperatures are considered to be the key design constraints that must be observed.

Future work on this subject could include modifying the source code, adding specific correlations for helical coil steam generators, and comparing the results, as well as quantifying uncertainties in the SGTR event. Main parameters in the quantification of uncertainties would be reactor power, single phase and two-phase discharge coefficients from the break, trip signals and delays as well as break size and location.

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Nomenclature

AC Power - Alternating Current Power

CNSC - Canadian Nuclear Safety Commission

CFR - Code of Federal Regulations

CNV - Containment Vessel

DHRS - Decay Heat Removal System

ECCS - Emergency Core Cooling System

FIV - Flow Induced Vibrations

FSAR - Final Safety Analysis Report

FWIV - Feedwater Isolation Valve

HCSG - Helical Coil Steam Generator

IAEA - International Atomic Energy Agency

IAB - Inadvertent Actuation Block

ICSP - International Collaborative Standard Problem

INL - Idaho National Laboratory

LOCA - Loss of Coolant Accident

LTOP - Low Temperature Over Pressurization

MASLWR - Multi-Application Small Light-Water Reactor

MSIV - Main Steam Isolation Valve

NPM - NuScale Power Module

NSSS - Nuclear Steam Supply System

RCS - Reactor Coolant System

RELAP - Reactor Excursion Leak Analysis Program

RPV - Reactor Pressure Vessel

RRV - Reactor Recirculation Valve

RVV - Reactor Vent Valve

PWR - Pressurized Water Reactor

SG - Steam Generator

SGTR - Steam Generator Tube Rupture

SMR - Small Modular Reactor

U.S. NRC - United States Nuclear Regulatory Commission

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1 Introduction

Safety analysis is of great importance due to the nature and extent of events if an accident were to occur in a nuclear reactor. A considerable percentage of the radioactive elements contained in a large power reactor could be discharged into the environment in a populated region. Even if a succession of exceedingly unlikely catastrophes occurs, containment and safety systems are essential to avoid radioactive material escape into the atmosphere.[10]

In this context, the safety analysis of a chosen reactor, the NuScale SMR, is conducted via regulatory procedures. Original objective of the work is to assess the tube rupture event with the RELAP5-3D code and to do code-to-code comparison of the design with calculations done by the designer with NRELAP5. This modified version of RELAP5-3D, as well as the model used by the NuScale Power LLC. company are proprietary and not published by US NRC; The model developed may be different from what the company originally published. Model changes may play important role in the transients, considering user effects of the code. The written thesis is divided into several sections, which are further explained as follows: Chapter 2 discusses the rationale for development of small modular reactors. Chapter 2 also discusses regulatory viewpoints in the licensing of nuclear reactors in the United States and Canada. Chapter 3 presents the Multi Application MASLWR, an early prototype of NuScale, and earlier tests conducted with the prototype, as well as the obtained data and deduced conclusions. Chapter 4 describes the NuScale SMR reactor and its various systems and components. Helical coil steam generators are also explained in the Chapter 4. The RELAP5 code and model developed for the NuScale SMR are described in Chapter 5. Chapter 6 describes steam generator tube rupture events, causes, earlier examples and the tube rupture model for the NuScale Reactor. Chapter 7 consists of results for the achieved steady state, power decrease, tube rupture and tube rupture without decay heat removal system transients. Chapter 8 describes the deduced conclusions from the work and the last chapter, Chapter 9 shows the references.

2 Small Modular Reactors

The phrase "small modular reactor" (SMR) refers to plants that produce less energy than large-scale commercial reactors. The power ranges from approximately 10–300 MW, and these reactors are classified as Generation 3+ or Generation 4. As the globe continues to revolve around climate change, zero-carbon energy generation becomes increasingly critical.

The US Nuclear Regulatory Commission (NRC) and the Canadian Nuclear Safety Commission (CNSC) are now reviewing a variety of SMRs for numerous reasons. While the primary objective of certain reactors is to generate energy, such as supplying stable and affordable electricity to isolated settlements or urban areas, alternative objectives might include hydrogen generation. SMR providers also consider mining applications and operations that require a lot of heat, such as desalination. At this time, there are several SMR advancements occurring globally.

A brief list of designs currently being assessed by CNSC can be seen in Table 1.

2.1 Safety Analysis and its Importance

The term "safety analysis" refers to the activities that take place from the first conceptual design of a nuclear reactor through to the decommissioning of the nuclear reactor. The International Atomic Energy Agency (IAEA) and other regulatory organizations such as the US Nuclear Regulatory Commission (US NRC) and the Canadian Nuclear Safety Commission (CNSC, formerly AECB), have been working since the 1950's to create basic regulatory principles for generating nuclear energy in a safe way. To ensure the safety of the public and the environment, it is critical to anticipate and eliminate the occurrences that could cause the reactor to deviate from normal operation, as well as

Vendor	Name of design and cooling type	Electrical capacity (MW electrical)	Applied for	Review start date	Status
Terrestrial Energy Inc.	IMSR Integral Molten Salt Reactor	200	Phase 1	April 2016	Complete
			Phase 2	December 2018	Assessment in progress
Ultra Safe Nuclear Corporation	MMR-5 and MMR-10 High-temperature gas	5-10	Phase 1	December 2016	Complete
			Phase 2	June 2021	Assessment in progress
LeadCold Nuclear Inc.	SEALER Molten Lead	3	Phase 1	January 2017	On hold at vendor's request
ARC Nuclear Canada Inc.	ARC-100 Liquid Sodium	100	Phase 1	September 2017	Complete
Moltex Energy	Moltex Energy Stable Salt Reactor Molten Salt	300	Series Phase 1 and 2	December 2017	Phase 1 completed
SMR, LLC.	SMR-160 Pressurized Light Water	160	Phase 1	July 2018	Complete
NuScale Power, LLC	NuScale Integral pressurized water reactor	60	Phase 2*	January 2020	Assessment in progress
U-Battery Canada Ltd.	U-Battery High-temperature gas	4	Phase 1	Pending	Project start pending
GE-Hitachi Nuclear Energy	BWRX-300 boiling water reactor	300	Phase 2*	January 2020	Assessment in progress
X Energy, LLC	Xe-100 High-temperature gas	80	Phase 2*	July 2020	Assessment in progress

Table 1: Small modular reactor designs in review by the Canadian Nuclear Safety Commission (CNSC). [5]

those that could result in the release of highly radioactive material at some point during the plant's operational life.

In this manner, voluminous safety analysis documents are created by the reactor vendors for the regulatory bodies of the planned country of installation. Safety Analysis documents consist of several chapters, usually describing general information about the plant, site characteristics, design criteria, reactor, reactor coolant system (RCS), engineered safety systems, instrumentation and controls, electricity requirements, auxiliary systems, steam and conversion systems, radioactive waste management, radiation protection, conduct of operations, initial testing and operation, accident analysis, technical specifications, quality assurance, probabilistic safety analysis, severe accident analysis, and so on. This work is done for the licensing analysis for regulatory review purposes. Design assist and experimental analysis are out of scope of this thesis as they are not contributing to improving the design of the NuScale SMR nor were experiments for assessing the code.

The US NRC states a more rigid structure of needs defined by the law of 10 CFR 50. Appendices of 10 CFR 50 describe different requirements for manufacturing quality, installation of systems, and what is accepted as "nuclear grade". Safety analysis studies must evaluate the design and performance of structures, systems, and components, as well as their suitability for accident prevention and mitigation. For example, the analysis and assessment of the cooling performance of the emergency core cooling system (ECCS) after hypothetical loss-of-coolant accidents (LOCAs) must comply with the criteria of 10 CFR 50.46; Or, the facility's technical specifications must be based on the safety analysis and developed in compliance with 10 CFR 50.36. [13]

The Canadian Nuclear Safety Commission (CNSC) addresses and details the standards and guidelines for preparing and presenting a safety analysis that proves a nuclear facility's safety analysis needs and methods to use to the reactor vendors in the REGDOC-2.4.1, Deterministic Safety Analysis document. [15]

3 Multi-Application Small Light-Water Reactor

The Multi-Application Small Light-Water Reactor (MASLWR) Test Reactor is an integrated pressurized light water reactor that relies on natural circulation. The MASLWR was built at Oregon State University as a prototype to a NuScale Power Module. The MASLWR design layout is seen in Figure 1. The MASLWR nuclear steam supply system (NSSS) is housed inside the reactor vessel, with natural circulation driving the core flow. The steam generators, which are made up of banks of vertical helical tubes, are positioned in the top section of the vessel, outside of the hot leg chimney. The feedwater is totally vaporized within the tubes after traversing roughly 60% of the tube length, resulting in superheated steam at the steam generator outlet. The significance of this facility is in the tests undertaken to determine natural circulation stability, to simulate significant events, and to finally validate the present thermal-hydraulic computer codes in light of this evidence.

Conclusions related to Helical Coil Steam Generators (HCSG) derived from International Collaborative Standart Problem (ICSP) experiments done in the MASLWR facility could be summed up as the following: The majority of modern computer programs used for thermal-hydraulic analysis do not include suitable heat transfer and pressure drop correlations for the helical coil's interior and outer surfaces. [6] This problem is further complicated by the fact that the helical coil steam generator's inlet condition corresponds to single-phase flow, while the outlet condition corresponds to super-heated flow, necessitating not only the estimation of heat transfer and pressure drop under single- and two-phase flow conditions, but also dryout and post-dryout heat transfer. Additionally, numerous geometric and operating parameters such as the diameter of the tube, the diameter of the helix, the helical pitch, the flow regime (laminar, transition, and turbulent flow in single-phase fluid), the orientation of the helical tube (vertical upward/downward, inclined or horizontal), and the flow patterns (bubbly, slug, annular, and droplet flow) all affect the helical coil heat transfer and pressure drop. Likewise, for heat transfer and pressure loss, the entrance effect must be considered.

It is also noted that while a lumped SG tube model demonstrated more stable behavior, parallel channel instabilities could not be investigated.

4 The NuScale Small Modular Reactor

One of the main reasons that analyses are chosen to be done with NuScale SMR is that it is a competitor in the Canadian market for small modular reactors. This particular reactor is an advanced design that depends on passive mechanisms to protect against design basis accidents, and it incorporates HCSGs as a means of heat transfer.

The NuScale Reactor is composed of multiple components. Each of the facility's nuclear reactor units is called a NuScale Power Module (NPM), partially submerged in water. It may have up to 12 such modules, each of which generates 50 MWe. [1] Each

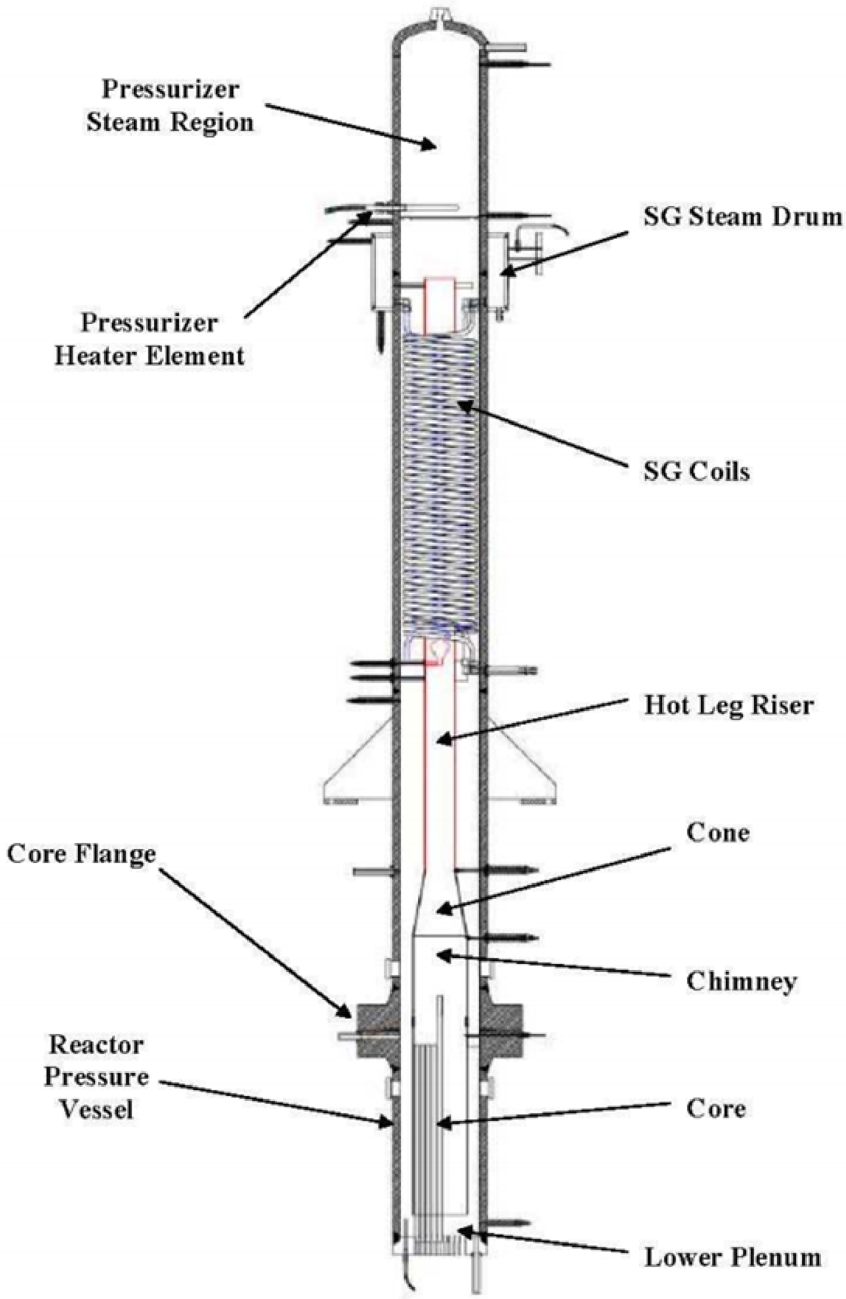


Figure 1: MASLWR Test Facility Schematic at Oregon State University [6]

module consists of a reactor pressure vessel (RPV) and is shrouded in a containment vessel (CNV). A total of 12 modules can be controlled from a single control room. The reactor body is seen in Figure 2 along with the flow of the primary coolant inside it. The core is located at the bottom, receiving the flow from the sides though the downcomer. With the core heating the primary coolant, the density of the water decreases. Density and elevation difference provide the drives for the flow in the primary system. After going through the core, primary fluid rises in the riser section. It continues its travels to the upper downcomer section, directed by the pressurized baffle plate. Here, primary coolant goes through two HCSGs, surrounding the upper section of the riser, which is a conspicuous feature of the NuScale design that the steam generators are integrated within the RPV. While the heat due to the temperature difference between the primary coolant and secondary coolant is transferred to the secondary side, fluid density increases in the primary side and coolant travels through the downcomer and upper plenum to reach back to the core.

In the secondary side, feedwater enters into HCSGs from feedwater tube sheets and moves in the tubes. Superheated steam exits once-through HCSG tubes and goes to the turbine for mechanical energy production. This cycle continues as the reactor operates under normal conditions as well as in unprecedented events.

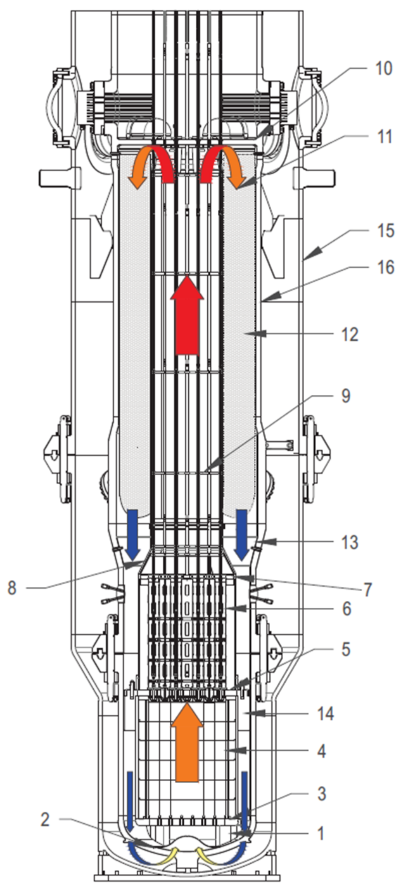
The flow rate of the coolant is dependent on the reactor power. A pressurizer located at the top part of the reactor provides stable pressure in the system. The reactor's upper head exists above that.

