TRANSIENT ANALYSIS OF THE NUSCALE POWER HELICAL-COIL
STEAM GENERATOR TUBE RUPTURE USING RELAP5-3D

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TRANSIENT ANALYSIS OF THE NUSCALE POWER HELICAL-COIL STEAM GENERATOR TUBE RUPTURE USING RELAP5-3D

BY

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A THESIS SUBMITTED IN PARTIAL FULFILLMENT OF THE REQUIREMENTS FOR THE DEGREE OF

MASTER OF SCIENCE

IN

MECHANICAL ENGINEERING AND APPLIED MECHANICS

UNIVERSITY OF RHODE ISLAND

2021
MASTER OF SCIENCE THESIS

OF

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2021
ABSTRACT

Nuclear energy is the largest source of carbon-free electricity in the United States, making up 20 percent of the electricity generated in the United States. The United States is the second-largest energy-consuming country globally, with fossil fuels being the largest electricity-producing source. With Climate Change at the head of the world’s most difficult circumstances, it is evident that nuclear power is a crucial and significant source of carbon-free energy to combat this crisis. The NuScale Power SMR can provide a cost-effective and safe solution to further the expansion of nuclear energy throughout the United States and the world. The nature of the buoyancy-driven natural circulation cooling design of the reactor primary systems and the modularity and scalability power plant system provide the innovations and technology needed to do so. There are few tools like RELAP5-3D that allow for the thermal-hydraulic transient analysis of nuclear reactors. Due to the minimum amount of open literature available on the transient analysis of the NuScale Power SMR, RELAP5-3D has been utilized to perform the steady-state and a steam generator tube-rupture transient calculation. The benchmark experiment for thermal-hydraulic calculation codes, called Edward’s pipe blowdown experiment, was first modeled to understand the basics of a transient two-phase flow model. This experiment was performed to acquire the essential modeling skills and techniques to build the model and perform the calculations of the NuScale Power SMR using RELAP5-3D.

The NuScale Power Small Modular Reactor (SMR) relies on buoyancy-driven natural circulation cooling to cool the reactor core and extract thermal energy for electricity generation. The natural convection phenomenon has been of research interest for
many years. NuScale Power LLC has only developed the SMR in recent years, and this integral Pressurized Water Reactor (iPWR) is the first nuclear reactor to utilize this phenomenon. Therefore, there is an increased interest in performing the transient analysis of the thermal-hydraulics of this reactor to understand conditions in which the natural circulation cooling inside the reactor system may be disrupted. There have been minimal published resources on this topic to date, making this research necessary for the growth and future of SMRs and natural circulation cooling of nuclear reactors. The innovations and designs of the NuScale Power SMR have allowed for enhanced safety, cost, scalability, modularity, time of construction, ease of transportation, and standardized manufacturing process of SMRs and nuclear power plants. RELAP5-3D was utilized to develop the model of the NuScale Power SMR and perform steady-state and transient analysis calculations of the reactor. This model was developed using the publicly available design data and parameters released by the U.S. NRC for the NuScale Power Final Safety Analysis Report (FSAR). The steady-state conditions of the reactor were modeled to simulate the reactor operation conditions in preparation for the transient analysis calculation. A tube rupture of the secondary steam generator was simulated for the transient analysis calculation to understand if the natural circulation cooling would be disrupted and if the secondary coolant would rise to dangerous levels proposing system failures.

The steady-state model simulated the proper reactor operational conditions, exhibiting higher mass flow rates than the best estimate flow rate specified in NuScale FSAR. The core temperatures were on the higher end of the temperature range but were still within the operational conditions, with the pressure controlled at 1850 psia. The
forward flow energy loss coefficients proposed a particular issue in manipulating the code to obtain the core's correct mass flow rates and temperatures. It was found that the loss coefficients could be changed in a manner that lowered the mass flow rates closer to the best estimate flow rate, but the temperatures would, in turn, increase. Because the mass flow rate specified in NuScale FSAR was the best estimate value, the author concluded that the steady-state model was sufficient for the tube rupture model. The tube rupture was modeled using a single junction that connected the primary and secondary steam generators. The model was created to simulate a single helical coil steam generator tube being ruptured. Depressurization was not seen on the primary because the pressurizer was modeled as a pressure boundary condition at 1850 psia. A mass flow rate of approximately 36 lbm/s was seen through the tube rupture to the secondary side of the system. The water level did not increase significantly, but the liquid void fraction increased slightly. The flow through the rupture was choked because of the flashing of the liquid at high temperature and pressure to vapor at the lower pressure. It was found that instabilities and oscillations occurred very quickly on the primary and secondary sides, but the natural circulation flow was not disrupted.
ACKNOWLEDGMENTS

The resources needed to perform this research came from multiple different locations and sources. An enormous thank you would like to be given to Dr. Nassersharif of the Mechanical, Industrial, and Systems Engineering department at the University of Rhode Island, for the constant supervision, insight, and advisement throughout the entire process. I would also like to acknowledge Patrick Freitag, a graduate student before me who initially modeled the NuScale SMR using RELAP5. Patrick’s model of NuScale Power’s SMR was beneficial in assisting with the modeling process and facilitated this research and thesis. I would also like to acknowledge the U.S. Department of Energy (DOE) and June Cook of the Idaho National Laboratory (INL) for assistance in obtaining a license for the RELAP5-3D code. The author would like to thank Doreen Jones and Ken Jones of Applied Programming Technology, Inc. (APT) and Chester Gingrich of the NRC for assisting in obtaining the no-cost educational institution license to the AptPlot software. Finally, acknowledgment would like to be provided to the United States Nuclear Regulatory Commission (NRC) and NuScale Power LLC in providing the crucial and essential design parameters, data, conditions, and tables of the NuScale Power SMR, along with information needed for the Edwards Pipe Blowdown Experiment to complete this research professionally.
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<tr>
<td>SMR</td>
<td>Small Modular Reactor</td>
</tr>
<tr>
<td>NPM</td>
<td>NuScale Power Module</td>
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<tr>
<td>HCSG</td>
<td>Helical Coil Steam Generator</td>
</tr>
<tr>
<td>NSSS</td>
<td>Nuclear Steam Supply System</td>
</tr>
<tr>
<td>iPWR</td>
<td>integral Pressurized Water Reactor</td>
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<tr>
<td>PWR</td>
<td>Pressurized Water Reactor</td>
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<tr>
<td>LCOA</td>
<td>Loss of Coolant Accident</td>
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<tr>
<td>NRC</td>
<td>Nuclear Regulatory Commission</td>
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<tr>
<td>DOE</td>
<td>Department of Energy</td>
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<tr>
<td>DOE-NE</td>
<td>DOE Office of Nuclear Energy</td>
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<td>DOE-NR</td>
<td>DOE Office of Naval Reactors</td>
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<tr>
<td>INL</td>
<td>Idaho National Laboratory</td>
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<tr>
<td>FSAR</td>
<td>Final Safety Analysis Report</td>
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<tr>
<td>DCA</td>
<td>Design Certification Application</td>
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<tr>
<td>EIA</td>
<td>Energy Information Administration</td>
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<tr>
<td>IPCC</td>
<td>Intergovernmental Panel on Climate Change</td>
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<tr>
<td>RELAP</td>
<td>Reactor Excursion and Leak Analysis Program</td>
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<tr>
<td>IRUG</td>
<td>International RELAP Users Group</td>
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<tr>
<td>SNAP</td>
<td>Symbolic Nuclear Analysis Package</td>
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<tr>
<td>APT</td>
<td>Applied Programming Technology</td>
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<tr>
<td>OSU</td>
<td>Oregon State University</td>
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<tr>
<td>Acronym</td>
<td>Description</td>
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<tr>
<td>MASLWR</td>
<td>Multi-Application Small Light Water Reactor</td>
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<td>US-EPR</td>
<td>U.S. Evolutionary Power Reactor</td>
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<tr>
<td>US-APWR</td>
<td>U.S. Advanced Pressurized Water Reactor</td>
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<tr>
<td>ECCS</td>
<td>Emergency Core Cooling System</td>
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<tr>
<td>DHRS</td>
<td>Decay Heat Removal System</td>
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<tr>
<td>RPV</td>
<td>Reactor Pressure Vessel</td>
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<tr>
<td>CNV</td>
<td>Containment Vessel</td>
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<td>CRAs</td>
<td>Control Rod Assemblies</td>
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<td>UHS</td>
<td>Ultimate Heat Sink</td>
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<td>MPS</td>
<td>Module Protection System</td>
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<td>MSIVs.</td>
<td>Main Steam Isolation Valves</td>
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<td>FWIVs.</td>
<td>Feedwater Isolation Valves</td>
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<td>U-235</td>
<td>Uranium 235</td>
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<tr>
<td>DCF</td>
<td>Discounted Cash Flow</td>
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<td>NOAK</td>
<td>Nth-Of-A-Kind</td>
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<tr>
<td>CCGTs</td>
<td>Combined-Cycle Gas Turbines</td>
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<td>LTO</td>
<td>Long-Term Operation</td>
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LIST OF UNITS

MWt    Megawatt Thermal
MWe    Megawatt Electric
MW     Megawatt
W      Watt
PSIA   Pounds per Square Inch Absolute
MPa    Megapascal
kPa    Kilopascal
m³     Cubic Meter
m²     Square Meter
m      Meter
mm     Millimeter
°F     Degrees Fahrenheit
°C     Degrees Celsius
K      Kelvin
lbm/s  Pound Mass per Second
ppm    Parts per Million
BTU    British Thermal Unit
USD/kWh U.S. Dollar per Kilowatt hour
USD/MWh U.S. Dollar per Megawatt hour
1 INTRODUCTION AND BACKGROUND

1.1 CLIMATE CRISIS

Since the late 19th century, the Earth's average surface temperature has risen approximately 2.12°F or 1.18 °C. This rise in Earth’s temperature has resulted from the increase in greenhouse gases (mainly carbon dioxide) emitted into the upper atmosphere [1]. The majority of the warming has occurred in the last 40 years, with the ten warmest years on record since 2005. An even more frightening statement is that nine of the ten warmest years since 1880 have occurred since 2005 [2] and seven of these ten warmest years have occurred since 2014 [3]. The warmest temperatures recorded on Earth have occurred in recent years (2016 and 2020). Climate change and warming are occurring due to an overabundance of greenhouse gases in the atmosphere, attributed to air pollutants produced through the combustion of fossil fuels during the last 140 years. Combusting fossil fuels release harmful byproducts and the air pollutants like carbon dioxide, sulfur, nitric oxide, volatile organic compounds, and several other pollutants and particulate matter, as seen below in Figure 1 [4].
Carbon dioxide is the most abundant byproduct of the pollutants produced by the combustion of fossil fuels and accounts for 60-90 percent of the mass of fuels burnt on the planet [4]. The primary fossil fuels that emit carbon dioxide are coal, natural gas, and oil. Figure 2 shows the breakdown of the air pollutants emitted into the atmosphere in 2019. The significant percent difference between carbon dioxide and the other greenhouse gases accounts for the 6,558 million metric tons of carbon dioxide equivalent emissions [5]. Carbon dioxide has contributed largely to the Human Enhanced Greenhouse Effect. Figure 3 shows the significant and relentless increase of carbon dioxide emissions on the planet since 1950. The graph shows the carbon dioxide levels as units of parts per million (ppm) over the last millennia. For more than 800,000 years, the planet's carbon dioxide levels have not exceeded approximately 300 ppm until 1950. Since 1950, the levels of carbon dioxide present in Earth’s atmosphere have risen exponentially. In 2013, the carbon dioxide levels on the planet surpassed 400 ppm for the first time in millennia. This rise in carbon dioxide
conveys a remarkably constant relationship with the combustion of fossil fuels and the correspondence that 60 percent of fossil fuel emissions stay in the air [6].

Figure 2 – Breakdown by Pollutant of the U.S. Greenhouse Gas Emissions in 2019 [5]

Figure 3 – Atmospheric Carbon Dioxide Levels Over the Past Millennia [6]
The Earth’s atmosphere naturally houses greenhouse gases that allow for the Natural Greenhouse Effect to regulate and protect the planet and all living beings. The atmosphere is made up of natural greenhouse gases that reflect harmful radiation from the sun into space while also absorbing the optimal amount of solar radiation and heat to regulate the weather and temperatures at the planet's surface. When there is an overabundance of greenhouse gases or air pollutants in the atmosphere, the solar radiation and heat that normally would bounce off the Earth’s surface and escape back into space are now being absorbed by overabundant air pollutants in the atmosphere. As a result, this heat is being trapped within the atmosphere and heating the planet's surface, which is known as the Human Enhanced Greenhouse Effect. The process of the Natural Greenhouse Effect compared to the Human Enhanced Greenhouse Effect can be seen below in Figure 4.

![Figure 4 – Natural vs. Human Enhanced Greenhouse Effect](image-url)

Figure 4 – Natural vs. Human Enhanced Greenhouse Effect [7]
Energy sources utilizing the combustion of fossil fuels to produce electricity have been mainly attributed to the overabundance of greenhouse gases in the atmosphere. Electricity production accounts for 25 percent of the greenhouse gas emissions in 2019, making it the second-highest greenhouse gas-producing sector in the United States behind the transportation sector at 29 percent, as seen in Figure 5. This is primarily attributed to the approximate 62 percent of the electricity generation in the United States from combusting fossil fuels [8]. The electricity sector in the United States has an enormous impact on the rise of greenhouse gases in the atmosphere of the Earth.

Figure 5 – Breakdown by Sector of the U.S. Greenhouse Gas Emissions in 2019 [8]

Scientists have predicted the time left until the effects of this phenomenon are irreversible and detrimental to the planet. Based on new prediction models, it has been
reported that the threshold for dangerous warming will be surpassed with an increase of 1.5°C and will likely occur between 2027 and 2042 [9]. This prediction presents a much narrower timespan than the estimation of now and 2052 by the Intergovernmental Panel on Climate Change (IPCC) [9]. It is apparent that electricity-producing energy sources that do not emit greenhouse gases and especially carbon dioxide, need to be developed quickly if the threshold of a 1.5°C increase should not be crossed. The expansion and advancement of electricity-producing energy sources like nuclear power plants could profoundly reduce the need to burn fossil fuels and reduce the time until the threshold for dangerous warming is reached.
2 CURRENT ENERGY USAGE AND IMPORTANCE OF NUCLEAR POWER

2.1 COMPARISON OF ENERGY USAGE IN THE UNITED STATES

The United States is the second-highest energy-consuming country in the world behind China [10], making it an important country to focus on regarding the breakdown of this large quantity of energy. According to the U.S. Energy Information Administration (EIA), in 2018, the United States produced approximately 95.754 quadrillion BTUs of energy compared to Russia, the next highest energy-producing country, at 63.463 quadrillion BTUs [10]. This is a significant amount of energy when comparing these values to other countries worldwide. The primary sources of electricity generation in the United States come from natural gas, coal, petroleum, nuclear, and renewable energies. As discussed, natural gas, coal, and petroleum are all considered fossil fuel combusting sources of energy, which are all contributors to greenhouse gas emissions into the atmosphere. These three sources make up 60 percent of the electricity generated in the United States, while nuclear and renewable energies make up the other 40 percent of electricity produced. These sources make up the 4.12 trillion kilowatt-hours of electricity produced by the United States in 2020 [11]. This equates to 2.472 trillion kilowatt-hours of electricity generated by fossil fuel-burning energy sources. This is an immense amount of energy to be produced while emitting greenhouse gases into the atmosphere. Besides the various other renewable energy sources that make up the other 20 percent, nuclear power solely
makes up the other 20 percent or one-fifth of this 4.12 trillion kilowatt-hour of electricity produced in the United States. This makes nuclear power the single largest carbon-free emitting electricity source. Therefore, it is evident that nuclear power is one of the most crucial carbon-free energy sources to focus on in the fight against the climate crisis, and there is ample room for increased use of nuclear power. The United States and the world need to reduce and eliminate fossil fuels as a means of electricity production. With fossil fuels making up 60 percent of the electricity generated in the United States, while renewables and nuclear make up the other 40 percent, it is evident that there is a capacity for renewable and nuclear energy to expand to reduce fossil fuel usage. The breakdown of these electricity-producing energy sources from 2020 can be seen in Figure 6 below.

Figure 6 – Sources of U.S. Electricity Generation in 2020 [11]
2.2 CAPACITY FACTOR AND LCOE COMPARISON OF ENERGIES

Capacity factor is the ratio of electricity produced and generated over a given period (usually one year) to the total amount of energy that could have been generated at a continuous full-power operation throughout the same given period [12]. Nuclear power plants operate 24/7 for 365 days a year while only refueling every 18 to 24 months. Due to the low frequency of refueling and the 24/7 operation, nuclear power plants achieve an average capacity factor of more than 93 percent, making it by far the most reliable and economical electricity source commercially available [13]. This is unlike many other sources of electricity, like fossil fuel-driven power plants, which often come online at times to meet grid energy needs and do not operate in a 24/7 fashion. Figure 7 below from the Department of Energy, shows a 93.5 percent capacity factor for nuclear power plants in 2019. This is practically double that of natural gas at 56.8 percent, which is the next power plant with the highest capacity factor of an energy source. This huge difference shows how reliable and economically practical nuclear power plants are compared to fossil fuel power plants.
When looking at the scope for the future of energy in the United States and around the globe, economic decisions play the most critical role. The Levelized Cost Of Energy (LCOE) compares different power plant technologies that do not resemble the same size, lifespan, capital cost, risk, return, and capacities. LCOE equates the lifetime costs of a power plant to the energy produced over the plant's lifespan. In the United States, it is often measured in USD/kWh or USD/MWh units. LCOE takes the investment expenditures, operation and maintenance (O&M) costs, the capacity factor, discount rates, lifespan of the power plant, and electricity generation to effectively understand and compare the different forms of electricity-producing power plants [15].

The IEA released the 2020 edition of the Projected Costs of Generating Electricity in December 2020, comparing the LCOEs of many forms of energy using the discounted
cash flow (DCF) method and taking the discount rates of low carbon electricity systems into account. All LCOE values presented in this section have a discount rate of seven percent. A critical insight to this report is that LCOEs of low carbon electricity-producing energy sources, like nuclear power, are dropping and are increasingly falling below the LCOEs of fossil fuel combusting energy sources. The 2020 edition on the projected costs of generating electricity shows lower expected costs for electricity production of new nuclear power plants than the 2015 edition. This report provides LCOE values of nuclear power plants for nth-of-a-kind (NOAK) plants to be completed by or after 2025 [16]. This would include a power plant like the NuScale Power SMR, ready to sell its modules to customers by 2027 [17]. Nuclear plants are expected to have a lower LCOE median value of 69 USD/MWh than coal’s median value of 88 USD/MWh. Natural gas-based combined-cycle gas turbines (CCGTs) are the only fossil fuel-based power plant that is competitive in some regions to nuclear plants. The LCOE of these plants varies greatly depending on the prices for natural gas and the cost of carbon emissions in individual regions. The median LCOE value for natural gas-based CCGTs is projected to be 71 USD/MWh, with a maximum value of 107 USD/MWh and a minimum value of 42 USD/MWh. This minimum projected value of natural gas-based CCGTs is identical to the minimum projected value of nuclear power plants. Yet, the median LCOE value of the CCGTs is slightly less affordable than the 69 USD/MWh LCOE value for nuclear plants MWh [16]. The report also provides the LCOE values for electricity produced by long-term operation (LTO) nuclear plants by lifetime extension. The median LCOE value for these plants
is not only the most affordable low-carbon generation energy source. Still, it undoubtedly has the lowest median LCOE value of 32 USD/MWh [16].

