2018

Transient Thermal-hydraulic Simulation of a Small modular Reactor in RELAP 5

Patrick Freitag
University of Rhode Island, patrick_freitag@my.uri.edu

TRANSIENT THERMAL HYDRAULIC SIMUATION OF A SMALL

MODULAR REACTOR IN RELAP 5

BY

PATRICK FREITAG

A THESIS SUBMITTED IN PARTIAL FULFILLMENT OF THE

REQUIREMENTS FOR THE DEGREE OF

MASTER OF SCIENCE

IN

MECANICAL ENGINEERING

UNIVERSITY OF RHODE ISLAND

2018

MASTER OF SCIENCE THESIS

OF

Patrick Freitag

APPROVED:

Thesis Committee:

Major Professor Bahram Nassersharif

Co-Major Professor Cameron Goodwin

Hamouda Ghonem

Arijit Bose

Nasser H. Zawia

DEAN OF THE GRADUATE SCHOOL

UNIVERSITY OF RHODE ISLAND

2018

Abstract

This thesis analyzes and evaluates relevant thermal-hydraulic features of the integral pressurized water reactor for a new design of nuclear power plant. The chosen design is the NuScale small modular reactor. This reactor has a thermal power of 160 MW and operates usually with more reactors of its kind in a common power plant. The NuScale design is currently in the licensing process from the Nuclear Regulatory Commission. The first part of this thesis deals with basic knowledge about nuclear fission, SMR technology, and the power plant steam cycle. The second part is about the simulation software RELAP 5, which uses a one-dimensional model to simulate nuclear power systems. It describes how to program the different components, which are needed to simulate the NuScale system. In addition, the two fluid model is introduced which is the basis for the RELAP 5 thermal hydraulic simulations. The final part is about the simulation and the evaluation of the SMR. The NuScale design criteria were looked up in the final safety analysis report, which is used for licensing at the NRC. The results show that the steady state values of the

simulation matches with the data from the FSAR of the NuScale design. Therefore it can be said that a reactor, which only runs via natural circulation, works and all the heat which is produced by the core is transferred to the secondary cycle of the SMR. The findings of this thesis confirm the benefits of the NuScale SMR design and suggest further theoretical and later experimental investigations.

Acknowledgments

At this point I would like to thank my major-professors Dr. Bahram Nassersharif and Dr. Cameron Goodwin for thier supervision. And I would like to thank Dr. Robert Martin for his help in programming RELAP 5.

Contents

List of Figures

List of Tables

List of Abbreviation

SMR	Small Modular Reactor
IAEA	International Atomic Energy Agency
IPWR	Integral Pressurized Water Reactor
LMR	Liquid Metal Cooled Reactor
HTGR	High-Temperature, Gas-Cooled Reactors
MSR	Molten Salt Reactors
M	Mass Number
Z	Atomic Number
U	Uranium
n	NeutRon
Kr	Krypton

Ba	Barium
Pu	Plutonium
H2O	Water
UO2	Uranium Dioxide
US	United States of America
DOE	US Department of Energy
INEEL	Idahos National Environment & Engineering Laboratory
OSU	Oregon State University
NRC	Nuclear Regulatory Commission
DHRS	Decay Heat Removal System
ECCS	Emergency Core Cooling System
LTC	Long-Term Cooling
T-S	Temperature-Entropy
CO2	Carbon Dioxide
SO2	Sulfur Dioxide

RELAP Reactor Excursion and Leak Analysis Program

NPM NuScale Power Module

1 Introduction

Energy and energy availability are two important topics for the future. Above all is the ever increasing public need for long-term economic and ecological energy supply which has become an ever greater challenge for scientists and engineers in many parts of the world in recent years. Nuclear fission energy production has ensured efficient and clean energy supply in many parts of the world for over 70 years. But even this technology continues to evolve and so in addition to ever larger nuclear power plants, so-called small modular reactors (SMR) are being developed. Not only are these SMRs much smaller in size, they also have much more application potential. They are also affordable in terms of startup Gst. The goal of these developments are safe and efficient small modular reactors which are designed for much improved safety. These reactors are tested and further developed in thermal-hydraulic simulation models. RELAP 5 is one such of these thermal-hydraulic simulation models and is used worldwide to test many types of nuclear reactors in a cost-effective manner and, importantly, without security risk. In order to perform these simulations, a basic understanding of nuclear energy, SMRs, and thermal-hydraulic must first be aquired in order to analyse the results calculated by RELAP 5. After that, the reactor model is analyzed by comparing RELAP 5 calculated results to design data, hand calculations, and experimental data when available. When the simulations start, the first required series of tests are carried out in order to see how the reactor model behaves in steady state. Then based on these results, various accident scenarios can be carried out and subsequently evaluated. The aim of the simulations are to gain insight into the behavior of the small modular reactor.

2 Small modular Reactor

Small modular reactors (SMR) are small nuclear reactors with low electric power. These reactors have an equivalent electric power of less than 300 MW, according to the IAEA classification and are an opportunity for new clean and economical energy production. Many SMR modules are combined into one power plant and can be switched on and off depending on the energy demand. Also new modules can be added to the plant to increase the energy output while other modules are still working. Because of their small size, small modular reactors can be used to produce energy in low populated regions like islands, deserts or jungles. These reactors are also an opportunity for developing countries because of the lower investment costs. Also, a well-developed infrastructure is unnecessary for SMRs which is usually needed to run a conventional nuclear power plant. Therefore SMRs are a good, long-term solution to the general energy production for the future. Small modular reactors can be distinguished into four different types:

- Integral Pressurized Water Reactors, IPWRs

- Liquid Metal-Cooled Reactors, LMRs
- High Temperature, Gas-cooled Reactors, HTGRs
- Molten Salt Reactors, MSRs

This differentiation is based on the cooling of the reactor core in the SMRs [1][2][3][4][5].

2.1 Integral Pressurized Water Reactors, IPWRs

The neutrons in an integral pressurized water reactor are moderated with light water. In addition to that, the light water is also used as cooland in the reactor primary cycle. An IPWR operates at a temperature level of about 300°C depending on how high the vapor pressure is in the cooling circuit. Enriched uranium (U^{235}) must be used as fuel because of the higher neutron absorption cross section of the water. The enriched uranium causes a greater number of nuclear fissions of the uranium which leads to a higher production of neutrons in the nuclear process. This balances the absorption losses. As shown in Figure 2.1 the reactor includes the reactor core, steam generators, pressurizer and cooling supply lines. All of these components are inside a large reactor pressure vessel. The cooling cycle of an IPWR is powered either from a pump inside the reactor pressure vessel or from natural circulation [1][2][3][4].

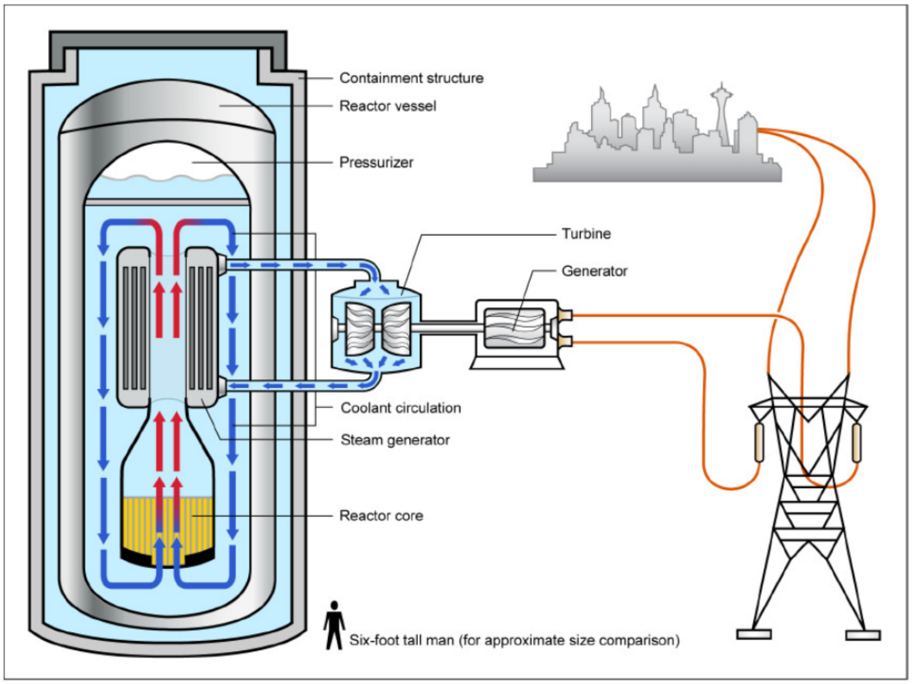

Figure 2.1: Schematic representation of a IPWR. [6]

2.2 Liquid Metal-Cooled Reactors, LMRs

Liquid metal-cooled reactors are cooled by metals such as sodium or lead bismuth as the primary coolant. These metals have high boiling-points and high thermal conductivity, thus they operate at high temperatures of approximately 750°C and at ambient pressure. The circultion of the metal inside the reactor is powered by electromagnetic pumps or natural circulation. A second cooling system which also uses liquid metal as coolant is installed between the primary cooling system and the stream generators. This safety system is installed so that only non-radioactive metal can react with water in the case of steam generator leakage. All LMRs are fast neutron-reactors thus a moderator is not needed in this system. Liquid metal-cooled reactors use the full energy potential of uranium compared to conventional power reactors which use only one percent of the uranium energy [1][2][3][4].

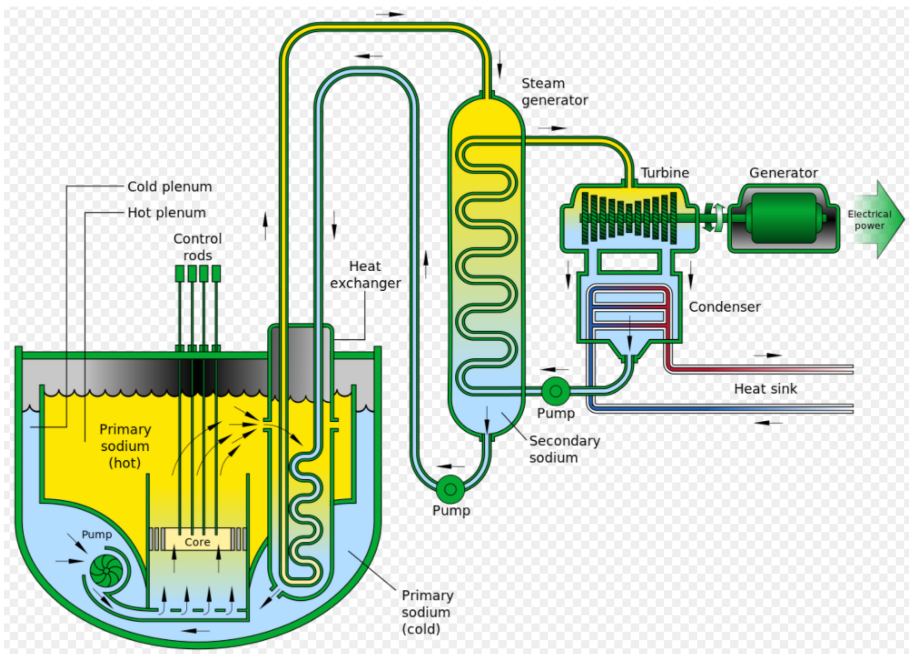

Figure 2.2: Schematic representation of a LMR. [7]

2.3 High-Temperature, Gas-Cooled Reactors, HTGRs

High-temperature, gas-cooled reactors are operated with pressures greater than 7 MPa and tempeatures up to 1000°C, which is higher than in other reactor types. This is possible by the use of gas as the coolant and graphite as the moderator inside the reactor core. The fuel elements consist of graphite into which the uranium, in the form of many smaller coated particles, is embedded. The ceramic coating of uranium particles serves to retain the fission products. Usually Helium is used as the coolant because of high temperatures in the primary reactor system. [1][2][3][4].

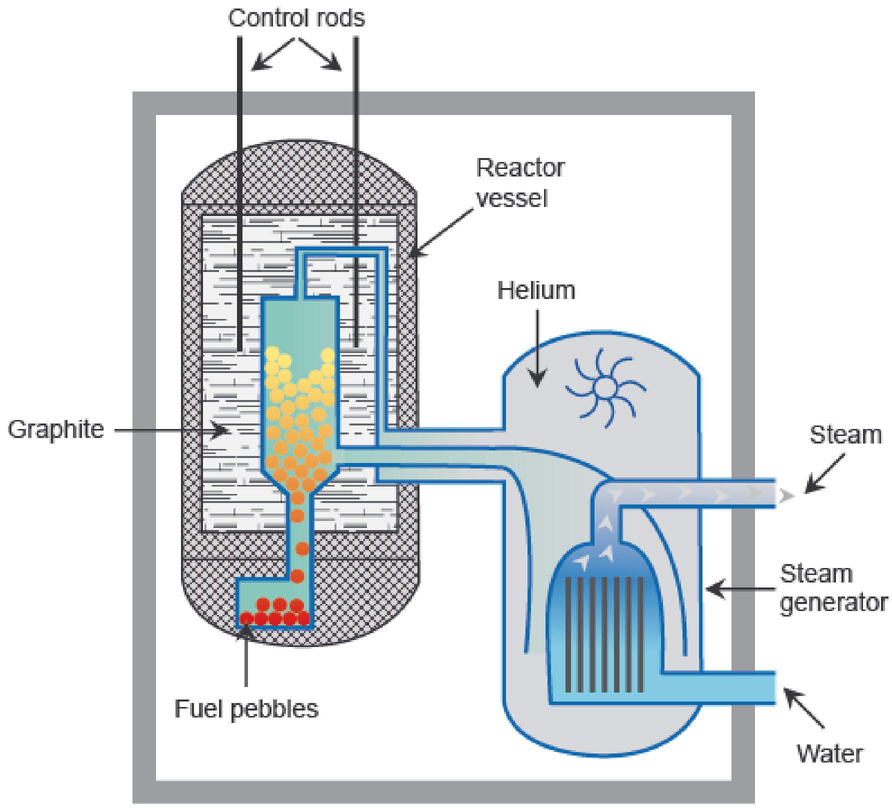

Figure 2.3: Schematic representation of a HTGR. [8]

2.4 Molten Salt Reactors, MSRs

In molten salt reactors, a molten salt which consists of fuel, cooling liquid and fission products, is used to run the nuclear reaction and to transport the produced heat. The molten salt circulates between the core and a heat exchanger. Only in the core is the nuclear moderation triggered by the existing moderating graphite and thus heat energy released. Outside the core the molten salt is subcritical. A second cooling cycle which also uses molten salt is used to transport the heat to the steam generator. MSRs can be operated at temperatures up to 1400°C. At higher temperatures, the molten salt is unstable [1][2].