Unlike a PWR's containment vessel, which is steel-enforced concrete, NuScale's compact containment vessel is made of steel entirely.

The reactor core is fueled with the well-known UO_2 fuel. The enrichment level of the fuel is low, up to 5 percent. A total of 37 assemblies with typical 17×17 geometry provide the power. The moderator and coolant material are light water. [1]

4.1 Reactor Geometry and Parameters

The following information was obtained through the 5^{th} Revision of the Final Safety Analysis Report of NuScale SMR, which was submitted to the U.S. Nuclear Regulatory Commission. Table 2 shows the main reactor parameters of power, system pressure, inlet outlet temperatures, core heat flux, flow area and heat transfer surface area on the fuel. Table 3 describes main reactor geometries which are used to model the reactor. Table 4 describes reactor vessel (RV) design parameters such as design pressure and temperature, length of RPV, thickness of wall at different sections.



No.	Stage
1	Core support blocks in downcomer
2	Downcomer to lower plenum turn
3	Lower core plate
4	Core
5	Upper core plate
6	Control rod assembly guide tubes
7	Control rod assembly guide tube support plate
8	Riser transition
9	Control rod drive shaft support
10	Pressurizer baffle
11	Upper riser turn to annulus
12	Downcomer through steam generator
13	Downcomer transition
14	Upper core support blocks
15	Containment vessel
16	Reactor vessel

Figure 2: The NuScale Reactor Coolant System.[3]

Reactor Parameter	**Imperial Unit**	**SI Unit**
Core thermal output	160 MWth	160 MWth
System pressure	1850 psia	12.76 MPa
Inlet temperature	497 °F	531.5 K
Core average temperature	543 °F	557 K
Average temperature rise in the core	100 °F	56 K
Core bypass flow (%)(best estimate)	7.3	7.3
Average linear power density	2.5 kW/ft	8.2 kW/m
Peak linear power for normal operating conditions	5 kW/ft	16.4 kW/m
Normal operation peak heat flux	170,088 Btu/hr-ft^2	536.558 kW/m^2
Total heat flux hot channel factor, FQ	2	2
Heat transfer area on fuel surface	6275.6 ft^2	583.022 m^2
Normal operation core average heat flux	85,044 Btu/hr-ft^2	268.28 kW/m^2
Core flow area	9.79 ft^2	0.9095 m^2
Core average coolant velocity	2.7 ft/s	0.823 m/s

Table 2: NuScale Reactor Parameters [2]

4.2 NuScale Safety Systems

This section will discuss the passive safety systems integrated into the NuScale reactor. The reactor is equipped with two primary safety systems to protect against Design Basis Events. The Decay Heat Removal System is the first of these, while the Emergency Core Cooling System is the second. Each safety system is described in detail in the sections that follow.

4.2.1 Decay Heat Removal System (DHRS)

The NuScale Power Module has a Decay Heat Removal System (DHRS) that removes core decay heat which is around 10 MW-thermal after the shutdown and decreases to 1.1 MW-thermal in a day. The DHRS system is comprised of two DHRS trains linked to the reactor. Each train of the DHRS is linked to one of the building's two NPM steam generators. The DHRS pipes are connected to the corresponding SG's main steam and feedwater lines. The DHRS steam input pipe connects to the system's main steam line, which is placed upstream of the system's main steam isolation valve. The DHR system's pipework is routed to two parallel DHR actuator valves.

Each train has an opening between the actuation valves and the DHRS passive con-

RCS Region	**Volume $m^3(ft^3)$**	**RCS Sub-region**	**Average Flow Area $m^2(ft^2)$**	**Length m(ft)**
Riser	17.98(635)	Lower riser and transition	2.31(24.9)	2.87(9.4)
		Upper riser and riser turn	1.43(15.4)	7.93(26)
Downcomer	33.95(1199)	Downcomer (including steam generators)	2.39(25.7)	14(46)
Core	2.52(89)	Fuel assemblies	0.96(10.3)	2.4(7.9)
		Reflector cooling channel	0.084(0.9)	2.4(7.9)
Pressurizer	16.37(578)	Pressurizer heaters / main steam plenums	3.353(36.1)	0.52(1.7)
		Cylindrical pressurizer	5.70(61.4)	2.1(6.9)
		Reactor pressure vessel top head	3.83(41.2)	0.67(2.2)

Table 3: Reactor Geometry Parameters [2].

densers, which assists in maintaining a controlled water flow during operation. Following that, a new length of pipe is created and routed along the exterior of the containment vessel to a train-specific DHRS passive condenser. The DHRS passive condenser's output is routed to the feedwater line servicing the associated SG, where it is connected to the feedwater line downstream of the main feed isolation valves to complete the loop.

The DHRS is always in standby mode during normal power operations, with each train of DHRS being isolated from the associated main steam lines through the closing of the DHRS actuation valves on the main steam lines. On each train, these four valves, each with two valves, are connected in parallel and always maintained closed. [1]

When the MSIVs and FWIVs are actuated, they are closed and the DHRS actuation valves are opened. The DHRS actuation valves are intended to open in the event of a control power interruption, whether caused by control system actuation or a power loss.

Actuation allows the water column in the DHRS piping to drain into the feedwater system piping and plenum, and steam from the SG to flow into the DHRS piping and passive condenser of the DHRS. The passive condenser condenses steam by transferring heat to the reactor pool. As a consequence of this procedure, condensate is continuously pumped from the passive condenser to the related feedwater line and into

Design Parameter	**Imperial Unit Value**	**SI Unit Value**
Design pressure	2100 psia	14.48 MPa
Design temperature	650 °F	616.48 K
Overall height, bottom of alignment feature to top of CRDM latch housing section	778 inches	1976.12 cm
Inside diameter of lower RPV section, cylindrical region, without clad	96.5 inches	245.11 cm
Outside diameter of lower RPV section, cylindrical region, without clad	105 inches	266.7 cm
Inside diameter of upper RPV section, cylindrical region, without clad	104.5 inches	265.43 cm
Outside diameter of upper RPV section, cylindrical region, without clad	112.5 inches	285.75 cm
Inside diameter of pressurizer, cylindrical region, without clad	106.5 inches	270.51 cm
Outside diameter of pressurizer, cylindrical region, without clad	115.5 inches	293.37 cm
Inside diameter of upper head without clad	104.5 inches	265.43 cm
Outside diameter of upper head without clad	112.5 inches	285.75 cm
Inner clad thickness	0.25 inches	0.64 cm
Outer clad thickness	0.125 inches	0.318 cm

Table 4: NuScale Reactor Pressure Vessel Parameters [3].

the associated SG. [3] DHRS is simulated for the tube rupture event.

4.2.2 Emergency Core Cooling System (ECCS)

The ECCS is a critical component of the NuScale Power Plant's safety system because it responds to LOCAs in a safe manner and serves as a component of both the reactor coolant and containment vessel pressure limits. The ECCS, in combination with the containment heat removal function, offers core decay heat removal in the case of a coolant loss that exceeds makeup capabilities.

Three reactor vent valves (RVVs) are positioned on the top head of the RPV, two reactor recirculation valves (RRVs) are fixed on the RPV's side, and accompanying actuators are situated on the upper CNV. During normal plant operation, all five valves are closed; they open to activate the system in the event of an accident. The RVVs

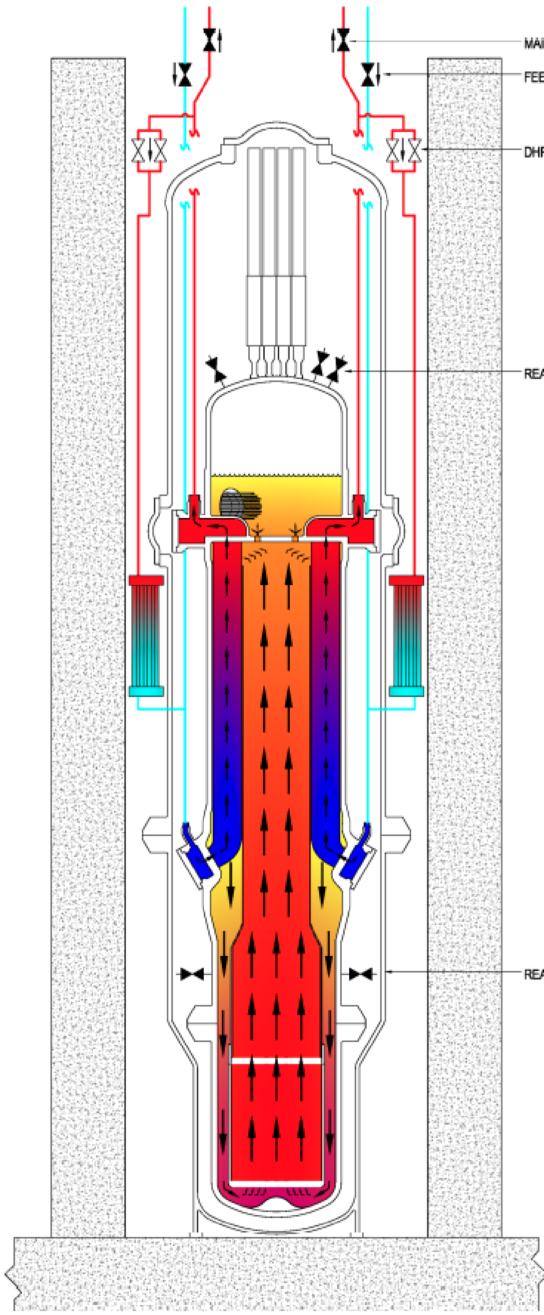


Figure 3: NuScale Decay Heat Removal System.[1]

Parameter	Imperial Unit Value	SI Unit Value
Internal Pressure	2100 psia	14.48 MPa
Actuation valve external pressure	60 psia	0.413 MPa
Passive condenser external pressure	27 psia	0.186 MPa
Temperature	650 °F	616.5 K
Number of condensers	2	
Total number of tubes per condenser	80	
Tube wall outer diameter	1.315 inches	3.34 cm
Tube wall thickness	0.109 inches	0.277 cm
Tube external surface area per condenser	258.2ft^2	23.99 m2
Fouling factor	0.0005 hr-ft^2-°F/BTU	0.00285 W/m^{2}K

Table 5: Decay Heat Removal System Design Data [3]

release steam from the RPV into the CNV, where it condenses and settles as liquid condensate at the containment's bottom. The RRVs enable collected coolant to be reintroduced into the RPV for recirculation and core cooling. The ECCS is totally passive in nature, with heat being transferred through the CNV wall to the reactor pool. The RRV penetrations are positioned on the side of the RPV in such a way that when the system is operated, the coolant level in the RPV stays above the core and the fuel remains covered. [1] ECCS is simulated in the model for tube rupture without availability of the decay heat removal system but not for the tube rupture transient itself.

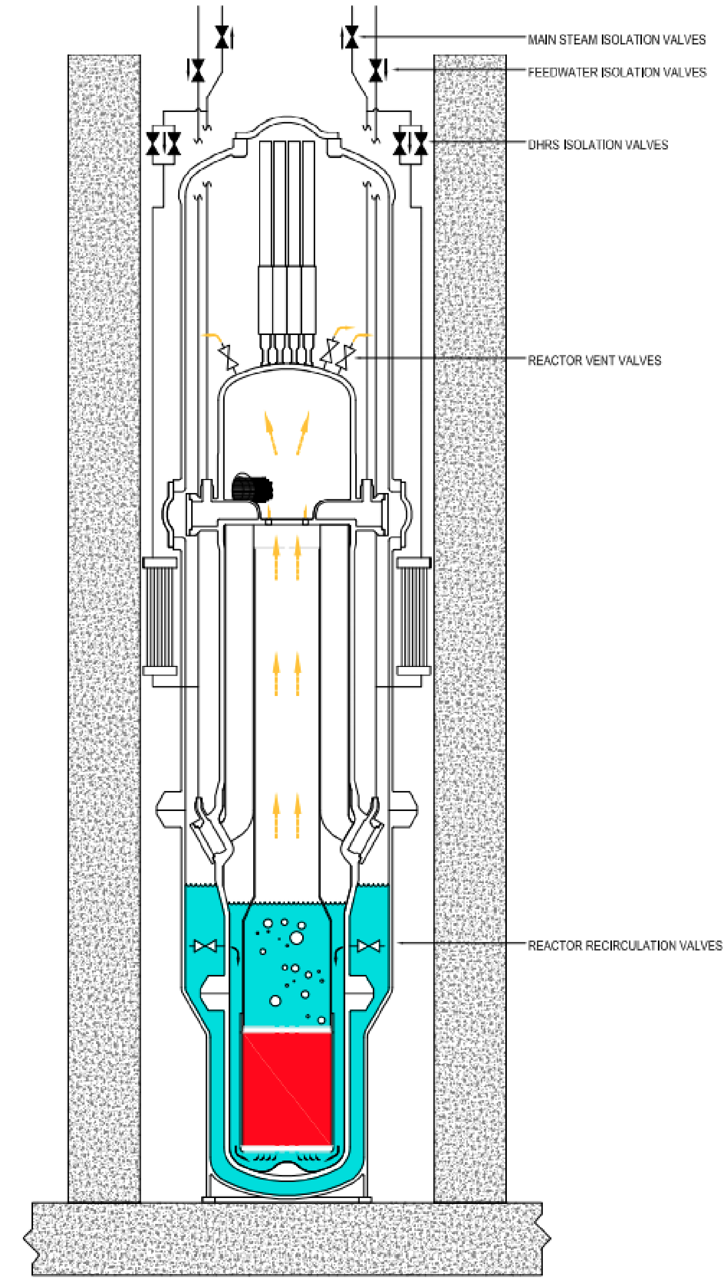


Figure 4: NuScale Emergency Core Cooling System.[1]

Steam Generator Design Data		
Parameter	**Imperial Unit**	**SI Unit**
Type	Helical, once-through	
Total number of helical tubes per NPM	1380	
Number of helical tube columns per NPM	21	
Internal pressure - secondary	2100 psia	14.48 MPa
External pressure - primary	2100 psia	14.48 MPa
External pressure - SG piping in containment	1000 psia	6.9 MPa
Internal temperature - secondary	650 °F	616.5 K
External temperature - primary	650 °F	616.5 K
External temperature - SG piping in containment	550 °F	561 K
Tube wall outer diameter	0.625 inches	1.5875 cm
Tube wall thickness	0.050 inches	0.0127 cm
Steam tubesheet thickness, without clad	4.0 inches	10.16 cm
Feed tubesheet thickness, without clad	6.0 inches	15.24 cm
Steam tubesheet thickness, with clad	4.625 inches	11.75 cm
Feed tubesheet thickness, with clad	6.625 inches	16.83 cm
Heat transfer surface area	17928 ft^2	1665.56 m^2
Fouling factor	0.0001 hr-ft^2-°F/BTU	0.00057 W/m^2K
Minimum SG tube transition bend radius	>6.250 inches	>15.875 cm

Table 6: NuScale Steam Generator Design Data [3]

4.3 Helical Coil Steam Generators

Helical coil steam generators (HCSG) are a sophisticated design that has gained popularity in recent years, despite the fact that they were invented in the early 1900s. Essentially, a helical coil is a torus that revolves around a virtual single line at the center. Between two separate fluids, referred to as shell side and tube side, lies a layer of stainless steel. While main fluid travels from top to bottom via the shell side, secondary fluid travels from bottom to top through the helical tube side. Obviously, the configuration of the geometry is determined by the application and other requirements.

In such a system, heat transport occurs by conduction.

Main parameters for designing a helical coil heat exchanger are: tube diameter, number of tubes, wall thickness of tubes, tube length, total coil rise, tube inlet and outlet temperatures, primary side inlet and outlet temperatures, design pressure of the system, pitch diameter, angle of curvature and so on. Figure 5 shows TF-2 test SGs designed and tested by NuScale in the NIST(MASLWR) Test Reactor.