Fossil fuels power plants have been around for so long and have been challenging to defer away from because of their relatively low Levelized Cost of Energy (LCOE). Nuclear power plants not only have carbon-free emissions, but they now also have lower LCOEs values than virtually all fossil fuel power plants. With modern advancements and innovations of nuclear power plants, the economic decision to build nuclear power plants over fossil fuel power plants is crucial to the health of our planet. Still, it is also economically practical and affordable for any country or company investing in its new energy sources.
3 THE NUSCALE POWER SMR DESIGN

3.1 INTRODUCTION

The NuScale Power SMR is the first advanced small modular reactor to pass the Design Certification Application (DCA) and obtain design approval from the U.S. Nuclear Regulatory Commission (NRC) in 2020. NuScale Power LLC refers to their SMR technology as the NuScale Power Module (NPM). It is rated at an output of 160 megawatt thermal (MWt) or approximately 50 megawatts of electricity (MWe) for a single module. The total rated gross power output for the 12-module system is 1,920 MWt or 600 MWe total electrical output [18]. Recent claims by NuScale Power LLC explain that a single NPM can generate 77 MWe or 924 MWe for the full-scale 12-module power plant [19]. The reactor core of an NPM consists of 37 fuel assemblies and 16 control rod assemblies (CRAs). The CRAs are created to have a regulating bank and a shutdown break. During normal operation, the regulating bank controls reactivity, while the shutdown bank is used during routine shutdowns. The fuel assembly has been designed for a standard 17x17 PWR fuel assembly with 24 guide tube locations where the control rods are inserted. The fuel assembly design is like normal PWRs, besides half the height of a standard fuel assembly and only five spacer grids. The NuScale Power SMR utilizes uranium dioxide with gadolinium oxide as a burnable absorber homogeneously mixed in the fuel of select fuel rods with a typical U-235 enrichment of less than 4.95 percent [18]. A single NPM is made up of systems, subsystems, and components that comprise the modularized and transportable nuclear steam supply system (NSSS). Each NPM consists of its skid-mounted steam
turbine-generator and condenser. At the same time, it is also installed below grade in a seismically robust, steel-lined concrete pool to enhance the plant's safety [20]. The NuScale NSSS is a passive small modular pressurized water reactor utilizing light water as the coolant. This NuScale Power specific design utilizes an integral power module made up of a reactor core, two steam generator bundles, and a pressurizer integrated within the reactor pressure vessel (RPV). A single reactor vessel is housed within the containment vessel (CNV) made of compact steel and immediately surrounds the RPV. This design eliminates external piping typically used in nuclear reactors, connecting the steam generators and the pressurizer to the RPV [18]. The pressurizer maintains a constant pressure of 1850 psia within the system. It is located at the top of the RPV, separated by a thick baffle plate with a diameter of four inches. The RPV is approximately 66 feet high, 9 feet in diameter, and weighs 700 tons. Figure 8 shows a simplified schematic of the primary systems and flow directions of a single NPM.
Figure 9, shown below, depicts the layout of the NPMs within the reactor building along with the associated components and equipment to maintain the facility properly. This figure only shows six of the potential 12 modules that can be purchased and installed to the power plant for additional energy production as needed.
Figure 9 – Layout of NPMs within Reactor Building [22]

The notable differences between this iPWR and a typical large PWR are the means of cooling and the steam generator system. Each NPM is connected to the same components of the secondary loop, just as any other power plant would produce steam and turn a turbine to produce electricity. The steam is then cooled and condensed back into a liquid form to be sent through the feedwater heaters and back through the feedwater pipes to the steam generator. The NPMs produce steam in an opposite manner to the typical large PWR. The primary and secondary systems' functionality can be seen in Figure 10 below.
Figure 10 - Schematic of a Single NuScale Power Module and Associated Secondary Equipment [18]

3.2 NATURAL CIRCULATION COOLING OF THE CORE

The NuScale Power SMR is the first iPWR to be developed on the concept and principles of buoyancy-driven natural circulation to extract the heat from the fission reaction and cool the reactor core. This process relies on water becoming less dense as it reaches higher temperatures and becomes more dense as it decreases to lower temperatures. The primary coolant increases in temperature and becomes less dense as it flows through the reactor's core, which causes the coolant to rise vertically upward through the central riser to the top of the RPV. This less dense coolant continues to
rise until it reaches the top of the riser, where it is met by the upper plenum. The coolant is then directed downward and is met by the primary steam generator, where it flows around the helical coil steam generator tubes. As it flows past the 1,380 tubes of the secondary steam generator, the heat is extracted and transferred to the secondary side, which superheats the feed water into steam sent through the steam pipe to the turbine to be converted to electricity. As a result, this decreases the temperature of the coolant on the primary side, therefore increasing the coolant’s density. As the density of the coolant increases, it continues to fall through the downcomer to the bottom of the reactor to repeat the process over again. Figure 11 shows a cutaway view of an NPM with a color scheme of the coolant temperatures as it flows throughout the systems to understand the natural circulation cooling process.
Due to this modern and innovative reactor design, the need for AC or DC electrical pumps to control the coolant flow through the core is eliminated. Electrical pumps are one of the primary sources of failure in any power system. They propose a particular danger when used throughout nuclear power plants, where controlling and monitoring the core's temperature or the fission reactions within the core is crucial.
3.3 HELICAL COIL STEAM GENERATOR (HCSG) SYSTEM

A common feature between almost all power plants is energy generation through conductive or convective heat transfer. This is typically done to convert liquid water into a vapor form or steam. This generated steam is then used to power and turn a turbine to produce electricity. This is the case for any power plant when the overall purpose of the plant is to produce electricity, regardless of the heat source which does so. A heat exchanger is present in typical PWRs or any nuclear power plant to extract the heat and assist with this process. The heat exchangers comprise the various number of straight-through, once-through, or U-shaped tubes used to extract the heat from the primary coolant system of the reactor to the secondary system where the steam is formed. The tubes are relatively simple to manufacture, and they are typically friction or pressure-fit to a tube sheet. This proposes a potential point of failure when the tubes expand and cause plastic deformation, leading to the tube walls becoming enfeebled [23]. These steam generator tube designs require a larger surface area to allow for a sufficient amount of heat to be transferred to the secondary side of the plant. Therefore, a larger amount of space must be present to accommodate the longer-length tubes [23]. This will, in turn, contribute to the restrictions of producing a nuclear reactor of smaller size.

NuScale Power LLC has designed and produced an iPWR comprising a steam generator system that can avoid these difficulties of lower thermal efficiencies, points of potential failure, and surface area restrictions of typical straight-through, once-through, or U-shaped tubes of steam generators. The Helical Coil Steam Generator
(HCSG) is one of the advanced innovations of the NuScale Power SMR. It comprises 1,380 helical coil-shaped tubes that wrap around the reactor to extract the thermal heat from the primary convective loop. The HCSGs are designed to provide the highest heat transfer surface area in a small volume. The HCSG is a once-through counter-flow design that allows for the generation of superheated steam within the tubes on the secondary side due to the high thermal efficiency from the natural circulation flow on the primary side of the reactor [19]. The geometry of the steam generator system allows for a shallow pressure drop serving to maximize the natural circulation cooling flow of the primary coolant system [19].

To better understand this design and geometry, an image of the helical coil steam generator bundle from the OSU-MASLWR test facility can be seen in Figure 12. The OSU-MASLWR test facility is an integral test facility constructed by the Oregon State University (OSU) under a U.S. Department of Energy grant to examine and understand the natural circulation phenomena that characterize the MASLWR design steady-state and transient conditions. The scale of this facility consists of a 1:3 length scale, 1:254.7 volume scale, and a 1:1 time scale [24]. The MASLWR reactor is manufactured entirely out of stainless steel and is designed for total pressure and temperature prototype operation and provides an essential visual and understanding of how a full-scale SMR of this type will operate once manufactured and built. The entire experimental facility can be seen below in Figure 13.
Figure 12 – OSU-MASLWR Test Facility Helical Coil Steam Generator Bundle [24]

Figure 13 – OSU-MASLWR Experimental Test Facility [24]
3.4 SAFETY ASPECTS AND FEATURES

As stated in the previous section, the NPMs are installed below ground in a seismically robust, steel-lined concrete pool to enhance the plant's safety [20]. This also allows for the capabilities of the power plant to be built in locations like islands, which are not ordinarily suitable for typical, large PWRs built above ground. Islands usually have a smaller amount of landmass than the locations where large PWRs are built, eliminating the possibility of large reactors being built there. Islands also experience natural disasters, which can have detrimental effects on reactor buildings built above ground. These are significant reasons that restrict nuclear power expansion to islands with large populations across the globe. A small iPWR like the NuScale Power SMR expands this possibility for nuclear energy. The NuScale Power SMR would take up a fraction of the landmass needed to build a nuclear reactor while also providing the means for a nuclear power plant to be built at vulnerable or difficult locations.

There are essential safety features of the NuScale Power SMR to be highlighted, like the Emergency Core Cooling System (ECCS) and the Decay Heat Removal System (DHRS). The ECCS safety feature exists in the case of a Loss of Coolant Accident (LCOA) to remove heat through the containment vessel to rapidly reduce the containment pressure and temperature. The ECCS utilizes three independent reactor vent valves and two independent reactor recirculation valves. During a successful actuation of the ECCS two of the three reactor vent valves must open, and one of the two-reactor recirculation valves must open. This system allows for the steam on the
inside surface of the containment vessel to condense into a liquid form, which is passively cooled by conduction and convection of heat to the surrounding reactor pool water [25].

The Decay Heat Removal System is a safety system that utilizes an additional heat exchanger that removes heat from the primary coolant, providing cooling to the secondary loop of the reactor for a non-LOCA event when the feedwater of the secondary loop is not available. The safety system is a two-phase, closed-loop natural circulation cooling system [25]. The main steam line of the steam generator is connected to the DHRS steam inlet piping [26]. If an accident were to occur or if DHRS control power was lost, the Module Protection System (MPS) opens the DHRS actuation valves using an actuation signal. The MPS monitors the systems and plant parameters and will automatically initiate signals for conditions that do not match normal operational limits if the MPS actuates the DHRS, its valves open and the Feedwater Isolation Valves (FWIVs) and the Main Steam Isolation Valves (MSIVs) all close [26]. This causes the steam exiting the helical coil steam generator tubes to be sent to the DHRS passive condensers, condensing the water to liquid form and flowing it to the feedwater lines. The DHRS heat exchangers are located in the reactor pool and operate as the ultimate heat sink (UHS) for this NuScale Power-specific design. Therefore, the natural circulation cooling process continues, and the DHRS safety system removes the potentially dangerous decay heat [26].
3.5 MANUFACTURABILITY AND SCALABILITY

The NuScale Power Plant has been designed to be small, modularized, scalable, and factory built. The power plants are built based on more assembly instead of more construction of the plant at the physical site. Large portions of the projects are to be manufactured off-site and assembled on-site, unlike typical nuclear reactors, which are too large to be manufactured off-site and transported to the site. This is an enormous advantage of the small, modularized design of the NPM and the overall power plant. The NPM and other systems and components of the plant can be mass-produced in a warehouse and shipped by truck, rail, or ship to the site's location. Some of these other components and systems which can also be fabricated and assembled off-site include the turbine-generators, chemical control processes, and other modular systems.

NuScale Power LLC is the first company to introduce a nuclear reactor mass-produced in a warehouse and shipped to the on-site location. This process increases efficiency, drastically expands the demographic of buyers, standardizes the manufacturing process, and lowers the cost [27]. Figure 14 and Figure 15 shows conceptual images of an NPM and other components being shipped to the on-site location by truck and by ship.
Figure 14 – NPM Shipped by Truck [27]

Figure 15 – NPM and Other Components Shipped by Ship [27]
The NuScale Power SMR power plants allow one to 12 NPMs to operate within a single Reactor Building. To meet the diverse energy needs of potential buyers across the globe, the company offers this scalability of their power plants, allowing for smaller power plant solutions in four-module and six-module configurations, with other configurations possible. This individual operation of the NPMs allows for diversified and flexible power solutions for various customers [19].

3.6 IMPORTANCE OF THE NUSCALE POWER SMR

Allowing for the modularized and scalable options of the power plants, the demographic of buyers and customers increases tremendously for nuclear power plants of this type. It may be impractical for less developed countries and islands to afford a typical, large nuclear reactor that can cost up to ten billion dollars and use many landmasses. The initial upfront costs of large reactors defer many potential customers away from nuclear power. These buyers usually defer to fossil fuel power plants because they exhibit lower upfront costs. The cost of the NuScale Power SMR can be a solution to these customers for lower upfront costs while choosing energy with carbon-free emissions. As discussed previously, the LCOE value of a reactor like the NuScale Power SMR is projected to be lower than all fossil fuel power plants, besides the CCGTs plants that exhibit a comparable LCOE value to nuclear plants. On top of this, nuclear power plants exhibit capacity factors far higher than any electricity-producing fossil fuel energy source. For the reasons discussed, the NuScale Power SMR is designed to provide exceptional safety while standardizing the manufacturing process and lowering the costs to build the power plant. With scientists
predicting the detrimental effects of climate change by 2027 to 2042, a solution to reducing fossil fuel power plants must be created quickly, and the NuScale Power SMR power plant can be that solution. The power plants can be built in approximately three years. They can further expand nuclear energy in the United States and globally while eliminating the need to combust fossil fuels to produce energy.
4 COMPARISON OF THE SMR TO A TRADITIONAL LARGE PWR

Many outstanding features of the NuScale Power SMR differ it from a typical, larger pressurized-water reactor (PWR). The most prominent characteristics would be the size, manufacturability, and cooling of the core. As discussed in section 3 of this report, the NuScale iPWR relies on the natural circulation of the coolant to regulate the core temperature and transfer heat to the steam generator. In Table 1, the main reactor and core parameters are listed. The differences are outstandingly apparent between the NuScale Power SMR and a large PWR like the U.S. Evolutionary Power Reactor (US-EPR) and the U.S. Advanced Pressurized Water Reactor (US-APWR).

Important parameters to be noted are the average temperature rise in the core and the estimated flow through the reactor's core. The average rise in temperature is relatively higher in the NuScale iPWR than the large PWRs, while the flow through the core is significantly lower than the large PWRs. These conditions and parameters result from the natural circulation cooling design of the NuScale iPWR, compared to the traditional method of pumping coolant at specific velocities through the core of large PWRs. Another clear difference is the significantly smaller-sized core of the iPWR. The fuel column length is only about 79 inches compared to 160 inches and 165.4 inches for US-EPR and US-APWR. The core flow area for the large PWRs is practically seven times the size of the core flow area for the NuScale iPWR. All these differences have resulted from the innovative technology of the NuScale Power SMR to cool the core and extract the thermal energy. Natural circulation cooling is extremely sensitive to flow, temperature, and pressure changes, leading to the
geometries applied to the reactor.

Table 1 – Comparison of Main Reactor and Core Parameters of NuScale iPWR to Large PWR [28]

<table>
<thead>
<tr>
<th>Reactor Parameter</th>
<th>NuScale iPWR</th>
<th>US-EPR</th>
<th>US-APWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core thermal output (MWt)</td>
<td>160</td>
<td>4590</td>
<td>4451</td>
</tr>
<tr>
<td>System pressure (psia)</td>
<td>1850</td>
<td>2250</td>
<td>2250</td>
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<tr>
<td>Number of loops</td>
<td>N/A</td>
<td>4</td>
<td>4</td>
</tr>
<tr>
<td>Inlet temperature (°F)</td>
<td>497</td>
<td>563.4</td>
<td>550.6</td>
</tr>
<tr>
<td>Core average temperature (°F)</td>
<td>543</td>
<td>596.8</td>
<td>588.8</td>
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<tr>
<td>Average temperature rise in core (°F)</td>
<td>100</td>
<td>62.7</td>
<td>72.1</td>
</tr>
<tr>
<td>Minimum design flow (lb/hr)</td>
<td>4.27E+06</td>
<td>1.73E+08</td>
<td>1.68E+08</td>
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<tr>
<td>Maximum design flow (lb/hr)</td>
<td>5.24E+06</td>
<td>1.95E+08</td>
<td>1.88E+08</td>
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<tr>
<td>Best estimate flow (lb/hr)</td>
<td>4.66E+06</td>
<td>1.80E+08</td>
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<tr>
<td>Core bypass flow (%)</td>
<td>8.5</td>
<td>5.5</td>
<td>9</td>
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<tr>
<td>Normal operation peak heat flux (10^6 Btu/hr-ft²)</td>
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<td>0.46</td>
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<td>Normal operation core average heat flux (Btu/hr-ft²)</td>
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<td>177,036</td>
<td>162,000</td>
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<td>Core flow area (ft²)</td>
<td>9.79</td>
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<td>Core average coolant velocity (ft/sec)</td>
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<td>Equivalent diameter of active core (in)</td>
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<td>Number of fuel assemblies</td>
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<td>Effective fuel length (in.)</td>
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<td>Rods per fuel assembly</td>
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<td>264</td>
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<tr>
<td>Number of grids per assembly</td>
<td>5</td>
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<td>Cladding outside diameter (in.)</td>
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<tr>
<td>Fuel column length (in.)</td>
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<td>Fuel pellet diameter (in.)</td>
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Table 2 – Safety Systems and Components Required to Protect the Reactor Core - NuScale SMR Comparison with Other Facilities [18]

<table>
<thead>
<tr>
<th>Safety System or Component</th>
<th>Typical PWR</th>
<th>NuScale iPWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Pressure Vessel</td>
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<td>X</td>
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<tr>
<td>Containment Vessel</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>Reactor Coolant System</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>Decay Heat Removal System</td>
<td>X</td>
<td>X</td>
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<td>Emergency Core Cooling System</td>
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<td>Control Rod Drive System</td>
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<td>Containment Isolation System</td>
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<tr>
<td>Ultimate Heat Sink</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>Residual Heat Removal System</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Safety Injection System</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Refueling Water Storage Tank</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Condensate Storage Tank</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Auxiliary Feedwater System</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Emergency Service Water System</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Hydrogen Recombiner or Ignition System</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Containment Spray System</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Reactor Coolant Pumps</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Safety-Related Electrical Distribution System</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Alternative Off-Site Power</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Emergency Diesel Generators</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Safety-Related Class 1E Battery System</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Anticipated Transient Without Scram (ATWS) System</td>
<td>X</td>
<td></td>
</tr>
</tbody>
</table>

Table 2 shows the incredible amount of safety systems and components eliminated through the innovative designs of the NuScale Power SMR. Essential safety systems and components to be noted that are not required within the NuScale Power SMR systems are the Reactor Coolant Pumps, Residual Heat Removal System, Auxiliary Feedwater System, Containment Spray System, Alternative Off-Site Power, and Emergency Diesel Generators. The use of buoyancy-driven natural circulation cooling
of the primary system and safety features like the ECCS and the DHRS have led to many of these safety systems and components providing no use and being eliminated from the system and power plant. Additional features and components can often produce more potential points of failure within a system, which is another benefit of the NuScale Power SMR. A breakdown of the exact differences in features, components, parameters, and geometries of a typical PWR to the NuScale Power iPWR are displayed in Table 3 below.

Table 3 – NuScale SMR Plant Comparison with Other Facilities [18]

<table>
<thead>
<tr>
<th>NuScale Plant Parameter or Feature (per NPM)</th>
<th>Typical PWR</th>
<th>NuScale iPWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nominal gross electrical output (MWe)</td>
<td>1,186</td>
<td>50</td>
</tr>
<tr>
<td>Core thermal output (MWt)</td>
<td>3,411</td>
<td>160</td>
</tr>
<tr>
<td>Number of fuel assemblies</td>
<td>193</td>
<td>37</td>
</tr>
<tr>
<td>Fuel assembly lattice</td>
<td>-17x17</td>
<td>17x17</td>
</tr>
<tr>
<td>Effective fuel length (ft)</td>
<td>12</td>
<td>6.56</td>
</tr>
<tr>
<td>Fuel rods per fuel assembly</td>
<td>264</td>
<td>264</td>
</tr>
<tr>
<td>Average linear heat rate (kW/ft)</td>
<td>5.4</td>
<td>2.5</td>
</tr>
<tr>
<td>Number of Control Rod Assemblies</td>
<td>53</td>
<td>16</td>
</tr>
<tr>
<td>Design life (years)</td>
<td>40</td>
<td>60</td>
</tr>
<tr>
<td><strong>Reactor Coolant System</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Number of heat transfer loops</td>
<td>4</td>
<td>No External Loops</td>
</tr>
<tr>
<td>Reactor Coolant Pipes (in.)</td>
<td>27.5-31</td>
<td>None</td>
</tr>
<tr>
<td>Operating pressure (psia)</td>
<td>2,250</td>
<td>1,850</td>
</tr>
<tr>
<td>Hot leg temperature (°F)</td>
<td>618</td>
<td>590</td>
</tr>
<tr>
<td><strong>Reactor Vessel</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Vessel inner diameter (in.)</td>
<td>173</td>
<td>107.5</td>
</tr>
<tr>
<td>Thermal shielding- and reflector design</td>
<td>Neutron pad design</td>
<td>Stacked stainless steel reflector blocks</td>
</tr>
<tr>
<td>In-core instrumentation</td>
<td>Bottom mounted</td>
<td>Top mounted</td>
</tr>
<tr>
<td><strong>Steam Generator</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Number</td>
<td>4</td>
<td>2</td>
</tr>
<tr>
<td>Type</td>
<td>Vertical U-tube</td>
<td>Helical coil</td>
</tr>
<tr>
<td>Heat transfer area (ft²)</td>
<td>55,000</td>
<td>Approximately</td>
</tr>
<tr>
<td></td>
<td>PCCV</td>
<td>Steel Pressure Vessel</td>
</tr>
<tr>
<td>------------------------------</td>
<td>-----------------------</td>
<td>-----------------------</td>
</tr>
<tr>
<td>Type</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Inner diameter (ft-in.)</td>
<td>140-0</td>
<td>14-2</td>
</tr>
<tr>
<td>Height (ft-in.)</td>
<td>205-0 (inner)</td>
<td>75-8.5 (outer)</td>
</tr>
<tr>
<td>Containment Spray Pumps</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>High Pressure Safety Injection Pumps</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>Low Pressure Safety Injection Pumps</td>
<td>2</td>
<td>None</td>
</tr>
<tr>
<td>Accumulators</td>
<td>4</td>
<td>None</td>
</tr>
<tr>
<td>I&amp;C System type</td>
<td>Analog</td>
<td>Digital</td>
</tr>
<tr>
<td>Emergency Diesel Generators</td>
<td>2</td>
<td>None</td>
</tr>
</tbody>
</table>

| Turbine Type                 | 1800 rpm, Tandem Compound Six Flow | 3,600 rpm, 10 stage with Superheat |
| Emergency Feedwater Pumps    | 3                     | None                  |
| Charging Pumps (CVCS pumps)  | 2                     | 2                     |
| Used for Safety Injection    | Yes                   | No                    |
| Volume Control Tank          | 1                     | 0                     |
| Reactor Component Cooling Water Pumps | 4              | 6 total for 12 NPMs   |

Besides comparing the main reactor and the core parameters, looking at the lead time to manufacture, build, and install the NuScale iPWR compared to large PWRs like the US-EPR and US-APWR are significantly different. As discussed previously, there is limited time until the impacts of climate change are irreversible. Therefore, the lead time to bring one of these reactors to operation and produce electricity is critical. Typically, large PWRs can take approximately ten years to build and cost tens of billions of dollars. The NuScale iPWR can be manufactured, shipped, installed, and
produce electricity in only about three years. This period is a significant decrease in lead time and could profoundly affect the future sources of energy for the world. The first NuScale Power modules will be sold to their clients by 2027, and the first module will be operational by 2029, with the remaining modules to come online for a full-scale operation by 2030 [17].
5 RELAP5-3D MODELING

5.1 RELAP5-3D EXPLAINED

Reactors Excursion and Leak Analysis Program, also known as RELAP5-3D, was developed at Idaho National Laboratory (INL) around 1966. RELAP5-3D is a powerful multidimensional thermal-hydraulic transient simulation tool that provides its users with the ability to model coupled behavior of nuclear reactor coolant systems and cores for transient analysis calculations and Loss of Coolant Accidents (LOCAs) that may occur within the system [29]. RELAP5-3D can also be used for reactor safety analysis, design of reactors, plant operator training, and education for university students. After developing the first nuclear reactors, the Nuclear Regulatory Commission (NRC) realized the crucial need for reactor safety analysis software. Therefore, in 1966, INL scientists started to develop the Reactor Excursion and Leak Analysis Program (RELAP) to combat the need of modeling reactor coolant and core behavior in a pressurized water reactor. Since then, the NRC and Department of Energy (DOE) have provided funding and support in the continued development of RELAP by increasing the complexity of the code to keep the modeling as realistic as possible. This continued growth of the code has allowed for an array of reactor designs and various power systems to be modeled by the program [29].