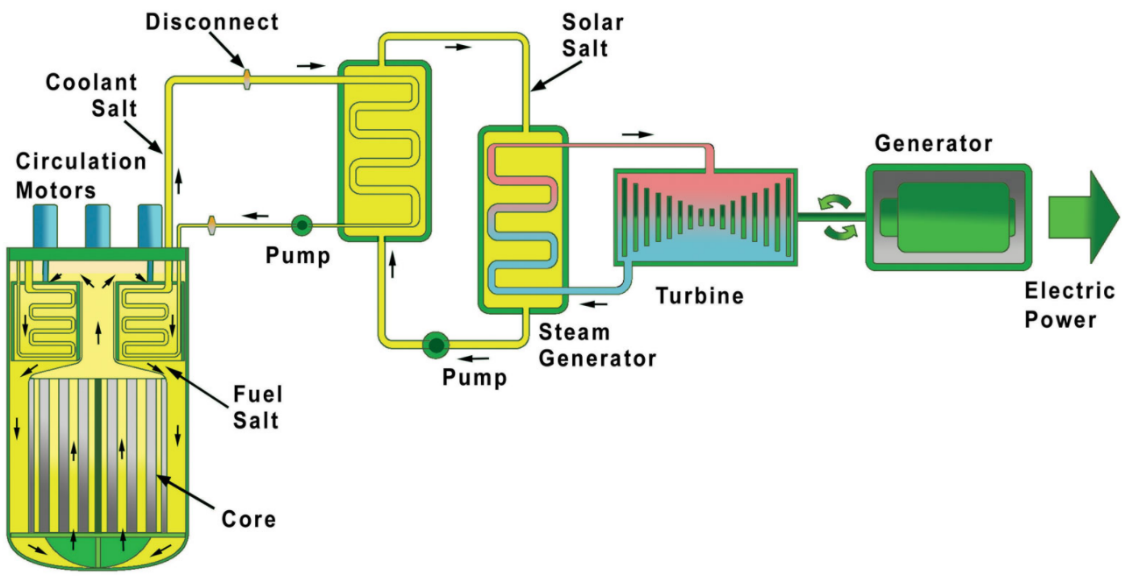

Figure 2.4: Schematic representation of a MSR. [9]

3 Basics of nuclear Fission

The principle of nuclear energy generation is based on the capture of a neutron by a fissle heavy atomic nucleus (e.g. uranium). By this capture, the nucleus is excited resulting in fission. Products of fission include two fragments (e.g. krypton and barium) and two to three new neutrons. In addition energy is released during the fission process, which is ultimately used to generate electrical energy. This chapter will deal with the basic topics of nuclear energy generation [10][11][12].

3.1 Structure of Atomic Nuclei

To begin with, it is important to consider the components of an atomic nucleus. Atoms consist of a nucleus and an electronic shell. While the core consists of positively charged protons and neutral neutrons, the shell consists exclusively of negatively charged electrons. Protons and electrons have a mass and an electrical charge, as can be seen in Table 3.1. Neutrons have a mass but no charge. The mass number (A) is the number of nucleons (protons and neutrons)in the nucleus.

The Z-number (number of protons) determines the position of the atom in the periodic table of elements. In the symbol 3.1, it is seen how the numbers of an element can be read [10][11][12].

$$^{A}_{Z}Element \tag{3.1}$$

Table 3.1: Mass and charge of the components of an atomic nucleus [10]

	Mass [kg]	Charge [C]
Proton	$1.6726 \cdot 10^{-27}$	$1.6022 \cdot 10^{-19}$
Neutron	$1.6726 \cdot 10^{-27}$	0
Electron	$9.1094 \cdot 10^{-31}$	$-1.6022 \cdot 10^{-19}$

3.2 Binding Energy

The components of the nucleus are held together with the strong force which works between the nucleons (protons and neutrons) in the nucleus. Between proton and proton, neutron and neutron and neutron and proton, the nuclear strong force is about equally strong. Nuclear strong forces only work at very short distances of less than $2 \cdot 10^{-15}$m, but then these are much stronger than all other interaction mechanisms. Binding energy is the energy required to separate the nucleons from the nucleus. The separation of the nucleons requires energy, that means, the sum total mass of nucleons is larger than the mass of the nucleus. As can be seen clearly in Figure 3.1, the binding energy per nucleon in the heaviest atomic nuclei (e.g. uranium) is about 7.5 MeV. It can also be seen that the binding energy per nucleon of uranium is less than the binding energy per nucleon of medium-heavy atomic nuclei, which corresponds to approximately 8 MeV (e.g, Fe-56).

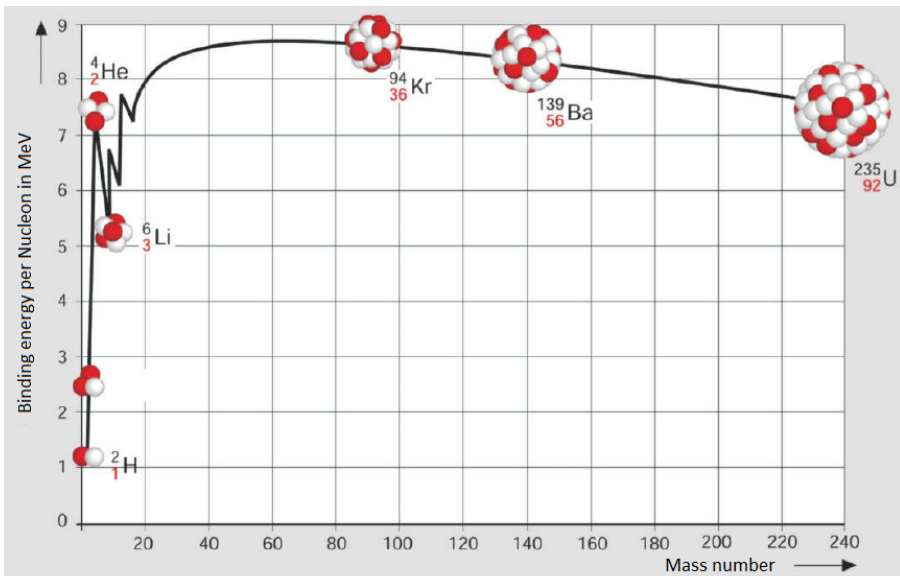

Figure 3.1: Binding energy over Atomic mass number [10]

For example, if uranium is split into two medium-heavy atomic nuclei, the binding energy per nucleon of the fission products is larger than the uranium nuclus binding energy per nucleon. That means, energy must be released, as shown in Equation

3.2. In this equation both sides of the nuclear reaction equation, not only number of nucleons but also energy must be conserved. It can also be seen that, in a fission, not only energy, but also three new neutrons are released, which can then again fission new uranium nuclei [10][11][12].

$$^{235}_{2}U +^{1}_{0} n \rightarrow^{94}_{36} Kr +^{139}_{56} Ba + 3 \cdot^{1}_{0} n + 210MeV \tag{3.2}$$

3.3 Mass Defect

According to Einstein's theory of relativity E=m·c^2, mass and energy are proportional. This means that, on both sides of the nuclear reaction equation 3.2, not only the same number of nucleons but also the same energy must be present. In in equation 3.2 with the atomic masses for: ^{235}U=235,04392996amu, ^{1}n=1,00866492amu, ^{94}Kr=143,92295281amu ,^{139}Ba=88,91763058amu and c=299 792 458 m/s, 1amu = 1,6605·$^{-27}$ kg

$$\Delta m = m_A(^{235}U) + m_A(^{1}n) - m_A(^{94}Kr) - m_A(^{139}Ba) - 3 \cdot m_A(^{1}n) \tag{3.3}$$

$$\Delta m = (235,04392996 - 143,92295281 - 88,91763058 - 2 \cdot 1,00866492) \cdot amu \tag{3.4}$$

$$\Delta m = 0,21338354amu \tag{3.5}$$

From this example, it is seen that energy must be released on the product side of the equation 3.2 to satisfy the equation. On the other hand, it can also be seen that the binding energy has a direct influence on the mass of atomic nuclei. The required energy can now be determined by the theory of relativity [10][11][12].

$$\Delta E = 0,21338354 amu \cdot c^2 \tag{3.6}$$

$$\Delta E = 210\ MeV \tag{3.7}$$

3.4 Neutron Reactions

In a nuclear reactor, many different neutron-nucleus nuclear reactions occur. In this section, however, only the four most important reactions will be discussed.

- Neutron absorption with nuclear fission
- Neutron absorption with γ-ray emission
- Neutron elastic collision
- Neutron inelastic collision

Neutrons can be divided into two groups: thermal and fast neutrons. A fast neutron has an energy of >1.0 MeV. In addition, the atomic nuclei can be distinguished into fissile and non-fissile nuclei. Fissile nuclei include ^{235}U and ^{239}Pu. Non-fissile but fissionable nuclei include ^{238}U and ^{232}Th. For the fission of an atomic nucleus, the critical energy must be overcome. This results in the following condition for a fission:

$$Binding\ \ energy + Kin.\,Energy\ \ of\ \ the\ \ last\ \ neutron > Critical\ \ energy \quad (3.8)$$

In the case of fissile atomic nuclei, the required critical energy is smaller than the binding energy of the last neutron. In the case of atomic nuclei which are fissionable, the binding energy of the last neutron is smaller than the critical enery at the compound nucleus Therefore, the neutron kinetic energy must be larger then the differnce, in order for fission to occur [10] [11][12].

3.5 Cross Section

The cross section is a quantity which determines the probability of a particular reaction of an atomic nucleus upon capture of a neutron. The cross section is dependent on the neutron energy. In Figure 3.2 and 3.3 it is very easy to see the magnitude of a given cross section when a neutron with a certain energy excites an atomic nucleus.

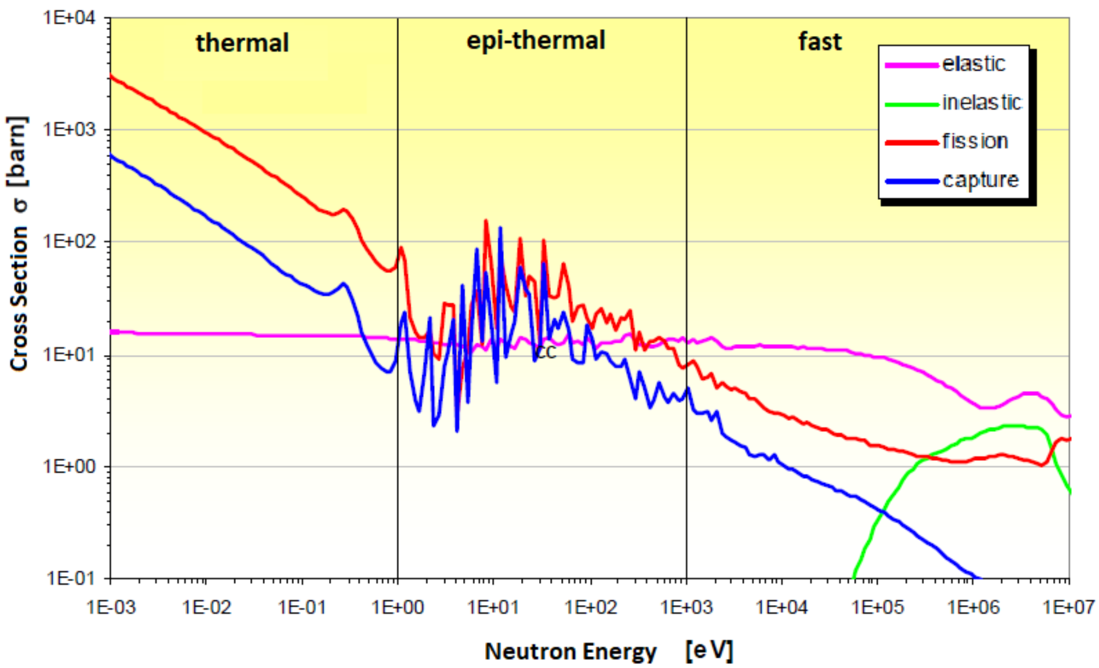

Figure 3.2: Cross sections for Uranium 235. [10]

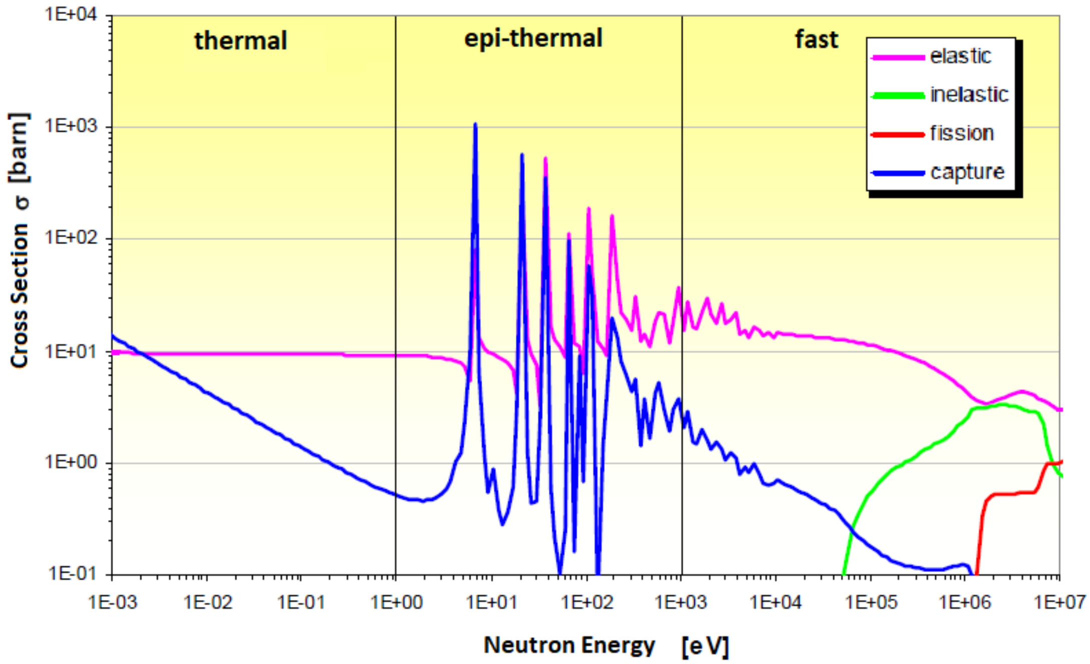

Figure 3.3: Cross sections for Uranium 238. [10]

As can be seen clearly from the figures, a fission occurs at ^{235}U even at low neutron energies. For ^{238}U, a certain neutron energy (velocity) is necessary to trigger a fission or to improve the probability of a fission [10][11][12].