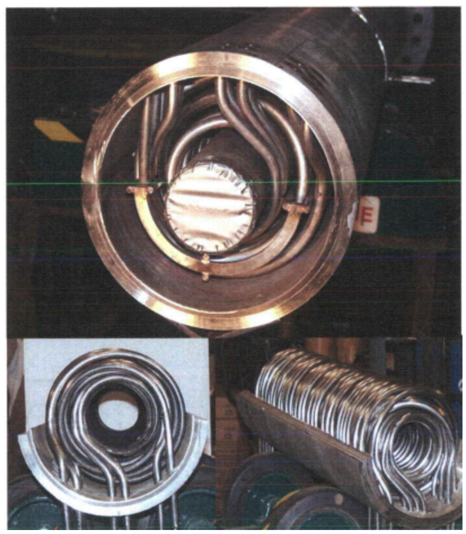


Figure 5: Helical Coil Steam Generators in TF-2 NIST Test Reactor

(Prabhanjan et al. 2002) describes, based on the experimental work on the comparison of the heat transfer coefficient, that a helical coil heat exchanger has a larger heat transfer coeeficient than a straight tube heat exchanger of the same dimensions. Heat transfer coefficients increase as the surrounding fluid temperature rises, most likely owing to buoyancy effects on the heat exchangers. [14] The coil shape and the fluid flow rate have an impact on the fluid's temperature increase. Prabhanjan notes that approximately 16 to 42 percent increase is seen in the experiments.[14]

Helical coil heat exchangers perform substantially better in heat transfer than straight tubes. The higher heat transfer coefficient occurs as a result of the coil's curvature,

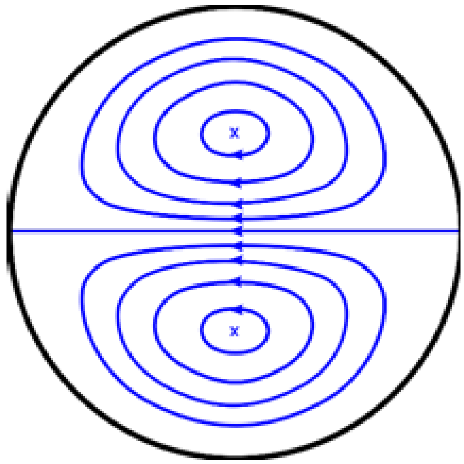


Figure 6: Schematic of a pair of Dean vortices that form in curved pipes.

which exerts centrifugal forces on the circulating fluid, resulting in secondary flows. Apart from increasing the heat transfer coefficient, generated secondary flow improves mixing and frictional pressure drop, particularly in laminar flows. The number of secondary flows occurring inside the tubes, Dean vortices, can be seen in Figure 6. Naturally, in a compact design where area is of concern, HCSGs are of choice compared to U-tube SGs. Another factor to consider is the quantity of water that these SGs can hold. U-tube SGs contain more water than HCSGs. This is important because U-tubes can withstand a reduction in secondary side inventory for a longer period of time than HCSGs. This is not a concern due to NuScale's powerful passive ECCS. Another benefit of HCSGs is their strong resistance to flow-induced vibrations (FIV), which are a significant cause of tube ruptures.

Previous studies in the literature used the RELAP5-3D code with HCSGs. [7] [19] Nonetheless, the code's simulation capabilities for HCSGs may be enhanced further. In the MASLWR IAEA collaboration work, for modeling HCSGs, contributors have used several different techniques such as: Heat transfer surface area increase, and increase in the heat transfer coefficient, using a fouling factor greater than 1.0. [6]

5 RELAP5 Code and Model Development

RELAP5 code is developed by Idaho National Laboratory (INL) since 1970's for thermal hydraulic analysis of nuclear systems. Today, the code has been expanded to function with a variety of fluids and a variety of simulated scenarios. [16] For this analysis, latest RELAP5-3D version 4.4.2 (released in June 2018) is used. A license for RELAP5-3D is obtained through an agreement between McMaster University and Idaho National Laboratory, as a university participant. Code allows users to simulate a wide variety of transients such as small and large break loss of coolant accidents, especially in Light Water Reactors (LWR).

Based on INL's RELAP5 and through the validation and testing prototype of NuScale, MASLWR at Oregon State University, the NuScale Power LLC. company developed proprietary NRELAP5 code for simulating important phenomena related to the NuScale Power Module in a more accurate way. NRELAP5 is validated through experimental tests conducted at the Oregon State University (MASLWR), at the SIET TF-1&2 facility in Italy for HCSGs as well as with CHF tests conducted in Stern Labs. Thermal hydraulic tests were conducted at these facilities. [6] [12] Since NRELAP5 is unavailable for use, RELAP5-3D was chosen to achieve the closest results.

Many system thermal hydraulic codes work in a similar way in that the system is divided into different nodes and connections. Pressure, temperature, quality parameters are input for volumes, and either mass flow rate or velocity terms should be input for junctions initially. Hydrodynamic structures are used for the flow in the system. An area, length, or volume should be entered for all hydrodynamic structures existing in the model. A list of hydrodynamic structures can be found in the table below.

Time-dependent volumes and junctions can be used for simulating inlet and outlet conditions. This component behaves as a source or sink. For example, since a turbine is not in the scope of work, turbine parameters (outlet temperature, pressure, quality) values can be simulated as a time-dependent volume, and the steam generator outlet is connected to time-dependent volumes. Likewise, time-dependent junctions are connections that can be used to provide a stable mass flow rate to volumes.

A branch component can be used to simulate when a lot of junctions should be connected to a volume at the same time. Each of the junctions may have different flow areas and velocities.

The valve component is used for simulating valves. Check valves, servo valves controlled by a control system, motor valves that depend on different trip parameters as well as opening and closing speeds, and trip valves activated when certain conditions are met can be used. Heat structures simulate the heat transfer occurring between

Component	**Input Name**
Single volume	snglvol
Time-dependent volume	tmdpvol
Single Junction	sngljun
Time-dependent junction	tmdpjun
Branch	branch
Separator	separatr
Pipe	pipe
Annulus	annulus
Pressurizer	prizer
Feedwater heater	fwhtr
Jetmixer	jetmixer
Turbine	turbine
ECC mixer	eccmix
Valve	Valve
Multiple Junction	mtlpjun

Table 7: RELAP hydrodynamic components

different components. The code calculates the heat transfer according to input values of wall thickness, mesh nodalization (both axially and radially), type of heat structure (rectangular, cylindrical, or spherical), surface area of each node, and heated and wetted diameters, thermal properties such as thermal conductivity and heat capacity of the material. A good way to simulate heat exchangers is to use counter flow on the opposite sides of the flow. Thermal properties of materials can be entered through temperature dependent tables for materials that are used in reactors, such as stainless steel, clad materials such as Zircaloy, UO_2, helium.

Volume	**Name**	**Node**	**Length, m(ft)**	**Flow Area, $m^2(ft^2)$**
315	Spray Valve Tmdpvol	1	1.0(3.28)	1(10.77)
314	Spray Valve Pipe	6	0.1(0.33)	0.002027 (0.02182)
302	Pressurizer	1	0.671(2.2)	3.83(41.2)
302	Pressurizer	1	2.103(6.9)	5.69(61.2)
302	Pressurizer	1	0.52(1.7)	3.35(36.1)
301	Upper Plenum	1	0.67(2.198)	3.83(41.2)
106,201	Upper Riser, Upper Downcomer	20	0.382(1.253)	2.39(25.7)
104,203	Middle Riser, Middle Downcomer	1	0.54(1.772)	1.43(15.4)
104,203	Middle Riser, Middle Downcomer	5	0.466(1.529)	2.31(24.9)
101-102, 103,205	Core Exit, Bypass Exit, Lower Downcomer	1	0.2(0.656)	0.9095(9.79), 2.39(25.7)
111,112-114, 115, 205	Core Channels, Bypass Channel, Lower Downcomer	10	0.2(0.656)	0.02586(0.2784), 0.3114(3.352), 0.0836(0.9), 2.39(25.7)
116-117, 118, 205	Core Entrance, Bypass Entrance, Lower Downcomer	1	0.2(0.656)	0.3114(3.352), 0.0836(0.9), 2.39(25.7)
107	Lower Plenum	1	0.5(1.640)	3.83(41.2)

Table 8: NuScale primary system RELAP5-3D model components and geometry.

A RELAP5-3D model for NuScale reactor is created through use of hydrodynamic and heat structure components. Table 8 describes each item of the geometry information in the primary system. The core is simulated with one channel with pipe component, having the entirety of the assemblies and flow area in the early nodalization. Later, this is changed and changes made are described. Middle riser, upper riser, pressurizer sections are modeled with pipe component, whereas upper, middle and lower downcomer components were created with an annulus component with given values in Table 8, which can be used for downward flow pipes. Heat structures are attached to core, inserting power into the system. Ten axial nodes and eight radial nodes were used.

Five radial nodes were used for simulating the fuel pellet, one node for gap material between fuel pellet and fuel sheath, and two radial nodes for sheath material. Thermal properties for each material are given with a heat conductivity and heat capacity table which is dependent on the temperature value. Default UO_2 and Zr sheathing tables are used since M5 zirconium alloy sheating produced by Framatome, used in NuScale reactor are proprietary. [8]

Table 9 shows the grouping of the radial power profile in the beginning of the cycle. Four groups of fuel assemblies are assigned and Figure 11 shows the clustering. Blue represents the hottest assembly in the core, whereas other three groups include 12 assemblies each. Axial Power Distribution is obtained from the FSAR at the beginning of the cycle and can be seen in the Figure 10. It is inserted into the power profile in the heat structures. Cross flow between the core volumes is added with multiple junction component. This results in a better simulation of the mixing in the core channels.

Blue	**Red**	**Green**	**Yellow**
111	**112**	**113**	**114**
1.137	0.991	0.999	0.998333

Table 9: Radial Power Profile for Core Channels in BOC

Table 3 reactor geometry information was obtained from the Final Safety Analysis Report. [3] (Hoffer et. al 2012)'s work [7] is used to model HCSGs. To put it simply, long helices of tubes are lumped into one single tube in terms of flow area and modelled as inclined tubes. HCSGs have a total heat transfer surface of 17,928 square feet. Length of tubes are 79.4 ft (around 24.2 meter) with an inclination of 18.4 degrees. Since HCSGs heat transfer efficiency is more than U-tube SGs as described in the HCSG section, a factor of 30% heat transfer surface area increase is applied to the model. This factor is also used in literature for modeling HCSGs in IRIS Reactor by (Zaman et al. 2017). [19] It is also seen in the ICSP experiments of MASLWR facility, factors of increase in between 20-100% are applied to simulate HCSG conditions better. [6] Given that RELAP5-3D does not include HCSG-specific phenomena, outlet conditions for HCSGs may also affect SGTR results.

Feedwater is simulated with a time-dependent-volume component and connected to SG piping with two separate time-dependent junctions to deliver to SGs a given mass flow rate with a stable condition. The turbine is simulated with a time-dependent-volume component similarly. The SG outlet piping is connected with the main steam isolation valve to the turbine.

Early nodalization was implemented to achieve steady state results. The results were not satisfying, and later changed based on the work of (Skolik et al. 2021) [18] . In this work, it is seen that slice nodalization is applied for natural circulation stabilities in the system, and all parts of the model for the primary side, containment vessel, and reactor pool volumes are doubled, one of them for upward flow and the other one for simulating downward flow, connected with branches at the top and bottom. Following changes based on [18] work are made,

- The core nodalization has been changed from a single core channel and bypass channels to four core channels and one bypass channel.
- The hottest core channel is simulating the hottest assembly in the core, and the rest of the three core channels are simulating an average of 12 assemblies in the core. This results in a total of 37 assemblies in the core. Length of active core is given as 6.56 ft (2 meter) in the FSAR.
- Components in the upward and downward flow changed to have the same number of nodes as well as the length of nodes.
- A primary mass flow rate control system is added.
- (Skolik et al. 2021) combined SGs into a single body, whereas this work requires the existence of two SGs due to the nature of the failure of one of them.

Since the NuScale Power Module does not have reactor coolant pumps, the mass flow rate of the primary system depends on the power of the reactor. As there is more

Primary System Temperatures and Flow Rates							
Reactor Power		**Primary Flow**		**Primary Coolant Temperature**			
%	MWt	%	(kg/s)	Core dT	T_cold (F)	T_avg (F)	T_hot (F)
Best Estimate Flow							
0	0	0-12	0-68.5	0		426	
15	24	48	280	32	529	543	558
50	80	76	444	67	512	543	574
75	120	89	521.6	85	504	543	583
100	160	100	587.0	100	497	543	590.1
Minimum Design Flow (kg/s)							
100	160	91.7	538.5	107.6	487.4	538.7	590.1
Maximum Design Flow (kg/s)							
100	160	112.5	660.5	89.65	507.8	548.9	590

Table 10: NuScale Primary System Flow Rates

power transferred to the coolant, the flow rate increases due to the larger temperature difference in the riser region. Instabilities may occur during such a system, and these instabilities are addressed in the Final Safety Analysis Report. To put it simply, if there is a less than 5 degree Fahrenheit temperature difference in the riser region, the reactor primary flow rate may be unstable and the reactor is not operable.

A control system in the upper riser with a servo valve component is added to achieve the desired mass flow rate of 587 kg/s. (Flow rate best estimated with full power) The control system works as a PID controller. Measured mass flow rate is subtracted from target mass flow rate, and with a proportional integral value, the valve opens and closes its stem position to achieve the targeted mass flow rate. A table is added for primary flow rates and corresponding temperatures on the primary side. This can be seen in the Table 10.

Pressurizer modeling during transient plays an important role. Pressurizer heaters consist of two heaters, each of them having 400 kW electrical power. A heat structure component is added to the bottom volume of the pressurizer to simulate heat addition in the case of activation of heaters. When the pressure drops, the increased steam produced by heaters provides additional pressure. When pressure increases, the spray valve which is connected to top volume of pressurizer is activated and steam condenses to liquid, providing pressure drop. Setpoints for these control are as the following,

- Pressurizer heater is activated when pressure decreases to 1780 psia, and then it is deactivated at 1850 psia.

- Pressurizer spray valve is activated at 1920 psia, and stops when pressure decreases to 1850 psia. (70 psia range)

Pressurizer spray valve is transferred through a pipe component which has downward flow, with a flow area of 0.02181 square feet and length of 1 feet for each of the nodes.

It is observed in the steady state run that the pressurizer level is lower than expected. A spurious liquid adding volume is added to increase the pressurizer level to 6.48 ft of expected value at the steady state values (60%). Later, it is increased to 7.34 ft (68%) with the biases applied for the SGTR transient.

DHRS modeling is critical for heat removal during a reactor shutdown scenario. While certain parameters, such as the heat transfer surface area, tube thickness, tube diameter and the number of tubes in the DHRS, are specified, the detailed model for the DHRS is not included in the FSAR. The length of tubes and piping system is assumed in this case based on the work of (Skolik et al. 2021) [18]. The DHRS has a flow area of 0.013 m^2, or 0.14 square feet, and a total length of 42.65 feet for the heat-exchanger section. Each DHRS train has a heat transfer surface area of 280.2 square feet, with a heat structure connected to the reactor pool. Additionally, piping is included to connect the DHRS system to the steam generators' inlet and outlet. DHRS actuation valves are added between piping of SG outlet and DHRS heat exchanger section.

Containment and reactor pool modeling is based on the parameters provided in the FSAR. The containment vessel is a stainless steel vessel surrounding the RPV and comprising ECCS valves, sitting inside a huge reactor pool which is the ultimate heat sink. CNV also provides insulation of RPV, since it is vacuumed and removes the need of simulating heat structures between RPV and outer sections. Usually in PWRs, CNV is a large concrete structure with several different components inside such as reactor core and piping, steam generators, RCS pumps, CNV water collection sump, CNV spray system, High and Low Pressure Injection Systems, Hydrogen Removal Systems and so on. Containment design parameters and specifications are given in the Chapter 6, Engineered Safety Systems of NuScale FSAR. Figure 12 depicts the containment geometry. As (Skolik et al. 2021) stated, for improving natural circulation stability, slice nodalization is also applied to CNV and reactor pool components. [18].

A total length of 908 inches elevation from the ground is given for the containment vessel, approximately 76 ft which is around 23 meters with an inner diameter of 14.17 ft (4.32 meters), and outer diameter of 14.75 ft(4.5 meters). While modeling the containment structure, the top and bottom sections of the containment are simulated as branch components, whereas upward and downward flow are simulated as different pipe components. Both pipe components have the exact same length and nodes as the RPV but a different flow area. Another difference between CNV and RPV in the model is the branches section. Branch components at the top and bottom are horizontal volumes for simulating natural circulation in CNV better, and they have a length of 10 ft each, with flow areas of 650 square feet at the top and 10.0 square feet at the bottom. Normally, the length of top CNV branch is expected to be 14.0 ft, considering the CNV diameter. Due to loop closure errors that are encountered with RELAP5, the length of the top branch is decreased and for compensation, flow area is proportionally increased. It is important to stress that the lengths given here are not vertical lengths but horizontal.