In 1996, INL decided to copyright the non-NRC-funded parts of the RELAP code, which led to the release of the RELAP-3D version in 1998 [29]. Since 1998, the DOE Office of Nuclear Energy (DOE-NE) and the Naval Reactors (DOE-NR) have funded the work performed on RELAP, which then led to the formation of the International
RELAP Users Group (IRUG) in the same year to support non-government users like students at universities and employees of the commercial nuclear industry [29]. Commercial nuclear reactor vendors such as NuScale Power LLC have used and continue to use RELAP5-3D to perform design analyses and obtain the NRC Design Certification Approval for new reactor designs like the NuScale Power SMR.

5.2 COMPATIBLE AND ASSOCIATED PROGRAMS

Many RELAP5-3D users have significantly benefited from the formation of the International RELAP Users Group. There are multiple levels of the IRUG membership that is a part of obtaining a license to RELAP5-3D. As a University Participant, a no-fee license to the executable code of RELAP5-3D can be acquired with the restriction that the code is only used for educational purposes at the university. Compared to the three other membership levels, the drawback to the University Participant membership level is that staff assistance is not provided [30]. Information and assistance with RELAP5-3D are tremendously limited because of the restrictions of obtaining a license to this program. The cost of the higher-level memberships is prohibitive, but staff assistance is provided with those. This was the greatest drawback of the University Participant membership level. Besides obtaining the license to RELAP5-3D and becoming an IRUG member, other associated programs have been advantageous in easing the deciphering of RELAP5-3D data. Due to the complexities of the code and the strict copyrighted licensing, there is no simple way to analyze and plot the data of the output and restart-plot files produced by RELAP5-3D.
The most compatible program to perform the data retrieval, analysis, and plotting of the RELAP5-3D data is AptPlot. AptPlot is a free Pure-Java 2D plotting tool designed to help perform data analysis and create professional-quality graphs and plots of numerical data. The program contains the capability of extensive scripting and GUI support for the analysis and manipulation of data sets and files [31]. AptPlot has been created to be a drop-in replacement to the NRC Analysis Code version of Grace called AcGrace, which has been altered to allow for direct interfaces to multiple analysis codes, NRC Databank files, the Symbolic Nuclear Analysis Package (SNAP), and to allow for a simpler means of performing calculation and analysis using these types of data files. Grace has been a powerful tool in providing extensive plotting and data analysis capabilities for several years yet presents limitations for users utilizing Microsoft Windows platforms. The Grace software was specifically written and tailored to Unix machines. Any user utilizing operating systems that do not provide similar functionalities to Unix operating systems had difficulties employing AcGrace [31]. Therefore, AptPlot was developed by Applied Programming Technology (APT) using Java programming language to allow for ease of use, installing, and maintaining the software for operating systems that do not provide Unix-like functionalities like Microsoft Windows operating systems. AptPlot has simplified the data analysis throughout this research and has been utilized in obtaining the professional quality plots needed for the presentation of data throughout this thesis. It has eased the use of RELAP5-3D tremendously and allowed for ample time to be saved throughout the process.
6 EDWARD’S PIPE BLOWDOWN EXPERIMENT

MODEL (BENCHMARK EXPERIMENT)

6.1 INTRODUCTION

The Edward’s Pipe Blowdown Experiment was used as a prerequisite benchmark experiment to gain the proper RELAP5-3D modeling knowledge and experience needed to develop the NuScale Power SMR model. This experiment has been extensively studied in the past. It has been used as a fundamental benchmark problem for two-phase flow codes due to the simple geometry of the pipe and the multitude of phenomena it covers [32]. The experiment was originally used to validate all the advancements and modifications to the RELAP5 code, including the early development of the hydrodynamic and critical flow models [33]. This experiment was originally performed by A.R. Edwards and T.P. O’Brien [34] in 1970 to study the phenomena associated with the depressurization of water reactors and consisted of a straight four-meter-long steel pipe filled with water, pressurized to 7000.0 kPa, and heated to 502.0 K. A glass disk was inserted at one end of the pipe to be used as the location of the rupture for the blowdown. The pipe area was 1.0956E-4 m$^2$ giving the ruptured disk or orifice an exit area of 0.95317E-4 m$^2$ due to a 13 percent area reduction from the remaining fragments of the disk leftover at the rupture location. The pipe diameter was 73 mm, and the exact length of the pipe was 4.09 m. The flow process of the liquid in the pipe and the phenomena occurring at the discharge of the orifice provide essential phenomena to be investigated using a transient two-phase flow model like RELAP5-3D. The following section will cover the development of the
Edward’s Pipe Blowdown Experiment model using RELAP5-3D, which has been modeled and performed to simulate the RELAP5 calculated data by K. E. Carlson, V. H. Ransom, and R. J. Wagner in their published paper named “The Application of RELAP5 to a Pipe Blowdown Experiment” [33]. This calculated data obtained through the execution of the RELAP5-3D code will be plotted and compared to the experimental data found by A.R. Edwards and T.P. O’Brien. The experimental data obtained by Edwards and O’Brien can be seen in Figure 16. This data has been compared to the calculated data obtained by RELAP5-3D displayed in Figure 17.

### 6.2 RELAP5-3D MODEL OF EXPERIMENT

The same geometry and conditions have been used in the RELAP5-3D model as the original experiment performed by A.R. Edwards and T.P. O’Brien, as described above in section 6.1. The nodalization diagram in Figure 16 below depicts the components, volumes, and junctions used to perform the calculation. A 20-volume pipe is used for component 111 to simulate the apparatus that is heated and pressurized. Component 112 represents the orifice or exit area with the 13 percent area reduction, which is ruptured during the experiment and is modeled using a single junction component. Component 113 represents the atmosphere outside the pipe and is modeled as a boundary condition using a time-dependent volume. Component 111 was modeled using the specified conditions and geometries explained in section 6.1.

The pipe's volume flow area and length were specified to 0.0041854 m$^2$ and 0.2048 m on cards CCC0101 and CCC0301, respectively, to compensate for the pipe being
broken up into 20 separate volumes. The volume on card CCC0401 was left as 0.0 m³ to allow RELAP5-3D to calculate these values. The inclination angle on card CCC0601 was left as 0.0 degrees because the experiment is performed using a horizontal straight pipe. Edwards and O’Brien only generally specified the pipe used during the experiment to be steel. Therefore, a wall roughness for commercial steel or wrought iron of 0.045E⁻³ m was used and was obtained from the Engineering toolbox website. The wall roughness was specified on CCC0801 which also provides the value of 0.073 m for the hydraulic diameter of the pipe.

Extensive use of the manuals and appendices of RELAP5-3D led to an understanding of how RELAP5-3D interprets the use of the process models, which are activated with the volume control flags cards and the junction control flags cards. The different process models and schemes are activated, deactivated, or specified using the packed format words tlpvbfe and jefvcahs for most components. The exact meaning of each digit of the packed format words varies from component to component but is generally similar and typically used for the models’ activation (or deactivation). Due to the simplicity of Edward’s Pipe Blowdown Experiment, most of these control flags were deactivated. When specifying the initial volume conditions with the packed format word ēbt, there are options using card CCC1201. The initial conditions specified for the pipe component are pressure and temperature at 7000.0 kPa, and 502.0 K. Understanding how the process models alter or modify the RELAP5-3D code calculations is highly beneficial when modeling a more complex system like the NuScale Power SMR.
The single-junction component connects the 20th volume of the pipe and the time-dependent volume representing the atmosphere. This is the location at the end of the pipe where the rupture of the glass disk occurs. On card CCC0101, the junction area is specified to be 0.95317E-4, compensating for a 13 percent area reduction of the pipe area because glass fragments are leftover in the orifice after the rupture. The form losses have been neglected and left as 0.0 due to the simplicity of this experiment. This card also can specify the discharge coefficients, which turned out to be an essential factor when modeling this experiment. These values were imputed to be 0.5. The third input used for the single junction is the initial junction conditions, all specified to be 0.0. The words used on this card specify the velocity to be calculated instead of mass flow rate, the initial liquid velocity, the initial vapor velocity, and the interface velocity. These are all inputted as 0.0 to allow RELAP5-3D to calculate these values.

The time-dependent volume component is used as a boundary condition and represents the atmosphere outside of the pipe where the break is flowing to. The volume flow area, length, and volume are inputted on card CCC0101. Due to this component acting as a boundary condition, the values of 1.0 m² for flow area, 0.0 m for length, and a very large value of 1.0E6 for the volume were used. These values seemed to function well with how RELAP5-3D interpreted the boundary condition. Only one of these values can be left as 0.0, and the length was chosen to allow RELAP5-3D to calculate this value. The card CCC0102 provided the information for the volume orientation,
and all these values were left as 0.0. CCC0103 provided the information for the wall roughness, hydraulic diameter, and the packed format word tlpvbf. These values were all input as zeros. CCC0200 is the control word \( \varepsilon_{bt} \) and was input as 102 to specify pressure and static quality on the following card, CCC0201. The atmospheric pressure of 101325.0 Pa and the static quality of 0.999 were used for this boundary condition to compensate for the vapor which was seen to flash out of the rupture of the glass disk.

The nodalization diagram of this model can be seen below in Figure 16.

![Figure 16 – Edward’s Pipe Blowdown Nodalization Diagram](image)

6.3 RESULTS OF BENCHMARK EXPERIMENT

Figure 17 depicts the experimental results obtained by Carlson, Ransom, and Wagner through their simulation of the original Edward’s Pipe Blowdown Experiment performed by A.R. Edwards and T.P. O’Brien in 1970. At zero time before the glass disk rupture on the end of the pipe, the pressure is 7.0 MPa or 7000.0 kPa for the volume of the pipe closest to the rupture. The plot shows the short-term pressure transient of this volume closest to the rupture. After approximately 0.0035 seconds, the pipe experiences a drastic decline in pressure. In 0.0035 seconds, the pressure drops from 7.0 MPa to approximately 1.5 MPa. Immediate depressurization at the volume closest to the rupture is expected in this short period. The choking
phenomenon occurs at the orifice of the pipe before equalizing with ambient pressure. The pressure increased up to about 2.5 MPa after oscillating for this short period. This choking phenomenon can be seen to only occur in under 0.008 seconds before equalizing with the ambient pressure and temperature outside of the pipe.

![Edward’s Experiment Short Term Pressure Transient for Volume 01 of Pipe](image)

Figure 17 – Edward’s Pipe Blowdown Experiment Experimental Short-term Pressure vs. Time Data [33]

Figure 18 shows the RELAP5-3D calculated data from the simulation of this experiment. The model created in RELAP5-3D was identical to the experiment performed by Carlson, Ransom, and Wagner, and the short-term pressure data for the volume closest to the glass disk rupture can also be seen. The calculated data exhibits a close relationship to the experimental results. In approximately 0.0035 seconds, the pressure drops dramatically from 7.0 MPa to almost 1.5 MPa. Choking through the
orifice appears to be present during the oscillations until rising to about 2.5 MPa. This depressurization process and choking phenomena closely resemble the results seen by Edwards and O’Brien during the actual experiment.

![PRESS VS TIME VOLUME 01 0.005 SECONDS](image)

Figure 18 – RELAP5-3D Calculated Edward’s Pipe Blowdown Experiment Short-term Pressure vs. Time Data

The Edwards Pipe Blowdown Experiment was a vital and straightforward benchmark study to understand how the RELAP5-3D code interprets various inputs, process models, and phenomena that occur due to the specified input. This was a pivotal step to ensure the skills were acquired to properly model a complex iPWR like the NuScale Power SMR, which utilizes natural circulation cooling. Natural circulation cooling of a nuclear reactor proposes complex and sensitive phenomena that must be carefully
and strategically modeled to ensure RELAP5-3D is interpreting the model provided by the user correctly. The author recommends that any novice or inexperienced RELAP5-3D user perform this experiment as a benchmark to ensure a more complicated system can be modeled precisely.
7 STEADY-STATE MODEL OF THE NUSCALE POWER SMR USING RELAP5-3D

7.1 DEVELOPMENT OF THE STEADY-STATE MODEL

This section of the report will detail the development of the steady-state model of the NuScale Power SMR. The primary source of information and data used in developing the model of the NuScale Power SMR was acquired from the NuScale Power LLC Revision 5 Final Safety Analysis Report (FSAR) submitted to the NRC in July 2020 for the Design Certification Application (DCA). Most of this data can be seen in tabular form below or seen in Figure 19, which depicts the nodalization diagram of the reactor model. The information and data used from the NuScale FSAR are as follows:

- Geometrical data of the primary and secondary components throughout the entire reactor system, which includes lengths, flow areas, and volumes
- Diagrams, descriptions, and visuals of the reactor core, coolant systems, and secondary systems
- Primary and secondary system parameters for operating conditions, which includes pressures, temperatures, velocities, void fractions, and mass flow rates
- All information and data was used for both the steady-state calculations and transient calculations of the helical coil tube rupture simulation

All other data was either calculated with the provided data of the NuScale FSAR, found through experimental simulations or intuitively assumed by the author. Most assumptions were made based on similar literature published on iPWR, which utilized the RELAP5-3D program as the transient analysis tool. The assumptions used by the
author were tested during the simulations of the reactor model to verify the validation of the data. Examples of the calculated data are the hydraulic diameters of the system components using the flow areas and the surface area of the helical coil steam generator tubes for the 30 percent increase. Examples of the assumptions used, adjusted accordingly after experimental simulations, where forward and reverse energy flow loss coefficients were changed.

**NuScale Power SMR Design Parameters**

The design parameters, geometries, and conditions utilized throughout the NuScale Power SMR model can be seen in the following tables. The pressure, temperature, geometries, and thermal output are different values and parameters met and provided by NuScale Power LLC. At the same time, the mass flow rates through the core and the overall systems are best estimates as a result of the other calculated parameters.
Table 4 - Reactor Core Design Parameters and Data [28]

<table>
<thead>
<tr>
<th>Reactor Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core thermal output (MWt)</td>
<td>160</td>
</tr>
<tr>
<td>System pressure (psia)</td>
<td>1850</td>
</tr>
<tr>
<td>Inlet temperature (°F)</td>
<td>497</td>
</tr>
<tr>
<td>Core average temperature (°F)</td>
<td>543</td>
</tr>
<tr>
<td>Average temperature rise in core (°F)</td>
<td>100</td>
</tr>
<tr>
<td>Best estimate flow (lb/hr)</td>
<td>4.66E+06</td>
</tr>
<tr>
<td>Core bypass flow (%)(best estimate)</td>
<td>7.3</td>
</tr>
<tr>
<td>Average linear power density (kw/ft)</td>
<td>2.5</td>
</tr>
<tr>
<td>Heat transfer area on fuel surface (ft²)</td>
<td>6275.6</td>
</tr>
<tr>
<td>Core average coolant velocity (ft/sec)</td>
<td>2.7</td>
</tr>
<tr>
<td>Core flow area (ft²)</td>
<td>9.79</td>
</tr>
<tr>
<td>Diameter of active core (ft)</td>
<td>4.94</td>
</tr>
<tr>
<td>Height of active core (ft)</td>
<td>6.57</td>
</tr>
<tr>
<td>Height-to-diameter ratio of active core</td>
<td>1.33</td>
</tr>
<tr>
<td>Fuel design</td>
<td>NuFuel HTP2™</td>
</tr>
<tr>
<td>Number of fuel assemblies</td>
<td>37</td>
</tr>
<tr>
<td>Rods per fuel assembly</td>
<td>264</td>
</tr>
<tr>
<td>Fuel assembly Length (in)</td>
<td>95.89</td>
</tr>
</tbody>
</table>

Table 5 – Steam Generator Design Parameters and Data [21]

<table>
<thead>
<tr>
<th>Steam Generator Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Type</td>
<td>Helical, once-through</td>
</tr>
<tr>
<td>Total number of helical tubes per NPM</td>
<td>1,380</td>
</tr>
<tr>
<td>Number of helical tube columns per NPM</td>
<td>21</td>
</tr>
<tr>
<td>Internal pressure - secondary (psia)</td>
<td>2100</td>
</tr>
<tr>
<td>External pressure - primary (psia)</td>
<td>2100</td>
</tr>
<tr>
<td>External pressure - SG piping in containment (psia)</td>
<td>1000</td>
</tr>
<tr>
<td>Internal temperature - secondary (°F)</td>
<td>650</td>
</tr>
<tr>
<td>External temperature - primary (°F)</td>
<td>650</td>
</tr>
<tr>
<td>External temperature - SG piping in containment (°F)</td>
<td>550</td>
</tr>
<tr>
<td>Tube wall outer diameter (inches)</td>
<td>0.625</td>
</tr>
<tr>
<td>Tube wall thickness (inches)</td>
<td>0.05</td>
</tr>
<tr>
<td>Total heat transfer area (ft²)</td>
<td>17928</td>
</tr>
</tbody>
</table>
Table 6 – SMR Primary System Geometry Data [28]

<table>
<thead>
<tr>
<th>RCS Region</th>
<th>Total RCS Region Volume (ft³)</th>
<th>RCS Sub-region Description</th>
<th>Average Flow Area (ft²)</th>
<th>Length (ft)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Riser</td>
<td>635</td>
<td>Lower riser and transition</td>
<td>24.9</td>
<td>9.4</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Upper riser and riser turn</td>
<td>15.4</td>
<td>26</td>
</tr>
<tr>
<td>Downcomer</td>
<td>1199</td>
<td>Downcomer (including steam generators)</td>
<td>25.7</td>
<td>46</td>
</tr>
<tr>
<td>Core</td>
<td>89</td>
<td>Fuel assemblies</td>
<td>10.3</td>
<td>7.9</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Reflector cooling channel</td>
<td>0.9</td>
<td>7.9</td>
</tr>
<tr>
<td>Pressurizer</td>
<td>578</td>
<td>Pressurizer heaters / main steam plenums</td>
<td>36.1</td>
<td>1.7</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Cylindrical pressurizer</td>
<td>61.4</td>
<td>6.9</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Reactor pressure vessel top head</td>
<td>41.2</td>
<td>2.2</td>
</tr>
</tbody>
</table>

Table 7 – SMR Primary System Volume Data [21]

<table>
<thead>
<tr>
<th>RCS Region</th>
<th>Nominal Volume (ft³)*</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hot Leg (lower riser, riser transition, upper riser, riser supports)</td>
<td>635</td>
</tr>
<tr>
<td>Cold Leg [feedwater plenums, downcomer transition, downcomer (lower riser), core barrel, RPV bottom head, flow diverter]</td>
<td>578</td>
</tr>
<tr>
<td>Core Region (fuel assembly region and reflector cooling channels)</td>
<td>89</td>
</tr>
<tr>
<td>SG Region</td>
<td>621</td>
</tr>
<tr>
<td>PZR Region (main steam plenums, PZR, RPV top head)</td>
<td>578</td>
</tr>
<tr>
<td>PZR Region, cylindrical (main steam plenums and PZR)</td>
<td>487</td>
</tr>
</tbody>
</table>

*Volumes are rounded to the nearest cubic foot.

The model utilizes various hydrodynamic components, junctions, volumes, heat structures, and process models. The primary loop of the reactor has been modeled to a
higher degree of detail than the secondary loop to simulate the natural circulation cooling properly. The secondary loop is modeled to represent the steam generator helical coil tubes, the steam pipe system, and feedwater subsystems. The model's focus is to accurately simulate the natural circulation cooling phenomena during the normal operating conditions of the NuScale Power SMR. The steady-state model was created to anticipate and prepare for the transient analysis of a rupture of a helical-coil steam generator tube.

Several hydrodynamic components were used throughout the model, including branches, annuluses, time-dependent volumes, pipes, time-dependent junctions, and single-junctions. The components modeled on the primary side of the reactor are the lower plenum, the left and right core of the reactor, the fuel rods, the lower riser, the middle riser, the upper risers, the upper plenum, the pressurizer, the primary steam generator, and the downcomers. The components modeled on the secondary side are the feedwater source, the feedwater pipe, the secondary steam generator or the steam generator helical coil tubes, the steam pipe, and the sink. The components that were modeled with a branch are the lower plenum and the upper plenum. The components modeled with an annulus are the steam generator downcomer 1 (cold leg 1) and the steam generator downcomer 2 (cold leg 2). The components modeled with a time-dependent volume are the time-dependent volume that replaces the pressurizer, the feedwater source, and the sink. The components that were modeled with a pipe are the left core, the right core, the lower riser, the middle riser, the upper riser 1, the upper riser 2, the primary steam generator, the pressurizer, the feedwater pipe, the secondary
steam generator, and the steam pipe. The junction of the water source to the feedwater pipe is modeled using a time-dependent junction. In contrast, all other junctions between components are modeled using single junctions besides where branch components are present.