3.6 Moderation

The neutrons generated during nuclear fission have initial energies of approximatly 2 MeV and thus are fast neutrons. Since the fission cross section, is higher in the thermal energy range by several orders of magnitude than the fast range, a moderator is used to slow down the new, fast neutrons by about 7 orders of magnitude on the energy-scale before they trigger new fissions. In the various reactors, graphite and water (light or heavy) are used as moderators. Figure 3.4 shows the three energy ranges in which a neutron can be found [10][11][12].

- thermal range: neutron energy < 1 eV
- resonance range: 1 eV $\leq$ neutron energy $\leq 10^5$ eV
- fast range: neutron energy $> 10^5$ eV

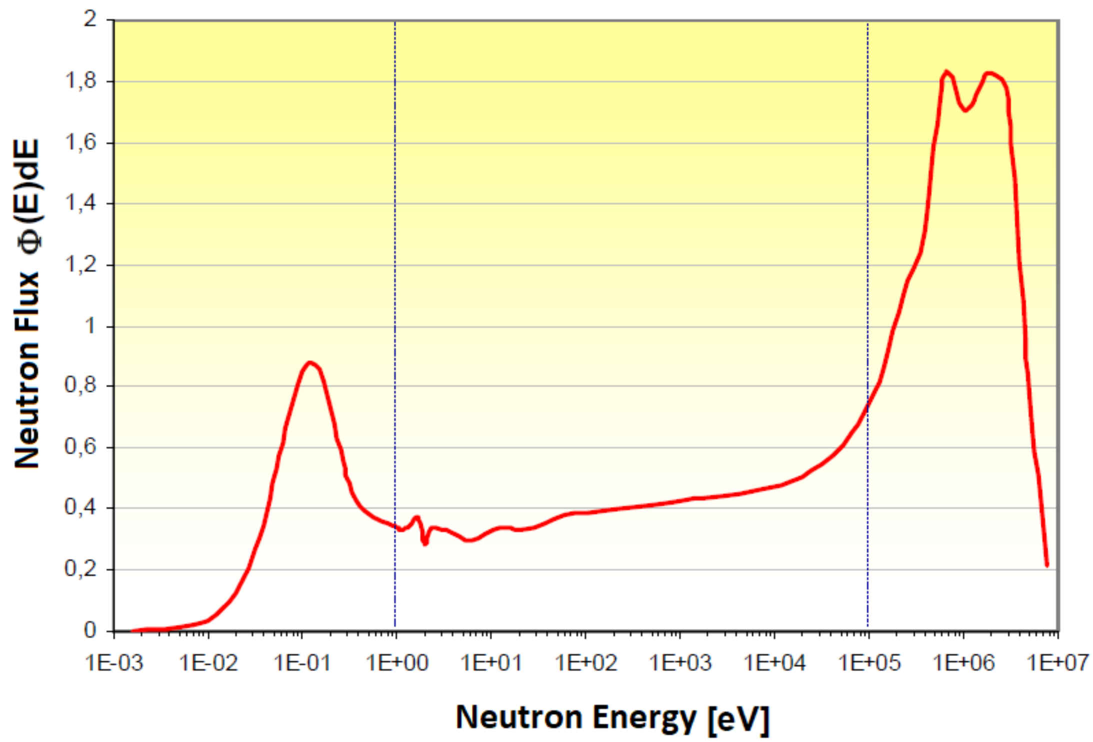

Figure 3.4: Neutron energy spectrum of a reactor [10]

3.7 Neutron Life Cycle

On the subject of safety it is necessary to consider the neutron generation cycle and the so-called multiplication factor, k, in a nuclear reactor. The approximate course of the neutron generation cycle can be described as follows. If there are N thermal neutrons in a reactor, they are affected by different events in the generation cycle.

Some of the neutrons are absorbed by the moderator (f: Thermal Utilization Factor) before they enter the fuel. The remaining neutrons then dissipate fissions in the fuel or are absorbed by the fuel (η: Thermal Fission Factor). This results in new fast neutrons, which can trigger fast fissions (ε: Fast Fission Factor). Some of these fast neutrons can leave the reactor and the rest (P_f Fast non-leakage Probability) of these fast neutrons is now slowed down in the moderator. Some are lost in the resonance range / braking range (p: Resonance Escape Probability). The result are thermal neutrons which again are able to leave the reactor (P_t thermal non-leakage Probability). This also decreases the number of neutrons. In the end, N$\cdot f \cdot \eta \cdot \varepsilon \cdot p \cdot P_f \cdot P_t$ neutrons remain in the reactor. The multiplication factor, k, is defined as k=$f \cdot \eta \cdot \varepsilon \cdot p \cdot P_f \cdot P_t$ and represents the generation cycle of all neutrons present in the reactor. Figure 3.5 illustrates this process again with 1000 Neutrons [10].

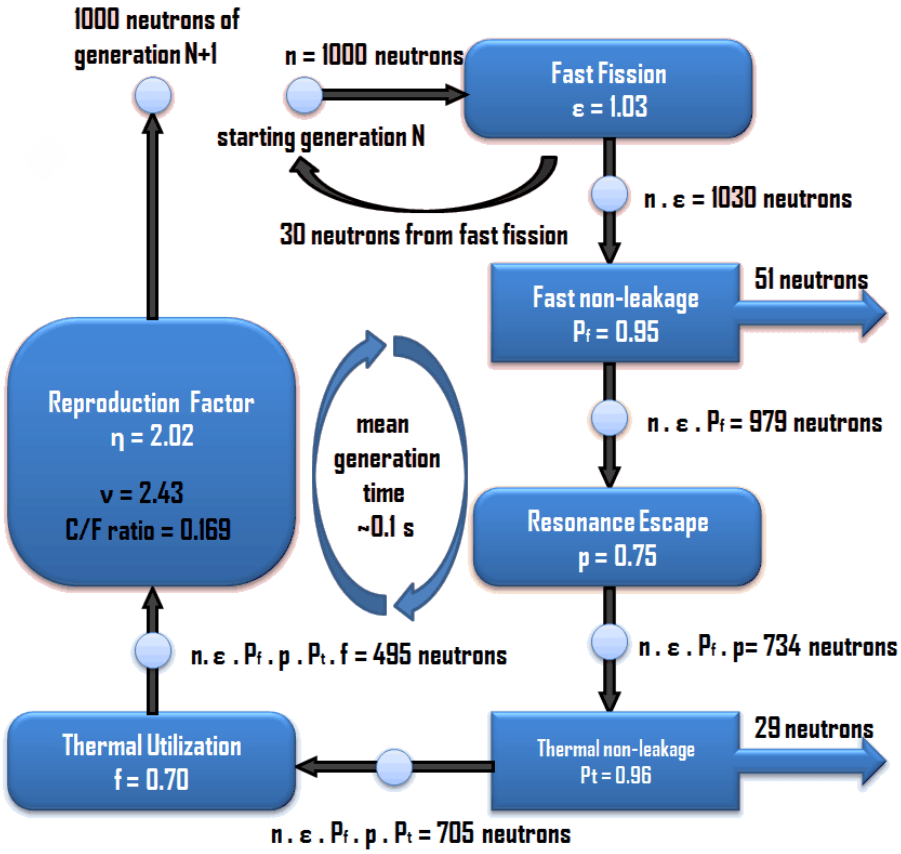

Figure 3.5: Illustration of the Neutron generation cycle. [10]

Thus, the power of the reactor is dependent on the multiplication factor k, since when k > 1 more neutrons are available for fission than in the previous cycle. Through different values of k, a nuclear reactor can be divided into three states. In addition to that the so-called reactivity, ρ, represents the deviation from the critical state and is calculated with $\rho = 1 - \frac{1}{k}$.

- subcritical: k < 1, $\rho < 0$
- critical: k = 1, $\rho = 0$
- supercritical: k > 1, $\rho > 0$

At the various values of the reactivity (ρ), the power of the reactor may either rise ($\rho > 0$), fall ($\rho < 0$), or remain at a steady level ($\rho = 0$). The neutron population is the key factor for control and safety in the nuclear reactor. This is achieved by the control rods in the reactor. The control rods are arranged between the individual uranium fuel elements and can be retracted and extended as required. The control rods are made of cadmium, boron or a similar material that has a high thermal neutron absorption cross section. Thus depending on how far the control rods are extended or retracted into the core of the nuclear reactor, the multiplication factor, k, and the reactivity, ρ, are influenced. In the same way the power of the reactor is influenced. The energy released during nuclear fission is released as heat energy. This must be transported through the various cooling cycles from the core to the turbine, where it is finally converted into electrical energy via a generator. The control rods regulate the released energy and thus also the total electrical energy production of the nuclear reactor [11][12].

4 Conventional Pressurized Light Water Reactors

Pressurized water reactors are the world's most widely used types of nuclear power plants. More than 70 percent of all nuclear power plants are designed as pressurized water reactors. A pressurized light water reactor is a nuclear reactor type which uses water as a coolant and as a moderator. As implied from the name of the reactor, the used water is under high pressure, which has an effect on its thermodynamic properties. In the case of light water reactors, normal water (H_2O) is used as a coolant, compared with deuterium in heavy water reactors. The rated power of a pressurized water reactor is between 700 MW_{el} and 1400 MW_{el}. The most important component of a pressurized water reactors is the reactor core in which the nuclear reaction takes place. The core is made out of different fuel assemblies, some of which are provided with control elements. The uranium enrichment (U_{235}) of the fuel in the fuel elements is between 1,9-4,8 percent. The fuel elements consist of the approximately 4 m long fuel rods. The fuel rods consist, in turn, of one column of fuel tablets consisting of sintered uranium dioxide (UO_2). These fuel columns are encased in gas tight and pressure-tight welded zircaloy tubes. The control elements for power control and fast disconnection in the pressurized water reactor consist of control rods made of neutron-absorbing Indium-Cadmium alloy. Figure 4.1 shows the structure of a pressurized water reactor.

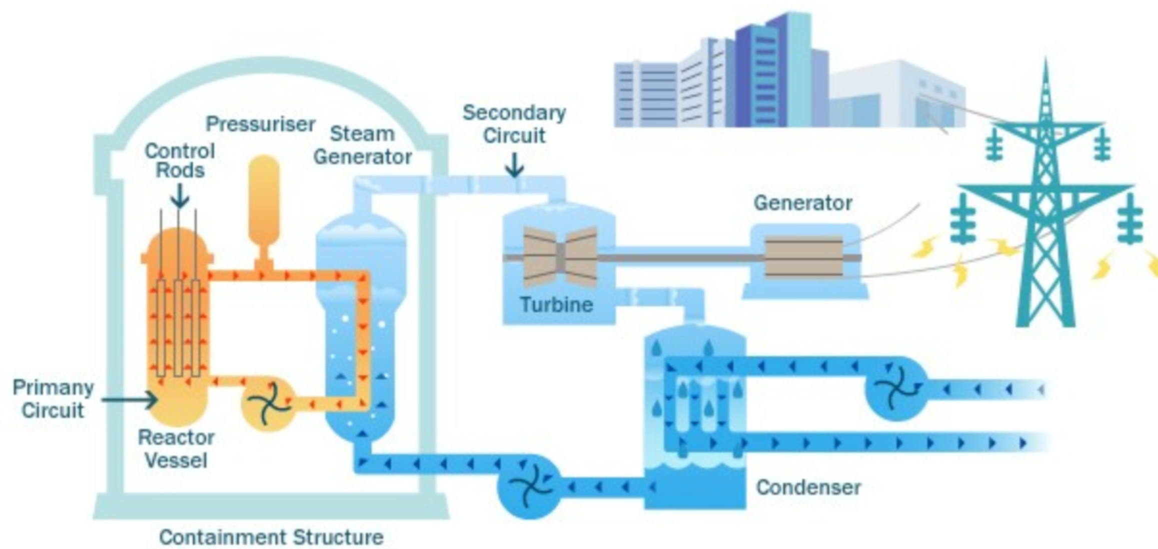

Figure 4.1: Structure of Pressurized Water Reactor [13]

As can be seen very clearly from the figure, the reactor has a total of three circuits for converting the heat energy from the uranium into electrical energy. These three circuits are called:

- Primary circuit
- Secondary circuit

- Cooling circuit

The primary circuit is located exclusively in a safety container. The separation of primary and secondary cycle is the task of the radioactive contaminated water exclusively within a safety container in the primary circuit. The advantage of this design compared to the boiling water reactor is that no radiation protection measures are necessary in the machine house where the turbines and the generator are located [14][15][16].