Assumption made for calculating top branch flow area, is that the control rod insertion system and several components exist in that area although RPV is not located in that space. The three RRVs and two RSVs are located above the RPV and are linked to the containment top branch. RSVs have 3 inches of inner diameter and 4 inches of outer diameter, with pressure setpoints of 2075 psia and 2100 psia. When the valves open, 10% blowdown occurs from either of the valves. The minimum amount of steam that can be released from safety valves is 65,536 lbm/hr at 10% over pressure, which is the equivalent of 18.2 lbm/s of steam. These two valves are only simulated for the scenario of SGTR without DHRS. Two reactor recirculation valves (RRV) are also added in the CNV model, connected from the CNV upward flow pipe component to the RPV middle downcomer.

The initial condition for containment hydrodynamic structures is 0.09 psia (620 Pa), with a static quality of 1.0. The reactor building is also considered and filled with air, instead of water. The flow area of the reactor pool is input as 2600 square feet for each pipe component, with a length of 70 ft. The initial conditions for the reactor pool are 14.7 psia (100 kPa) and 100 F (310 K).

The ECCS valve actuation signals are given as the following:

High CNV level actuation 252 inches (equivalent to 21 feet, 6.4 meter) above the reactor pool floor, riser water low level actuation 30 ft (equivalent to 9.14 m), low RCS pressure 800 psia (5.515 MPa), RPV low temperature and high pressure (LTOP) actuation, which is a function of the RCS cold temperature, is not required here because it is associated with reactor startup and has no effect on the SGTR findings.

Reactor Vent Valves are designed to interrupt opening of ECCS systems inadvertently. As described in the Chapter 6, Engineered Safety Systems, ECCS section, Inadvertent Actuation Block (IAB) features a spring-loaded pressure-differential system. The spring system in the ECCS valves needs to have a pressure difference of less than 1300 psid between the two media to be actuated. For example, if the reactor pressure is at 2000 psi, containment vessel pressure needs to be at 700 psi to be ECCS actuation even if ECCS signals are actuated (which are described above). Also, if reactor pressure can be reduced gradually, IAB releases at 950$\pm$ 50 psi. It is also noted that IAB does not interrupt valve opening for initial pressure of 900 psid or below. [4]

It is shown in Figure 12 how RSVs and RRVs are connected. In the model, they are linked to the Containment Top Branch component. Sensitivity studies for containment volume, power and pressure were conducted for SGTR without DHRS event. HCSG flow rate and heat transfer surface area sensitivity was conducted for achieving steady state flow rate in the steam generators.When a multiplier used for simulating HCSGs, oscillations occurred in the secondary side in the steady state runs. Due to that, instead, surface area increase method is used.

Design Conditions	**Imperial Unit Value**	**SI Unit Value**
Internal Design Pressure	1050 psia	7.24 MPa
External Design Pressure	60 psia	0.414 MPa
Design Temperature	550 °F	561 K
Design Maximum Containment Leakage	17.5 ft^3/hr	137 cm^3/s
UHS Pool Water Temperature	212 °F	373.15 K
Reactor Building Air Temperature	65 - 85 °F	292-303 K
Normal Operating Conditions (nominal)		
Internal CNV Pressure	0.09 psia	620 Pa
External CNV Pressure	60 psia	0.414 MPa
CNV Temperature (Atmosphere)	100 °F	311 K
UHS Pool Water Level	68 - 69 ft	20.7-21.03 m
UHS Building Elevation	93 - 94 ft	28.3-28.6 m
UHS Pool Water Volume	4 million gallons	15.14 million m^3
UHS Pool Water (Avg) Temperature	100 °F	311 K
Reactor Building Air Temperature	75 ±10 °F	297±6 K
Lowest Service Temperature	40 °F	277.6 K

Table 11: Containment and Reactor Pool Parameters

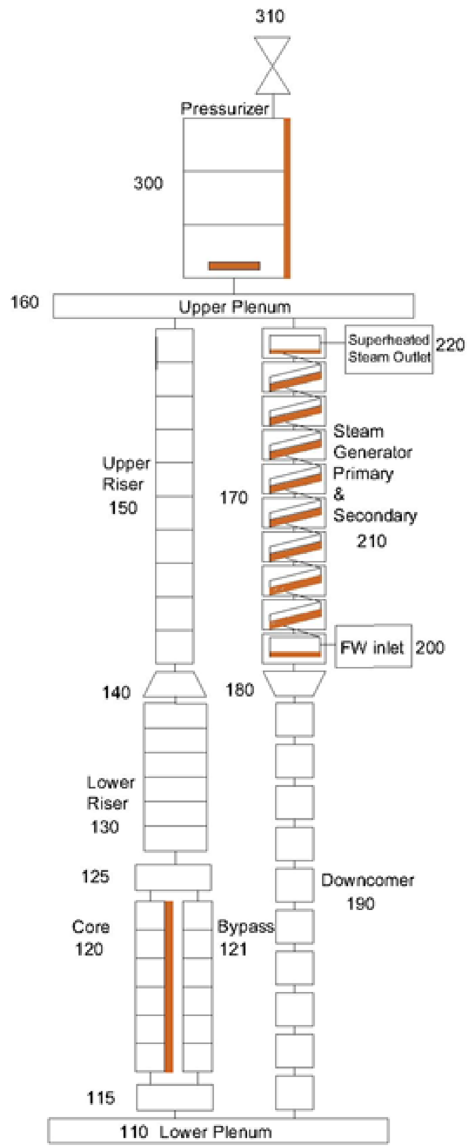


Figure 7: Early nodalization.

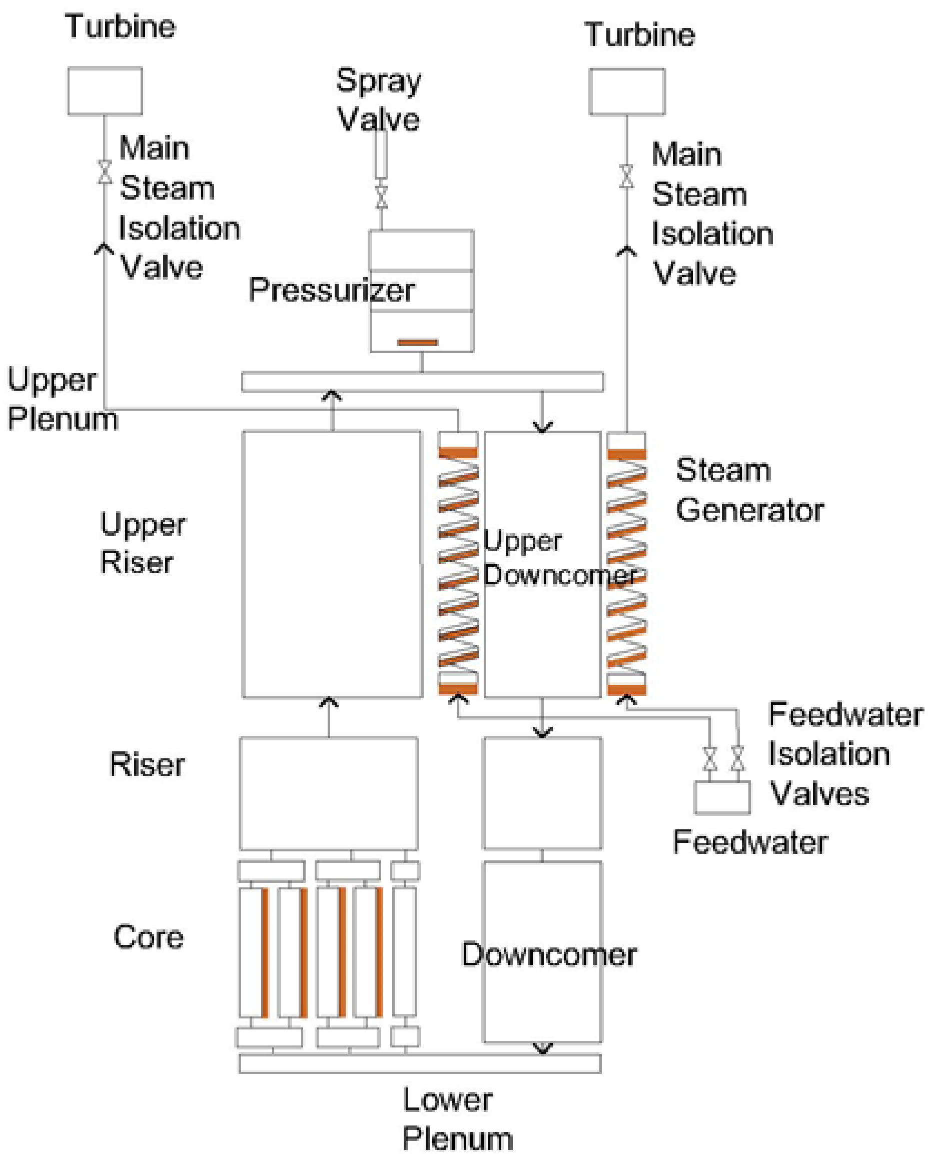


Figure 8: Improved nodalization for natural circulation stability.

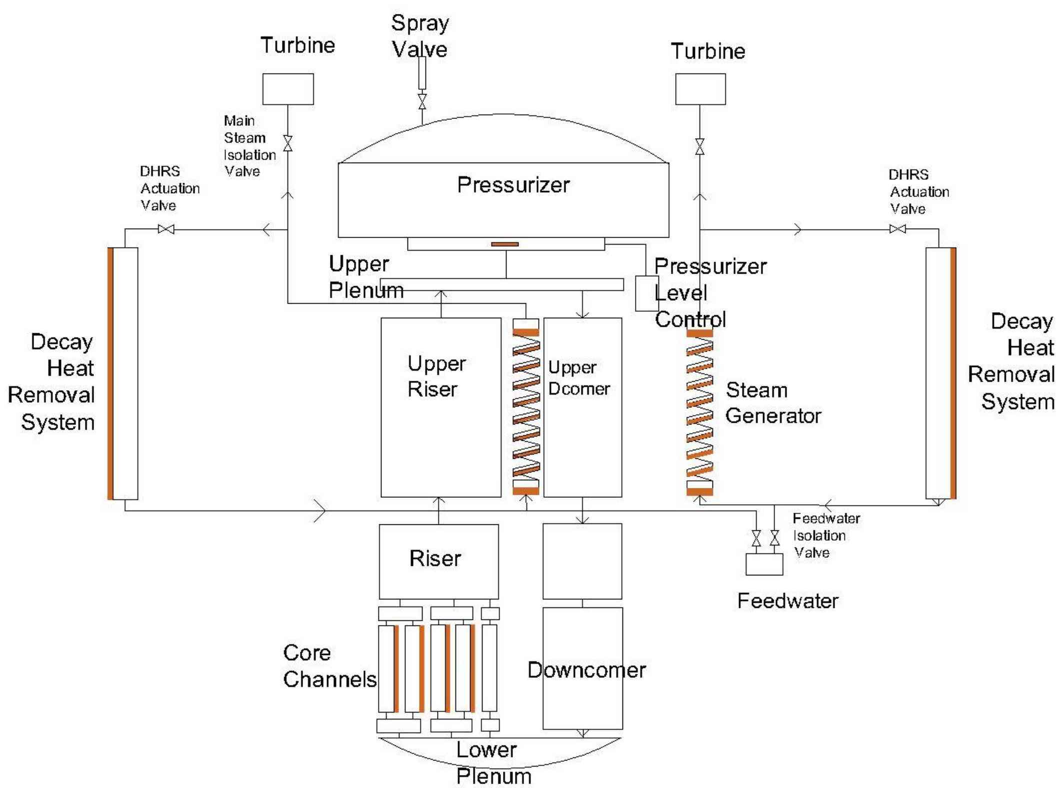


Figure 9: Nodalization with Decay Heat Removal System Added

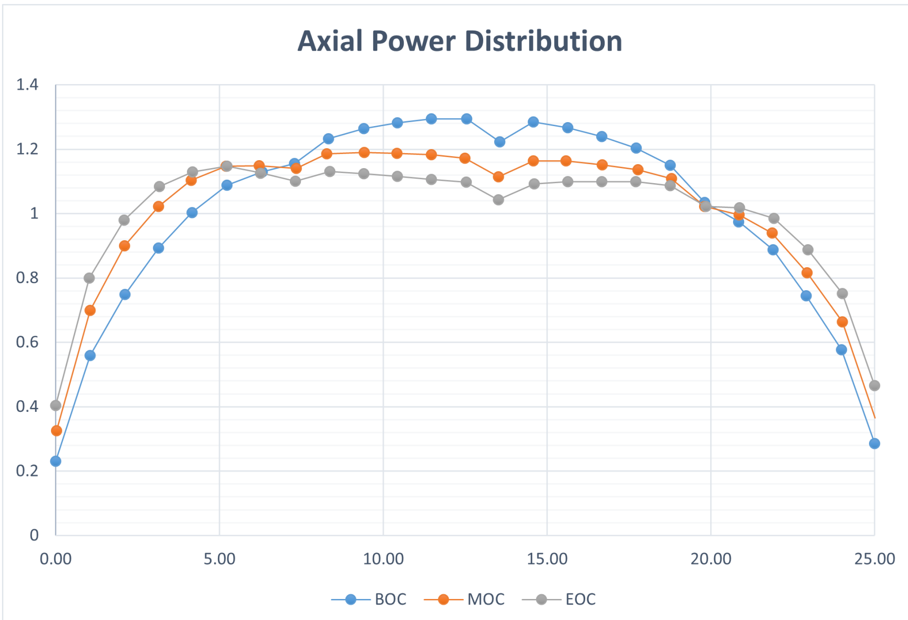


Figure 10: Axial power profile in the NuScale power module [2].

		0.895	0.911	0.888		
	1.033	1.137	0.957	1.135	1.033	
0.888	1.135	1.054	0.967	1.054	1.136	0.895
0.911	0.957	0.967	1.091	0.967	0.957	0.911
0.895	1.136	1.054	0.967	1.054	1.135	0.888
	1.033	1.135	0.957	1.137	1.033	
		0.888	0.911	0.895		

Figure 11: Radial power profile in beginning of cycle.

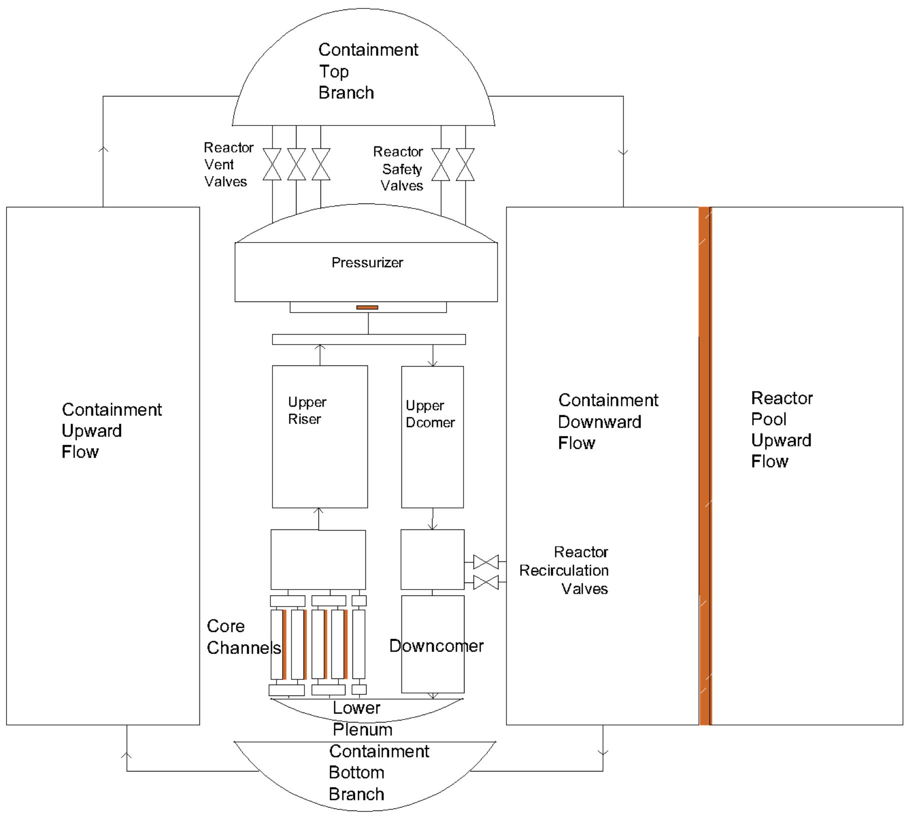


Figure 12: Containment vessel and ECCS modeling.