The lower plenum is input as component 100 and utilizes a single-volume branch, creating connections between the two downcomers and the left and right cores. The left and right cores are modeled identically as components 110 and 120, respectively, and both pipes utilize an eight-volume pipe. The heat structure labeled as component 810 represents the fuel rods within the core and utilizes eight heat structures. The lower riser is modeled with a seven-volume pipe (labeled component 200) and is connected to the left and right core using two single-junction components or known to RELAP5-3D as a sngljun component. The next component is the middle riser (component 210), which is modeled using a five-volume pipe and is connected to the lower riser and the upper risers using single junctions, just as the cores are connected to the lower riser. The upper riser is split into two pipes with 25 volumes each and is labeled as components 220 and 221. The upper risers are connected to the upper plenum (component 350), another branch that connects the pressurizer and the primary steam generator. This branch is implemented to simulate the downward turn area for the coolant of the upper reactor vessel to the steam generators. The pressurizer is labeled as component 360 and modeled using a six-volume pipe as done in the open literature by Skolik, et al. [26]. The pressurizer was provided a self-initialization option control to properly control the operating pressure of 1850 psia throughout the
system. This required a new time-dependent volume (component 361) to be modeled, replacing the pressurizer and connecting itself to the pressurizer using a single-junction component (component 362). The primary steam generator is labeled as component 400 and is modeled using a 15-volume pipe for simplification. The heat structure labeled as component 820 has been utilized to simulate the heat transfer between the steam generator of the primary loop to the helical coil steam generator tubes of the secondary side (secondary steam generator), which is labeled as component 401. Another single-junction component connects the primary steam generator to the two downcomers labeled as component 500 and 501, respectively. The downcomer was split into two separate pipes to allow the length to be larger than the flow area as recommended by SCDAP/RELAP5 Development Team, which is also performed by Skolik, et al. [26]. The downcomers conclude the final components of the primary loop. They are connected to the lower plenum branch to simulate the natural circulation cycle's lower turn area, which flows upwards through the core to repeat the process over again.

The feedwater source of the secondary loop, labeled as component 740, is modeled with time-dependent volume or known as tmdpvol to RELAP5-3D. This represents the boundary condition of the feedwater subsystem. The feedwater source is connected to the feedwater pipe using a time-dependent junction known as tmdpjun to RELAP5-3D and is described as component 750. The feedwater pipe or component 760 is modeled using 25 volumes, with three of the volumes orientated horizontally and the remaining 22 volumes oriented vertically upwards towards the vertically oriented secondary
steam generator. The feedwater pipe is connected to the secondary steam generator with a single junction component (component 770). The secondary steam generator is then connected to the steam pipe (component 780) with another single junction (component 775). The steam pipe consists of 25 volumes. The first six volumes are orientated vertically, and the last 19 are orientated horizontally to simulate the piping to the secondary building of the power plant where the turbine is located. The final connection of the secondary loop again utilizes a single junction and connects the steam pipe to boundary condition represented by another time-dependent volume labeled as component 800. All hydrodynamic components and heat structures used in RELAP5-3D to model the NuScale Power SMR can be seen in the nodalization diagram depicted in Figure 19.

Due to the differences of the NuScale Power SMR to a typical PWR, the names of the components throughout the system are not as straightforward or standard compared to typical PWRs, especially when attempting to translate them to a RELAP5-3D model. This results from the unique designs of the NuScale Power SMR with natural circulation cooling of the core and the helical coil steam generator.
Figure 19 – Nodalization Diagram of the NuScale Power SMR RELAP5-3D Steady-State Model

7.2 PROCESS MODEL USE AND INTERPRETATION BY RELAP5-3D

Code inputs that impact the accuracy of the steady-state calculation of this model are the process models specified for the volumes and junctions. Various process models can alter the calculations performed by RELAP5-3D code. The process models are used to simulate processes that have to do with large spatial gradients or when complexity is high and empirical models are needed [35]. Some of these processes are only for specific components, but the majority of the process models are generalized.
Some of the process models that are general to most components are the area change model, choked flow, reflood model, condensable or noncondensable option, water packing scheme, CCFL option, mixture level tracking option, along with the vertical and thermal stratification models. The process models are specified on a specific input card of each component, typically on cards CCC1001 and CCC1101. The application of these models falls onto the user to be inputted correctly. As briefly discussed in section 6, the process models are inputted using packed word formats for each volume and junction component using tlpvbfe and jefvcahs, respectively. These packed word formats are considered volume control flags and junction control flags according to the RELAP5-3D manual and appendices. Each letter is referred to as a digit by RELAP5-3D, and each digit corresponds to a different process model. The digits are entered typically to activate or deactivate a process model with additional options for various process models. The digit zero for the packed words usually deactivates models, but this does not always hold for every model, option, or scheme. For example, in the packed word jefvcahs, the digit ‘c’ pertains to applying (or not applying) the choked-flow model. In this case, the choke flow model is activated using zero and is deactivated using one. Some of the process models that should be activated (or deactivated) are recommended explicitly by the RELAP5-3D manuals and appendices. In contrast, others should be activated (or deactivated) with a degree of strategy and intuitive thinking to input them correctly. Inputting the process models at the correct locations and situations allows RELAP5-3D to accurately perform the mass, momentum, and energy calculations [35]. This is especially important when
modeling a peculiar phenomenon like natural circulation cooling within the NuScale Power SMR.

The use of the digit zero for the input value of the process models was used for most component volumes and junctions throughout the model shown by Freitag [36]. This is of particular interest when specific systems and subsystems of the reactor would experience various phenomena and should have specific process models applied. As discussed previously, the choked flow model is activated with the digit zero. This means the choked flow model would have been applied to every junction within the reactor system. The RELAP5-3D manuals and appendices recommended the water packing scheme and the vertical stratification model should be applied when modeling a pressurizer within a reactor model. Both models would be applied to the pressurizer if all digits are zero and applied to every other volume modeled in the reactor. The water packing scheme and the vertical stratification model can only be applied to vertically orientated components. The NuScale Power SMR is composed of various components both horizontally and vertically orientated. Another essential model that should be used correctly and can alter the equations applied is the area change model. This model consists of three options: smooth area change, full abrupt area change, and partial abrupt area change. The smooth area change is used for junctions without an area change. The full abrupt area change model utilizes $K_{\text{m,m}}$, area apportioning at a branch, restricted junction area, and extra interphase drag. The partial abrupt area change model does not utilize $K_{\text{m,m}}$, but includes area apportioning at a branch,
restricted junction area, and extra interphase drag. The partial and full abrupt area change model is recommended to be used at branches [37].

Due to the unique phenomena occurring within a natural circulation cooling system and the limited capabilities of the RELAP5-3D code for this SMR, the best options for the volume and junction control flags were not necessarily chosen with the recommendations of the RELAP5-3D manuals and appendices. The volume control flags, tlpvkle, chosen for the pipe volumes of the left and right cores (components 110 and 120), were inputted as 0011100. This means the thermal front tracking model is deactivated, the mixture level tracking is deactivated, the water packing scheme is deactivated, the rod bundle interphase friction model is applied, the wall friction effects are applied, and the nonequilibrium calculation is specified. The junction control flags, jefvcahs, chosen for the pipe junctions of the left and right cores, were inputted as 00001000. The digit ‘j’ would apply or not apply the jet junction model for a junction component, but this process model is not used. The following digits after ‘j’ mean that the modified PV term in energy equations is not applied, the CCFL option is not applied, the horizontal stratification entrainment/pull-through model is not used, the choking model is not applied, the smooth area change model is applied, the nonhomogeneous (two-velocity momentum equations) option is activated, and the momentum flux in both the to volume and the from volume is applied. Many of the other components throughout the model had similar volume, and junction flags applied to them. One notable difference from the core volume control flags to the other components is digit ‘b.’ The rod-bundle interphase friction model is applied to
the core. Still, all the other components besides the secondary steam generator (component 401) utilize the pipe interphase friction model for digit ‘b.’ The rod bundle interphase friction model is also applied to the secondary steam generator heat exchanger [26]. Figure 20 can be seen below to understand better how the volume and junction control flags are implemented and input for the reactor core using the RELAP5-3D input code.

```
*Crdf tlpvbfe Vol# (X-Cood Vol Cont Flags)
1101001 001100 8
*
*Crdf jefvcahs Jun# (Junct Cont Flags)
1101101 00001000 7
```

Figure 20 - Volume and Junction Control Flags of Reactor Core

**7.3 Z-COORDINATE PLACEMENT OF MODEL COMPONENTS**

The z-axis coordinates for the components must be orientated strategically to simulate the natural circulation cooling properly. The components representing the flow upwards through the core and risers, and the components representing the flow downwards through the steam generator and downcomers to the lower plenum of the reactor, must be orientated on the z-axis at identical heights or lengths. If an additional component with a length were added to the downcomer side, then the length of the opposing flows would no longer be identical, and a z-coordinate error would be seen. As discussed, it can be essential to avoid larger flow areas than the volume lengths [26]. The author’s responsibility has fallen on creating the model as accurately as
possible with the information released by NuScale FSAR. The model created by Freitag [36] does this strategically to simulate the natural circulation cooling of the primary loop. The length stack-ups of the components along the z-axis can be modeled in many different ways. Like Freitag [36], the core, the lower riser, the middle riser, and the upper riser are all the components stacked up to make up the coolant flow upwards through the primary loop. The primary steam generator and downcomers are the components that comprise the flow of the coolant downwards. The NuScale Power SMR model has been created to allow for the components representing the upward flow to be the same length as the components representing the downward flow of the reactor. This stack-up does not account for the lower plenum (component 100), the upper plenum (component 350), the pressurizer (component 360), and the time-dependent volume of the pressurizer. This is because the lower plenum is a branch at the lowest z-coordinate with all other components stacked on top of this component. The upper plenum is also a branch with the pressurizer and the time-dependent volume of the pressurizer stacked above at the highest z-coordinates. These components would not alter the lengths of the upward and downward flows.

7.4 MODEL OF THE HELICAL-COIL STEAM GENERATOR TUBES
As discussed previously, natural convection cooling requires smooth transitions in geometries with very low restrictions to the flow of the coolant to allow for optimal operating conditions and heat transfer. The heat is transferred from the primary loop by conduction to the secondary loop through the helical-coil steam generator tubes to form superheated steam. The helical coil tubes are a large bundle of tubes angled
upward with relatively small diameters to maximize the heat transfer. Modeling the NuScale Power SMR steam generator in RELAP5-3D was complex due to the 1,380 helical coil tubes. RELAP5-3D does not have the option for a helical coil geometry. It has been simplified by using a pipe with all 15 volumes orientated at 16.5 degrees to represent the inclination angle of the tubes [26]. It has been seen in the open literature that only modeling the helical coil steam generator tubes at the 16.5-degree inclination angle is not enough to properly represent this system and the heat transfer of this geometry [26]. It has also been seen that the mass flow rates must be increased accordingly, or the helical coil steam generator heat transfer surface area must be increased. Similar studies were also performed on this iPWR model that verified these findings. Model adjustments were needed to properly model the geometry of the helical coil tubes and the complexities of the natural circulation cooling. Therefore, the heat transfer surface area of the heat structure 820 representing the helical coil steam generator tubes was increased by 30 percent to lower the temperatures and reduce the flow rates of the system [26]. This resulted in optimal operational temperatures within the core and throughout the entire reactor. However, mass flow rates were still higher than the best estimate flow as provided in the NuScale FSAR. This led to lowering the mass flow rates, even with the pressure, temperatures, and void fractions all at steady-state conditions of the reactor.
7.5 MANIPULATION OF FLOW ENERGY LOSS COEFFICIENTS

Patrick Freitag [36] modeled the NuScale Power SMR system to a degree of complexity. Still, he could not fully and accurately simulate the design operating parameters and conditions released by NuScale Power LLC to the U.S. NRC for the Design Certification Application. After analyzing the research performed by Patrick Freitag, conclusions were made as to why the steady-state conditions could not be accurately simulated. First, the focus was the frictional losses and, more particularly, the flow energy loss coefficients through the core. The reasoning behind this part of the study was due to the sensitivity of natural circulation to flow changes and restrictions, along with the system exhibiting high flow rates. The card pertaining to forward and reverse flow energy loss coefficients for the pipe, annulus, or pressurizer components (CCC0901) was manipulated in many ways to understand the resultant differences in temperature, void fraction, velocity, and mass flow rate. These experiments were performed with the pressure of the system controlled and held constant by the pressurizer system at 1850 psia. The author initially anticipated that the flow energy loss coefficients only needed to be manipulated for the components that represent the reactor's core. In contrast, the rest of the primary loop components’ loss coefficients could be controlled at 0.0 during each variation. Adjusting the flow energy loss coefficients within the core influenced the inlet, average, and outlet temperatures of the core and the mass flow rates of the system. The core exit temperature was calculated by RELAP5-3D to be closer to the \( T_{\text{in}} \) value but still within the range of 497°F to 590°F, as seen in Figure 21.
The best estimate mass flow rate was achieved at higher flow energy loss coefficients within the core but boiling within the upper volumes of the core and risers also occurred. After many iterations and experiments in manipulating the flow energy loss coefficients from values of 0.0 to 3.0 within the reactor core, it was found that it was not enough to only apply flow energy loss coefficients to the reactor's core. Therefore, the author decided that the flow energy loss coefficients needed to be applied throughout the reactor primary loop systems to simulate the operational temperatures and mass flow rates correctly. This is deemed a critical factor in obtaining the reactor’s specified design parameters for a reactor utilizing natural convection. This was understood through further experimental simulations in manipulating the flow energy loss coefficients [38]. Vijayan, et al. [38] observed that mass flow rates can be
reduced by increasing the flow energy loss coefficients throughout a reactor that utilizes natural circulation cooling and observed that the effects of the flow energy loss coefficients can only be determined experimentally, which helped validate the basis of this study.

Flow energy loss coefficient values between 0.1 to 2.0 were applied to the components of the primary system. It was also seen by Vijayan, et al. [38] that applying flow energy loss coefficients to hot leg components of a natural circulation system can cause instabilities, while applying flow energy loss coefficients to cold leg systems can ensue a stabilizing effect. This led the author to apply lower and higher flow energy loss coefficients throughout primary and secondary systems. A loss coefficient of 0.0 was applied to all components of the secondary side of the reactor model. The downcomers (components 500 and 501) had a loss coefficient of 0.6. The left and right core (components 110 and 120) also had a loss coefficient of 0.6. A loss coefficient of 1.0 was applied to the primary steam generator (component 400). This was done because a reduction of the mass flow rate occurs through an increase in flow energy loss coefficients. The 1.0 loss coefficient was chosen for the primary steam generator to allow more heat transfer through the helical coil steam tubes and to, in turn, lower the temperatures through the downcomer to the core of the reactor. This was done similarly within the core of the reactor model. A loss coefficient of 0.3 was applied to all other components and junctions besides the pressurizer system components. These components consisted of the lower riser, middle riser, upper risers, lower plenum, upper plenum, and the junctions to and from. The flow energy loss
coefficients were applied to the various components to ensure the system's stability while also applying loss coefficients consistently throughout the system. The primary system components needed to have low loss coefficients, yet the difference between these loss coefficients and the cold leg loss coefficients could not be too large. These findings led to the final values chosen for the flow energy loss coefficients throughout the primary system and can be seen in tabular form in Table 8.

Table 8 – Primary Loop System Components Flow Energy Loss Coefficients

<table>
<thead>
<tr>
<th>Component Name</th>
<th>Component Number</th>
<th>Flow Energy Loss Coefficient, $A_F$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Lower Plenum</td>
<td>Component 100</td>
<td>0.3</td>
</tr>
<tr>
<td>Left Core</td>
<td>Component 110</td>
<td>0.6</td>
</tr>
<tr>
<td>Right Core</td>
<td>Component 120</td>
<td>0.6</td>
</tr>
<tr>
<td>Left Core SNGLJUN</td>
<td>Component 130</td>
<td>0.3</td>
</tr>
<tr>
<td>Right Core SNGLJUN</td>
<td>Component 140</td>
<td>0.3</td>
</tr>
<tr>
<td>Lower Riser</td>
<td>Component 200</td>
<td>0.3</td>
</tr>
<tr>
<td>Lower Riser SNGLJUN</td>
<td>Component 205</td>
<td>0.3</td>
</tr>
<tr>
<td>Middle Riser</td>
<td>Component 210</td>
<td>0.3</td>
</tr>
<tr>
<td>Middle Riser SNGLJUN 1</td>
<td>Component 215</td>
<td>0.3</td>
</tr>
<tr>
<td>Middle Riser SNGLJUN 2</td>
<td>Component 216</td>
<td>0.3</td>
</tr>
<tr>
<td>Upper Riser 1</td>
<td>Component 220</td>
<td>0.3</td>
</tr>
<tr>
<td>Upper Riser 2</td>
<td>Component 221</td>
<td>0.3</td>
</tr>
<tr>
<td>Upper Plenum</td>
<td>Component 350</td>
<td>0.3</td>
</tr>
<tr>
<td>Pressurizer</td>
<td>Component 360</td>
<td>0.0</td>
</tr>
<tr>
<td>TMDPVOL for Pressurizer</td>
<td>Component 361</td>
<td>0.0</td>
</tr>
<tr>
<td>TMDPVOL Pressurizer SNGLJUN</td>
<td>Component 362</td>
<td>0.0</td>
</tr>
<tr>
<td>Primary SG</td>
<td>Component 400</td>
<td>1.0</td>
</tr>
<tr>
<td>Primary SG SNGLJUN 1</td>
<td>Component 405</td>
<td>0.6</td>
</tr>
<tr>
<td>Primary SG SNGLJUN 2</td>
<td>Component 406</td>
<td>0.6</td>
</tr>
<tr>
<td>Downcomer 1</td>
<td>Component 500</td>
<td>0.6</td>
</tr>
<tr>
<td>Downcomer 2</td>
<td>Component 501</td>
<td>0.6</td>
</tr>
</tbody>
</table>
Manipulating the form-loss-coefficients throughout a system comprised of over 25 components was deemed a complex task and was not the focus of this research. After many attempts to obtain the proper mass flow rates while also staying within the operational temperature range of the reactor coolant system, the author decided the best estimate mass flow rate would be a result of the normal operating pressure and temperatures that were calculated to be identical to the NuScale FSAR released design parameters and conditions. It was concluded that the RELAP5-3D code might need to be more complex for the natural circulation phenomena or that a more complex 3D model would be needed to obtain the best estimate mass flow rate of 1294.44 lbm/s.

7.6 RESULTS OF THE STEADY-STATE MODEL

The plots presented below depict the data calculated by RELAP5-3D for the model created to simulate the steady-state operation conditions of the NuScale Power SMR. The development of the model has been discussed in the previous sections. The most critical parameters have been plotted against time, and the most critical components within the system have been chosen to be presented. The simulation has been run for 1,500 seconds with a minimum time step of 1.0-6 and a maximum time step of 0.001 to allow for any unseen issues to transpire and keep the estimated error low. The core of the reactor is split into two pipe components labeled left core and right core. These two components are modeled identically, and for presentation, the left core data has been displayed in the plots below.
Table 9 - RELAP5-3D Calculated Results of Reactor Parameters [28]

<table>
<thead>
<tr>
<th>Steady State Model</th>
<th>Parameter</th>
<th>Result</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Core Inlet Temperature (°F)</td>
<td>521.17</td>
</tr>
<tr>
<td></td>
<td>Core Average Temperature (°F)</td>
<td>559.82</td>
</tr>
<tr>
<td></td>
<td>Core Outlet Temperature (°F)</td>
<td>588.37</td>
</tr>
<tr>
<td></td>
<td>System Pressure (psia)</td>
<td>1850.0</td>
</tr>
<tr>
<td></td>
<td>Core Average Coolant Velocity (ft/s)</td>
<td>3.41</td>
</tr>
<tr>
<td></td>
<td>Core thermal output (MWt)</td>
<td>160.0</td>
</tr>
<tr>
<td></td>
<td>Best estimate flow (lbm/s)</td>
<td>1765.0</td>
</tr>
<tr>
<td></td>
<td>Heat transfer area on fuel surface (ft²)</td>
<td>6275.6</td>
</tr>
<tr>
<td></td>
<td>Core flow area (ft²)</td>
<td>9.79</td>
</tr>
<tr>
<td></td>
<td>Diameter of active core (ft)</td>
<td>4.94</td>
</tr>
<tr>
<td></td>
<td>Height of active core (ft)</td>
<td>6.57</td>
</tr>
<tr>
<td></td>
<td>Outer Tube Surface Area (Calculated) (ft²)</td>
<td>5453.12</td>
</tr>
<tr>
<td></td>
<td>SG Surface Area (30% Increase)</td>
<td>7089.053</td>
</tr>
</tbody>
</table>
Plots of Pressure Versus Time Plots for Important Components:

Figure 22 – Pressurizer Pressure vs. Time RELAP5-3D Steady-State Calculation

Figure 22 displays the pressurizer system pressure versus time throughout the steady-state calculation. The system pressure is appropriately controlled throughout the duration of time at approximately 1850 psia for all six volumes of the pressurizer component. The pressure within the core is at the optimal operation conditions with an average of about 1863 PSI between the eight volumes, as seen in Figure 23.
Figure 23 – Left Core Pressure vs. Time RELAP5-3D Steady-State Calculation
Figure 24 – Secondary Steam Generator Pressure vs. Time RELAP5-3D Steady-State Calculation

Figure 24 shows the pressure over time of the secondary steam generator or the helical coil steam generator tubes. The plot displays the pressure of the secondary side, which was input as 1000 psia. All 25 volumes of the secondary steam generator only experience pressure with a high of 1007 psia for volume one and a low of about 999 psia. This is only a slight pressure drop and is expected as the flow moves through the helical coil tubes and extracts the heat from the primary side.
Plots of Temperature Versus Time Plots for Important Components:

Figure 25 – Lower Plenum Temperature vs. Time RELAP5-3D Steady-State Calculation

Figure 25 displays the temperature of the lower plenum throughout the 1,500-second simulation. The lower plenum is a branch component representing the lower turn area of the reactor where the downcomers flow. The area angles the downward flow upward to flow the coolant through the core. The lower plenum temperature represents the inlet temperature of the core. The core inlet temperature was calculated to be approximately 521°F. This is 24°F above the specified 497°F inlet core temperature by the NuScale FSAR but is well within the operating range.
Figure 26 shows the temperature of the coolant flowing through the eight volumes of the core. The first volume exhibits a temperature of approximately 530°F. The eighth volume of the core exhibits a temperature of 588°F, representing the core's exit temperature. The average temperature of the core is approximately 560°F, which is about 17°F above the specified 543°F average core temperature by the NuScale FSAR. The core experiences a temperature increase of 67°F from the lowest volume to the highest volume.
Figure 27 shows the temperatures of the 15 volumes that make up the primary steam generator. The temperature drops to 576°F in the first volume of the steam generator. As more heat is extracted to the secondary loop through each volume, the temperature drops from 576°F to 521°F in volume 15. There is an approximate drop of 55°F from volume one to volume 15 of the steam generator, and the temperature of 521°F resembles the inlet temperature of the core.
Figure 28 – Secondary Steam Generator Temperature vs. Time RELAP5-3D Steady-State Calculation

Figure 28 is displayed to understand the temperature increase through helical coil tubes on the secondary side, which extracts the thermal heat of the primary loop to form superheated vapor within the secondary loop. The secondary side has an operating temperature of 500°F. Volume one of the helical coil tubes exhibits a temperature of 477°F. Volume two immediately increases up to 525°F with volume three following a similar trend and increasing to 540°F. The final 12 volumes of the helical coil tubes exhibit similar temperatures of about 546°F, equating to an increase in temperature of approximately 69°F from volume one to volume 15. With the
secondary loop operating at 1000 psia, the saturation steam temperature would be approximately 546°F. This means that volumes four through 15 (the final 12 volumes) are experiencing a degree of boiling, and superheated vapor is forming within these volumes to be sent upward through the steam pipe. To further understand this, the liquid and vapor void fraction plots of the helical coil tubes can be seen in Figure 31 and Figure 32, respectively.