4.1 Primary Circuit

The primary circuit consists of the reactor core vessel in which the uranium fuel rods are located, the steam generators, the circulating pumps and connecting pressure pipes between these components. The entire primary circuit is surrounded by a protective cover made of reinforced concrete. The nuclear fission in the reactor core vessel produces heat energy and thereby heats the cooling water of the primary circuit. This heat energy is transported to the steam generator by means of the pumps and the high pressure water. There, the heat energy is transferred to the secondary circuit and the primary cooling water is thereby cooled. Afterwards, the water is fed back into the reactor core and the process begins again. As already mentioned, the water in the primary cycle is under a very high pressure of about 155 to 160 bar and has an evaporation temperature of 350°C. Since only temperatures of 290 to 325°C are reached in the primary circuit, there is no phase change of the coolant [10][14][15][17].

4.2 Secondary Circuit

The secondary circuit is a current clausius rankine power plant process. The water pressure is increased by a feed water pump. Thereafter, the water is directed into the steam generator, in which primary and secondary flow meet. There, the water changes its aggregate state from liquid to gaseous. This steam now drives turbines that are connected to a gernerator, which finally generates electrical energy. In the turbines, the steam is expanded to a lower pressure and is passed into a further heat exchanger in which the steam is again liquefied. The water is then returned to the feed water pump and the process starts again. The secondary circuit is operated at a pressure level of 60 to 70 bar. The evaporation temperature at these pressures is about 280°C. Thus, the pressure difference between primary and secondary side of the steam generator ensures that the primary circuit, which is under high pressure, triggers a phase change of the cooling liquid of the secondary circuit [10][14][15][17].

4.3 Cooling Circuit

The cooling circuit is used to ensure the liquefaction of the cooling water in the secondary circuit and to remove the waste heat which is not usable from the secondary circuit. Cooling circuit and secondary circuit meet at the second heat exchanger. For this last cycle, cooling water is required, depending on the location of the nuclear power plant, either from the sea or from a river. With the aid of a pump, the cooling water enters the second heat exchanger of the secondary circuit and is subsequently passed into the cooling tower. By this, it is then possible to dissipate the waste heat, which is not usable, into the environment [10][14][15][17].

4.4 Power Plant Example

The following Table 4.1 shows technical data of a typical pressurized water reactor as it is built around the world.

Table 4.1: Technical Data of a Pressurized Water Reactor [10]

Data	Value
Thermal Power	3850 MW
Electrical Gross Power	1363 MW
Electrical Net Power	1290 MW
Own Demand Power	73 MW
Net Efficiency	33 %
Number of Fuel assemblies	193
Number of Fuel rods per Fuel element	300
Nuclear fuel amount	103 t of Uranium
Power density	93 kW/l
Spec. Fuel line	37.4 kW/kg of Uranium
Coolant flow rate	19732 kg/s
Coolant pressure	158 bar
Coolant inlet temperature	291°C
Coolant outlet temperature	326°C
Steam pressure	63.4 bar
Steam temperature	280°C
Steam flow rate	2128 kg/s

Pressurized water reactors of this type are used worldwide and produce energy for the industries and the population of the respective countries. Some examples can be seen in Figure 4.2.

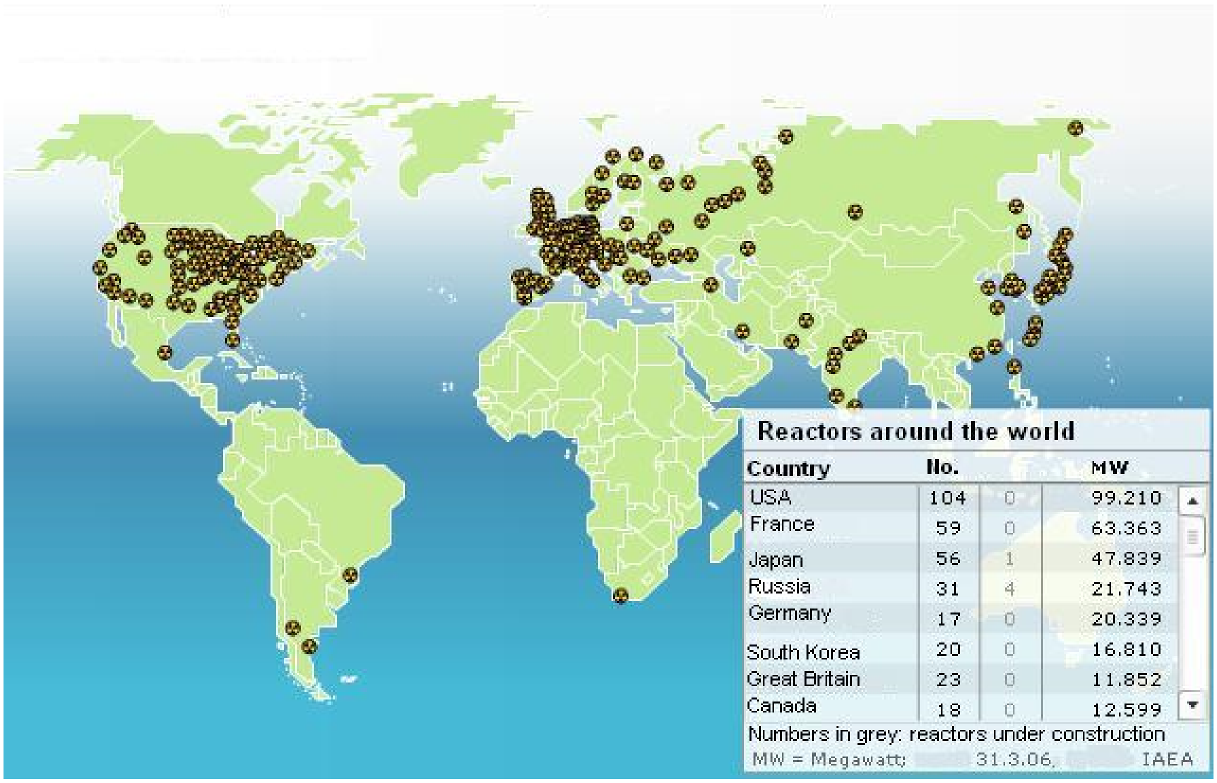

Figure 4.2: Locations of Nuclear Reactors worldwide. [18]

5 NuScale Systems

An example of a manufacturer of SMRs is the US-company NuScale. This enterprise is specialized in the design and development of integral pressurized water reactors (IPWRs). The company, founded in 2007, predicts that its technology will be commercially available by the year 2025, and will contribute a large share to clean energy generation [19][20].

5.1 NuScale Incorporated

The origins of NuScale Incorporated go back to the year 2000. That year the US Department of Energy (DOE) funded the research and development of a SMR. Idaho National Environment & Engineering Laboratory (INEEL) was commissioned with the project, being supported by the Oregon State University (OSU), which was at that time leading the development of passive safety systems and natural circulation for nuclear power plant cooling. After completation of the research project in 2003, the OSU scientists continued with the development of a SMR with natural circulation.

Finally they built a 1:3 version as a test facility for their design of a small modular reactor. Following the construction of the test facility the scientists founded NuScale in 2007. In exchange for a small equity in their new company they inherited related patents from the university. In 2008, NuScale began the certification of its SMR at the Nuclear Regulatory Commission (NRC). The design for the certification included a 50 MW_{el} module, which can be operated either independent or in cooperation with other modules to generate electric energy and was thus the first company to submit plans for a small reactor to the NRC. In 2011, NuScale employed 100 people in three cities (Tigard, Oregon; Richland, Washington and Corvallis, Oregon). In November 2014, NuScale announced that the first SMR nuclear power plant in the United States will be located in Idaho and therefore submitted drafts to the Nuclear Regulatory Commission in January 2017. If approved, the first facility with an SMR system will be completed by 2026 [19][20].

5.2 NuScale Small Modular Reactor

A NuScale SMR is a integral pressurized water reactor that can operate as a stand-alone unit or in a system of up to twelve SMR modules. All SMR units of the reactor system are enclosed in a high-strength containment vessel and work in a common pool filled with water, which contributes a great part to the safety of the reactors. Each vessel is called module and is equipped with its own steam turbine-generator. Figure 5.1 shows schematically how a NuScale SMR is constructed.

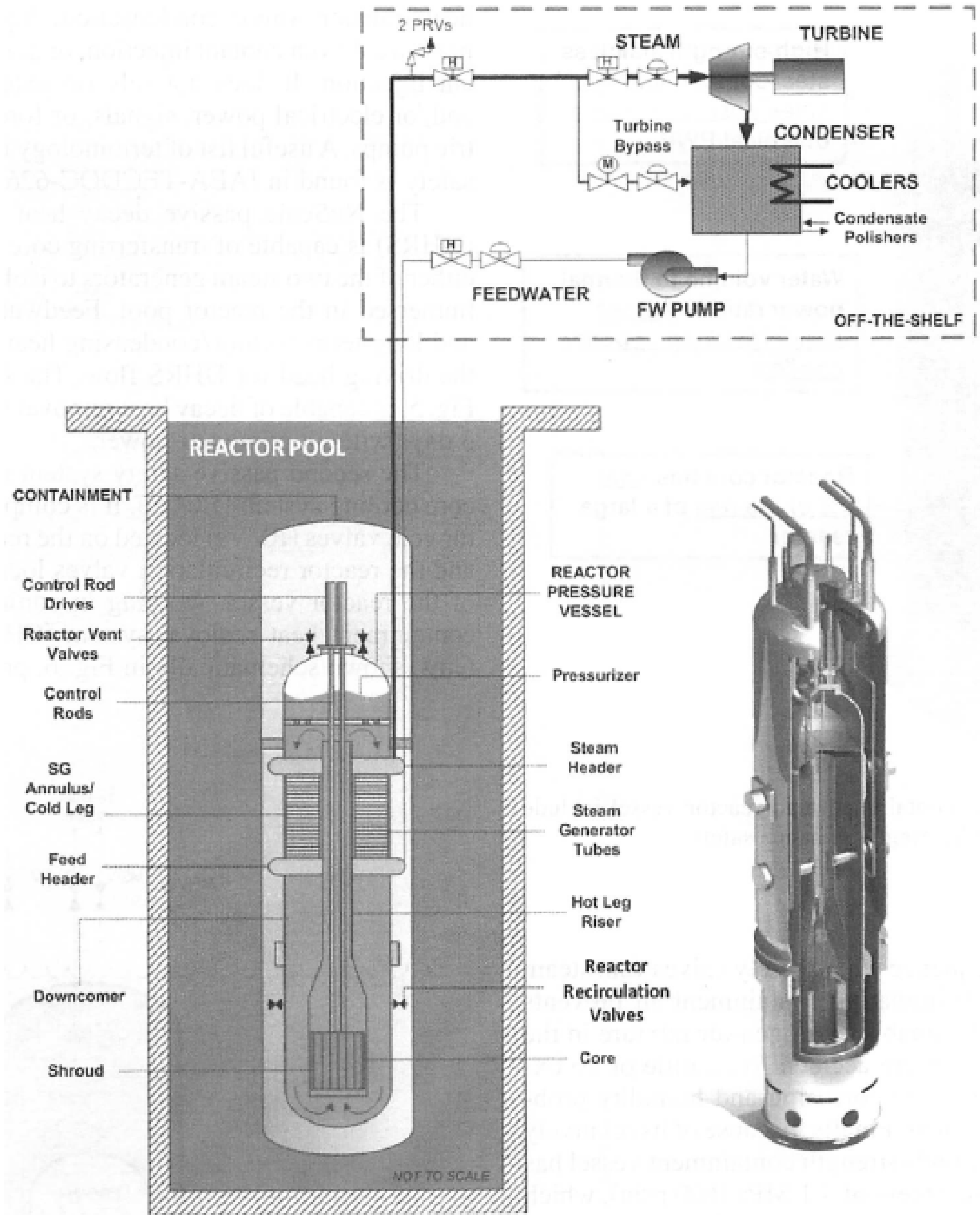

Figure 5.1: Schematic Construction of a NuScale SMR [19]

Due to the small size of the components of the reactor system compared to conventional light water reactors, the reactor, pumps and turbines are easy to transport, install and maintain. The NuScale SMR uses two cooling cycles to convert the heat energy of the core into electrical energy through generators. Each module consists of a reactor core, in which the fission reaction takes place. The reactor core is surrounded by a high-pressure safety vessel which has a height of 13.7 m and a diameter of 2.7 m and operates at a nominal operating pressure of 12.8 MPa. The reactor containment vessel, with a height of 19.8 m and a diameter of 4.4 m, can withstand pressures of up to 4 MPa without failing during accident scenarios. While the steam generators are located in the upper part of the high-pressure safety vessel, the core is installed in the lower part. Uranium oxide (UO_2) with an enrichment of 5 % is used as the fuel of the SMR in the core and is capable to produce 160 MW_t power. The primary cycle in the high-pressure safety vessel works according to the principle of natural circulation, for this reason no pumps are needed to allow cooling water to flow through the reactor core. The cooling water is heated by the nuclear reaction as it crosses the core. By heating, the density of the cooling water decreases and it rises upwards inside the closed cycle. As soon as the heated water reaches the upper part of the high-pressure safety vessel, it flows through the steam generators and is cooled there. The density of the cooling water increases again and is drawn back by gravity to the bottom of the vessel. There it is again heated by the reactor core and the process continues. The cooling water of the primary circuit is kept separate from the secondary circuit with the help of the steam generators in order to prevent nuclear contamination of the pumps and turbines. The heat of the primary circuit is transferred to the cooling water of the secondary circuit via the hundreds of pipes in the steam generator. During this process, the secondary circuit coolant is evaporated. The steam is used to drive turbines, which operate via a shaft to a generator which then produces electrical energy. After the turbines have been passed, the steam loses its energy. The steam is liquefied again in a condenser and is then pumped back into the steam generator with a feed water pump, where it begins the cycle again. The high-pressure safety vessel and reactor comtainment vessel comprises various design features that serve to increase safety and efficiency. On the one hand, a vacuum is generated in the space between the two vessels, which ensures that the pressure safety vessel does not have to be isolated. On the other hand, the NuScale SMR has two passive safety systems which, in the event of an accident, would lead to further cooling of the reactor. These two safety features are called Decay Heat Removal System (DHRS) and Emergency Core Cooling System (ECCS). Generally, the passive safety systems provide for cooling the core, using natural convection, to remove the core decay heat when the normal feedwater system is not available. Table 11.2 shows technical data of a NuScale SMR [19][20].