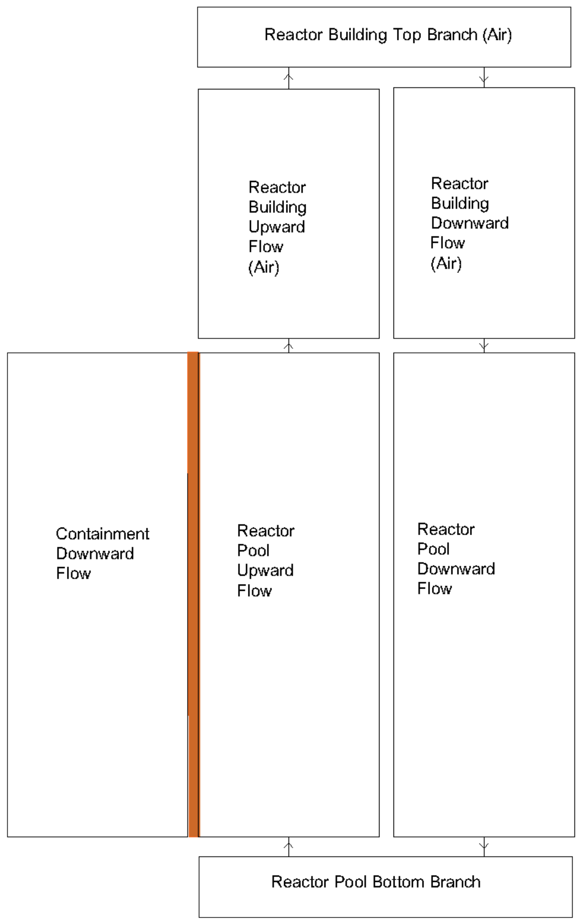


Figure 13: Reactor pool modeling.

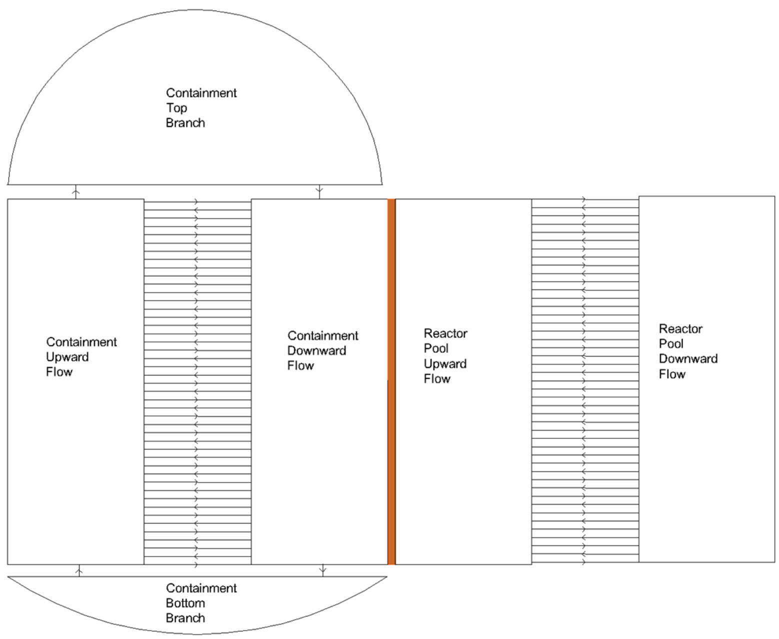


Figure 14: Containment and reactor pool multiple junction connections.

6 SGTR Events and SGTR Methodology

6.1 SGTR Events and Causes

This section describes SGTR events briefly, and also the model used for simulating SGTR event in the NuScale reactor. (Macdonald et. al, 1996) notes that in PWRs, SGTR accidents are the most common accidents.[11] A steam generator tube rupture event can be described as a crack in one or several of the tubes of a steam generator, causing the loss of the primary seal, which is one of the boundaries for the defense in depth concept. The size of the break, the location of the break, and the number of tubes involved are all important parameters to consider for the event. Causes of accidents have been investigated in the past, and several reasons for SGTR events are given below.

- Events of outer diameter stress corrosion cracking
- Flow induced vibration fatigue cracking
- Foreign objects in the tubes
- Wastage, fretting, denting, pitting of the tubes

To diminish the number of accidents or to mitigate the consequences, several precautions are taken, such as maintenance and inspection of secondary sides, tube inspections, as well as controlling water chemistry. If the tubes are deemed not fit for service, then a portion of the tubes may be plugged. This obviously decreases the total amount of flow going through the system and decreases heat transfer occurring between the two sides. On a side note, as stated earlier, HCSGs can withstand flow-induced vibrations better than U-tube steam generators.

Year	Plant	Location	Flow Rate (kg/s)	Rupture /Leak	Cause
1975	Point Beach 1	Wisconsin	7.875	Rupture	Wastage
1976	Surry 2	Virginia	20.79	Rupture	PWSCC
1979	Prairie Island 1	Minnesota	24.57	Rupture	Loose parts
1982	Ginna	New York	39.59	Rupture	Loose parts and tube wear
1987	North Anna 1	Virginia	37.8	Rupture	High cycle fatigue
1989	McGuire 1	North Carolina	31.5	Rupture	ODSCC
1993	Palo Verde 2	Arizona	15.12	Rupture	ODSCC
2000	Indian Point 2	New York	5.67	Rupture	PWSCC

Table 12: Some of the past steam generator tube rupture events [11]

Although these events appear to have occurred in the past, more recently the San Onofre Nuclear Generating Station located in California suffered from tube-to-tube wear in the replaced U-tube steam generators in the years of 2011 and 2012. [17] Two units were eventually permanently shut down by Southern California Edison, resulting in higher emissions and higher utility bills in the state of California.

6.2 SGTR Model

Modeling of the SGTR event and applied biases is defined in this section. The main objective of the model is to maximize primary side pressure and assess the event according to the acceptance criteria for non-LOCA events. The radiological consequences of mass release from primary side to secondary side as well as how much of a total mass could be released are not in the scope of this work. At first a model of such is thought

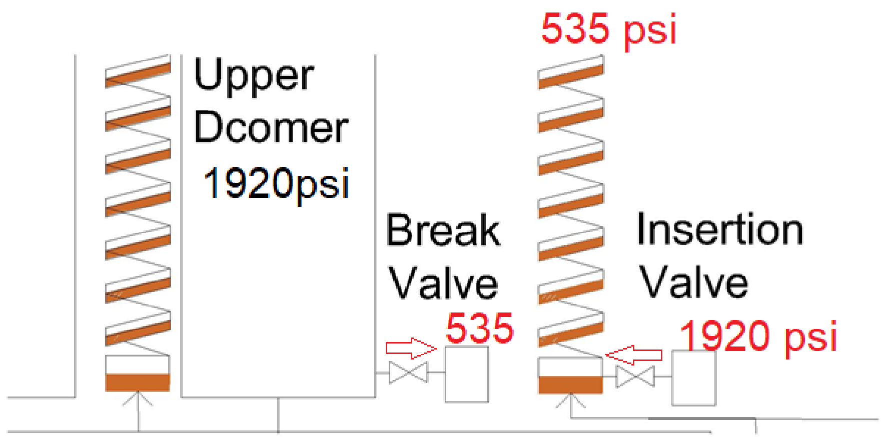


Figure 15: Break Valve RELAP Model for SGTR Event

and implemented for SGTR event in the NuScale, considering general Loss of Coolant Accident (LOCA) break modeling.

Break mass flow is diverted from Upper Downcomer's the most bottom volume, 201-20 to a tmdpvol component. A control system measures the mass flow of this and from another tmdpvol source term, same amount of mass with given temperature and pressure is added into Steam generators lowest section, Component 400's first volume. Connection of such system diagram is given in Figure 15. The reason for choosing the most bottom volume in this case would be that the pressure differential between both sides would be greater than at other locations that are connected.

Later, it is seen that there are discrepancies in the results, and model is changed to direct connection between the last volume of the upper downcomer and first volume

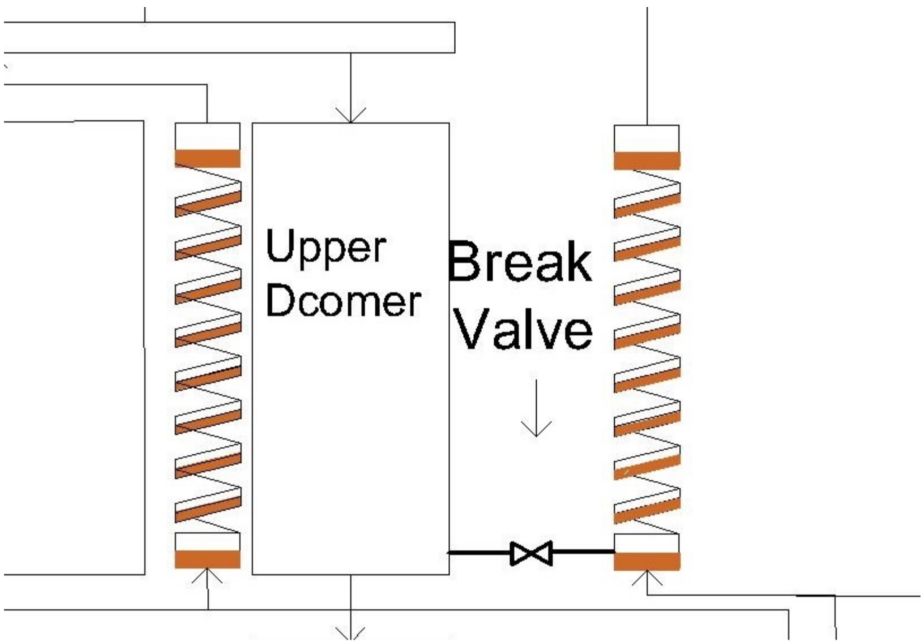


Figure 16: Changed break valve RELAP model for SGTR event

of the SG-1 which are at the same level as can be seen in Figure 16. With the break opening, it is also simulated that loss of AC power occurs at the same time, losing feedwater and turbine.

Description	Nominal	Bias
Core Power	160 MWt	163.2 MWt (+2%)
Pressurizer Pressure	1850 psia (12.76 MPa)	1920 psia (+70 psia)
SG pressure	500 psia (3.45 MPa)	535 psia (+35 psia)
Feedwater Temperature	300 F (422.04 K)	290 F (-10 F) (416.48 K)
Pressurizer Level	60%	68% (+8%)
Location of Break	-	Bottom of SG

Table 13: SGTR Biases

Several biases are applied at the start of the event. High core power is related to mass release in that it causes greater pressure differential between the main and secondary systems, resulting in a high break flow. Applying given biases are conservative in terms of main and secondary side effects. Higher pressurizer pressure results in a higher secondary side pressure, as well as delayed low pressurizer pressure trip.

A scram signal is created with the given below levels for different components,

- Pressure of pressurizer lower than 1600 psia, (2.0 seconds delay)
- Pressure of pressurizer higher than 2000 psia, (2.0 seconds delay)
- Pressurizer level lower than 35.0 percent, (2.0 seconds delay)

If any of the above occurs, the reactor is tripped with a given delay and the decay power curve is initiated for the reactor which can be seen in Figure 17. Decay heat power input is given in the FSAR, with the assumption of the highest-value control rod being stuck and not inserted in the core.

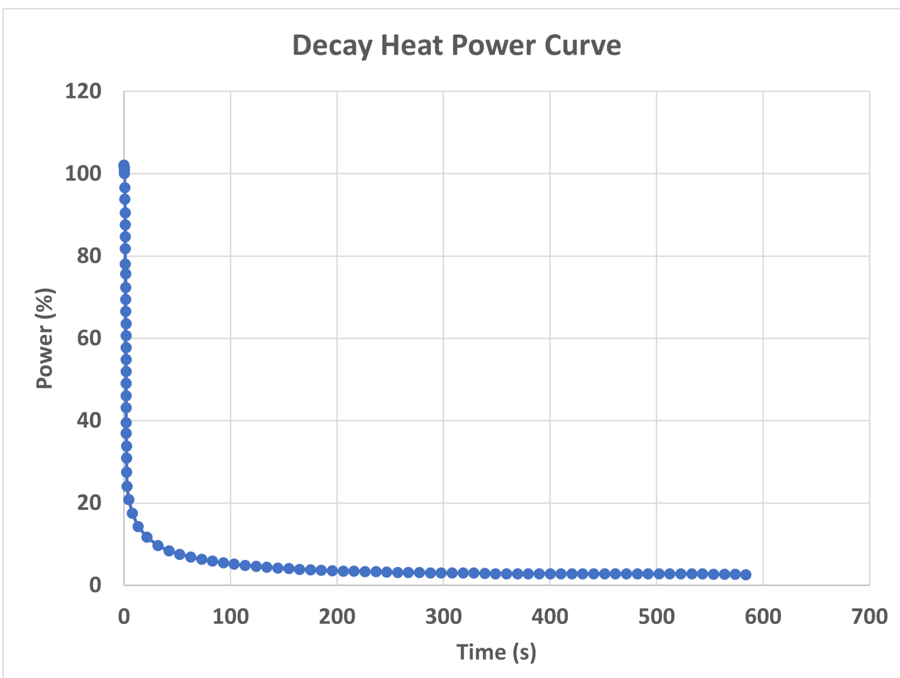


Figure 17: Decay heat power after reactor shutdown

In the SGTR event, turbine stop valves and feedwater isolation valves are activated with loss of AC power. Main steam isolation valves are postulated to be successfully closed after reactor trip. Pressurizer pressure control and level control systems are deactivated for the transient.

If this event is considered to challenge fuel integrity and DHRS does not successfully cool the system, there is a need to simulate ECCS systems as well. (Reactor Vent Valves, Reactor Recirculation Valves). It is expected that the DHRS will be able to cool down the reactor. Later, for a more severe scenario which is SGTR without DHRS, ECCS components are simulated.

A steady state file is run for several thousand seconds to establish stability in the system. Afterwards, using a restart file of the steady state file, a transient file is run. A restart problem is simply an extension of a previous calculation, beginning from the exact conditions present at a restart edit in that calculation. With given trip and initial conditions which are input above, for maximizing the primary side pressure, break size is chosen as 16% tube area split break, which is 0.00375 sq-ft.

A recent Masters thesis was published in the open literature for transient analysis of HCSG tube rupture in NuScale with RELAP5-3D by Johnson, P. Kyle, 2021. [9] Since the model used in the assessment is different in the sense of nodalization of the primary system, modeling of the DHRS system for tube rupture event in this model, modeling reactor core power in transient as well as modeling the pressurizer of the primary system, the findings of the study cannot be compared directly to the ones presented here.

7 Results and Discussion

7.1 Steady State Results

Steady state results are given below. Comparison with the reference values of the Final Safety Analysis Report is added with a table.

Parameter	Reference	Calculation
Core Power (MWth)	160	160
Pressure (MPa)	12.76	12.76
Primary flow rate (kg/s)	587.0	587.6
Core Inlet Temperature (K)	531.5	531.8
Core Outlet Temperature (K)	587.04	583.7
Secondary Inlet Temperature (K)	422.04	422.04
Secondary Outlet Temperature (K)	574.8	575.1
Secondary flow rate (kg/s)	67.06	74.5

Table 14: Steady state results and comparison to reference values.

In the steady state, core power and pressure are introduced to the system. The primary mass flow rate of 587 kg/s is obtained with a control system, which is the best-estimate primary flow rate in the FSAR. In a conventional PWR, a primary reactor coolant pump controls the primary flow rate. Since NuScale lacks primary RCS pumps, the flow resistance of the reactor's components is critical for flow rate. The core inlet and outlet temperatures are 531.8 K and 583.7 K. Temperature is 3 degrees cooler than the original 587 K. This is mainly due to the secondary side flow rate being increased. The secondary flow rate was increased because there was void fraction occurrence in the hottest assembly with the lower flow rates than 76.6 kg/s. This could be because HCSG specific phenomena such as heat transfer and pressure drop correlations are not included in RELAP5-3D. The model of HCSGs in RELAP5-3D was previously described in the model definition. For the primary mass flow rate, the calculation starts with the initial condition of 587 kg/s. There is a slight oscillation occurring in the mass flow rate at the start of event, though this is not important as the code tries to converge to a result with level control system, heaters, spray valve and so on.