Plots of Liquid Void Fraction Versus Time for Important Components:

![Liquid Void Fraction vs Time](image)

Figure 29 – Left Core Liquid Void Fraction vs. Time RELAP5-3D Steady-State Calculation
Figure 30 – Primary Steam Generator Liquid Void Fraction vs. Time RELAP5-3D Steady-State Calculation

Liquid void fractions for the core and the primary steam generator can be seen above in Figure 29 and Figure 30, respectively. For the entire duration of the simulation, the liquid void fractions are held at 1.0. This is as expected during steady-state operations for any nuclear reactor.
As discussed above, the liquid void fraction for the volumes of the secondary steam generator can be seen in Figure 31. Volumes one through four all exhibit a 100 percent liquid void fraction. This would mean that the bottom four volumes of the helical coil tubes are constantly filled with total liquid water from the feedwater. The following volumes slowly decrease in liquid void fraction until being practically 100 percent vapor within volume 15. To better understand the data and occurrences within the secondary loop, the vapor void fractions can be seen below in Figure 32 and Figure 33 for the secondary steam generator and steam pipe, respectively.
Figure 32 – Secondary Steam Generator Vapor Void Fraction vs. Time RELAP5-3D Steady-State Calculation

Figure 32 shows the vapor from within the volumes of the helical coil tubes. Volumes one through four exhibit a vapor void fraction of 0.0, meaning these volumes are at 100% liquid water, the opposite of the liquid void fraction shown prior. It is not until volume five (the yellow bottom line) that vapor forms in the helical coil tubes. Volume five experiences a vapor void fraction of about 0.1, and each volume moving
upward through the tubes has an increasing vapor void fraction until reaching volume 15, which exhibits almost 100% vapor.

Figure 33 – Steam Pipe Vapor Void Fraction vs. Time RELAP5-3D Steady-State Calculation

Figure 33 shows the steam pipe of the secondary loop. All volumes exhibit a vapor void fraction above 0.975 with an average of about 0.98. These are the results expected for a steam pipe within a power plant. The remaining void fraction of 0.02 would most likely comprise liquid droplets caused by condensation within the pipe. This is common to occur and shows further validation of this steady-state model.
This is an important parameter to monitor for any nuclear reactor due to the input used to model the pressurizer. The self-initialization option control card 147 was used to allow the pressurizer to control the system pressure at 1850 psia at all times. To the RELAP5-3D code, this card replaces the actual pressurizer modeled with six volumes, with a time-dependent volume and uses a single junction to connect the two components. The vapor void fraction of the pressurizer time-dependent volume is depicted in Figure 34. The void fraction of this volume can be seen to be approximately 67 percent vapor and 33 percent liquid.
Plots of Velocity Versus Time for Important Components:

Figure 35 – Left Core Velocity vs. Time RELAP5-3D Steady-State Calculation
The velocities throughout the primary system have also been presented above in Figure 35 and Figure 36. The velocity through the left core exhibits an average value of about 3.9 ft/s through the reactor's core and approximately 1.45 ft/s through the primary steam generator.
Plots of Mass Flow Rates Versus Time for Important Components:

Liquid Mass Flow Rate vs Time

Figure 37 – Left Core Mass Flow Rate vs. Time RELAP5-3D Steady-State Calculation
Figure 37 shows the mass flow rate through the left core of the reactor. Due to the model's core being split into two pipes, half the total flow rate can be seen in this plot. As previously discussed, the author has decided that the best estimate flow rate is a result of the other system parameters and conditions. The mass flow rate through the left core is approximately 882.5 lb./s, and the total mass flow rate through the system can be seen in Figure 38 for the primary steam generator at about 1765.0 lb./s. This is slightly higher than the specified best estimate flow rate of 1294.44 lb./s by the NuScale FSAR. It is possible to adjust the core model while configuring the best flow.
energy loss coefficients throughout the system to simulate the best estimate mass flow rate properly. Further experimental data and trials would have to be obtained and analyzed to understand better. The RELAP5-3D code may not have the most advanced capabilities of modeling natural circulation cooling within a nuclear power plant and may present limitations in calculating the best estimate flow.

Figure 39 – Secondary Steam Generator Mass Flow Rate vs. Time RELAP5-3D Steady-State Calculation

Figure 39 shows the mass flow rate of the secondary steam generator or the helical coil tubes. This is displayed to understand the flow rates of the secondary loop and
expressly understand how the flow rates change as heat is extracted from the primary loop. Volumes one through eight are the lower volumes of the secondary steam generator, and all exhibit the calculated flow rate of about 200 lb./s. Volumes nine through 14 show an apparent decrease in flow rates from approximately 180 lb./s down to 90 lb./s. As heat is extracted and vapor forms in the upper volumes of the helical coil tubes, less liquid will be present, which would lower the mass flow rate of the liquid and increase the vapor flow rate. A mass flow rate of 147.81 lb./s was input into the model for the second loop of the system. The secondary loop exhibited a flow rate of approximately 200 lb./s. This was not an issue because benefits were seen in controlling and predicting the natural circulation cooling parameters of the primary loop when the flow rate of the feedwater source was increased, as done by Skolik, et al. [26].
Plots of Power Versus Time for the Core of the Reactor:

Core Power vs Time
Left Core

Figure 40 – Left Core Power vs. Time RELAP5-3D Steady-State Calculation
The core power outputs for the left and right core are displayed in Figure 40 and Figure 41, respectively. These plots have been presented using SI units of Watts for convenience and comparison to the reported data. The reported power output of the NuScale FSAR is 160 MWt for a single operational NuScale Power SMR module. The model developed using RELAP5-3D represents a single NPM. The two pipes representing the left and right core have been broken up into eight volumes, meaning the core comprises 16 volumes within the RELAP5-3D model. Each volume exhibits approximately 10.0 MWt of power and can be seen in Figure 40 and Figure 41. With 16 volumes making up the two cores, this equates to 160.0 MWt of power produced.
by the core. This is identical to the specified power output reported in the NuScale FSAR.
8 HELICAL-COIL TUBE RUPTURE RELAP5-3D MODEL
OF THE NUSCALE POWER SMR

8.1 DEVELOPMENT OF THE HELICAL-COIL STEAM GENERATOR TUBE RUPTURE MODEL

The helical-coil steam-generator tube-rupture model was developed based on the NuScale Power SMR steady-state model discussed above in section 7. The author wanted to simulate a rupture at a weld point of one of the 1,380 helical coil tubes that make up the once-through steam generator. As stated, the secondary steam generator represented the helical coil tubes of the steam generator. It was thought that the weld points of the helical coil tubes proposed a potential failure point in the future life of the reactor. The author believed only one tube would be a practical failure, and failure at multiple tube weld points would be less likely. There has been minimal literature published on the tube rupture of the helical coil once-through steam generator, which motivated the need for this study.

Multiple methods were initially used and implemented to model the rupture correctly. Some methods proposed issues within the RELAP5-3D code, while others provided valid results. The author initially modeled the tube rupture with a tee or a branch component, but issues and errors arose within this model. As discussed in section 7.3, the z-coordinate orientation of the components was strategically laid out to simulate the upward and downward flow through the reactor core. The lengths and elevation changes of the upward flow components and the downward flow components were
placed or stacked at identical points along the z-axis to simulate the natural circulation of the coolant properly. For example, if an additional component with a length needed to be implemented on the downcomer side, then an exact length of this component needed to be added to the riser side. This was problematic because the steady-state model was created to simulate the exact parameters and geometries released in the NuScale FSAR. This was an obstacle when implementing a new component at the steam generator on the downcomer side of the primary loop. A z-coordinate error was seen when implementing the hydrodynamic branch component to connect the primary steam generator (component 400) to the secondary steam generator (component 401).

The author utilized the knowledge obtained through Edward’s pipe blowdown experiment to simulate a similar pipe rupture scenario within the steam generators. Therefore, a single junction component is chosen to allow for a proper junction or rupture between the primary and the secondary steam generators. The single-junction component does not add length as the other hydrodynamic components would. This allowed for the connection at volume one of the primary steam generator and volume 15 of the secondary steam generator without a z-coordinate error. The single-junction (component 402) at these volumes can be seen in the tube rupture nodalization diagrams of Figure 42 and Figure 43. The abrupt area change model was initially applied to this component. Still, this model only allows the junction area to be equivalent to the smallest flow area of the two connecting volumes. The smallest flow area of the two components would be 0.785 ft², but this was not the rupture area of one of the helical coil tubes. Therefore, this process model was changed to the smooth area
change, which does not have a limitation on the chosen junction area. This allowed for the tube rupture area of 0.0021 ft² to be implemented into the input card correctly. If there were a failure at a weld point of the helical coil tubes, it would likely initially be a smaller crack and not be equivalent to the entire cross-sectional area of the tube. Due to the minimal open literature published on this study, it was decided that the complete break of one tube at the weld point would be the focus of this study. Until further evidence and experiments are performed to understand the degree or size of a potential break at one of the weld points, it is hard to predict the percentage of the tube area that would be opened and exposed to the primary loop. The choked flow model was applied to the single junction to model the break properly, allowing the abrupt area change model to be removed and the correct process model to be implemented. The abrupt area change model cannot be activated while the choke flow model is activated. The nonhomogeneous option was applied because of the naturally circulating water flowing through the hole to a mostly vapor-filled secondary steam generator and a steam pipe. Like the rest of the modeled components, the momentum flux in both the ‘to’ volume and the ‘from’ volume was specified at the single junction. This allowed for the geometry of the break to be correctly modeled while applying the appropriate process models.
Figure 42 – Helical-Coil Steam Generator Tube Rupture Nodalization Diagram
Figure 43 – Nodalization Diagram of the Main Components used in the Helical-Coil Steam Generator Tube Rupture

Figure 44 – Nodalization Diagram of the Surrounding Volumes and Junctions of the Helical-Coil Steam Generator Tube Rupture Location
8.2 RESULTS OF THE HELICAL-COIL STEAM GENERATOR TUBE RUPTURE

The plots depicted below show the critical parameters and conditions of the helical-coil steam generator tube rupture. RELAP5-3D does not have the capabilities to model a steam generator of helical coil geometry. Therefore, modifications to the inputs were made to properly simulate this geometry and phenomena to occur, as discussed in section 7.4. The simulation of the tube rupture was run and plotted for 2,000 seconds. The tube rupture plots display the time along the x-axis of 1500 seconds to 3500 seconds. The reasoning for this is because the tube rupture needed to be run as a restart problem in RELAP5-3D to use the data of the steady-state calculations. Therefore, the results of the tube rupture are a continuation of the steady-state data file, which starts at 1500 seconds. The main difference between the steady-state model and the tube rupture model is the addition of the single junction to simulate the break. The initial plots are shown below display the conditions and parameters of the primary coolant system to show the results and effects on the natural circulation cooling after the tube rupture was initiated. Following these plots will be the plotted data of the steam generators and the steam pipe to understand the phenomena occurring within the secondary loop.
Primary Loop Results of Tube Rupture:

Figure 45 – Power vs. Time of the Left Core RELAP5-3D Tube Rupture Calculation
The first plots that are displayed in Figure 45 and Figure 46 are the plots of the left and right core power output over time. Both the left and right core power output have been shown to display the gross output of 160 MWt for the core of one NuScale Power Module. As done in section 7.6, SI units were chosen to be used for these plots to be consistent with the units used in the NuScale FSAR. AptPlot displays the power unit of Watts for these plots. The purpose of displaying these plots before any others is to show the power output of the core has not been changed due to the rupture of the helical coil tube. Short-term oscillations of the transient can be seen immediately after the tube rupture before converging back to the thermal power output of $1e+07$ W or 10 MW per volume of the core. All the volumes exhibit an identical power output and are
plotted over each other, causing the figure to appear as only one line. Figure 45 and Figure 46 both depict eight volumes each, making up the 16 volumes of the core model, as discussed during the analysis of the core power output of the steady-state results. The power output of each volume is identical at the value of 10 MW. With all 16 volumes at 10 MW each, the gross power output totals 160 MWt as specified in the NuScale FSAR and shows the tube rupture does not disrupt the reactor's power output.

Figure 47 displays the pressure of the left core during the tube rupture transient calculation. The pressure immediately drops and exhibits slight oscillations for a short

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Figure 47 – Pressure vs. Time of the Left Core RELAP5-3D Tube Rupture Calculation
period. The immediate drop and oscillations are not significant and do not pose a concern but do show that there is an immediate impact on the primary system pressure of the reactor. The core pressure drops from approximately 1865 psia to 1850 psia before converging back to the same values calculated during the steady-state simulations.

Figure 48 – Pressure vs. Time of the Pressurizer RELAP5-3D Tube Rupture Calculation

Figure 48 displays the pressure over time of the pressurizer component. The pressurizer also drops slightly when the tube rupture is initiated. It is a very minimum
drop, but the pressurizer does experience a small drop of 0.1 psia. This trend is similar to the pressure activity of the core during the tube rupture simulation. The pressurizer is input into the model using a self-initialization option control card to control the primary system pressure at 1850 psia. If the system experiences any pressure drop or rises above 1850 psia, the pressurizer model will immediately adjust the pressure back to 1850 psia. This is believed to be the reasoning behind depressurization not occurring to the primary system during the helical coil tube rupture, theoretically allowing the mass to flow between the secondary system, eventually causing the primary system to depressurize.
Figure 49 – Mass Flow Rate vs. Time of the Left Core RELAP5-3D Tube Rupture Calculation

Figure 49 shows the left core mass flow rate during the tube rupture simulation. As discussed previously, the mass flow rate data can only be plotted at the junctions when using the AptPlot software to decipher the RELAP5-3D data. The mass flow rates of the seven junctions connecting the eight volumes of the left core are shown above. All the junctions exhibit identical flow rates plotted over each other, causing the figure to appear as only one line. After the initiation of the tube rupture at 1500 seconds, oscillations occur during the transient calculation before converging back to a constant value at about 1700 seconds. The plot shows that the mass flow rate through the core
converges to a value of 881.28 lb./s, equating to a total system mass flow rate of approximately 1762.6 lb./s. This mass flow rate is similar to the value calculated for the left core during the steady-state simulation, which exhibited a value of approximately 882.5 lb./s. The tube rupture flow rate through the left core is slightly higher than the steady-state calculation by about 1.2 lb./s, which is not of significant concern. This data shows that the natural circulation cooling of the primary loop was not disrupted during the helical coil tube rupture simulation.

Figure 50 – Temperature vs. Time of the Left Core RELAP5-3D Tube Rupture Calculation
Figure 50 shows the temperatures of the eight volumes of the left core during the tube rupture simulation. A similar trend to the mass flow rate of the left core can be seen. Initial oscillations occur before converging to similar values of the steady-state calculation. In this case, the temperatures are identical to the temperature values of the steady-state calculation. These temperatures are again within the temperature range of the operational conditions of the reactor, showing no disruption to the natural circulation cooling of the primary system.

**Secondary Loop Tube Rupture Results:**

The critical parameters are plotted below to understand how the conditions changed within the steam generators and the steam pipe of the secondary loop of the reactor. As stated, an essential factor to be noted is that the AptPlot software, which has been utilized to produce these plots, only has the capability of plotting the mass flow rates for any junction component but does not have the capability of plotting the mass flow rates for individual volumes. Therefore, the data analyzed and depicted for the mass flow rates in this section is only for the junctions between volumes of the components. Void fractions, temperatures, pressures, and all other parameters have the capability of being plotted for both the volumes and junctions. The helical coil tube rupture results are plotted from 1500 seconds to 3500 seconds. The transient calculation was run as a restart type of problem within RELAP5-3D to start the calculation at the last restart dump of the steady-state calculation. To compare the rate of energy transferred to the steam generator during the steady-state calculation and tube rupture calculation, the steady-state results from time 1000 seconds to time 1500 seconds are shown in Figure
74, Figure 75, Figure 76, and Figure 77. During the steady-state calculation, a minimum time step of 1e-6 seconds and a maximum time step of 0.001 seconds were used.

Figure 51 – Liquid Temperature vs. Time of Secondary Steam Generator Volume 15

RELAP5-3D Tube Rupture Calculation
Figure 52 – Liquid Temperature vs. Time of Steam Pipe Volume One RELAP5-3D Tube Rupture Calculation
Figure 53 – Pressure vs. Time of Steam Pipe Volume One RELAP5-3D Tube Rupture Calculation
Figure 54 – Pressure vs. Time of Secondary Steam Generator Volume 15 RELAP5-3D Tube Rupture Calculation

Figure 51, Figure 52, Figure 53, and Figure 54 show that calculated pressures and temperatures of the secondary steam generator volume 15 and the steam pipe volume one have not been altered by the helical coil tube rupture and are identical to the steady-state calculation.
Figure 55 – Mass Flow Rate vs. Time of Tube Rupture RELAP5-3D Calculation

Figure 55 shows the mass flow rate of the single junction component used to model the break of the helical coil steam generator tube. As expected, there are slight oscillations seen at the start of the transient calculation. The oscillations cease after about 200 seconds at time 1700 along the x-axis of the plot. The flow rate converges to a value of 35.77 lbm/s and stays constant at this value until the calculation terminates at 3500 seconds.
Figure 56 depicts the liquid mass flow rate through a single junction (component 402) used to simulate the helical coil tube rupture placed between volume 1 of the primary steam generator and volume 15 of the secondary steam generator. The plot shows the same flow rate as the overall flow rate shown in Figure 54 above. This plot shows that the mass flowing through the rupture is fully liquid.
Figure 57 – Vapor Mass Flow Rate vs. Time of Tube Rupture RELAP5-3D Calculation

Figure 57 displays the vapor mass flow rate through the rupture to volume 15 of the secondary steam generator. This plot is shown to confirm further that the mass flowing through the break is 100 percent liquid. The vapor mass flow rate at the junction is 0.0 lb./s.
As discussed, the mass flow rate can only be plotted at the junctions of the hydrodynamic components. Therefore, the mass flow rate cannot be plotted in volume 15 of the secondary steam generator, where the liquid flows from the primary steam generator. The closest junction to resemble the effects of the situation is the single junction (component 775), displayed in Figure 58. It is orientated in the upward flow direction that connects volume 15 of the secondary steam generator to volume one of the steam pipe. This junction and volume one of the steam pipe experience very
similar effects to volume 15 of the secondary steam generator. An increase of 35.77 lb./s liquid flow rate from 200 lb./s to 235.77 lb./s is experienced at this junction.

Figure 59 – Mass Flow Rate vs. Time of Steam Pipe Junction One RELAP5-3D Tube Rupture Calculation

Volume one of the steam pipe and volume 15 of the secondary steam generator showed similar behavior during the tube rupture. Figure 59 displays the junction one of the steam pipe, which connects volume one to volume two. Like junction 775 depicted in Figure 58 above, this junction experiences a liquid mass flow rate increase of 35.77 lb./s, totaling the liquid flow rate to 235.77 lb./s from volume one to volume two within the steam pipe.
To further understand if the steam pipe continues to carry this flow rate through to the other volumes, Figure 60 is shown. The plot displays junctions two through nine of the steam pipe following junction one, shown in Figure 59. The same liquid flow rate of 235.77 lb./s is also seen at these junctions.
The mass flow rate through the junction connecting volume 14 of the secondary steam generator to volume 15, where the tube rupture occurs, does not exhibit a change in flow rate, as shown in Figure 61. A flow rate of 200 lb./s occurs during the steady-state calculation and the tube rupture transient calculation. The flow rate of 200 lb./s is also exhibited in the previous 13 volumes of the secondary steam generator.
Volume one of the steam pipe is the closest volume to tube rupture, located immediately upward along the flow direction. This means any increase or decrease in the mass flow rate from the tube rupture or volume 15 of the secondary steam generator will also be experienced in this volume. Figure 62 shows the increase of liquid void fraction within this volume. The liquid void fraction experienced in this volume during the steady-state calculation was approximately 1.68 percent liquid. This void fraction increased to about 2.59 percent, meaning it experienced an increase of about 0.91 percent. This is a low increase in void fraction but does show the
direction where the liquid mass flowing through the break of the tube is moving. The following figures depict the increase in the liquid void fraction throughout the volumes of the steam pipe.