Table 5.1: Technical Data of a NuScale SMR

Data	Value
Thermal Power	160 MW
Electrical Gross Power	50 MW
Dimensions	19.8 m x 4.4 m
Weight	700 t
Cost	5100 $/KW
Fuel	5.15 t of Uranium

5.3 Decay Heat Removal System

The decay heat removal system or short DHRS is one of the two passive safety systems of the NuScale SMR. If the normal feedwater system of the secondary cycle is not available, it is possible to cool the reactor using the DHRS. For this purpose, capacitors, which are located on the outer wall of the high-strength containment vessel, are used. It is also necessary to close the valves, which connect the steam generators of the primary cycle to the secondary cycle, and to open the DHRS valves. After opening, the cooling water of the DHRS cycle is able to transfer the decay heat to the capacitors via the steam generators of the primary cycle. These then give the decay heat to the pool, into which the entire reactor vessel has been immersed. Natural convection also plays an important role in this process, since the primary cycle is driven by this process. The DHRS is able to remove decay heat for a minimum of 3 days without pumps or power. Figure 5.2 illustrates the discribeld process of the decay heat removal system in a NuScale SMR works [19][20].

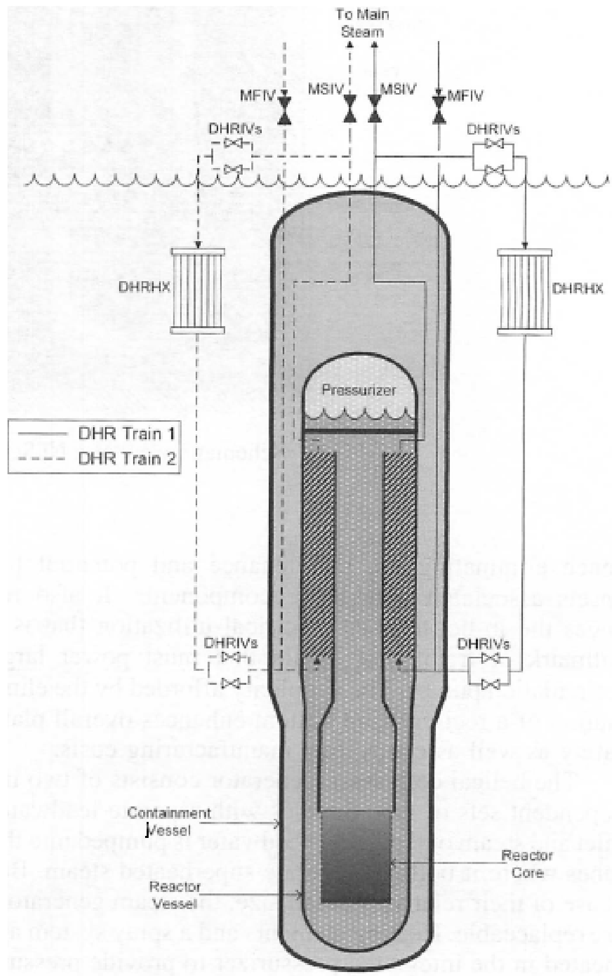

Figure 5.2: Decay Heat Removal System in a NuScale SMR [19]

5.4 Emergency Core Cooling System

The emergency core cooling system or short ECCS is the second passive safety system of the NuScale SMR. If the normal feedwater system of the secondary cycle or the decay heat removal system (DHRS) are not available, it is possible to cool the reactor using the ECCS. In the head of the core reactor vessel, ventilation valves are installed, which can be opened if necessary and thus lead to a pressure drop in the core reactor vessel. It is also necessary to close the valves, which connect the steam generators of the primary cycle to the secondary cycle. The cooling water of the primary cycle begins to boil as a result of the pressure drop and changes its state from liquid to gaseous when it passes throuth the core. The steam enters the high-pressure safety vessel from the vent valves. Through the heat exchange between the steam and the water of the reactor pool, the steam is again liquefied and the core heat is released to the environment. The re-liquefied cooling water collects in the lower area of the high-pressure safety vessel. When a certain level of liquid has been reached in the high-pressure safety vessel the recirculation valves which are installed on the sides of the core reactor vessel are opened. Through these valves, the condensate returns to the vessel and passes through the core, where it evaporates. Thus, the process can begin anew. This process makes use of natural convection, since it also works without pumps. The difference to the natural convection in normal primary cycle is that here a phase change of the cooling water takes place. But in general, the driving force is the density difference in the different phases of the process. Figure 5.3 illustrates the discribeld process of the emergency core cooling system in a NuScale SMR works [19][20].

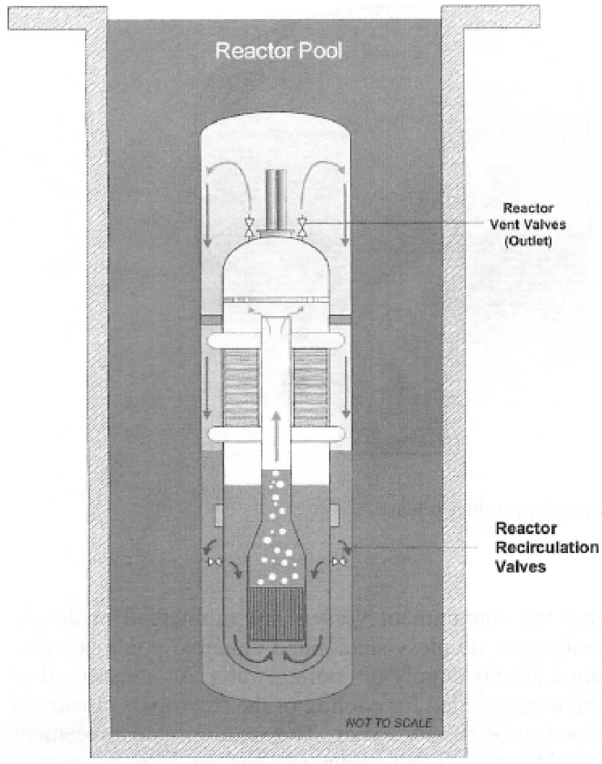

Figure 5.3: Emergency Eore Cooling System in a NuScale SMR [19]

5.5 Behavior of the Pool

As previously described, each reactor module of an SMR system is submerged in a common pool of water. In this pool, there is water to cool the reactors in accident scenarios for another 30 days after the incident. This is only possible if the water pool is not refilled during this time, which is possible without any problems. For this purpose, it is possible to use rotary pumps. One of the advantages of this design is that the water needed for cooling is present at all times and does not have to be first transported to the reactor. Furthermore, the pool is underground, which is an advantage in terms of extreme events and offers significant protection against earthquakes, floods, tornadoes and aircraft impacts. Figure 5.4 shows how the pool behaves in the event of an accident. The figure shows the NuScale design to provide long-term cooling (LTC) for the case of a complete station blackout without additional cooling or water addition to the reactor water pool.

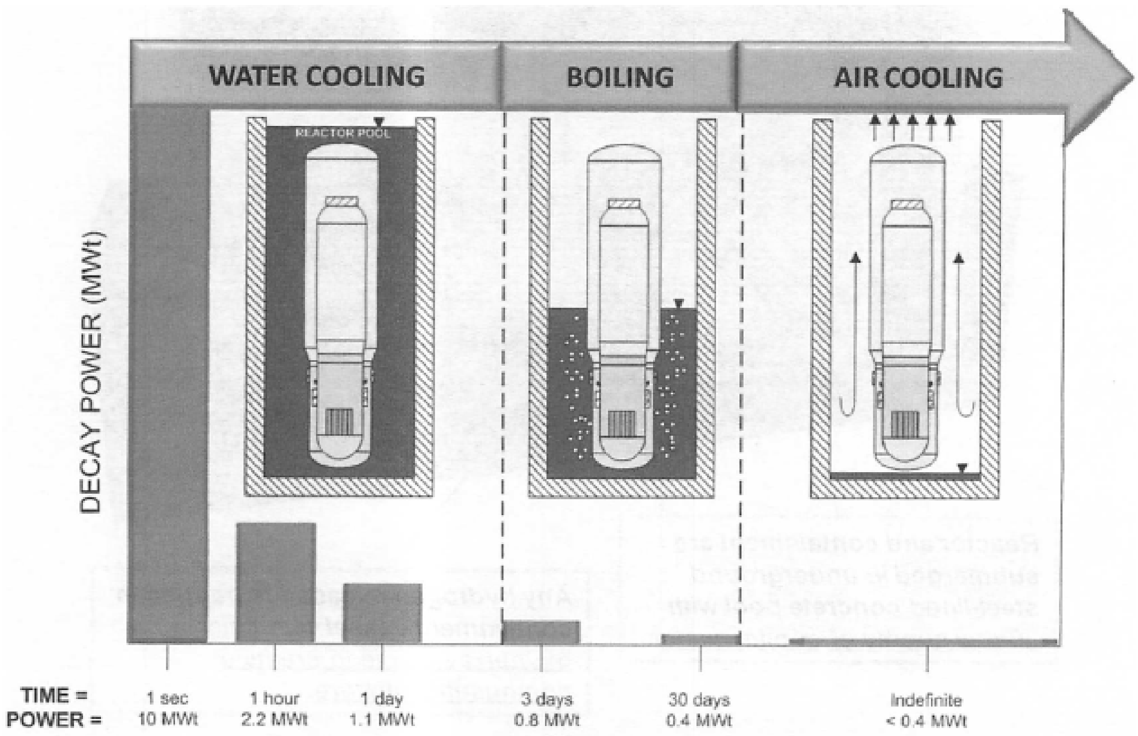

Figure 5.4: Pool Behavior in a NuScale SMR [19]

The figure shows the three phases of LTC defined in terms of heat transfer mechanisms. In addition, it shows how much heat is released from the reactor in the different phases of the LTC. The first phase is called the water cooling phase. In this phase of the LTC, the entire reactor is submerged in the water pool one of the passive safety systems (ECCS/DHRS) would be in operation. The heat generated in the core is then released via the containment vessel to the pool and thus the core is cooled. If the pool is not cooled and no new water is introduced into the pool, the liquid level will drop over time as a result of evaporation and subsequent saturated liquid boiling in the pool. Estimates indicate that the liquid level of the pool has reached the top of the containment vessel after about 3 days. It is also required that at the end of the first phase of the LTC the released core heat is less than 1 MW (thermal). In the second phase of the LTC, the liquid level of the pool is below the top of the containment vessels and above the bottom of the containment vessels. The second phase of the LTC of the reactor lasts from day 3 to day 30. At this time, the reactor is cooled by the boiling water of the pool, which then evaporates in time. Rough estimates suggest that all the water in the pool would evaporate after 30 days, without the addition of new water. At the end of the second phase of the LTC, the core power should not exceed 0.4 MW (thermal) per module. The last and third phase of long-term cooling of the reactor begins after 30 days. After this time, it is sufficient to cool the reactor by the natural convection with the surrounding air and the heat radiation from the outer surface of the containment vessels. The core heat at this time is only 400 kW per module [19][20].

6 Advantages of Small Modular Reactors

Small modular reactors are a further development of the old conventional nuclear reactors, which have already been built for decades. Through new technologies and innovations, SMRs have many advantages over the conventional large nuclear reactors. For example, light water reactors with an electrical output of 1400 MW with integral pressurized water reactors of 50 MW can be compared with one another. The advantages of the SMRs include:

- Size
- Transportation
- New Applications
- Manufacturing Process
- Safety

These points are among the most important advantages and will be further discussed for this reason [1][2][10][17][19].

6.1 Size

The difference in size is one of the main reasons why SMRs have an advantage compared to typical conventional reactors. The dimensions of a conventional reactor, such as is built for example in the United States of America, are 61 m x 37 m. In comparison, the dimensions of a NuScale SMR are only 19.8 m x 4.4 m. Thus, the conventional reactor is about 217 times larger than the SMR. The advantages due to size are unambiguous. On the one hand, the SMR does not need a separate building to work, for example, it can simply be placed next to the machine, which is to supply it with power. On the other hand, the energy losses that occur during the transport of the energy are eliminated since the SMR has to work locally and no large distances have to be bridged [1][2][10][17][19].

6.2 Manufacturing Process

Each conventional nuclear reactor is individually planned, licensed, tested and subsequently built, with great effort in a building. This building has to be planned and licensed after special conditions for nuclear power plants. This process makes the core actuator very expensive. Compared to small modular reactos which can be built on the assembly line at significantly better costs. In addition, SMRs can be manufactured in one place and mounted in a different location. This is made possible the smaller size and by the use of standardized components. This also eliminates the strict licensing procedures by authorities for each individual reactor since SMRs are all the same. The production of spare parts is also simpler and cheaper for small modular reactos, thanks to the use of standardized components. If a component is no longer working, it can be quickly replaced by a new component of the same type and does not have to be produced anew. For these reasons, it is very easy to derive the economic benefits of small modular reactors in terms of manufacturing [1][2][10][17][19].

6.3 Transportation

Another advantage of the small modular reactors is that they can be transported easily. This is made possible by the low weight and the demountability of the SMRs. In some regions of the world, it is difficult to produce energy or transport energy to these regions. Examples include islands in open Pacific Ocean, deserts or states where the energy grid is poorly developed. Due to the transportability of the small modular reactors it is also possible to supply these regions of the earth with sufficient energy. The individual parts of the reactors can be transported by ship, truck or plane to the respective continent, mounted there and then provide the energy supply. Thus, in the future it will be possible to supply every part of the earth with efficient cheap and clean energy. In special cases, submarines or rockets can also be used to transport the reactors to power a space station or an underwater laboratory [1][2][10][17][19].