The pressurizer pressure reaches to a stable value of 1850 psia(12.76 MPa) in the calculations. The secondary side inlet temperature is inserted into the system through a time-dependent volume with a time-dependent junction of 422 K. Coolant enters into the single inclined tube in the liquid phase. Steam generation begins around 40-50% of the way down the tube. Since steam is less dense than water, heat transfer decreases as the fluid travels through the tube. Steam temperature rises to 575 K at the tube exit. Both of the steam generators have the same mass flow rate of 37.25 kg/s in the steady state run.

7.2 Power Decrease

A power decrease event in the reactor and corresponding reactor parameters of core power, pressure, and primary mass flow rate are shown in Figures 18 and 19 to verify that the primary flow with natural circulation is functioning properly. While the reactor is operating at full power (160 MWth) for 10,000 seconds, power is reduced to 15% (24 MWth) in 1,000 seconds. Table 10 shows the power and associated primary flow rates. Figure 18 depicts the primary side pressure in the power decrease event. Figure 19 shows reactor power in the event. Although power decreases, the pressure control system is acting to stabilize the primary pressure at 1850 psia. There is a slight decrease in the pressure at the start of power decrease, and this is compensated by activating heaters. The mass flow rate initially stabilizes around 587 kg/s. After the power decrease, the flow rate decreases to 268 kg/s at the end of the event. In the power to flow rate table, the best-estimate flow rate is given as 280 kg/s for 15% power. The relative error of the flow rate is less than 5% in this case.

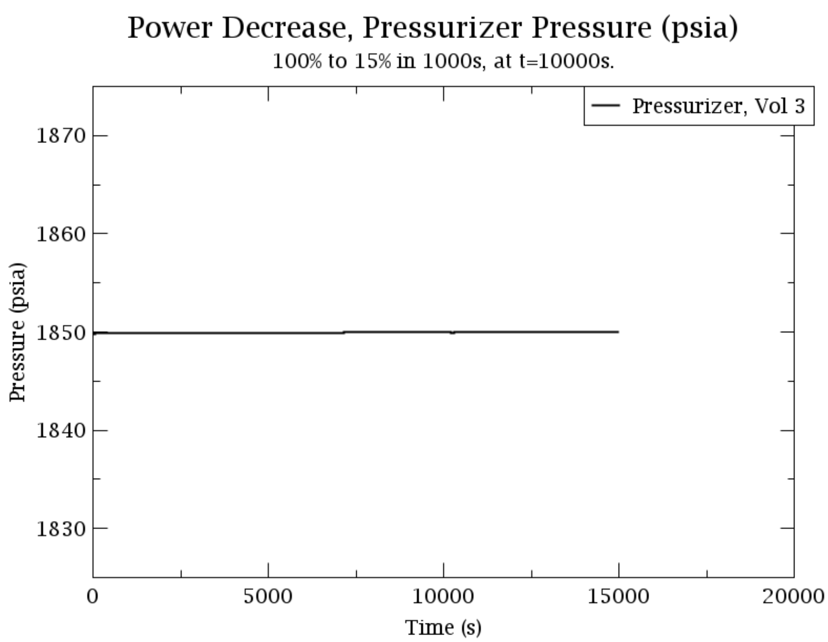


Figure 18: Primary pressure in a power decrease event.

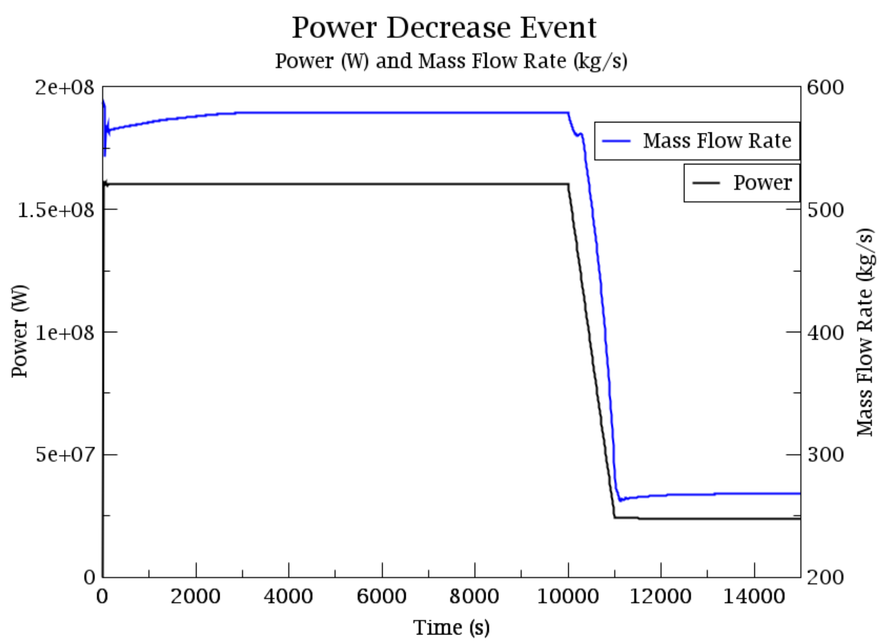


Figure 19: Primary flow rate in a power decrease event.

Event Time	Reference Time (s), NRELAP5	Calculation Time (s), RELAP5-3D
SGTR (16% tube area split break) at bottom of SG	0	0
Loss of AC Power	0	0
TSV Closure	0	0
High PZR Pressure (2000 psi)	6	5.5
DHRS Actuation (successfully opens in the reference values, DHRS1 fails in the calculations)	8	7.5
MSIV Closure Signal (successful)	8	7.5
Reactor Trip (successful)	8	7.5
Maximum RCS pressure	12	14.5
Maximum SG pressure reached	1385	1850

Table 15: Event Sequence Comparison for SGTR Event

7.3 Transient SGTR Results

In this section, transient SGTR results are shown with parameters that are heavily impacted in the event. A discussion of the results is also added in this part. The SGTR break occurs at $t = 0$ seconds, but shown in the Figures as $t = 3,000$ seconds. This is due to null transient section for 3,000 seconds to achieve steady state results. Transient results are an extension of the steady state file, thus $t = 3,000$ seconds in the figures depict $t = 0$ seconds in the transient. An event sequence comparison can be seen in Table 15.

Two of the most important parameters to depict are the primary and secondary side pressures in tube break event. Figure 21 shows the pressurizer pressure comparison with the reference values given in the FSAR. Pressure in the primary system initially spikes to a maximum of 2,158 psia in the reference results. With the closure of Turbine Stop valves, secondary side experience a heat transfer degradation. Water density increases and expands as heat removal decreases, resulting in an increase in primary pressure. In the calculations, at 5.5 seconds, the high pressurizer pressure limit trip initiates reactor trip, secondary side isolation, and module protection system activation with a two-second delay.

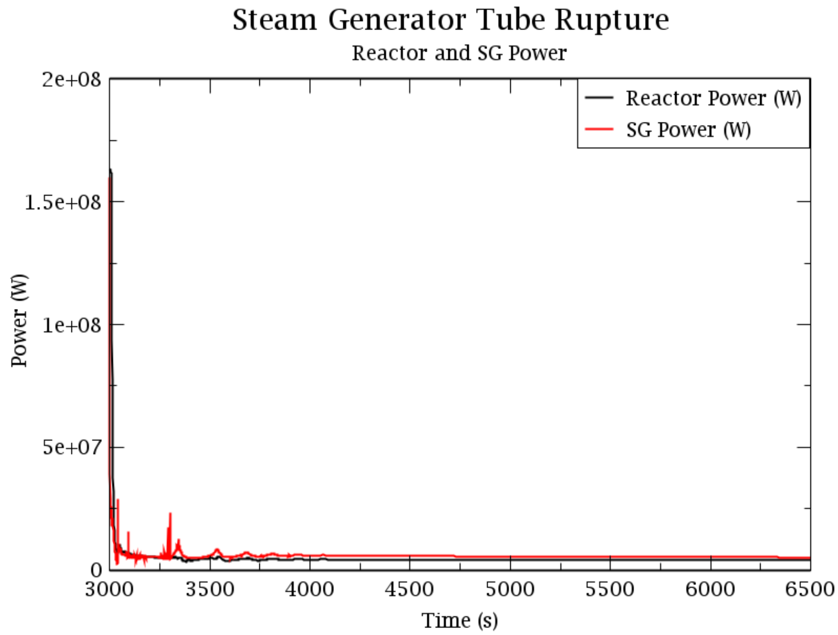


Figure 20: Reactor and steam generator power in a transient SGTR event.

Following a successful reactor trip, the power output rapidly decreases. The DHRS actuation valve is used to activate the trains, although the broken SGTR's DHRS actuation valve is assumed to fail open, but intact SG's DHRS successfully opens and steam from the outlet SGs is diverted to DHRS. Calculations reveal an initial pressure spike, reaching a maximum of 2,138 psia. When the intact DHRS is initiated, the pressure drops to around 1,900 psia and continues to fall. Primary pressure is slightly higher than reference values, this might be due to DHRS1 actuation valve being closed for the entirety of the event as well as secondary side heat removal discrepancies. The criterion for acceptance of the reactor pressure in the FSAR is specified as 120 % of the design pressure, or 2,520 psia.

The Secondary side pressure comparison is shown in the figure. The reference broken SG pressure shows a larger spike than calculations for broken SG at 1,575 psia

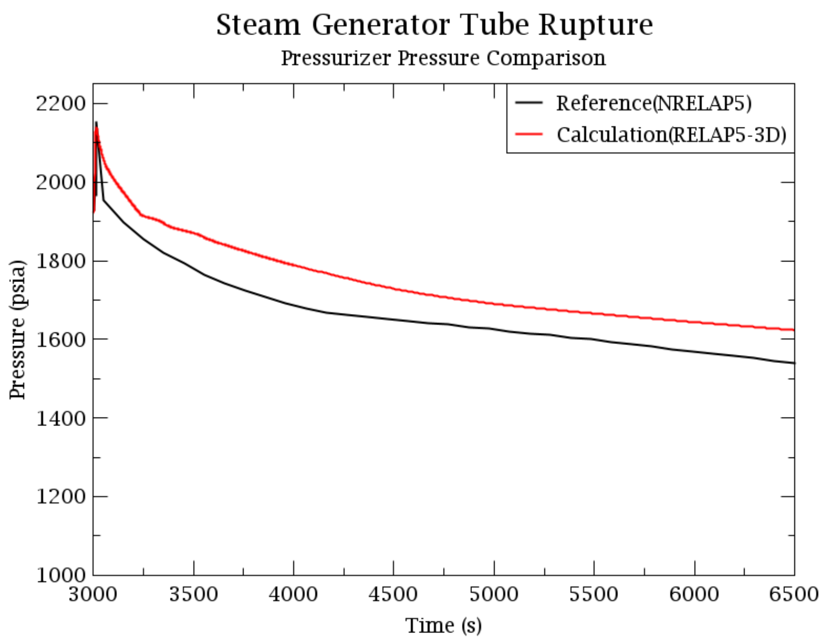


Figure 21: Pressurizer pressure in a transient SGTR event.

to 1,250 psia. The difference in the initial spike could be due to the increased mass flow rate of SGs, which is higher than the reference steady state values by more than 10%. The mass flow rate of SGs was increased in the steady state calculations to overcome the void occurrence in the core.

Another reason for the initial spike being larger in reference calculations could be the HCSG inside tube pressure correlations used in the NRELAP5 code, compared to the RELAP5-3D calculation. Intact SG pressures are depicted on the Figure 23. Pressure difference of 200 psia can be seen, while reference values show 1,375 psia, calculations point to 1,175 psia. Primary and broken SG pressures are equalized around 1,350 seconds in the reference values in the FSAR, compared to 1,850 seconds in the calculations. Although the slopes of the curves are similar, calculations show a higher-pressure level than reference values at the end of the event. The reason for

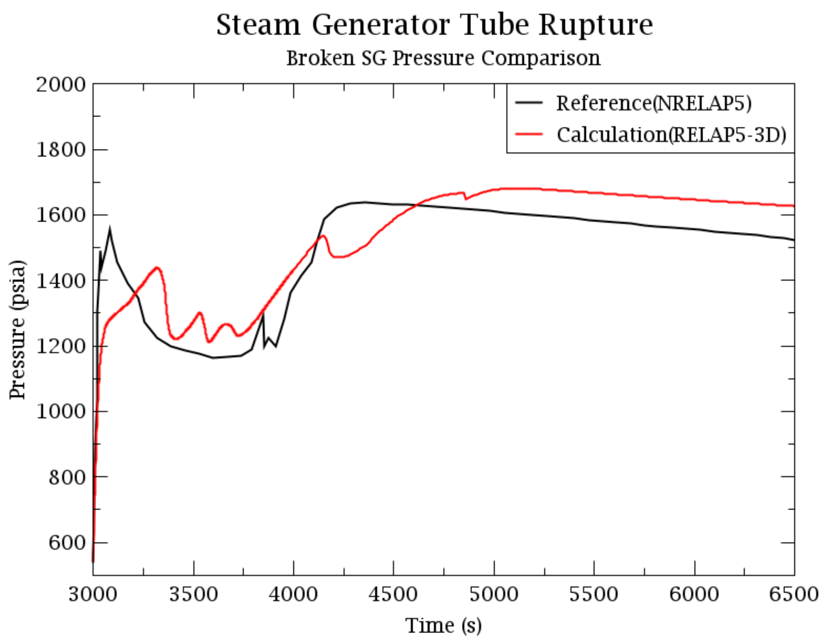


Figure 22: Faulted SG pressure in a transient SGTR event.

this difference could be due to the scenario used in the calculations where the DHRS1 actuation valve is never opened, and less heat removal occurs through the only intact DHRS2. In the reference values, faulted DHRS1 flow is able to transfer some heat to the reactor pool for around 750 seconds and can be seen in Figure 24.

Figures 24 and 25 depict the comparison of DHRS flow rates during the transient. While the reference intact DHRS shows a spike to 8 lb/s and stabilizes with a mass flow rate of around 5 lb/s to 4.0 lb/s at the end of event, the intact DHRS flow rate initially spikes at 15 lb/s and stabilizes at around 6.0 lb/s at the end. The broken SG's DHRS flow in the calculations is 0 lb/s as the DHRS1 actuation valve is closed throughout the event, whereas the faulted DHRS flow rate is spiking to 8 lb/s, continues with 4 lb/s at the start of event and diminishes to zero in 1,350 seconds. It is also seen in the reference values that DHRS flow starts 36 seconds after the event occurrence and 30

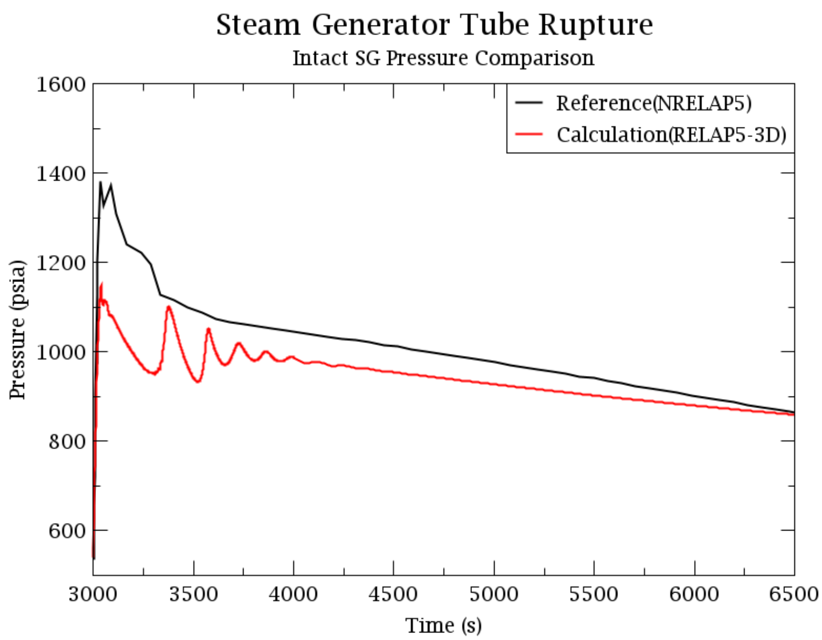


Figure 23: Intact SG pressure in a transient SGTR event.

seconds after the reactor trip signal. This is most likely due to DHRS activation taking 30 seconds to activate after the reactor trip signal is initiated, although in the event sequence it is given as 2 seconds following the reactor trip signal.