Figure 63 – Liquid Void Fraction vs. Time of Steam Pipe Volume Five RELAP5-3D Tube Rupture Calculation

Figure 63 displays the liquid void fraction of volume five of the steam pipe. As discussed previously, the steam pipe has six vertical volumes and 19 horizontal volumes. Volume five is the last before the 90-degree turn to the horizontal volumes at volume six. Volume five experienced a void fraction of 1.7 percent before the tube
rupture. Volume five has increased slightly compared to volume one but exhibits a similar liquid void fraction of about 2.61 percent liquid.

Figure 64 – Liquid Void Fraction vs. Time of Steam Pipe Volume 10 RELAP5-3D Tube Rupture Calculation

Figure 64, Figure 65, Figure 66, Figure 67, and Figure 68 are shown to provide the data on the liquid void fractions of the horizontal volumes along the steam pipe. Figure 63 shows the liquid void fraction of volume 10 of the steam pipe. The liquid void fraction of volume 10 has increased to 2.83 percent liquid compared to 2.6
percent of volume one. This is a 0.94 percent increase from the liquid void fraction of 1.89 percent exhibited in volume 10 during the steady-state calculation.

Figure 65 – Liquid Void Fraction vs. Time of Steam Pipe Volume 15 RELAP5-3D Tube Rupture Calculation

Figure 65 shows the data on the liquid void fraction of volume 15 of the steam pipe. The liquid void fraction for this volume is 3.07 percent liquid compared to the 2.12 liquid void fraction experienced during the steady-state calculation. This is an increase of 0.95 percent liquid.
Figure 66 displays the liquid void fraction data for volume 20 of the steam pipe during the tube rupture transient calculation. The liquid void fraction of this volume during the steady-state calculation was 2.41 percent liquid. A percent liquid void fraction value of 3.38 percent is calculated for this transient calculation showing an increase of 0.97 percent liquid.
Volume 24 of the steam pipe experiences the highest liquid void fraction of any volume of the steam pipe. This data can be seen in Figure 67, which shows a liquid void fraction percentage of 3.82 percent. The percent liquid void fraction of this volume was 2.65 percent during the steady-state calculation. This is an increase of 1.17 percent, the highest increase out of any of the 25 volumes that make up the steam pipe. A trend with the liquid void fractions can be seen along with the volumes of the steam pipe. The liquid void fractions are at higher values farther along the steam pipe besides volume 25. Each increasing volume also experiences slightly higher increases.
in liquid void fractions during the tube rupture when compared to the preceding volumes.

Figure 68 – Liquid Void Fraction vs. Time of Steam Pipe Volume 25 RELAP5-3D Tube Rupture Calculation

An interesting situation occurs with volume 25 of the steam pipe, as shown in Figure 68. The liquid void fraction decreases back to a similar value exhibited in volume one of the pipe. This volume has the lowest percent liquid void fraction of 1.66 percent during the steady-state calculation. It increases to 2.56 percent during the tube rupture, which can be seen in Figure 68. This is also the lowest liquid void fraction of the
steam pipe during the transient calculation while also experiencing the lowest increase in liquid void fraction.

![Liquid Void Fraction vs Time](image)

Figure 69 – Liquid Void Fraction vs. Time of Secondary Steam Generator Volume 15

RELAP5-3D Tube Rupture Calculation

It can be seen in Figure 70, Figure 71, and Figure 72 below that the preceding volumes of the secondary steam generator below volume 15 exhibit an increase in vapor void fraction. In contrast, volume 15 (the location of the tube rupture) increases in liquid void fraction, as seen in Figure 68.
As discussed, volume 15 of the secondary steam generator (location of tube rupture) and volume one of the steam pipe experience very similar effects of the rupture. Figure 68 shows an almost identical increase in liquid void fraction as experienced by volume one of the steam pipe, shown in Figure 61. Although the liquid void fraction of volume 15 of the secondary steam generator was slightly higher than the liquid void fraction of volume one of the steam pipe during the steady-state calculation, both volumes increased to the same liquid void fraction of about 0.026 during the tube rupture simulation.

Figure 70 – Liquid Void Fraction vs. Time of Secondary Steam Generator Volume 14

RELAP5-3D Tube Rupture Calculation
Figure 70 shows the liquid void fraction of volume 14 of the secondary steam generator. A percent value of 5.29 percent liquid can be seen in this volume from the plot. This decreases 1.43 percent from the 6.72 percent liquid mass exhibited during the steady-state calculation.

Figure 71 shows the liquid void fraction of volume 10 of the secondary steam generator. The percent liquid void fraction value exhibited in this volume is 35.91 percent during the tube rupture. The percent liquid void fraction value exhibited
during the steady-state calculation was 41.66 percent, as expected. A decrease of 5.75 percent occurs during the tube rupture transient calculation.

![Liquid Void Fraction vs Time](image)

Figure 72 – Liquid Void Fraction vs. Time of Secondary Steam Generator Volume Five RELAP5-3D Tube Rupture Calculation

This trend continues downward along with the volumes of the secondary steam generator. Figure 72 depicts the data for the liquid void fraction of volume five. A percent value of 89.77 percent occurs during the tube rupture simulation. A decrease of 2.7 percent from the 92.47 percent liquid void fraction is exhibited in this volume during the steady-state simulation.
Figure 73 – Liquid Void Fraction vs. Time of Secondary Steam Generator Volumes One through Four RELAP5-3D Tube Rupture Calculation

Figure 73 displays the liquid void fractions for volumes one through four of the secondary steam generator. As seen during the steady-state calculation, these four volumes exhibit a percent liquid void fraction of 100 percent. This has not changed during the tube rupture simulation. This means volumes one through four are filled fully with liquid water during the steady-state and tube rupture calculations.
The rate of energy or heat transferred through the steam generator to the change in liquid void fraction throughout the secondary steam generator is explained. This parameter was examined previously to understand the power output of the core. It can also be viewed as the rate of energy transferred through the helical coil tubes to the secondary loop. Figure 74 shows the rate of energy transferred to volume 15 of the secondary steam generator during the steady-state and tube rupture. It is essential to see the change in the heat transferred during the steady-state calculation to the heat transferred during the tube rupture. Therefore, both have been plotted. The steady-state...
state calculation can be seen from the start of the plot (1000 seconds) up until 1500 seconds along the x-axis. The tube rupture calculation can be seen from 1500 seconds to 3500 seconds. There is a clear drop in energy transferred to volume 15 of the secondary steam generator. A decrease from approximately 28.78 MW to 12.16 MW of power or the energy transfer rate is experienced.

Figure 75 – Rate of Energy Transfer vs. Time of Secondary Steam Generator Volume 14 RELAP5-3D Tube Rupture Calculation

Figure 75 shows the rate at which energy is transferred through the steam generator to volume 14 of the secondary side. In contrast to volume 15, where the tube rupture is
located, volume 14 experiences an increase in energy transfer rate. Again, the steady-state calculation of the rate of heat transferred is also displayed on this plot. An increase to 30.14 MW from 23.92 MW is experienced in this volume.

![Steam Generator Energy Transfer](image)

Figure 76 – Rate of Energy Transfer vs. Time of Secondary Steam Generator Volume 10 RELAP5-3D Tube Rupture Calculation
To further understand why the liquid void fraction increases in volume 15 of the secondary steam generator and decreases in the other 14 volumes of secondary steam generator, the energy transfer rate to volumes 10 and five are displayed in Figure 76 and Figure 77, respectively. There is a trend with the rate of energy transferred to the decrease in the liquid void fraction of the lower volumes of the secondary steam generator. There is a decrease in the liquid void fraction in volumes five through 14 of the secondary steam generator. The rate of energy transferred increases in these volumes compared to the rate of energy transferred calculated during the steady-state.
calculation, but the rate of energy transferred to volume 15 of the secondary steam
generator decreases due to the tube rupture. If a higher rate of energy transfer is
experienced, then a higher percentage of vapor mass would form.
9 RELAP5-3D CODE IMPROVEMENTS

The author would like to provide suggestions to the RELAP5-3D code developers on the improvements that could be made to the code regarding the models discussed in this thesis. The model of the NuScale Power SMR was created to simulate the conditions and parameters of a single NPM. As discussed, this SMR relies on a natural convection system to cool the reactor’s core while utilizing a newly designed and patented once-through counter-flow helical coil steam generator. The combination of these systems proposed challenges in simulating the reactor’s operational conditions using RELAP5-3D. RELAP5-3D currently cannot model the helical coil geometry. The steam generator was simplified using a pipe hydrodynamic component to simulate the helical coil tube geometry and was assigned a 16.5-degree inclination angle. It was seen through open literature that the 16.5-degree inclination angle was not the only alteration needed, and the steam generator’s heat transfer surface area also needed to be increased. These were some of the alterations done to the model to simulate these systems and components but did not necessarily represent reality. Suppose the helical coil geometry is input to the RELAP5-3D code. In that case, the model within RELAP5-3D could more accurately represent the design of this steam generator, and the code could better predict the parameters and phenomena.

The other suggestion on the improvements that could be made to RELAP5-3D code pertains to the predictions of the parameters and conditions of natural circulation cooling systems. The mass flow rates calculated by RELAP5-3D for the natural circulation cooling system of the NPM were not entirely accurate. They did not
simulate the best estimate flow rate specified in the NuScale Power FSAR. These predictions could be due to the loss coefficients and friction factor correlations not being predicted reliably for a natural convection system. A similar scenario was seen by Hsun-Chia Lin [39] when a comparison between the experimental data of a natural circulation system was compared to the calculated data obtained using RELAP5-3D. Lin saw an over prediction of the natural circulation mass flow rates by RELAP5-3D. The flow rates of a natural circulation system could be better predicted if these correlations within the RELAP5-3D code are further examined and updated by the code developers.
10 CONCLUSION

The steady-state conditions of the NuScale Power SMR have been achieved with the consensus that RELAP5-3D calculates the best estimate mass flow rate due to the specified design pressures and temperatures. RELAP5-3D does not have advanced capabilities of modeling natural circulation cooling and does not have the option for the helical coil geometry for the steam generator tubes designed by NuScale Power LLC for their SMR. Buoyancy-driven natural circulation cooling is sensitive to flow restrictions, frictional losses, and changes in direction or geometries. Multiple inputs needed to be altered and manipulated to simulate the conditions and geometry of the helical-coil steam generator, which do not necessarily simulate the data and parameters of the reactor. Information in the open literature led to these changes to accurately simulate the NuScale SMR's operating conditions. As stated, all parameters and conditions of the reactor were accurately simulated besides the best estimate flow rate of 1294.44 lb./s provided in the NuScale FSAR. A best estimate flow rate of 1765.0 lb./s was calculated during the simulation. A more complex model or 3D modeling software may be needed to precisely obtain the best estimate flow rate of the natural circulation cooling phenomena.

The helical coil tube rupture transient analysis simulation was sought out due to the minimal literature published on this study to date. A single tube of the steam generator was ruptured during a restart problem of RELAP5-3D using the steady-state calculated data. The natural circulation cooling was not disrupted within the primary system. A liquid mass flow rate of 35.77 lb./s was calculated through the rupture of the tube to the secondary side of the steam generator. The data displayed in section 8.2 pertaining
to the tube rupture shows an increase in liquid void fractions within volume 15 of the secondary steam generator (where the rupture is located) and within the volumes and junctions of the steam pipe located upstream of the rupture. There is no significant increase in liquid void fraction within these volumes. Still, an increase of 0.91 percent within volume 15 of the secondary steam generator and volume one of the steam pipe is experienced. This liquid void fraction continues to increase along the steam pipe to a maximum increase of 1.17 percent, making up the 3.82 percent liquid void fraction within volume 24 of the steam pipe.

Figure 56 showed the liquid mass flow rate through the tube rupture. The mass through this break is 100 percent liquid flowing from the primary side. The slight increases of the liquid void fractions within the steam pipe and volume 15 of the secondary steam generator show the direction and location of this fluid. The steam pipe experiences higher increases of liquid void fractions to volume 24. Even though the increases are minimal, there is still a degree of radioactive liquid flowing through the rupture to the steam pipe and carrying the matter, which turns the turbine to produce electricity. This means that if a tube rupture of this sort would occur at this location, it can eventually fill up the steam pipe to concerning levels. It is also possible that radioactive fluid could flow to the secondary building of the reactor, which poses severe dangers and concerns. Due to the rupture, the energy transfer rate to volume 15 of the secondary steam generator decreases. The lower volumes of the secondary steam generator, in turn, experience a higher rate of energy transfer. The tube rupture alters the rate of energy transferred to the volumes of the secondary steam generator. Due to the actual flow rate of 200 lb./s through volume 15 of the secondary steam
generator, the liquid flashing from the rupture at the flow rate of 35.77 lb./s increases this flow rate to 235.77 lb./s and is forced upward to and through the steam pipe. This flashing phenomenon coincides with the low increase in liquid void fractions. Because of the flashing phenomena occurring and being restricted by the speed of sound, minimal liquid can flow into the secondary steam generator and the steam pipe. The choked flow process model is applied at the location of the tube rupture, which causes the flow to be choked through the ruptured tube by the amount of mass that is physically able to flow or leak through.

Suppose the secondary building housing the turbine and condenser were also modeled. In that case, a better understanding of what could potentially happen with the radioactive fluid flowing through the steam pipe could be analyzed. Another critical fact to be noted is that the primary side did not depressurize. This is thought to occur because of how the pressurizer is modeled, which may not allow the pressurizer to let the primary system pressure decrease or increase from 1850 psia. It is possible to remodel this component with valves to release pressure during the transient calculation. It was also thought that if this depressurization did occur, the reactor's safety features and sensors would trip, causing the reactor to shut down before any radioactive fluid can flow through the steam pipe. Again, more complex modeling of these components would be needed to understand this.
=NuScale Power's SMR
*
*NuScale SMR Model Input Code 4/22/2021
*
*Kyle P. Johnson
*
*--------------------------------------------------------
0000100 new transnt
*0000100 restart transnt
*
*--------------------------------------------------------
*crd101 Input Check/Run Option
0000101 run
*
*--------------------------------------------------------
*crd102 Input units Output units
0000102 british british
*
*--------------------------------------------------------
*crd103 Restart Input Control ncmpress MY-SMR TubeRupture
*0000103 -1 ncmpress
*
*--------------------------------------------------------
*Cr#
0000147 361 362 1850.0
*
*--------------------------------------------------------
*Cr#
0000140 0 0 0
*
*--------------------------------------------------------
crd201 End time Min dt Max dt ssd00 Minor ed Major ed Restart
0000201 1500.0 1.0-6 0.001 0003 500 10000 10000
SMR Primary Loop (Natural Circulation Cooling)

Component 100 - Lower Plenum Turn Area

*Crd#, Component name, Component type
1000000 "Low Plen" branch

*Crd#, Number of junctions, Mass flow rate specified
1000001 4 1
**Crd#, Flow area (ft²), Length (ft), Volume, Horz orientation, Vert orientation, Elevation change**

<table>
<thead>
<tr>
<th>Crd#</th>
<th>Flow area (ft²)</th>
<th>Length (ft)</th>
<th>Volume</th>
<th>Horz orientation</th>
<th>Vert orientation</th>
<th>Elevation change</th>
</tr>
</thead>
<tbody>
<tr>
<td>1000101</td>
<td>41.2</td>
<td>2.2</td>
<td>0.0</td>
<td>0.0</td>
<td>-90.0</td>
<td>-2.2</td>
</tr>
</tbody>
</table>

*End of Geometry and Orientation*

**Crd#, (Cont./Same as Last) Wall roughness (ft), Hydraul dia (ft), tlpvbf (X-Coord Vol Cont Flags) (Wall roughness for Commercial, new steel pipe or Iron, Wrought, new from Eng. Edge)**

<table>
<thead>
<tr>
<th>Crd#</th>
<th>Wall roughness</th>
<th>Hydraul dia</th>
<th>tlpvbf</th>
</tr>
</thead>
<tbody>
<tr>
<td>1000102</td>
<td>0.00015</td>
<td>7.243</td>
<td>0011000</td>
</tr>
</tbody>
</table>

**Crd#, Volume initial conditions (εbt), Pressure (psia), Temperature (F)(inlet to core temp)**

<table>
<thead>
<tr>
<th>Crd#</th>
<th>Volume initial conditions</th>
<th>Pressure (psia)</th>
<th>Temperature (F)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1000200</td>
<td>003</td>
<td>1850.0</td>
<td>497.0</td>
</tr>
</tbody>
</table>

*Junctions for Lower Plenum*

**Crd#, Outlet of low plen, Inlet to left core, Junct area (ft²), F loss, R loss, jefvcachs Subcool dis coeff, 2-phase dis coeff, Supheat dis coeff (v=1? for stratification entrainment/pullthrough for upward jun from horiz vol, choking model applied?)**

<table>
<thead>
<tr>
<th>Crd#</th>
<th>Outlet of low plen</th>
<th>Inlet to left core</th>
<th>Junct area (ft²)</th>
<th>F loss</th>
<th>R loss</th>
<th>jefvcachs Subcool dis coeff</th>
<th>2-phase dis coeff</th>
<th>Supheat dis coeff</th>
</tr>
</thead>
<tbody>
<tr>
<td>1001101</td>
<td>100010001</td>
<td>110010001</td>
<td>4.895</td>
<td>0.3</td>
<td>0.3</td>
<td>000010001</td>
<td>1.0</td>
<td>1.0</td>
</tr>
</tbody>
</table>

**Crd#, Outlet of low plen, Inlet to right core, Junct area (ft²), F loss, R loss, jefvcachs Subcool dis coeff, 2-phase dis coeff, Supheat dis coeff**

<table>
<thead>
<tr>
<th>Crd#</th>
<th>Outlet of low plen</th>
<th>Inlet to right core</th>
<th>Junct area (ft²)</th>
<th>F loss</th>
<th>R loss</th>
<th>jefvcachs Subcool dis coeff</th>
<th>2-phase dis coeff</th>
<th>Supheat dis coeff</th>
</tr>
</thead>
<tbody>
<tr>
<td>1002101</td>
<td>100010001</td>
<td>120010001</td>
<td>4.895</td>
<td>0.3</td>
<td>0.3</td>
<td>000010001</td>
<td>1.0</td>
<td>1.0</td>
</tr>
</tbody>
</table>

**Crd#, Outlet of SG DC1, Inlet to low plen, Junct area (ft²), F loss, R loss, jefvcachs Subcool dis coeff, 2-phase dis coeff, Supheat dis coeff**

<table>
<thead>
<tr>
<th>Crd#</th>
<th>Outlet of SG DC1</th>
<th>Inlet to low plen</th>
<th>Junct area (ft²)</th>
<th>F loss</th>
<th>R loss</th>
<th>jefvcachs Subcool dis coeff</th>
<th>2-phase dis coeff</th>
<th>Supheat dis coeff</th>
</tr>
</thead>
<tbody>
<tr>
<td>1003101</td>
<td>500200002</td>
<td>100010001</td>
<td>12.85</td>
<td>0.3</td>
<td>0.3</td>
<td>000010001</td>
<td>1.0</td>
<td>1.0</td>
</tr>
</tbody>
</table>

**Crd#, Outlet of SG DC2, Inlet to low plen, Junct area (ft²), F loss, R loss, jefvcachs Subcool dis coeff, 2-phase dis coeff, Supheat dis coeff**

<table>
<thead>
<tr>
<th>Crd#</th>
<th>Outlet of SG DC2</th>
<th>Inlet to low plen</th>
<th>Junct area (ft²)</th>
<th>F loss</th>
<th>R loss</th>
<th>jefvcachs Subcool dis coeff</th>
<th>2-phase dis coeff</th>
<th>Supheat dis coeff</th>
</tr>
</thead>
<tbody>
<tr>
<td>1004101</td>
<td>501200002</td>
<td>100010001</td>
<td>12.85</td>
<td>0.3</td>
<td>0.3</td>
<td>000010001</td>
<td>1.0</td>
<td>1.0</td>
</tr>
</tbody>
</table>

**Crd#, Liquid mass flow, Vapor mass flow, Interface velocity**

<table>
<thead>
<tr>
<th>Crd#</th>
<th>Liquid mass flow</th>
<th>Vapor mass flow</th>
<th>Interface velocity</th>
</tr>
</thead>
<tbody>
<tr>
<td>1001201</td>
<td>647.22</td>
<td>0.0</td>
<td>0.0</td>
</tr>
</tbody>
</table>
1002201  647.22  0.0  0.0
1003201  647.22  0.0  0.0
1004201  647.22  0.0  0.0
*  
*  

COMPONENT 110 - Core Left

*Crd#, Component name, Component type
1100000 "LCore" pipe

*Crd#, Number of volumes
1100001 8

*Crd#, Flow area (ft²), Vol# (9.79 ft^2 for whole core flow area)
1100101 4.895 8