6.4 New Applications

In addition, new applications for nuclear reactors are possible with small modular reactors. Large conventional light water reactors have hitherto only been used to generate large scale energy for large cities, densely populated regions and aircraft carriers. In other words, only where there is enough space to build a large reactor. In addition to that a large body of water has to be near the conventional nuclear power plant due to it large decay heat. Due to the considerably smaller size of an SMR, machines or small regions whose energy requirements are not so high can be supplied in the future. Examples of this are tunnel boring machines, large production plants or countries in Africa whose energy requirements are generally very low. Another factor for these new applications is the lower price of a small modular reactor compared to a conventional reactor [1][2][10][17][19].

6.5 Safety

Small modular reactors are safer than conventional pressurized water reactors due to their passive safety systems. One of the most important features of the passive safety of an SMR is the natural convention cooling cycle which continues to function even in the event of power or secondary systems failing, to cool the nuclear core of the reactor. In addition, the reactor core of the SMR is surrounded by several safety vessels and the entire reactor is placed in a water pool below the ground level. If an accident occurs during which the pool is completely emptied by the passive safety systems, but the reactor still needs to be cooled with water, the pool can be easily refilled with mechanical pumps and the cooling process can be continued[1][2][10][17][19].

7 Nuclear Energy Cycle

This chapter deals with the nuclear energy cycle in short how the released heat energy of the uranium is transferred into electrical energy. For this purpose, the primary topic of this chapter is the secondary cycle of a steam-electric power plant. The task of a power plant is to convert primary energy into electrical energy. It is a thermal power plant when the primary energy is first converted into heat energy and then transferred to a heat engine. Each thermal power plant consists of a heat generator and the heat engine. The heat generator converts the primary energy of the power plant into heat and the heat engine converts this heat into useful electrical energy. Depending on what primary energy a thermal power plant uses, it can be divided into three different types.

- Fuel power plants (Coal and Gas)
- Nuclear power plants
- Thermal Solar power plants

In the heat engine, the working fluid passes through a cycle and is heated. At a special point in the cycle, this heat is transferred to the secondary cycle via a steam generator. As a working medium water or steam is almost always used, due to its good thermodynamic properties and high heat capacity. The secondary cycle of a steam-electric power plant is commonly referred to as Rankine cycle and is named after the Scottish engineer William John Macquorn Rankine. The Rankine cycle is generally used for any type of power plant where steam is used to generate energy. The Rankine basic cycle can be divided into four primary components.

- Feed Pump.
- Steam Generator
- Steam Turbine
- Condenser

The construction scheme of the cycle can be seen in Figure 7.1.

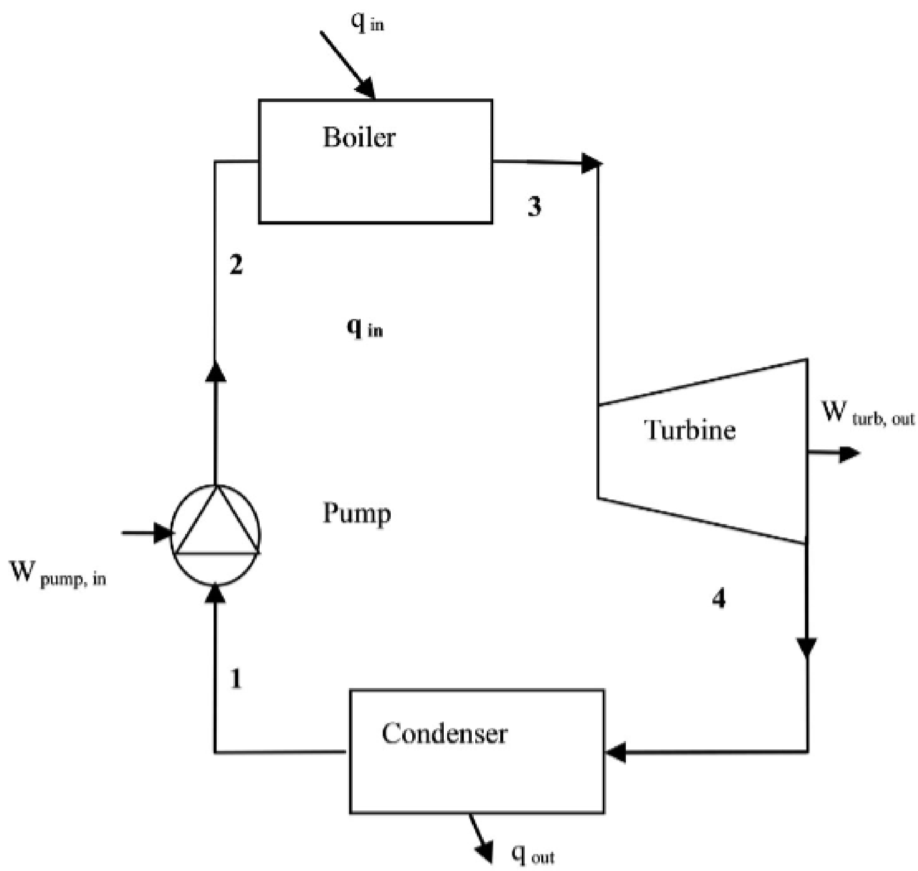

Figure 7.1: Construction scheme of the Rankine cycle [21]

It is also possible to divide the rankine cycle into an ideal and a real cycle. The difference lies in whether or not the feed pump and the steam turbine are considered as ideally working components [22][23][24].

7.1 Ideal Cycle

Figure 7.2 shows a Temperature-Entropy-diagram of an ideal Rankine cycle. As can be seen from this figure the cycle runs as follows.

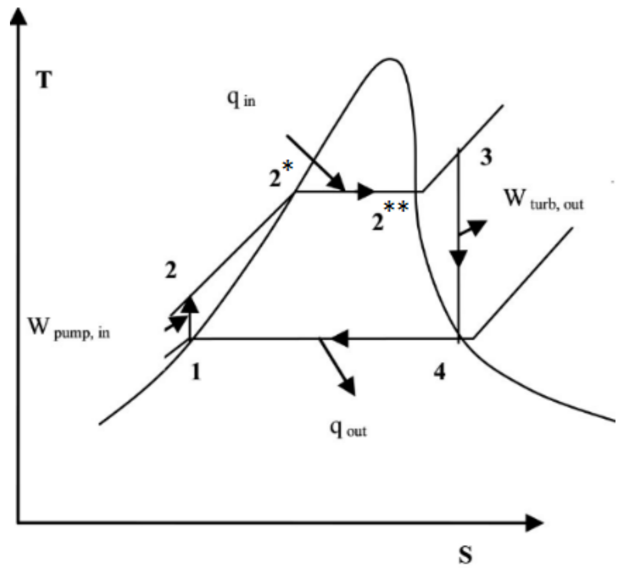

Figure 7.2: T-S-diagram of an ideal Rankine cycle [21]

- 1-2, adiabatic isentropic pressure increase by the feed pump
- 2-3, isobaric heat supply and superheating in the steam generator
- 3-4, adiabate, isentropic expansion of the steam in a steam turbine
- 4-1, isobaric condensation of the vapor in the condenser

When the first state is reached again, the cycle begins again. At the beginning of the ideal cycle (point 1), there is a liquid working medium that is compressed to a high pressure with the feed water pump (point 2). The pump has to do work, but this is done by the later expansion of the steam in the turbine. After the pressure increase, the working medium is directed into the steam generator, where it changes its state of aggregation. In this process, three subprocesses occur. First, the working medium is heated to the boiling point (point 2*) then it is evaporated (point 2**) and finally brought into a superheated state by further heating of the now vaporous working medium (point 3). The steam is now expanded in the steam turbine, while doing work and releases mechanical energy. With this mechanical energy on the one hand, the feed pump, but also the electric generator for power generation is operated. The working medium is partially condensed during this process (point 4). Subsequently, the remaining vapor is liquefied again in the condenser so that the working medium again assumes its initial state (point 1). After that the medium is again passed into the feedwater pump. The work done can be read directly from the T-S diagram. The area enclosed by the working curve of a cycle represents the work gained during the cycle. In order to be able to quantitatively evaluate the work achieved, the efficiency (η) of the cycle has to be considered. The efficiency of an ideal Rankine Cycle is defined as follows.

$$\eta_{th}^{rev} = \frac{w_{4-3}^{rev} - w_{2-1}^{rev}}{q_{3-2}} = \frac{w_t^{rev}}{q_{3-2}} \tag{7.1}$$

From the given work of the turbine ($w_{4\text{-}3}$), the required work of the feedwater pump ($w_{2\text{-}1}$) has to be deducted in order to determine the net work (w_t) of the process. The advantage of the rankine cycle is the large specific cycle work that results from the high specific volume difference between liquid and vapor. As a result, the feed water pump has to do little work to increase the pressure, while the turbine generates significantly more work in the expansion of the working fluid. This leads to relatively high efficiencies of up to 40 %.To calculate the efficiency level, extensive tables of the performance of the working fluid at different pressures and temperaturesare are required [22][23][24][25].

7.2 Real Cycle

In the real Rankine cycle, the feed water pump and the steam turbine are not working ideally. Now these components have an inner efficiency like the rest of the process. This means that not all the work that these components require is converted. Part of the work involved in operating the feed water pump and the steam turbine is lost inside these components. The losses can be seen in the following T-S diagram 7.3 with an increase of the thermodynamic value entropy.

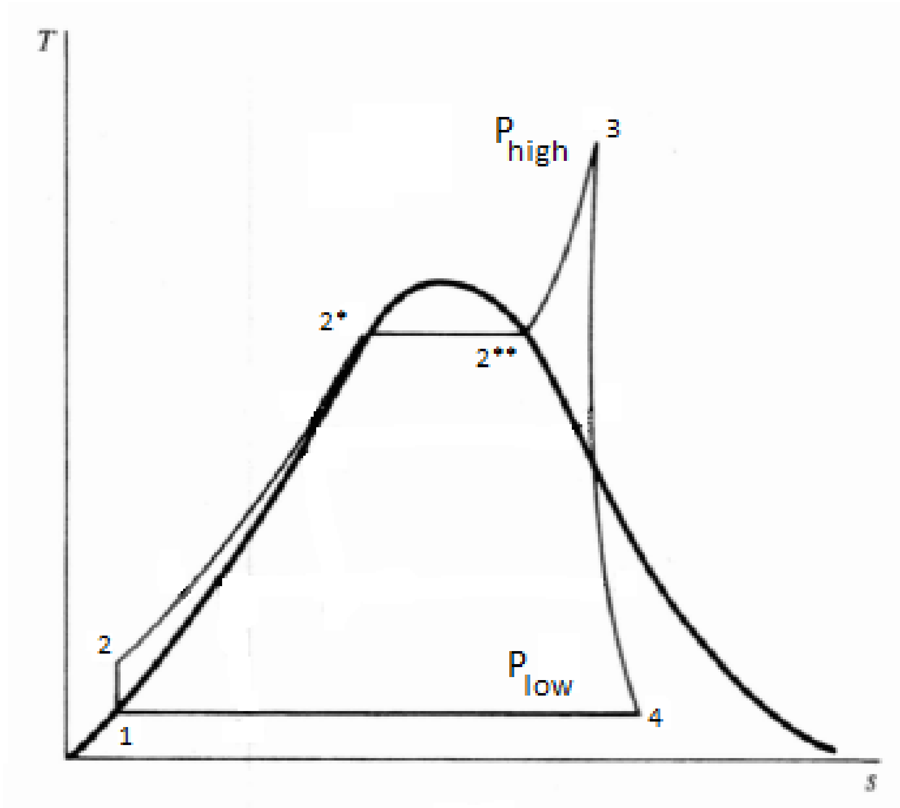

Figure 7.3: T-S-diagram of an real Rankine cycle

As seen in figure 7.3, the positions of points 2 and 4 in the diagram changed, due to the components which now no longer work ideally. The displacement of the points can be determined by means of the efficiency of the feedwater pump and steam turbine and the tables for the thermodynamic behavior of the working medium. The relationship between ideal and real work of the components is described in the following equations 7.3 and 7.2.

$$w_{tur} = w_{tur}^{rev} \cdot \eta_{tur} \tag{7.2}$$

$$w_{pum} = \frac{w_{pum}^{rev}}{\eta_{pum}} \tag{7.3}$$

It can be clearly seen from the equations that the feedwater pump requires more work in non-ideal behavior to bring the working fluid to the desired pressure and the steam turbine produces less work for the cycle in non-ideal behavior than in ideal behavior. The reason for this is that the losses during operation of the components are expressed as heat losses, friction, and in refluxes of the fluid via the blades in the feedwater pump and the steam turbine. The heat and the fluid refluxes have an influence on the working medium and change the thermodynamic properties of the working medium. Among other things, the heat increases the specific volume of the fluid and the refluxes increase the mass of the working medium. The feed water pump must now compress a higher volume resulting in a higher workload and thus a lower efficiency. For turbines on the other hand more volume is available for expansion due to the higher specific volume and thus more work can be generated. As a result of these processes, the surface area of the area enclosed by the working curve of the cyclic process and thus also the work won and efficiency change. All this can be seen from Figure 7.2 and in Figure 7.3 [22][23][24][26].

7.3 Optimization

The rankine cycle can be improved by various operational and design steps. These steps can not only gain more work but also increase the efficiency of the entire cycle. One of the operational steps is the increase in steam parameters. So the increase of temperature and pressure at the highest point of the cycle. Constructive steps include reheating and regenerative feedwater preheating [23][24].

7.3.1 Reheating

The maximum temperature increase is limited by the thermal capacity of the components, thus an increase in efficiency by increasing the steam parameters is limited. Reheating can circumvent this fact. In this process the steam is first expanded in a high pressure turbine to an intermediate pressure and then redirected to the steam generator. There the steam is overheated again and is then expanded in a low pressure turbine to the condensation pressure. Due to the reheating, the average temperature of the heat supply increases and thereby raises the temperature level of the steam in the steam generator. The energy loss during heat transfer from the heat generator to the heat engine is reduced. The effects of reheat are shown in Figure 7.4 in a T-S diagram. The reheat increases the overall efficiency of the cycle by about 10 % [23][24].