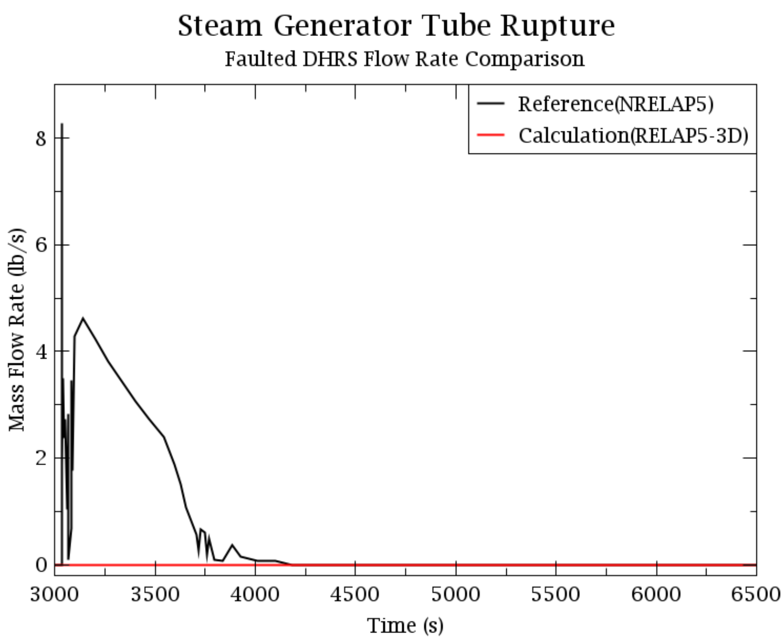


Figure 24: Faulted DHRS1 mass flow rate in a transient SGTR event.

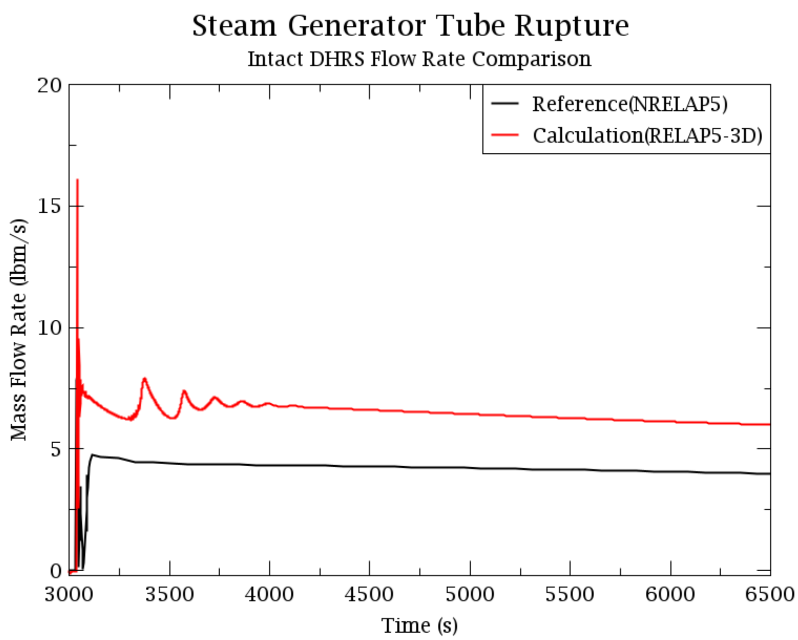


Figure 25: Intact DHRS mass flow rate in transient SGTR event.

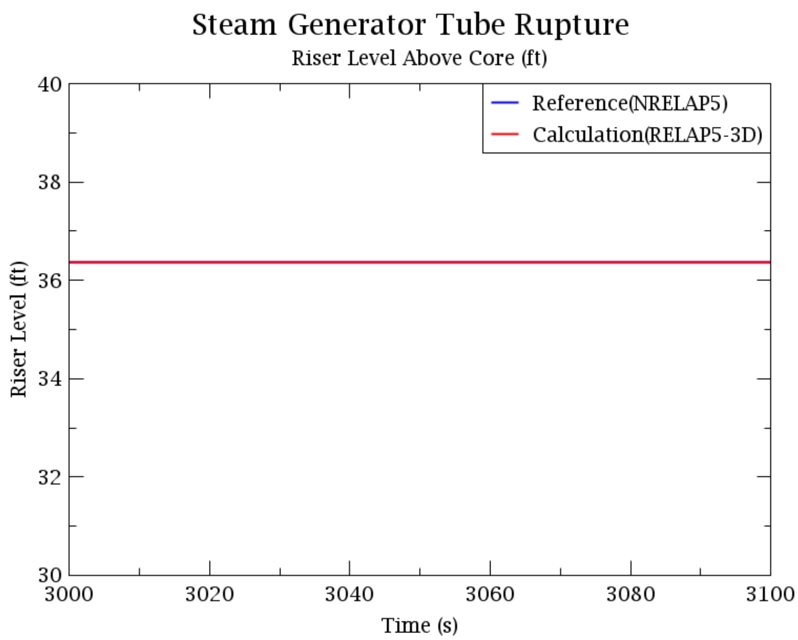


Figure 26: Riser water level in a transient SGTR event.

The water level above core does not decrease. A riser level comparison is shown in Figure 26 for the first 100 seconds of the event. Calculated peak clad temperature is given in Figure 27. The fuel average temperature comparison can be seen in Figure 28. Slight oscillations are visible in the calculated results and these are caused by primary flow rate changes.

Acceptance criteria for non-LOCA events are given in the FSAR as:

- Fuel integrity is not challenged by the event as water level above core is stable.

- Design pressure of primary side is 2,100 psia. Primary pressure should be maintained below 120% design value of 2,520 psia. In the event, pressure doesn't reach to 2,520 psia with the initial spike (2,138 psia) and safety of primary seal is not breached.

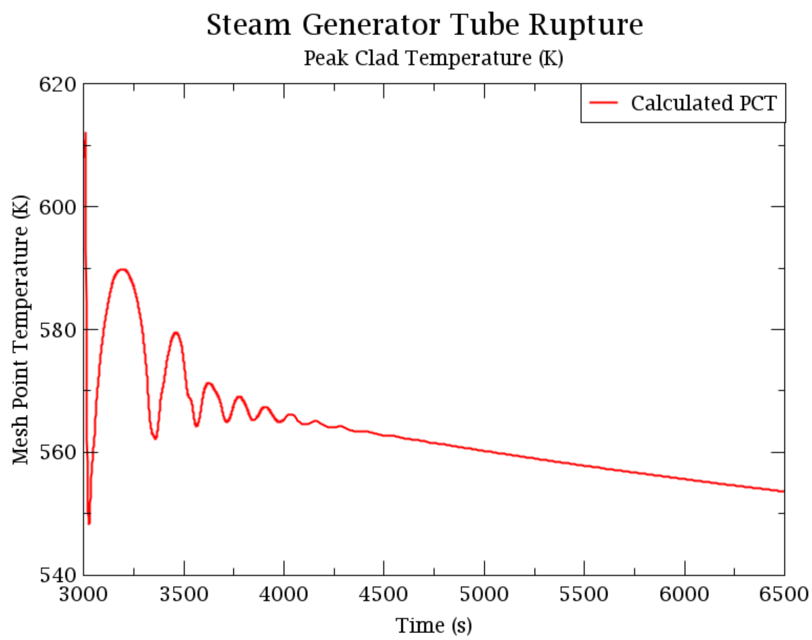


Figure 27: Peak clad temperature in hot channel in a transient SGTR event.

- Design pressure of secondary side is 2,100 psia. Again, the 120% design value of 2,520 psia is not reached in the steam pressure. Maximum pressure is seen after 1,850 seconds the event around 1,750 psia.

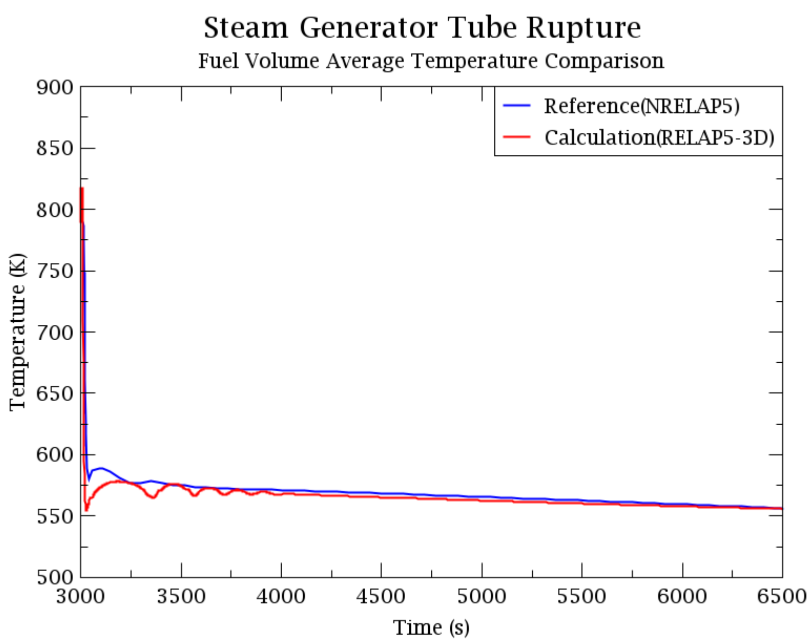


Figure 28: Fuel average temperature in a transient SGTR event.

7.4 Transient SGTR without DHRS Results

In this scenario, a transient run with postulated failure of both DHRS valves is simulated to observe ECCS response to the reactor. The closest scenario to this event in the FSAR is the Loss of Coolant Accident, although in the LOCA event, DHRS is available for the removal of decay heat and decreasing pressure on the RPV. In this chosen scenario, the pressure on the primary side steadily increases due to decay heat, and pressure can only be released in small amounts through reactor safety valves. In 10 CFR 50.46, the acceptance criterion for emergency core cooling systems for light-water nuclear power reactors is given as the following for the ECCS systems. Also in 10 CFR 50, Appendix K, ECCS Evaluation Models are specified.

- The calculated maximum fuel cladding temperature must not exceed 2,200 °F (1477.6 K),
- The calculated maximum total oxidation of cladding must not exceed 17% of the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam must not exceed 0.01 times the hypothetical maximum amount that could be generated.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period.

The event sequence is given in Table 16. The event starts with a single tube break in the bottom of the SGs with the consequent loss of AC power, similar to earlier section with break having 16% tube area. Only this time, both DHRS valves fail, leading to

Event Time	Calculation Time (s), RELAP5-3D
SGTR (16% tube area split break) at bottom of SG	0
Loss of AC Power	0
TSV Closure	0
High PZR Pressure (2000 psi)	6.6
MSIV Closure Signal (successful)	8.6
Reactor Trip (successful)	8.6
Maximum RCS pressure	11.0
DHRS Actuation (both DHRS fails)	36.6
CNV High Pressure Signal	3433
Low Riser Level Signal	6264
CNV High Water Level Signal	11605
IAB threshold lifted (ECCS Activation)	14647
Maximum CNV pressure	14652
Minimum Riser Level	14652
End Time	19000

Table 16: SGTR without DHRS event sequence.

an increase in pressure in the primary side, with no heat removal systems available. The reactor trips successfully with high pressurizer pressure trip and power decreases to decay heat levels. The Reactor Safety Valves with pressure set points of 2,075 and 2,100 psia are expected to release the excessive pressure to the Containment Vessel until one of the ECCS signals is activated through trips. However, a threshold is put to interrupt ECCS valves opening while reactor pressure is still at higher levels, which is explained previously in the model definition section for ECCS valves. Steam which is condensed through CNV walls that is in contact with reactor pool water, starts being collected at the bottom of the CNV and leads to increase in the CNV water level. In normal operation, the CNV is emptied and heat transfer between CNV and RPV is negligible. In this case, heat structures are added between downcomer and lower plenum section of the primary side, connected to upward flow CNV, considering the heat transfer occurring between collected water at the bottom which is touching to outer wall of RPV.

The Reactor Safety Valves mass flow rate is shown in Figure 29. Mass flow release from a single RSV with 12 lb/s and around 15 lb/s in some occasions, can be seen in the

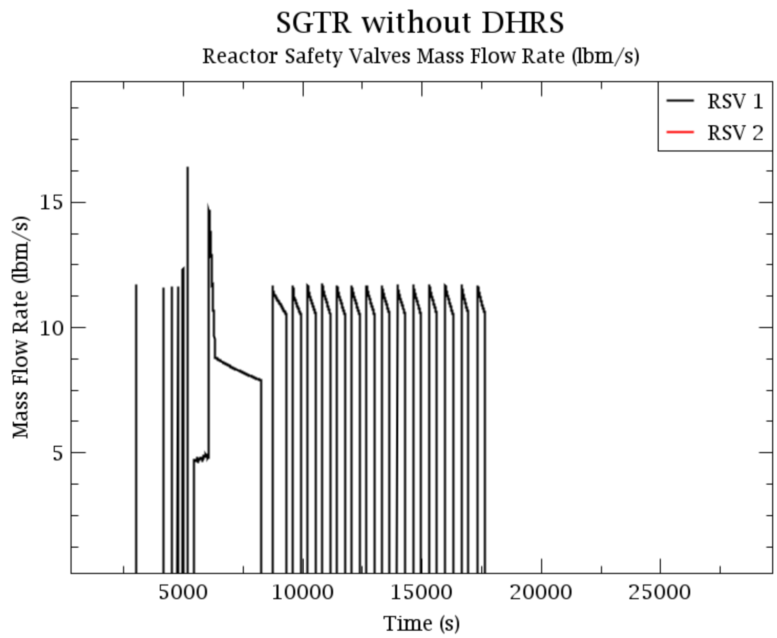


Figure 29: Reactor safety valve flow rates in a transient SGTR without DHRS event.

same figure when the pressure reaches to the set point of 2,075 psi and the valve stays open until pressure is reduced to 1,867 psi. A continuous release from the primary side is visible, due to the pressure increasing steadily from the void occurrence in the core due to decay heat. Void occurrence in the core is visible in Figure 30, as the void fraction is greater than 0.0.