*Crd#, Length (ft), Vol#
1100301 0.9875 8

*Crd#, Volume Vol#
1100401 0.0 8

*Crd#, Vertical inclination angle (degrees), Vol#
1100601 90.0 8

*Crd#, Elevation change (ft), Vol#
1100701 0.9875 8

*End of Geometry

*Crd#, Wall roughness (ft), Hydraul Dia (ft), Vol#
1100801 0.00015 2.496 8

*Crd#, Floss, Rloss, Jun#
1100901 0.6 0.6 7

*Crd#, tlpvbfe, Vol# (X-Coord Vol Cont Flags) (t=0 no thermal front tracking, l=0 no mixture level tracking,
p=1 no water packing scheme, b=1 for rod bundle
interphase friction model, v=1 for no vertical
stratification model, f=0 for wall friction effects)
1101001 0011100 8
*  
*Crd#, jefvcahs, Jun# (Junct Cont Flags)(c=1 no choke
model, h=0 nonhomogeneous, s=0 momentum flux both
ways)(Rest of models deactivated)
1101101 00001000 7
*  
*Crd#, Volume initial conditions (εbt), Pressure (psia),
Temperature (F), W4, W5, W6, Vol# (543 F is core avg
temp)
1101201 003 1850.0 543.0 0.0 0.0 0.0 8
*  
*Crd#, Mass flow rate specified (lbm/s)
1101300 1
*  
*Crd#, Liquid mass flow, Vapor mass flow, Interface
velocity, Jun#
1101301 647.22 0.0 0
*  
*  
*$
$  COMPONENT 120 - Core Right
$  
*  
*Crd#, Component name, Component type
1200000 "RCore" pipe
*  
*Crd#, Number of volumes
1200001 8
*  
*Crd#, Flow area (ft²), Vol#
1200101 4.895 8
*  
*Crd#, Length (ft), Vol#
1200301 0.9875 8
*  
*Crd#, Volume, Vol#
1200401 0.0 8
* Crd#, Vertical inclination angle (degrees), Vol# (90.0 orients inlet at bottom)
1200601 90.0 8
*
* Crd#, Elevation change, Vol#
1200701 0.9875 8
*
* End of Geometry
*
* Crd#, Wall roughness (ft), Hydraul Dia (ft), Vol#
1200801 0.00015 2.496 8
*
* Crd#, Floss, Rloss, Jun#
1200901 0.6 0.6 7
*
* Crd#, tlpvbfe, Vol# (X-Coord Vol Cont Flags) (t=0 no thermal front tracking, l=0 no mixture level tracking, p=1 no water packing scheme, v=1 for no vertical stratification model, b=1 for rod bundle interphase friction model, f=0 for wall friction effects, e=0 nonequilibrium calc)
1201001 0011100 8
*
* Crd#, jefvcahs, Jun# (Junct Cont Flags) (j=0 not used, e=0 no modified PV term, f=0 no CCFL model, v=0 not used, c=1 no choke model, a=0 smooth area change, h=0 nonhomogeneous, s=0 momentum flux both ways)
1201101 00001000 7
*
* Crd#, Volume initial conditions (εbt), Pressure (psia), Temperature (F), W4, W5, W6, Vol#
1201201 003 1850.0 543.0 0.0 0.0 0.0 8
*
* Crd#, Mass flow rate specified (lbm/s)
1201300 1
*
* Crd#, Liquid mass flow, Vapor mass flow, Interface velocity, Jun#
1201301 647.22 0.0 0.0 7
*
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$

$
COMPONENT 130 - Core Left Junction to Lower Riser

*Cr#, Component name, Component type
1300000 "LCoLRJu" sngljun

*Cr#, Outlet of L core, Inlet to low riser, Junct area, F loss, R loss, jefvcahs (j=0 no jet junction model, e=0 no PV term, f=0 no CCFL, v=0 no strat entrainment/pullthrough, c=1 no choke flow, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both dir)
1300101 110080002 200010001 4.895 0.3 0.3 00001000

*Cr#, Mass flow rate specified (lbm/s), Liquid mass flow, Vapor mass flow, Interface velocity
1300201 1 647.22 0.0 0.0

COMPONENT 140 - Core Right Connection to Lower Riser

*Cr#, Component name, Component type
1400000 "RCoLRJu" sngljun

*Cr#, Outlet of R core, Inlet to low riser, Junct area, F loss, R loss, jefvcahs (j=0 no jet junction model, e=0 no PV term, f=0 no CCFL, v=0 no strat entrainment/pullthrough, c=1 no choke flow, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both dir)
1400101 120080002 200010001 4.895 0.3 0.3 00001000

*Cr#, Mass flow rate specified (lbm/s), Liquid mass flow, Vapor mass flow, Interface velocity
1400201 1 647.22 0.0 0.0
**COMPONENT 200 - Lower Riser**

*Crd# Component name Component type
2000000 "Low Rise" pipe

*Crd# Number of volumes
2000001 7

*Geometry

*Crd# Flow area (ft^2) Vol#
2000101 24.9 7

*Crd# Length Vol#
2000301 1.0 7

*Crd# Volume Vol#
2000401 0.0 7

*Orientation

*Crd# Vertical inclination angle (degrees) Vol# (90.0 orients inlet at bottom, 0.0 horizontal)
2000601 90.0 7

*Crd# Elevation change?? Vol#
2000701 1.0 7

*Properties

*Crd# Wall roughness (ft) (Pat used 3.0)?? Hydraul Dia (ft) Vol#
2000801 0.00015 5.6306 7

*Crd# Floss Rloss Jun#
2000901 0.3 0.3 6

*Process Models

*Crd# tlpvbfe Vol# (X-Coord Vol Cont Flags) (t=0 no thermal front tracking, l=0 no mixture level tracking,
p=1 no water packing scheme, v=1 for no vertical stratification model, b=0 for interphase friction model, f=0 for wall friction effects, e=0 nonequilibrium calc)
2001001 0011000 7
*
*Crd# jefvcah???s Jun# (Junct Cont Flags) (j=0 not used, e=0 no modified PV term, f=0 no CCFL model, v=0 not used, c=1 no choke model, a=0 smooth area change, h=0 nonhomogeneous, s=0 momentum flux both ways)
2001101 00001000 6
*
*Values
*
*Crd# Volume initial conditions (εbt) Pressure (psia) Temperature (F) W4 W5 W6 Vol# (Find where Pat get 590psi) (Avg rise of 100F through core)
2001201 003 1850.0 590.0 0.0 0.0 0.0 7
*
*Crd# Mass flow rate specified (lbm/s)
2001300 1
*
*Crd# Liquid mass flow??? Vapor mass flow Interface velocity Jun#
2001301 1294.44 0.0 0.0 6
*
* COMPONENT 205 - Lower Riser Junction to Middle Riser
*
*Crd# Component name Component type
2050000 "LRMRJu" sngljun
*
*Crd# Outlet of low riser Inlet to mid riser Junct area F loss R loss jefv???cahs (j=0 no jet junction model, e=0 no PV term, f=0 no CCFL, v=0 no strat entrainment/pullthrough (Pat used v=3???), c=1 no choke flow, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both dir)
2050101 200070002 210010001 24.9 0.3 0.3 00001000
*Crd# Mass flow rate specified (lbm/s) Liquid mass flow??? Vapor mass flow Interface velocity
2050201 1 1294.44 0.0 0.0
*
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$
$
$ COMPONENT 210 - Middle Riser$
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$
*
*Crd# Component name Component type
2100000 "Mid Rise" pipe
*
*Crd# Number of volumes
2100001 5
*
*Geometry
*
*Crd# Flow area (ft^2) Vol#
2100101 24.900 1
2100102 22.525 2
2100103 20.150 3
2100104 17.775 4
2100105 15.400 5
*
*Crd# Length Vol#
2100301 0.88 1
2100302 0.88 2
2100303 0.88 3
2100304 0.88 4
2100305 0.88 5
*
*Crd# Volume Vol#
2100401 0.0 5
*
*Orientation
*
*Crd# Vertical inclination angle (degrees) Vol# (90.0 orients inlet at bottom, 0.0 horizontal)
2100601 90.0 5
*
*Crd# Elevation change??? Vol#
2100701 0.88 1
2100702 0.88 2
2100703 0.88 3
2100704 0.88 4
2100705 0.88 5

*Properties*

*Crd# Wall roughness (ft) Hydraul Dia (ft) Vol#*
2100801 0.00015 5.6306 1
2100802 0.00015 5.3553 2
2100803 0.00015 5.06515 3
2100804 0.00015 4.75729 4
2100805 0.00015 4.4281 5

*Crd# Floss Rloss Jun#*
2100901 0.3 0.3 4

*Process Models*

*Crd# tlpvber Vol# (X-Coord Vol Cont Flags)(t=0 no thermal front tracking, l=0 no mixture level tracking, p=1 no water packing scheme, v=1 for no vertical stratification model, b=0 for interphase friction model, f=0 for wall friction effects, e=0 nonequilibrium calc)*
2101001 0011000 5

*Crd# jefvcah???s Jun# (Junct Cont Flags)(j=0 not used, e=0 no modified PV term, f=0 no CCFL model, v=0 not used, c=1 no choke model, a=0 smooth area change, h=0 nonhomogeneous, s=0 momentum flux both ways)*
2101101 00001000 4

*Values*

*Crd# Volume initial conditions (εbt) Pressure??? (psia) Temperature??? (F) W4 W5 W6 Vol# (Find where Pat get 590F???)*
2101201 003 1850.0 590.0 0.0 0.0 0.0 5

*Crd# Mass flow rate specified (lbm/s)*
2101300 1

*Crd# Liquid mass flow??? Vapor mass flow Interface velocity Jun#*
2101301 1294.44 0.0 0.0 4
* COMPONENT 215 - Middle Riser (MR) Junction (Ju) to Upper Riser 1 (UR)
* Crd# Component name Component type
2150000 "MR UR J1" sngljun
* Crd# Outlet of mid riser Inlet to upp riser Junct area F loss R loss jefv???cahs (j=0 no jet junction model, e=0 no PV term, f=0 no CCFL, v=0 no strat entrainment/pullthrough (Pat used v=3???), c=1 no choke flow, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both dir)
2150101 210050002 220010001 7.7 0.3 0.3 00001000
* Crd# Mass flow rate specified (lbm/s) Liquid mass flow??? Vapor mass flow Interface velocity
2150201 1 647.22 0.0 0.0
*
* COMPONENT 216 - Middle Riser (MR) Junction (Ju) to Upper Riser 2 (UR)
* Crd# Component name Component type
2160000 "MR UR J2" sngljun
* Crd# Outlet of mid riser Inlet to upp riser Junct area F loss R loss jefv???cahs (j=0 no jet junction model, e=0 no PV term, f=0 no CCFL, v=0 no strat entrainment/pullthrough (Pat used v=3???), c=1 no choke flow, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both dir)
2160101 210050002 221010001 7.7 0.3 0.3 00001000
*
*Crd# Mass flow rate specified (lbm/s) Liquid mass flow??? Vapor mass flow Interface velocity
2160201 1 647.22 0.0 0.0
*
*

COMPONENT 220 - Upper Riser 1 (Start of Turn to Annulus?)

*Crd# Component name Component type
2200000 "Up Riser1" pipe

*Crd# Number of volumes
2200001 25

*Geometry

*Crd# Flow area (ft^2) Vol#
2200101 7.7 25

*Crd# Length Vol#
2200301 0.9924 25

*Crd# Volume Vol#
2200401 0.0 25

*Orientation

*Crd# Vertical inclination angle (degrees) Vol# (90.0 orients inlet at bottom, 0.0 horizontal)
2200601 90.0 25

*Crd# Elevation change??? Vol#
2200701 0.9924 25

*Properties

*Crd# Wall roughness (ft) (Pat used 1.5)??? Hydraul Dia (ft) Vol#
2200801 0.00015 3.131 25

*
*Crd# Floss Rloss Jun#
2200901 0.3 0.3 24
*
*Process Models
*
*Crd# tlvpbfe Vol# (X-Coord Vol Cont Flags)(t=0 no thermal front tracking, l=0 no mixture level tracking, p=1 no water packing scheme, v=1 for no vertical stratification model, b=0 for interphase friction model, f=0 for wall friction effects, e=0 nonequilibrium calc)
2201001 0011000 25
*
*Crd# jefvcahs Jun# (Junct Cont Flags)(j=0 not used, e=0 no modified PV term, f=0 no CCFL model, v=0 not used, c=1 no choke model, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both ways)
2201101 00001000 24
*
*Values
*
*Crd# Volume initial conditions (εbt) Pressure??? (psia) Temperature??? (F) W4 W5 W6 Vol# (Find where Pat get 590F???)
2201201 003 1850.0 590.0 0.0 0.0 0.0 25
*
*Crd# Mass flow rate specified (lbm/s)
22013001
*
*Crd# Liquid mass flow Vapor mass flow Interface velocity Jun#
2201301 647.22 0.0 0.0 24
*
*

COMPONENT 221 - Upper Riser 2 (Start of Turn to Annulus?)

*Cr# Component name Component type
2210000 "Up Rise2" pipe
*
*Crd# Number of volumes
2210001 25
*
*Geometry
*
*Crd# Flow area (ft^2) Vol#
2210101 7.7 25
*
*Crd# Length Vol#
2210301 0.9924 25
*
*Crd# Volume Vol#
2210401 0.0 25
*
*Orientation
*
*Crd# Vertical inclination angle (degrees) Vol# (90.0 orients inlet at bottom, 0.0 horizontal)
2210601 90.0 25
*
*Crd# Elevation change??? Vol#
2210701 0.9924 25
*
*Properties
*
*Crd# Wall roughness (ft) (Pat used 1.5)??? Hydraul Dia (ft) Vol#
2210801 0.00015 3.131 25
*
*Crd# Floss Rloss Jun#
2210901 0.3 0.3 24
*
*Process Models
*
*Crd# tlpvbfe Vol# (X-Coord Vol Cont Flags) (t=0 no thermal front tracking, l=0 no mixture level tracking, p=1 no water packing scheme, v=1 no vertical stratification model, b=0 for interphase friction model, f=0 for wall friction effects, e=0 nonequilibrium calc)
2211001 0011000 25
*
*Crd# jefvcahs Jun# (Junct Cont Flags) (j=0 not used, e=0 no modified PV term, f=0 no CCFL model, v=0 not used, c=1 no choke model, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both ways)
2211101 00001000 24
*
*Values
* *Crd# Volume initial conditions (εbt) Pressure?? (psia) Temperature?? (F) W4 W5 W6 Vol# (Find where Pat get 590F???)
2211201 003 1850.0 590.0 0.0 0.0 0.0 25
*
*Crd# Mass flow rate specified (lbm/s)
2211300 1
*
*Crd# Liquid mass flow Vapor mass flow Interface velocity Jun#
2211301 647.22 0.0 0.0 24
*

$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$ $$$$$$$$$$$$$$$$$$$
$ COMPONENT 350 - Upper Plenum / Turn to Annulus?
$ $$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$ $$$$$$$$$$$$$$$$$$$
*
*Crd# Component name Component type
3500000 "Up Plen" branch
*
*Crd# Number of junctions Mass flow rate specified
3500001 5 1
*
*Geometry and Orientation
*
*Crd# (X-Dir) Flow area (ft^2) Length (ft) Volume (ft^3) Horiz orientation Vert orientation Elevation change
3500101 41.2 1.7 0.0 0.0 90.0 1.7
*
*Crd# (Cont./Same as Last) Wall roughness (ft) Hydraul dia (ft) tipvbfe (X-Coord Vol Cont Flags)(t=0 no thermal front tracking, l=0 no mixture level tracking, p=1 no water packing scheme, v=1 for no vertical stratification model, b=0 for interphase friction model, f=0 for wall friction effects, e=0 nonequilibrium calc)
3500102 0.00015 7.24275 0011000
*
*Crd# Volume initial conditions (εbt) Pressure (psia) Temperature (F) (Where 590F from???)
3500200 003 1850.0 590.0
*
*Junctions for Upper Plenum*

*Crd#  From upp riser To upp plenum Junction area (ft^2)  Comment* 
loss R loss jeffvcahs (Subcool dis coeff W7, 2-phase dis coeff W8, Supheat dis coeff W9)=1.0
3501101 220250002 350010001 7.7 0.3 0.3 00001000

*Crd#  From pressurizer To upp Plenum Junction area (ft^2)  Comment* 
loss R loss jeffvcahs?? (Subcool dis coeff W7, 2-phase dis coeff W8, Supheat dis coeff W9)=1.0
3502101 360010001 350010002 23.16 0.3 0.3 00001000

*Crd#  From upp Plenum To SG primary Junction area (ft^2)  Comment* 
loss R loss jeffvcahs (Subcool dis coeff W7, 2-phase dis coeff W8, Supheat dis coeff W9)=1.0
3503101 350010001 400010001 12.85 0.3 0.3 00001000

*Crd#  From upp Plenum To SG primary Junction area (ft^2)  Comment* 
loss R loss jeffvcahs (Subcool dis coeff W7, 2-phase dis coeff W8, Supheat dis coeff W9)=1.0
3504101 350010001 400010001 12.85 0.3 0.3 00001000

*Crd#  From upp riser To upp plenum Junction area (ft^2)  Comment* 
loss R loss jeffvcahs (Subcool dis coeff W7, 2-phase dis coeff W8, Supheat dis coeff W9)=1.0
3505101 221250002 350010001 7.7 0.3 0.3 00001000

*Crd# Liquid mass flow Vapor mass flow Interface velocity* 
3501201 647.22 0.0 0.0
3502201 0.0 0.0 0.0
3503201 647.22 0.0 0.0
3504201 647.22 0.0 0.0
3505201 647.22 0.0 0.0

* COMPONENT 360 - Pressurizer* 

*Crd#, Component name, Component type* 
3600000 "Press" pipe
*Crd#, Number of volumes
3600001 6
*
*Crd#, Flow area (ft^2), Vol#
3600101 0.0 6
*
*Crd#, Length (ft), Vol#
3600301 0.2685 6
*
*Crd#, Volume (ft^3), Vol#
3600401 37.306 6
*
*Crd#, Vertical inclination angle (degrees), Vol#
3600601 90.0 6
*
*Crd#, Elevation change, Vol#
3600701 0.2685 6
*
*Crd#, Wall roughness (ft), Hydraul dia (ft), Vol#
3600801 0.00015 0.0 6
*
*Crd#, tlpvbfe, Vol# (X-Coord Vol Cont Flags) (t=0 no thermal front tracking, l=1 mixture level tracking on, p=0 water packing scheme on, v=0 vertical stratification model on, b=0 for interphase friction model, f=0 for wall friction effects, e=0 nonequilibrium calc)
3601001 0000000 6
*
*Crd#, jefvcahs, Jun# (j=0 no jet junction model, e=0 no PV term, f=0 no CCFL???, v=0 strat entrainment/pullthrough, c=1 no choke flow, a=0 smooth area change, h=0 nonhomogeneous, s=0 momentum flux both dir)
3601101 00001000 5
*
*Crd#, Volume initial conditions (εbt), Pressure (psia), Static quality, W4, W5, W6, Vol#
3601201 102 1850.0 0.2 0.0 0.0 0.0 6
*
*Crd#, Mass flow rate specified
3601300 1
*
*Crd#, (Junct initial conditions) Liquid mass flow, Vapor mass flow, Interface velocity, Jun#
3601301 0.0 0.0 0.0 5
*
*
COMPONENT 361 - TMDPVOL for Pressurizer

*Crd# Component name, Component type
3610000 "Pres TDV" tmdpvol

*Crd# Volume flow area W1, Length of volume W2, Volume of volume W3
3610101 200.0 0.0 200.0

*Crd# Horz angle W4, Vert angle W5, Elevation change W6
3610102 0.0 0.0 0.0

*Crd# Roughness W7, Hydraul diameter W8, (X-Cood Vol Cont Flags) tlpvbfe W9 (t=0 not used, l=0 not used, p=0 not used, v=0 not used, b=0 for interphase friction model, f=0 wall friction effects not used for tmdpvol, e=0 nonequilibrium calc)
3610103 0.0 200.0 0000000

*Crd# Volume initial conditions (εbt)
3610200 002

*Crd# tdigit Pressure (psia), Static quality
3610201 0.0 1850.0 0.2

*

COMPONENT 362 - TMDPVOL Pressuizer Connection

*Crd# Component name, Component type
3620000 "TDV Pres" sngljun

*Crd# Connection of Pressurizer to TMDPVOL, Junct area, F loss, R loss, jefvcahs (j=0 no jet junction model, e=0 no
PV term, f=0 no CCFL, v=0 no strat entrainment/pullthrough (Pat used v=3???), c=1 no choke flow, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both dir)

3620101 361010001 360010002 0.0 0.0 0.0 00001000
*
*Crd# Mass flow rate specified (lbm/s), Liquid mass flow,
Vapor mass flow, Interface velocity
3620201 1 0.0 0.0 0.0
*
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$ COMPONENT 400 - Steam Generator Primary / Downcomer Through S.G. Tubes
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
* Crd# Component name, Component type
4000000 "SG Prim" pipe
*
*Crd# Number of volumes
4000001 15
*
*Crd# Flow area (ft²), Vol#
4000101 25.7 15
*
*Crd# Length (ft), Vol#
4000301 1.61 15
*
*Crd# Volume, Vol#
4000401 0.0 15
*
*Crd# Vertical inclination angle (degrees), Vol#
4000601 -90.0 15
*
*Crd# Elevation change, Vol#
4000701 -1.61 15
*
*Crd# Wall roughness (ft), Hydraul Dia (ft), Vol#
4000801 0.00015 5.72034 15
*
*Crd# Floss, Rloss, Jun#
4000901 1.0 1.0 14
*Crd# tlpvbfe, Vol# (X-Coord Vol Cont Flags) (t=0 no thermal front tracking, l=0 no mixture level tracking, p=1 no water packing scheme, v=1 for no vertical stratification model, b=0 for interphase friction model, f=0 for wall friction effects, e=0 nonequilibrium calc)
4001001 0011000 15
*
*Crd# jefvcahs, Jun# (Junct Cont Flags) (j=0 not used, e=0 no modified PV term, f=0 no CCFL model, v=0 not used, c=1 no choke model, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both ways)
4001101 00001000 14
*
*Crd# Volume initial conditions (εbt) Pressure (psia) Temperature (F) ??? W4 W5 W6 Vol# (Where did Pat get 590F???)
4001201 003 1850.0 590.0 0.0 0.0 0.0 15
*
*Crd# Mass flow rate specified (lbm/s)
4001300 1
*
*Crd# Liquid mass flow, Vapor mass flow, Interface velocity, Jun#
4001301 1294.44 0.0 0.0 14
*