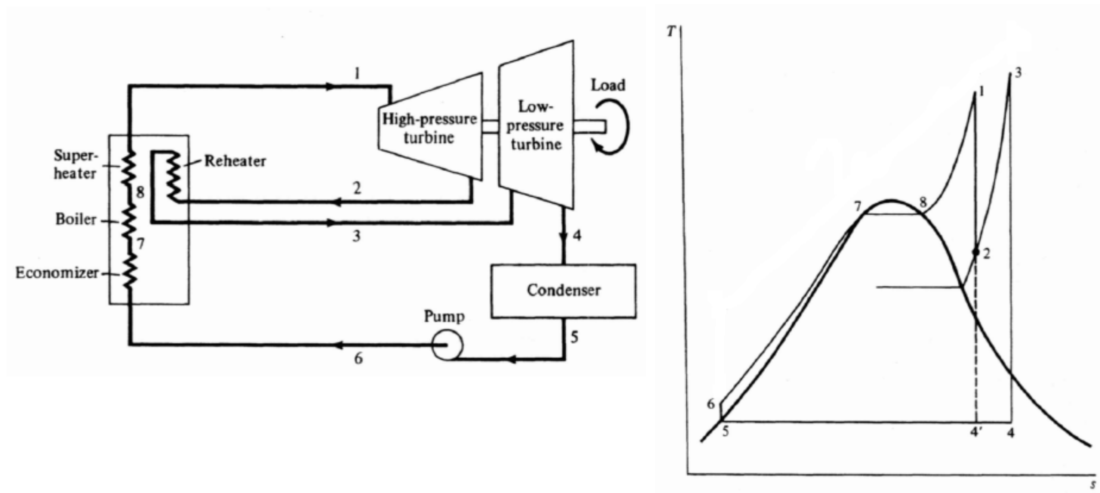

Figure 7.4: T-S-diagram of an Rankine cycle with Reheating

7.3.2 Regenerative Feedwater Preheating

Another way to increase the efficiency results from the preheating of the feedwater. In this process, a portion of the steam is removed from the turbine and used to preheat the liquid working medium. In the turbine enters a certain amount of steam which is expanded from a high pressure to an intermediate pressure, the so-called extraction pressure. Now, part of the steam is taken out of the turbine and fed to the feedwater preheater, while the remaining steam is expanded to the condensation pressure. The extracted steam transfers its heat energy to the liquid working medium and thus increases its temperature. The extracted steam condenses in the preheater and is then mixed with working fluid, which has been liquefied by the condenser. The feedwater preheat increases the temperature level of the steam in the steam generator and heat transfer losses are reduced. The effects of a regenerative feedwater preheater are shown in Figure 7.5 in a T-S diagram [23][24].

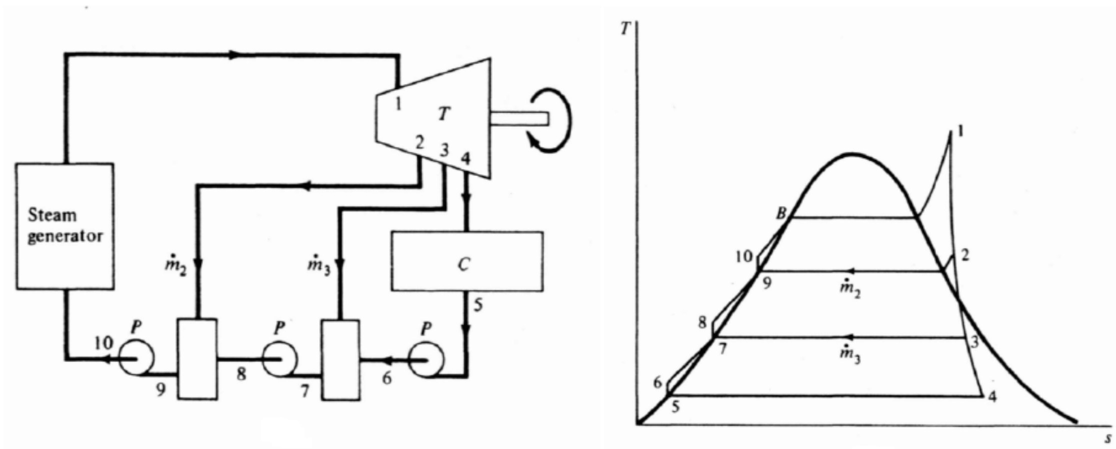

Figure 7.5: T-S-diagram of an Rankine cycle regenerative Feedwater preheating

7.4 Nuclear Cycle

In the nuclear energy cycle, the heat generator is the core in which uranium is fissioned and thus energy is released in the form of heat. This heat is transported to the heat engine via the primary cooling circuit. The circuits are connected to each other via the steam generator in which the heat transfer from the heat generator to the heat engine takes place. Since this heat engine cycle is in a steady state during operation of the nuclear power plant it can be assumed to be thermodynamic and not thermal-hydraulic as is the case with the primary cycle. Since nuclear power plants operate in comparison to other steam power plants only at a low temperature, the efficiency for this type of energy production is limited. The cycle of the heat engine must be adapted to these conditions, as high pressures and high temperatures are not possible. Superheating of the steam is limited, as a result of which the expansion of the steam in the steam turbine proceeds almost completely in the vapor curve. The great advantage of nuclear energy compared to other energy sources is the high energy density of uranium in the core. This surpasses fossil fuels by several orders of magnitude. For example, the energy density of hard coal is 34 MJ/kg as compared to the energy density of Uranium-235 at $79.39 \cdot 10^6$ MJ/kg. A disadvantage of the use of nuclear energy are the radioactive fission products.

Exposure to radiation must be reduced by very complex safety systems. In the event of an accident, no radioactive substances or radiation are released into the environment. Consequently, a nuclear power plant emits far less radioactivity than for example a coal power plant as coal contains natural radioactive isotopes that enter the environment from combustion. In addition a nuclear power plant produces no greenhouse gases including CO_2, SO_2 or N_2O [10][22][23][24].

8 Programming in RELAP 5

RELAP is an abbreviation for Reactor Excursion and Leak Analysis Program and is usually used for the simulation of the thermal-hydraulic behavior of the reactor coolant system and the core for various operational transients and postulated accidents that might occur in a nuclear reactor.

RELAP uses so called "cards" for the simulation of the models. Each card contains a special order: what the program has to do or under what conditions the program should run. For example, the card with the number 0000100 contains the name of the project and what kind of problem (steady state or transient) is simulated. With the help of these cards, every component of the system is modeled. Each new card implements a new part of the system. For example: a pipe, a pump, or other components for a thermal-hydraulic simulation. There are several cards, all with a different orders of how the new component is modeled, and under what conditions.

The modeling of a new componet starts with the card XXX0000. This card implements the name of the new component and what kind of component is modeled.

The XXX stands for the component number which is a variable, but a system of numbers would be an advantage. A system of numbers is especially important for very large systems, because otherwise the overview of the simulated system is very quickly lost. After naming the component, it is split into a chosen number of cells for a more exact modelling. Each cell contains a length and volume which is also chosen with a special card of RELAP. There are several other options for the cells, for example: height, angle, and roughness which are also very important for complex simulations. The most important thing about modeling components is that the component number and command number on the cards match. Otherwise an already finished component can get a new boundary condition, which with RELAP cannot work. With numbers for XXXYYYY, every card works. While XXX, as already said stands for the component number, YYYY is the command number for every simulated cell of the component. In addition to that, the temperature, the pressure, and the boundary conditions of the fluid inside each cell have to be chosen. Also, for this RELAP has special cards which have to be used. All modeled components can be connected to each other at so called "junctions". These also have to be modeled with RELAP cards like the modeled components. With the help of the junctions, it is possible for RELAP to simulate very complex systems of components. For example, a light water reactor, SMRs or any other thermal-hydraulic system, for very simple problems, like the "Edward's Pipe Problem", single junctions are not necessary, because it deals with only one componnent, which is not conected to anything else. A junction can be used to create a hole in the last cell of pipeline or any other component of the system. For example, the junction is modeled as a valve that can be opened and closed.

The following figure shows the beginning section of a RELAP code. The code shows how the cards are called and how certain commands are executed. It will only be clarified here how RELAP 5 is programmed [27][28][29][30][31].

```
*-------------------------------------------------------------------
*crdno          problem type       option
0000100              new          transnt    <= the beginning of the
*-------------------------------------------------------------------
                                              line is the card number
*crdno          inp-chk/run option
0000101               run
*-------------------------------------------------------------------
*crdno          input units        output units   <=  choosing the units
0000102            british            british            (here british)
*-------------------------------------------------------------------
*crdno    time 1     time 2
0000105   20.0       30.0
*-------------------------------------------------------------------
*crdno    noncond 1    noncond 2    noncond 3    noncond 4    noncond 5
0000110   nitrogen<= choosing of the not condensable substances
*-------------------------------------------------------------------
*crdno end time  min dt   max dt  ssd00    minor ed  major ed  restart
0000201  200.0  1.0e-6   0.05    0003      40      2000      2000
*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*
```

Figure 8.1: Scheme of RELAP 5 simulation. [31]

After modelling the components for the chosen system or the chosen problem the code has to be input in RELAP. For starting RELAP, a bat-data was written to execute the RELAP5 program with the right input data. This input data is the written code. For example the "Pipe blowdown Problem" . Before the start of the simulation, RELAP checks the code for mistakes, and will not start until every line is correct. If the code is correct, RELAP will start the simulation. It starts with calculating the steady state for the chosen system, or the chosen problem, until everything is in an equilibrium. After that, RELAP runs the code under the given boundaries and parameters. After RELAP has finished its simulations, it saves the data in three files. One file shows how RELAP has done the simulation, and also contains all simulation errors. The second file is the plot file, which contains all measurement data. This file has to be opened by a special software, called "AptPlot" . This software is able to read the binary file and can create plots for the evaluation. Although, the plot was made in another software, called "Origin" , because this program is specialized to create very complex and exact plots [27][28][29][30][31].

9 Thermal-hydraulics in RELAP 5

RELAP 5 uses mathematical models to analyze operational and accident scenarios that can occur in a nuclear fission-based power plant. With RELAP 5, it is possible to describe the physical processes that take place in a nuclear power plant and to analyze and evaluate them at the end. As a model, RELAP 5 uses a set of partial differential equations that can describe and predict certain phenomena in a range of applicability. The prediction capability of the models can be considered as a compelling criterion, because the set of partial differential equations can only describe and evaluate certain scenarios.

The general solution of this set of partial differential equations is very complex and very difficult to solve. An analytic solution of the nonliniear partial differential equations is generally not possible. The most commonly used approach to solving partial differential equations is discretization, thus solving a discretized version of the system.

RELAP 5 simulates the thermal-hydraulic behavior of water and steam in the power plant cycles for the transport of energy generated in the reactor. Despite the benefits of the discrete solution of these systems, some problems with this type of solution must be considered. The solution of discretized equations is subject to errors that must be monitored and solution steps controlled to reduce. Moreover, the interpretation of the results is a more notable problem. It should be noted that RELAP 5 is a mathematical model that is based on physics models to simulate nuclear power plants. As a starting point for the thermal-hydraulic calculations of the systems simulated in RELAP 5, the following space-time-dependent partial differential equations are considered.

- Conservation of Mass
- Conservation of Momentum
- Conservation of Energy

The nomenclature used hereinafter can be found in the appendices in chapter 15 [32][33][34][35].

9.1 Conservation of Mass

$$\frac{\partial}{\partial t}(\alpha_k A \rho_k) + \frac{\partial}{\partial x}(\alpha_k A \rho_k \upsilon_k) = A\Gamma_k \tag{9.1}$$

9.2 Conservation of Momentum

$$\begin{aligned}\frac{\partial}{\partial t}(\alpha_k A \rho_k \upsilon_k) + \frac{\partial}{\partial x}(\alpha_k A \rho_k \upsilon_k^2) = -\alpha_k A \frac{\partial P}{\partial x} + \alpha_k \rho_k B_x A \\ - \alpha_k \rho_k A\ FW_k \upsilon_k + A\Gamma_k \upsilon_i - \alpha_k \rho_k A\ FI_k(\upsilon_k + \upsilon_{k'}) \\ - C\alpha_k \alpha_{k'} \rho_m A \left[\frac{\partial}{\partial t}(\upsilon_k + \upsilon_{k'}) + \upsilon_{k'}\frac{\partial \upsilon_k}{\partial x} - \upsilon_k \frac{\partial \upsilon_{k'}}{\partial x}\right]\end{aligned} \tag{9.2}$$

9.3 Conservation of Energy

$$\frac{\partial}{\partial t}(\alpha_k A \rho_k u_k) + \frac{\partial}{\partial x}(\alpha_k A \rho_k u_k \upsilon_k) = -AP\frac{\partial \alpha_k}{\partial t} - P\frac{\partial}{\partial x}(\alpha_k A \upsilon_k)$$
$$+ AQ_{wk} + AQ_{ik} + A\Gamma_{ik} h_k^* + A\Gamma_{wk} h_k' + A\ DISS_k \tag{9.3}$$

The model represented by these partial differential equations is called a two-fluid model. In this model, each phase (liquid or gaseous) is considered separately and for each phase one of the conservation equations can be established. The model evaluates, compared to other models, thermal-hydraulic non-equilibria between the different phases by its basic equations. As a result, the two-fluid model is precise and can easily solve and evaluate thermal-hydraulic problems very precisely. In general, for the two-fluid model, it can be said that each phase has its own velocity, temperature and pressure. While the different velocities of the phases are caused by density differences, temperature differences between the phases are generally caused by the time delay of the energy transfer between the phases. The pressure difference of the different phases of the two-fluid model is generated by three effects:

- pressure differences due to surface energy of a curved interface

- pressure differences due to mass transfer
- pressure differences due to dynamic effects

The first effect is generally caused by the phases, since the simple existence of an interface results in a pressure difference from the total mechanical equilibrium between these phases. The second effect is caused by large mass flows between the phases due to phase change (high evaporation or condensation) at the interface. Due to the dynamics in which phase A has a greater pressure than phase B due to very rapid energy deposition or pressurization effects, the third effect is finally produced.