As the primary water level inside the RPV starts decreasing, a low riser signal occurs around 6,250 seconds. As the water level decreases, the minimum level of water above core is calculated as 1.2 ft at around 14,650 seconds. With the actuation of ECCS signals and the Inadvertent Block Valve pressure-differential being less than the threshold value of 1,300 psid, maximum containment pressure in the calculations is 1,030 psi, where the design limit of containment pressure is 1,050 psi. Before and after release of RVV and RRVs, containment and RPV pressures can be seen in Figure 31

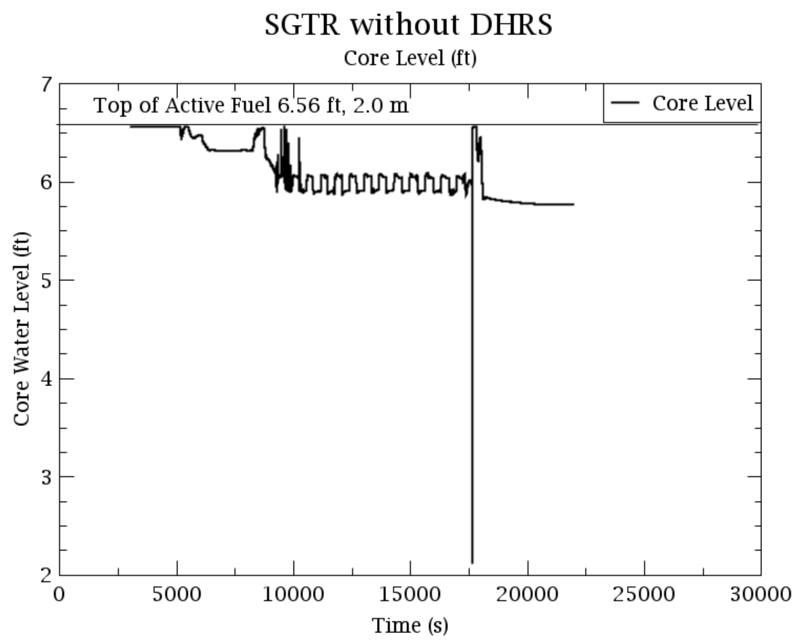


Figure 30: Core water level in a transient SGTR without DHRS event.

and Figure 32. The Containment water level and riser water level are shown in Figures 33 and 34. The core water level is tracked, and is shown in Figure 30. The maximum containment wall temperature is shown in Figure 35. The maximum containment wall temperature obtained in the results was 543.8 K, whereas the design wall temperature as given in the FSAR is 561 K. RVV and RRV mass flow rates are shown in Figures 36 and 37. The Reactor Vent Valves release more pressure to the Containment and decrease primary coolant inventory at around 14,647 seconds. The presence of water above the top of the core during the transient protects the core from CHF occurrence and eliminates the need to calculate zircaloy oxidation in the core. Around 17,500 seconds in the figure, equivalent to 14,500 seconds in the transient time, a sharp line occurs at the moment of ECCS activation. This is due to a massive amount of mass released from the primary side to containment, almost equalizing the inside and outside

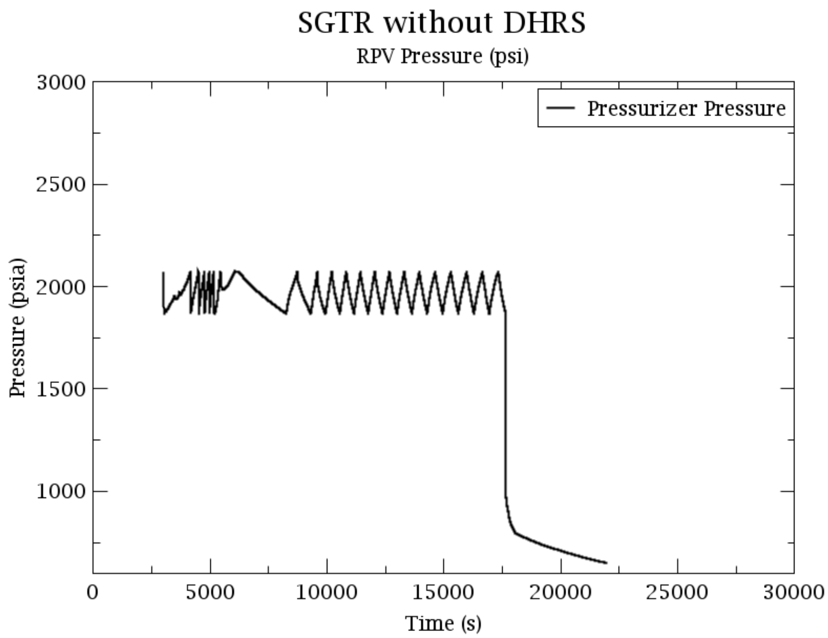


Figure 31: Pressurizer pressure in transient SGTR without DHRS event.

of RPV water. The slight difference in CNV and RPV water levels provides the natural circulation after ECCS actuation and long term cooling of the reactor.

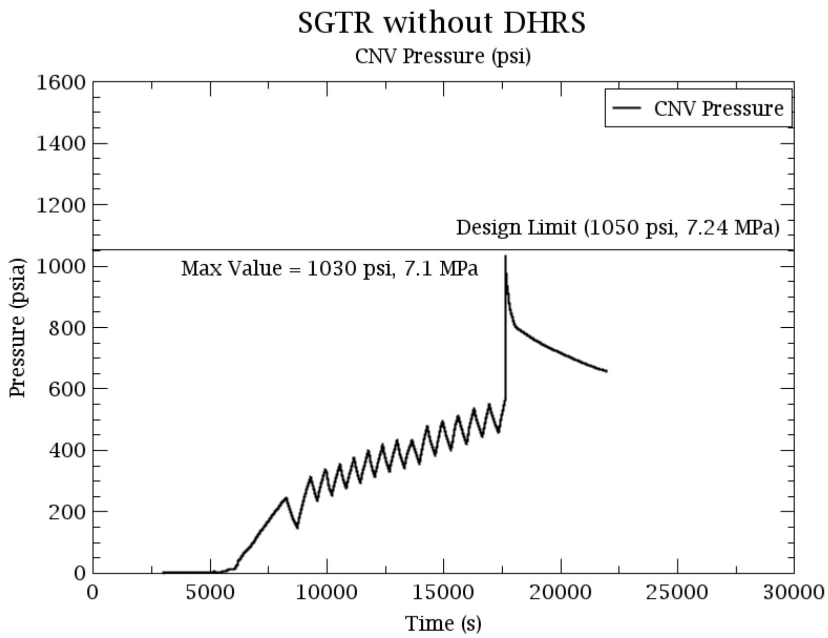


Figure 32: Containment pressure in a transient SGTR without DHRS event.

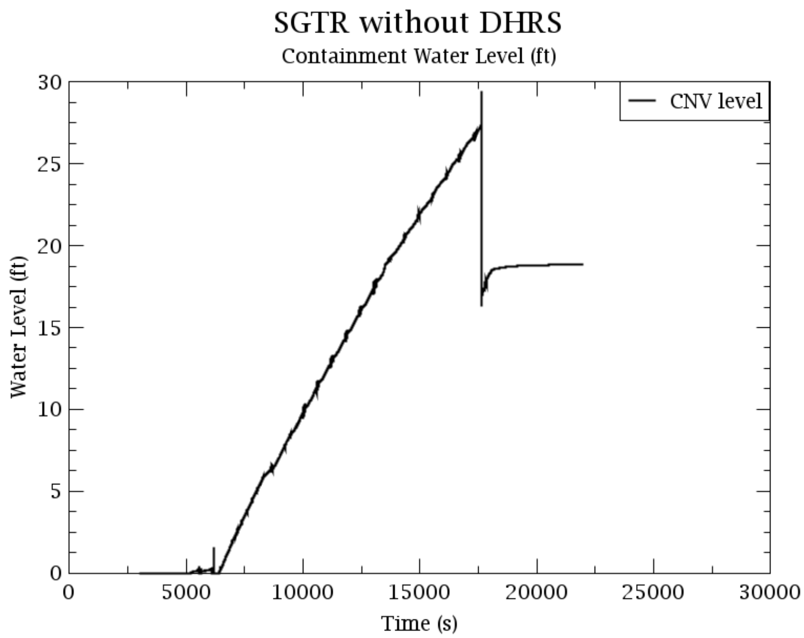


Figure 33: CNV level in a transient SGTR without DHRS event.

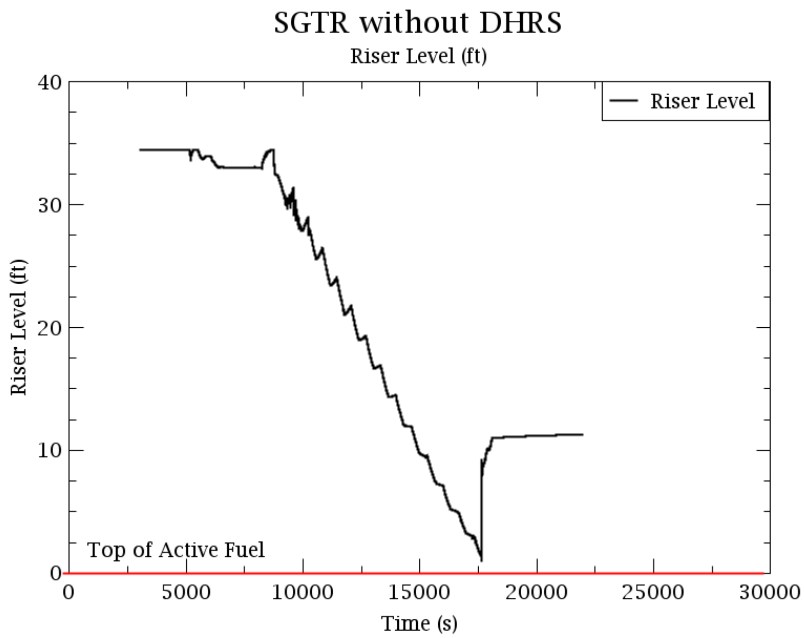


Figure 34: Riser level in a transient SGTR without DHRS event.

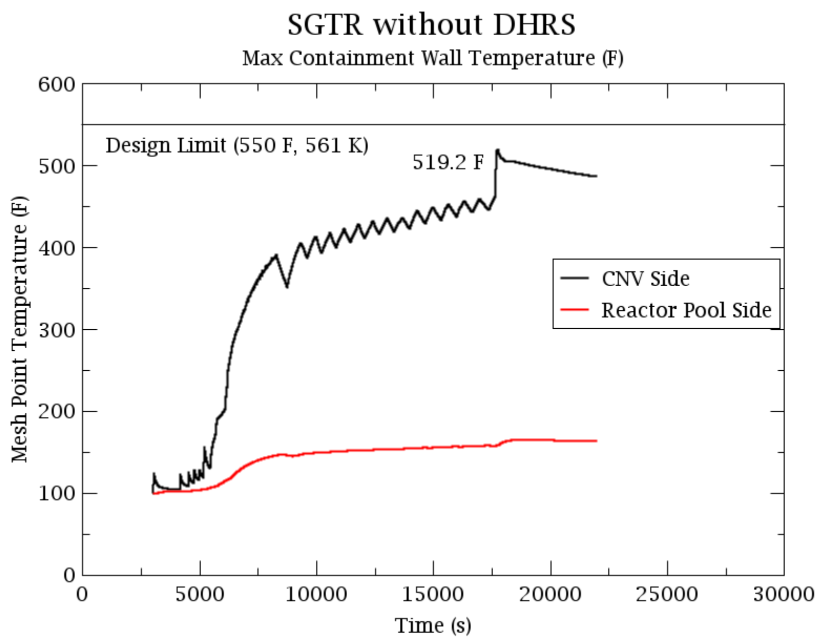


Figure 35: Containment wall temperature in a transient SGTR without DHRS event.

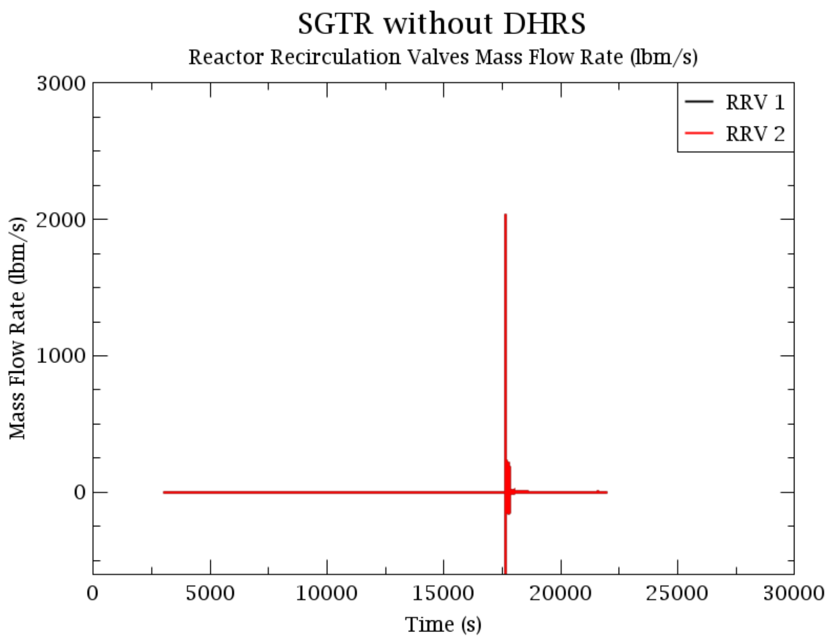


Figure 36: Reactor recirculation valve flow rates in a transient SGTR without DHRS event.

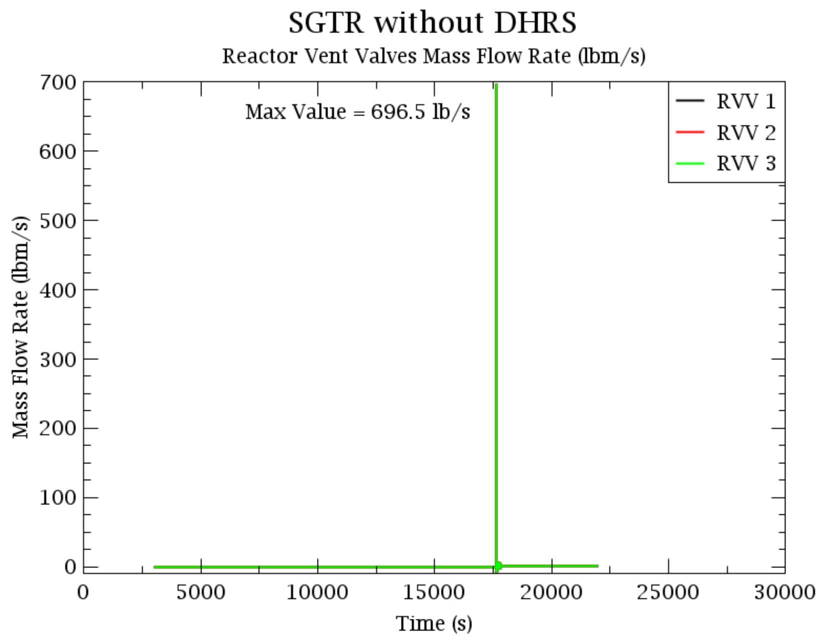


Figure 37: Reactor vent valve flow rates in a transient SGTR without DHRS event.

8 Conclusions

The main objective of this work was to assess a postulated Steam Generator Tube Rupture event in the NuScale SMR with evaluation of passive Decay Heat Removal System, using the industrially renown RELAP5-3D code. A model was developed based on the available data given in NuScale's Final Safety Analysis Report. Its core is modeled with three average channels, each having 12 assemblies and one hot channel having a single assembly with radially highest power fraction. HCSGs were modeled as double, lumped, inclined tubes with equivalent hydraulic diameter and flow area.

Steady state conditions were achieved and a power decrease event was simulated with the model for observing natural circulation flow rate of the system. Although there were no oscillations in the power decrease event, reactor shutdown and tube rupture caused oscillations in the primary mass flow rate. The break location of the single tube break with 16 percent flow area was chosen at the bottom of one of the steam generators. Moreover, several biases were applied in the start of transient, such as an increase in operating and secondary pressure, a decrease in feedwater temperature and an increase in pressurizer level. This increases the pressure differential between RPV and secondary side and causes the initial spike to be larger.

In the tube rupture event where single DHRS were available the simulation showed that the broken steam generator's initial pressure spike was slightly under-predicted and long-term pressure decrease was over-predicted with respect to the FSAR. Fuel average temperature and peak clad temperature were not challenged in the event. The intact DHRS mass flow rate was calculated as 6.5 lb/s compared to the reference value of 4.5 lb/s. Radiological consequences of an SGTR event and mass releases from the secondary side were not in the scope of this work. Design pressures for the primary and secondary sides were not challenged as the means of heat removal was sufficient for such an event.

A severe accident was also simulated in the absence of a decay heat removal system in order to observe the reactor's ECCS reaction to a severe accident. Although more sensitivity tests may be necessary, the containment pressure and wall temperature were predicted to be both below design levels. During the transient, the water in RPV was still above the core level when the maximum containment pressure occurred.

9 Recommendations

Several challenges were encountered during this work. The secondary side mass flow rate had to be increased from 67 to 74.5 kg/s to overcome void occurrence in the core in the steady state results. The higher mass flow rate in SGs may impact transient response of the system in the tube rupture event.

Modeling of the natural circulation proved challenging during transients. Modeling natural circulation may be improved in RELAP5-3D in the future works to establish a more stable flow rate after a transient.

HCSG specific correlations may also be incorporated and examined with existing experimental data to see improvements on modeling HCSG with RELAP5-3D. Also, oscillations were encountered in the secondary side flow while trying to implement a multiplier for increasing heat transfer efficiency of HCSGs. Instead, a surface area increase in the heat structures was implemented.

Valve flow areas of the RRV and RVVs might be different than what the original design has, which may indicate that the mass flow rates also be different from the design of the reactor. Containment modeling, especially flow obstructions between reactor pressure vessel and containment vessel may need further sensitivity studies. Volume of containment is significant for the reason it describes the level of containment and water level of riser in the transient.

Trip delays may be put into sensitivity studies, such as MSIV, FWIV closure, DHRS actuation.

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