COMPONENT 405 - S.G. Downcomer Junction 1 / Downcomer Transition 1

*Crd#, Component name, Component type
4050000 "SG DC J1" sngljun
*
*Crd#, Outlet of SG Prim, Inlet to SG DC1, Junct area, F loss, R loss, jefvcahs
4050101 400150002 500010001 12.85 0.6 0.6 00001000 1
*
*Crd#, Mass flow specified, Liquid mass flow, Vapor mass flow, Interface velocity
4050201 1 647.22 0.0 0.0
* COMPONENT 406 - S.G. Downcomer Junction 2 / Downcomer Transition 2 *
* Crd#, Component name, Component type
  4060000 "SG DC J2" sngljun *
* Crd#, Outlet of SG Prim, Inlet to SG DC2, Junct area, F loss, R loss, jefvcas
  4060101 400150002 501010001 12.85 0.6 0.6 00001000 *
* Crd#, Mass flow specified, Liquid mass flow, Vapor mass flow, Interface velocity
  4060201 1 647.22 0.0 0.0 *
* *
* COMPONENT 500 - S.G. Downcomer 1 (Cold Leg 1) *
* Crd#, Component name, Component type
  5000000 "SG DC1" annulus *
* Crd#, Number of volumes
  5000001 20 *
* Geometry *
* Crd#, Flow area (ft²), Vol#
  5000101 12.85 20 *
* Crd# Length (ft), Vol#
  5000301 0.998 20 *
* Crd#, Volume, Vol#
5000401 0.0 20
*
*Orientation
*
*Crd#, Vertical inclination angle (degrees), Vol# (-90.0 orients inlet at top)
5000601 -90.0 20
*
*Crd#, Elevation change, Vol#
5000701 -0.998 20
*
*Properties
*
*Crd#, Wall roughness (ft), Hydraul Dia (ft), Vol#
5000801 0.00015 4.0449 20
*
*Crd#, Floss, Rloss, Jun#
5000901 0.6 0.6 19
*
*Process Models
*
*Crd#, tlpvbfe, Vol# (X-Coord Vol Cont Flags) (t=0 no thermal front tracking, l=0 no mixture level tracking, p=1 no water packing scheme, v=1 for no vertical stratification model, b=0 for interphase friction model, f=0 for wall friction effects, e=0 nonequilibrium calc)
5001001 0011000 20
*
*Crd#, jefvcahs, Jun# (Junct Cont Flags) (j=0 not used, e=0 no modified PV term, f=0 no CCFL model, v=0 not used, c=1 no choke model, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both ways)
5001101 00001000 19
*
*Values
*
*Crd#, Volume initial conditions (εbt), Pressure (psia), Temperature (F), W4, W5, W6, Vol#
5001201 003 1850.0 497.0 0.0 0.0 0.0 20
*
*Crd#, Mass flow rate specified (lbm/s)
5001300 1
*
*Crd#, Liquid mass flow, Vapor mass flow, Interface velocity, Jun#
5001301 647.22 0.0 0.0 19
*
*  
*  
$\text{COMPONENT 501 - S.G. Downcomer 2 (Cold Leg 2)}$
$  

*Cr#, Component name, Component type
5010000 "SG DC2" annulus

*Cr#, Number of volumes
5010001 20

*Geometry

*Cr#, Flow area (ft²), Vol#
5010101 12.85 20

*Cr#, Length (ft), Vol#
5010301 0.998 20

*Cr#, Volume, Vol#
5010401 0.0 20

*Orientation

*Cr#, Vertical inclination angle (degrees), Vol# (90.0 orients inlet at top)
5010601 −90.0 20

*Cr#, Elevation change, Vol#
5010701 −0.998 20

*Properties

*Cr#, Wall roughness (ft), Hydraul Dia (ft), Vol#
5010801 0.00015 4.0449 20

*Cr#, Floss, Rloss, Jun#
5010901 0.6 0.6 19

*Process Models

*
*Crd#, tlpvbfe, Vol# (X-Coord Vol Cont Flags) (t=0 no thermal front tracking, l=0 no mixture level tracking, p=1 no water packing scheme, v=1 for no vertical stratification model, b=0 for interphase friction model, f=0 for wall friction effects, e=0 nonequilibrium calc)
5011001 0011000 20
*
*Crd#, jefvcahs, Jun# (Junct Cont Flags) (j=0 not used, e=0 no modified PV term, f=0 no CCFL model, v=0 not used, c=1 no choke model, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both ways)
5011101 00001000 1
*
*Values
*
*Crd#, Volume initial conditions (εbt), Pressure (psia), Temperature (F), W4, W5, W6, Vol#
5011201 003 1850.0 497.0 0.0 0.0 0.0 20
*
*Crd#, Mass flow rate specified (lbm/s)
5011300 1
*
*Crd#, Liquid mass flow, Vapor mass flow, Interface velocity, Jun#
5011301 647.22 0.0 0.0 19
*
*
*Secondary Loop
*
$COMPONENT 740 - Source Volume for Feedwater$
* Crd#, Component name, Component type
7400000 "WatSourc" tmdpvol
*
* Crd#, Volume flow area W1, Length of volume W2, Volume of volume W3
7400101 1.0e6 0.0 1.0e6
*
* Crd#, Horz angle W4, Vert angle W5, Elevation change W6
7400102 0.0 0.0 0.0
*
* Crd#, Wall roughness (ft) W7, Hydraul diameter (ft) W8, tlpvbfef W9 (X-Coord Vol Cont Flags) (t=0 not used, l=0 not used, p=0 not used, v=0 not used, b=0 for interphase friction model, f=0 wall friction effects not used for tmdpvol, e=0 nonequilibrium calc)
7400103 0.0 1.0e6 0000000
*
* Crd#, Volume initial conditions (εbt)
7400200 003
*
* Crd#, tdigit Pressure (psia), Temperature (F)
7400201 0.0 500.0 300.0
*
*
 COMPONENT 750 - Water Source (WS) Junction (J) with Feedwater Pipe (FWP)
*
* Crd#, Component name Component type
7500000 "WS J FWP" tmdpjun
*
* Crd#, Outlet of source volume, Inlet to feedwater pipe, Junct area, jefvcaho (All models not used for tmdpjun)
7500101 740010002 760010001 0.785 0000000
*
* Crd#, Mass flow rate specified (lbm/s)
7500200 1
*
*Crd#, Search variable (time), Liquid mass flow (lbm/s), Vapor mass flow (lbm/s), Interface velocity
7500201 0.0 0.0 0.0 0.0
7500202 1.0 201.0 0.0 0.0
*
*

COMPONENT 760 - Feedwater Pipe
*

*Crd#, Component name, Component type
7600000 "FW Pipe" pipe
*

*Crd#, Number of volumes
7600001 25
*

*Geometry
*
*Crd#, Flow area (ft²), Vol#
7600101 0.785 25
*
*Crd#, Length (ft), Vol#
7600301 1.0 25
*
*Crd#, Volume, Vol#
7600401 0.0 25
*

*Orientation
*
*Crd#, Vertical inclination angle (degrees), Vol# (-90.0 orients inlet at top, 0.0 horizontal)
7600601 0.0 19
7600602 45.0 20
7600603 90.0 25
*
*Crd#, Elevation change, Vol#
7600701 0.0 19
7600702 0.5 20
7600703 1.0 25
*

*Properties
*
*Crd#, Wall roughness (ft), Hydraul Dia (ft), Vol#
7600801 0.00015 0.9997 25
*
*Process Models
*
*Crd#, tlpvbfe, Vol# (X-Coord Vol Cont Flags) (t=0 no thermal front tracking, l=0 no mixture level tracking, p=1 no water packing scheme, v=1 for no vertical stratification model, b=0 for interphase friction model, f=0 for wall friction effects, e=0 nonequilibrium calc)
7601001 0011000 25
*
*Crd#, jefvcahs, Jun# (Junct Cont Flags) (j=0 not used, e=0 no modified PV term, f=0 no CCFL model, v=0 not used, c=1 no choke model, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both ways)
7601101 00001010 2
*
*Values
*
*Crd#, Volume initial conditions (εbt), Pressure (psia), Temperature (F), W4, W5, W6, Vol#
7601201 003 500.0 300.0 0.0 0.0 0.0 25
*
*Crd#, Mass flow rate specified (lbm/s)
7601300 1
*
*Crd#, Liquid mass flow, Vapor mass flow, Interface velocity, Jun#
7601301 147.81 0.0 0.0 24
*
*

COMPONENT 770 - Feedwater Pipe S.G. Secondary (SGS) Junction

*Crd#, Component name, Component type
7700000 "FW SGS J" sngljun
*
*Crd#, Outlet of FW Pipe, Inlet to SG secondary, Junct area, F loss, R loss, jefvcahs (j=0 no jet junction
model, e=0 no PV term, f=0 no CCFL, v=0 no strat
entrainment/pullthrough, c=1 no choke flow, a=0 smooth
area change, h=1 homogeneous, s=0 momentum flux both dir)
7700101 760250002 401010001 0.785 0.0 0.0 00001010
*
*Crd#, Mass flow rate specified (lbm/s), Liquid mass
flow, Vapor mass flow, Interface velocity
7700201 1 147.81 0.0 0.0
* 
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$ 
$ 
COMPONENT 402 - S.G. Prim to S.G. Sec
Connection
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
* 
*Crd#, Component name, Component type
*4020000 "SG1 SG2" snogljun
* 
*Crd#, Outlet of SG secondary, Inlet to steam pipe, Junct
area, F loss, R loss, jefvcahas (j=0 no jet junction
model, e=0 no PV term, f=0 no CCFL, v=0 no strat
entrainment/pullthrough, c=1 no choke flow, a=0 smooth
area change, h=1 homogeneous, s=0 momentum flux both dir)
*4020101 400010001 401150001 0.0021 0.0 0.0 00000000
*
*Crd#, Mass flow rate specified (lbm/s), Liquid mass
flow, Vapor mass flow, Interface velocity
*4020201 1 0.0 0.0 0.0
* 
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$ 
$ 
COMPONENT 401 - Steam Generator Secondary Loop
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
* 
*Crd#, Component name, Component type
4010000 "SG Sec" pipe
*
<table>
<thead>
<tr>
<th>Crd#, Number of volumes</th>
<th>4010001 15</th>
</tr>
</thead>
<tbody>
<tr>
<td>*Geometry</td>
<td></td>
</tr>
<tr>
<td>*Crd#, Flow area (ft²), Vol#</td>
<td>4010101 0.785 15</td>
</tr>
<tr>
<td>*Crd#, Length (ft), Vol#</td>
<td>4010301 1.61 15</td>
</tr>
<tr>
<td>*Crd#, Volume, Vol#</td>
<td>4010401 0.0 15</td>
</tr>
<tr>
<td>*Orientation</td>
<td></td>
</tr>
<tr>
<td>*Crd#, Vertical inclination angle (degrees), Vol#</td>
<td>-90.0 orients inlet at top</td>
</tr>
<tr>
<td>*Crd#, Elevation change, Vol#</td>
<td>4010701 1.61 15</td>
</tr>
<tr>
<td>*Properties</td>
<td></td>
</tr>
<tr>
<td>*Crd#, Wall roughness (ft), Hydraul Dia (ft), Vol#</td>
<td>4010801 0.00015 0.9997 15</td>
</tr>
<tr>
<td>*Crd#, Floss, Rloss, Jun#</td>
<td>4010901 0.05 0.5 14</td>
</tr>
<tr>
<td>*Process Models</td>
<td></td>
</tr>
<tr>
<td>*Crd#, tlpvbfe, Vol# (X-Coord Vol Cont Flags)</td>
<td>t=0 no thermal front tracking, l=1 mixture level tracking, p=1 no water packing scheme, v=1 for no vertical stratification model, b=0 for interphase friction model, f=0 for wall friction effects, e=0 nonequilibrium calc</td>
</tr>
<tr>
<td>*Crd#, jefvcahs, Jun# (Junct Cont Flags)</td>
<td>j=0 not used, e=0 no modified PV term, f=0 no CCFL model, v=0 not used, c=1 no choke model, a=0 smooth area change, h=0 nonhomogeneous, s=0 momentum flux both ways</td>
</tr>
<tr>
<td>*Values</td>
<td></td>
</tr>
</tbody>
</table>
**Crd#, Volume initial conditions (**bt), Pressure (psia), Temperature (F), W4, W5, W6, Vol# (Find where Pat get 500psi and 300F???)
4011201 003 1000.0 550.0 0.0 0.0 0.0 15

**Crd#, Mass flow rate specified (lbm/s)
4011300 1

**Crd#, Liquid mass flow, Vapor mass flow, Interface velocity, Jun#
4011301 147.81 0.0 0.0 14

COMPONENT 775 - S.G. to Steam Pipe Connection

COMPONENT 780 - Steam Pipe Secondary Loop
*Crd#, Component name, Component type
7800000 "Stm Pipe" pipe
*
*Crd#, Number of volumes
7800001 25
*
*Geometry
*
*Crd#, Flow area (ft²), Vol#
7800101 0.785 25
*
*Crd#, Length (ft), Vol#
7800301 1.0 25
*
*Crd#, Volume, Vol#
7800401 0.0 25
*
*Orientation
*
*Crd#, Vertical inclination angle (degrees), Vol# (-90.0
orients inlet at top, 0.0 horizontal)
7800601 90.0 5
7800602 45.0 6
7800603 0.0 25
*
*Crd#, Elevation change, Vol#
7800701 1.0 5
7800702 0.5 6
7800703 0.0 25
*
*Properties
*
*Crd#, Wall roughness (ft), Hydraul Dia (ft), Vol#
7800801 0.00015 0.9997 25
*
*Process Models
*
*Crd#, tlpvbfe, Vol# (X-Coord Vol Cont Flags) (t=0 no
thermal front tracking, l=0 no mixture level tracking
(keeping off for now, may need tho is not completely
vapor here), p=1 no water packing scheme, v=1 for no
vertical stratification model, b=0 for interphase
friction model, f=0 for wall friction effects, e=0
nonequilibrium calc)
7801001 0011000 25
*Crd#, jefvcahs, Jun# (Junct Cont Flags) (j=0 not used, e=0 no modified PV term, f=0 no CCFL model, v=0 not used, c=1 no choke model, a=0 smooth area change, h=0 nonhomogeneous, s=0 momentum flux both ways)
7801101 00001000 24
*
*Values
*
*Crd#, Volume initial conditions (εbt), Pressure (psia), Temperature (F), W4, W5, W6, Vol# (Find where Pat get 500psi and 575F???)
7801201 003 1000.0 550.0 0.0 0.0 0.0 25
*
*Crd#, Mass flow rate specified (lbm/s)
7801300 1
*
*Crd#, Liquid mass flow, Vapor mass flow, Interface velocity, Jun#
7801301 0.0 147.81 0.0 24
*
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$
$$
$$
COMPONENT 790 - Steam Pipe to Sink Junction
$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$
*Crd#, Component name, Component type
7900000 "SP Snk J" sngljun
*
*Crd#, Outlet of steam pipe, Inlet to sink Junct area, F loss, R loss, jefvcahs (j=0 no jet junction model, e=0 no PV term, f=0 no CCFL, v=0 no strat entrainment/pullthrough, c=1 no choke model, a=0 smooth area change, h=1 homogeneous, s=0 momentum flux both dir)
7900101 7802500002 800010001 0.785 0.0 0.0 0.0 00001010
*
*Crd#, Mass flow rate specified (lbm/s), Liquid mass flow, Vapor mass flow, Interface velocity
7900201 1 0.0 147.81 0.0
*
*
COMPONENT 800 - Sink

8000000 "Sink" tmdpvol

8000101 1.0e6 0.0 1.0e6

8000102 0.0 0.0 0.0

8000103 0.0 1128.378 0000000

8000200 002

8000201 0.0 1000.0 0.999

Heat Structures
COMPONENT 1820 - Steam Generator Tubes

*Cr# Heat struct # Mesh pts # (reflood or metal water reaction)??? Geometry rectangular SS init flag Left boundary coord (ft)
18201000 15 3 1 1 0.024

*Cr# Mesh location flag Mesh format flag
18201100 0 1

*Cr# # of intervals Right boundary coord (ft)
18201101 2 0.026

*Cr# Composition #??? Interval #
18201201 001 2

*Cr# Source value (Qi,input) Mesh interval #
18201301 0.0 2

*Cr# Initial temp flag
18201400 0

*Cr# Temperature Mesh pt #
18201401 560.0 3

*Cr# Left boundary volume Increment 1D Boundary cond
Surf area code Surf area/factor (ft^2)(Pat had 1195.2 OG code) Heat struct #
18201501 400010000 10000 1 1 7089.053 15

*Cr# Left boundary volume Increment 1D Boundary cond
Surf area code Surf area/factor (ft^2)(Pat had 1195.2 OG code) Heat struct #
*18201501 400010000 10000 1 1 7089.053 15

*Cr# Right boundary volume Increment 1D Boundary cond
Surf area code Surf area/factor (ft^2)(Pat had 1195.2 OG code) Heat struct #
18201601 401150000 -10000 1 1 7089.053 15

*Cr# Right boundary volume Increment 1D Boundary cond
Surf area code Surf area/factor (ft^2)(Pat had 1195.2 OG code) Heat struct #
*18201601 402010000 -10000 1 1 7089.053 15 *
*Crd# Source type Internal source mult Left mod heat mult Right mod heat mult Heat struct #
18201701 0 0.0 0.0 0.0 15 *
*
*Crd# 9 words for 18201801
18201800 0 *

*ADDITIONAL LEFT BOUNDARY CONDITION
*Crd# Heated equiv dia Heated length F Heated length R Grid spacer length F Grid spacer length R Grid loss coeff F Grid loss coeff Local boiling Heat struct # (Wrod 1 72.27e-3)
18201801 0.285 50.0 50.0 0.0 0.0 0.0 0.0 1.0 15 *
*Crd# 9 words for 18201901
18201900 0 *

*ADDITIONAL RIGHT BOUNDARY CONDITION
*Crd# Heated equiv dia Heated length F Heated length R Grid spacer length F Grid spacer length R Grid loss coeff F Grid loss coeff Local boiling Heat struct #
18201901 5.2e-2 50.0 50.0 0.0 0.0 0.0 0.0 1.0 15 *
*
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$#
#
COMPONENT 1810 - Fuel Rods
$#
$#
*
*Crd# Heat struct # Mesh pts # (reflood or metal water reaction)??? Geometry rectangular SS init flag Left boundary coord (ft)
18101000 8 2 1 1 0.0 *
*Crd# Mesh location flag Mesh format flag
18101100 0 1 *
*Crd# # of intervals Right boundary coord (ft)
18101101 1 0.0312 *

*Crd#  Composition #??? Interval #
18101201  005  1
*
*Crd#  Source value (Qi,input) Mesh interval #
18101301  1.0  1
*
*Crd#  Initial temp flag
18101400  0
*
*Crd# Temperature Mesh pt #
18101401  590.0  2
*
*Crd#  Left boundary volume Increment 1D Boundary cond
(110???) Surf area code Surf area/factor (ft^2) Heat
struct #
18101501  120010000 10000  1 0  392.14  8
*
*Crd#  Right boundary volume Increment 1D Boundary cond
(110???) Surf area code Surf area/factor (ft^2) Heat
struct #
18101601  110010000 10000  1 0  392.14  8
*
*Crd#  Source type??? Internal source mult Left mod heat
mult Right mod heat mult Heat struct #
18101701  100 0.125 0.0 0.0  0 8
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*Crd#  9 words for 18101801
18101800  0
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*Crd#  Heated equiv dia Heated length F Heated length R
Grid spacer length F Grid spacer length R Grid loss coeff
F Grid loss coeff Local boiling Heat struct #
18101801  0.041  50.0  50.0  0.0 0.0 0.0 0.1 0.0  8
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*Crd#  9 words for 18101901
18101900  0
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Grid spacer length F Grid spacer length R Grid loss coeff
F Grid loss coeff Local boiling Heat struct #
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*Crd#  Table type (power vs. time) No trip or factors
20210000 power
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* Thermal properties of cladding - composition 4 *

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20100400 tbl/fctn 1 1 *

*Thermal properties of cladding *

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20100402 392.0 2.2478e-3
20100403 752.0 2.7297e-3
20100404 1112.0 3.0508e-3
20100405 1472.0 3.5325e-3
20100406 1832.0 3.9078e-3
20100407 2192.0 4.2708e-3
20100408 2552.0 4.6339e-3
20100409 2912.0 4.9970e-3
20100410 3272.0 5.3501e-3
20100411 3632.0 5.7032e-3
20100412 3992.0 6.0563e-3

*Crd# Temperature Volumetric heat capacity
20100451 0.0 28.392
20100452 1480.3 34.476
20100453 1675.00 85.176
20100454 1787.5 34.370
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* Thermal properties of uo2 - composition 5 *

*Crd# Material type Therm Conduct flag Vol heat cap flag (uo2)
20100500 tbl/fctn 1 1 *

*Thermal properties of uo2 *

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20100502 188.6 1.284e-3
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*Crd# Temperature Volumetric heat capacity

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*Thermal properties of fuel gap(average core) - composition 6 *

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*Thermal properties of fuel gap
*
*Crd# Temperature Thermal conductivity
20100601 32.0 0.00031
20100602 5400.0 0.00031
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*Crd# Temperature Volumetric heat capacity
20100651 32.0 0.000075
20100652 5400.0 0.000075
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