In most cases, however, it can be assumed that the pressure in the phases is the same and thus there are no pressure differences. RELAP 5 takes this case in its calculations. From RELAP 5 manual volume 1 "the phasic pressures are assumed equal" [27]. Thus it is assumed:

$$P = P_f = P_g \tag{9.4}$$

A counter example, in which pressure differences are important is in research on explosive boiling. In these cases, a local momentum equation (e.g. Rayleigh momentum equation) is used to model the pressure difference of the two phases. Furthermore, it can be seen from the equations 9.1, 9.2 and 9.3 that the phases of the model are connected with each other via different parameters. Two of these values are Γ_k and Γ_k'. Γ_k and Γ_k' represent the mass transfer rate due to the phase change and the interaction force between the different phases. It can be assumed:

$$\Gamma_k = -\Gamma_{k'} \tag{9.5}$$

The model provides for dividing the mass transfer rate Γ_k into two different areas. On the one hand, the mass transfer rate at the interface between vapor and liquid in the mass of the fluid Γ_{ik} is considered. On the other hand, the mass transfer rate at the interface between vapor and liquid near the walls Γ_{wk} is considered.

$$\Gamma_k = \Gamma_{ik} + \Gamma_{wk} \tag{9.6}$$

Figure 9.1 shows the scheme of the model in terms of mass transfer rate Γ_k.

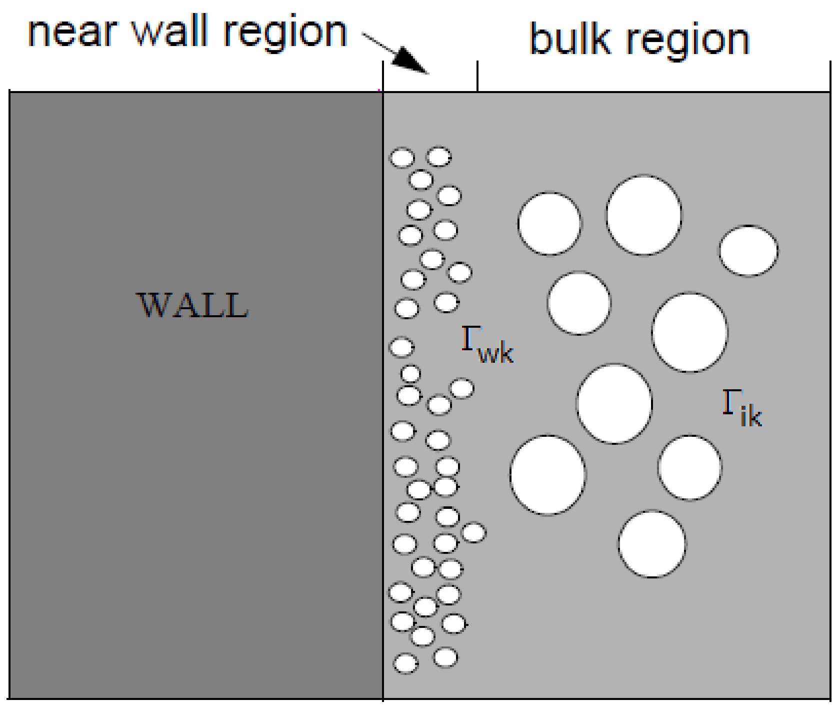

Figure 9.1: Mass transfer rate in RELAP 5 [27]

The expression of the momentum equation in RELAP 5 is reduced to Bernoulli equations for a steady, incompressible and frictionless flow. The reason for this is that impulse effects in the reactor safety analysis are regarded as secondary compared to mass and energy conservation. It follows that a less accurate formulation (compared to mass and energy conservation) is acceptable, especially since there are large momentum sources and sinks (pumps or turbines) in nuclear reactors. The reason for using the momentum equation in extended form is that it has made the development of the RELAP 5 numerical scheme easier. The force terms on the right sides of the equation 9.2 are:

- $\alpha_k A \frac{\partial P}{\partial x}$ = pressure gradient
- $\alpha_k \rho_k B_x A$ = body force
- $\alpha_k \rho_k A\ FW_k \upsilon_k$ = wall friction
- $A \Gamma_k \upsilon_i$ = momentum transfer due to interface mass transfer
- $\alpha_k \rho_k A\ FI_k(\upsilon_k + \upsilon_{k'})$ = interface frictional drag
- $C \alpha_k \alpha_{k'} \rho_m A \left[\frac{\partial}{\partial t}(\upsilon_k + \upsilon_{k'}) + \upsilon_{k'} \frac{\partial \upsilon_k}{\partial t} - \upsilon_k \frac{\partial \upsilon_{k'}}{\partial t} \right]$ = force due to virtual mass

The term FW_k belongs to the frictional resistance of the walls through which the fluid flows and is a product of different friction coefficients. The coefficient FI_k is part of the frictional resistance of the interface between the vapor and the liquid phase of the fluid. For the calculation of FI_k two different models are used (drift flux and drag coefficient), depending on whether the flow is a laminar or a turbulent flow.

The listed virtual mass term is simplified in the RELAP 5 encoding. It follows that the spatial derivative part of the expression is neglected in the calculations. The reason for this simplification is that numerical computation of the spatial derivative part of the expression can lead to inaccuracies, which can then lead to nonphysical properties in the numerical solution. The coefficient of the virtual mass in turn depends on the type of flow and is assumed in RELAP 5 as C>0,5. A C-value greater than 0,5 is suitable for bubbling or dispersed flows, unlike separate or stratified flows where a C-value of 0 is appropriate. The primary effect of the virtual mass term is the mixing sound velocity of the liquid-vapor fluid. Thus, it is acceptable that in the RELAP 5 model the simplified form is used . As seen in the next equation, 9.7, the conservation of momentum at the transition between phases requires that the values that participate in the momentum exchange add up to zero.

$$\begin{aligned}&\Gamma_k A \upsilon_{ik} - (\alpha_k \rho_k A)\, FI_k(\upsilon_k - \upsilon_{k'}) - C\alpha_k \alpha_{k'} \rho_m A \left[\frac{\partial}{\partial t}(\upsilon_k - \upsilon_{k'})\right] \\ &- \Gamma_{k'} A \upsilon_{ik'} - (\alpha_{k'} \rho_{k'} A)\, FI_{k'}(\upsilon_{k'} - \upsilon_k) - C\alpha_{k'} \alpha_k \rho_m A \left[\frac{\partial}{\partial t}(\upsilon_{k'} - \upsilon_k)\right] = 0\end{aligned} \tag{9.7}$$

The equation shows that all spatial derivatives of the momentum conservation have been removed, as described above. This special interface momentum balance results from the conservation of momentum for the different phases in equation 9.2.

In addition, for equation 9.7, it is assumed that the interfacial impulse transfer due to friction and due to the mass transfer add independently to zero.

The energy conservation equation represents the temporal and spatial evolution of energy in the simulated two phase system of RELAP 5. In the equation, the following simplifications are surpassed:

- Reynolds heat flux is neglected
- interfaces energy storage is neglected
- internal phasic heat transfer is neglected

In energy conservation as well, as in mass transfer, a differentiation is made between energy transfer near the wall and energy transfer in the fluid. The values Q_{wk} and $Q_{wk'}$ are heat transfer rates of the phases near the walls of the system. Q_w represents the total heat transfer near the walls of the system, as shown in equation 9.8.

$$Q_w = Q_{wk} + Q_{wk'} \tag{9.8}$$

The specific enthalpies h_k^* and $h_{k'}^*$ are related to the mass transfer between the different phases in the fluid region and the specific enthalpies h_k' and $h_{k'}'$ are in turn related to the mass transfer between the different phases around the walls of the system. Thus it is again distinguished whether the mass transfer and the resulting energy transfer occurs near the walls or in the fluid. As already discussed, the phase transition of the fluid consists of two parts. On the one hand the phase transition in the area of the fluid (Γ_{ik}) and on the other hand the phase transition in the area of the walls (Γ_{wk}) of the system. Each phase change also includes heat transfer effects, which can also be differentiated, depending on whether the heat transfer takes place in the fluid (Q_{ik}) or near the walls (Q_{wk}). For the energy exchange between the phases due to heat or mass transfer, it is necessary that this results in a total of zero, as clearly shown in equation 9.9.

$$Q_{ik} + Q_{ik'} + \Gamma_{ik}(h_k^* - h_{k'}^*) + \Gamma_{wk}(h_k' - h_{k'}') = 0 \tag{9.9}$$

The values for the heat transfer between the phases $(Q_{ik}\ \&\ Q_{ik'})$ can also be divided into two parts. On the one hand in the heat transfer $(Q_{ik}^B\ \&\ Q_{ik'}^B)$ in the fluid and on the other hand in the heat transfer near the walls $(Q_{ik}^W\ \&\ Q_{ik'}^W)$. The interphase heat transfer can be seen as shown in Figure 9.2. In addition, the total heat transfer between the phases can be expressed by equation 9.10.

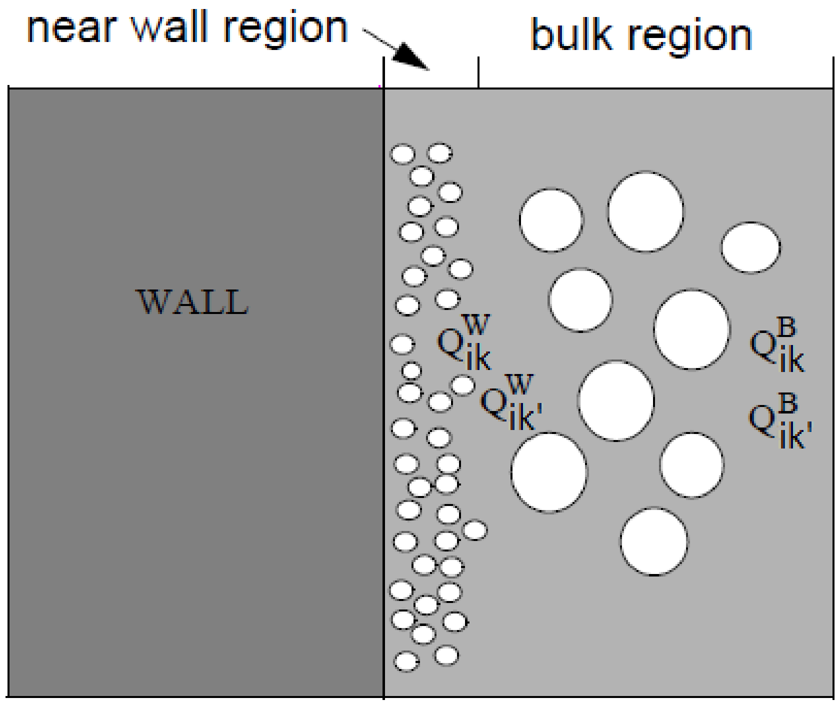

Figure 9.2: Heat transfer in RELAP 5 [27]

$$Q_{ik} = Q_{ik}^{B} + Q_{ik}^{W} \tag{9.10}$$

The heat transfer between the phases in the fluid occurs at the interface between the phases. The heat transfer takes place at the saturation temperature T_S and the total pressure P. In the equation for the interface heat transfer H_{ik} represents the interface heat transfer coefficient and T_k is the temperature of the phase.

$$Q_{ik}^{B} = H_{ik}(T_s - T_k) \tag{9.11}$$

Furthermore, it is assumed that the values of equation 9.9 on wall and between phases sum independently to zero. This can be seen from the next equations 9.12 and 9.13.

$$H_{ik}(T_s - T_k) + H_{ik'}(T_s - T_{k'}) + \Gamma_{ik}(h_k^* - h_{k'}^*) = 0 \tag{9.12}$$

$$Q_{ik}^{W} + Q_{ik'}^{W} + \Gamma_{wk}(h_k' - h_{k'}') = 0 \tag{9.13}$$

In order to determine the mass transfer rate near the walls, Q_{ik}^{W} = 0 is assumed for phase transitions from phase k to phase k' and $Q_{ik'}^{W}$ = 0 for phase transitions from phase k' to phase k. With the equation 9.13 and these assumptions, the mass transfer rates for the phases on the wall are:

$$\Gamma_{wk} = \frac{-Q_{ik}^{W}}{(h_k' - h_{k'}')} \tag{9.14}$$

Through this new expression in equation 9.15 and equation 9.11, the equation 9.10 can be expressed in a new general form:

$$Q_{ik} = H_{ik}(T_s - T_k) - \frac{\Gamma_{wk}}{(h_k' - h_{k'}')} \tag{9.15}$$

The last value of the energy conservation equation is the phasic energy dissipation term $DISS_k$. $DISS_k$ represents the sum of the energy dissipation effects through pumps, turbines or wall friction in the system. Equation 9.16 is shows how the phasic energy dissipation term expresses for wall friction losses. Dissipation effects due to mass transfer between the phases, friction between the phases, or virtual mass are neglected because these are very small in the energy conservation equation.

$$DISS_k = \alpha_k \rho_k FW v_k^2 \tag{9.16}$$

The losses of the different phases can be summed up again to a total value, as can be seen in equation 9.17.

$$DISS = DISS_k + DISS_{k'} \tag{9.17}$$

As can be seen from these equations and the made assumptions, the description of the model that RELAP 5 uses is very difficult, and it is very clear that the phases and the system are interconnected and interact with each other through these partial differential equations [27][32][33][34][35].

10 Natural Circulation

In conventional nuclear power plants, the circulation of the coolant of the primary cycle and the secondary cycle is operated by reactor coolant pumps or by feedwater pumps. The NuScale SMR uses the natural circulation of water in its primary cycle to dissipate its core heat. The natural circulation is generated exclusively by the arrangement of the core (heat source) and the steam generator (heat sink). Natural circulation, in comparison to the principles of conventional circulation methods, has the advantage that no external energy is needed to maintain the cycle. This is a particularly strong safety aspect of a NuScale small modular reactor. This chapter deals with the physical principles of natural circulation [19][36][37][38].

10.1 Physical Principle

The driving force of natural circulation is the density difference of the fluid present in the system. These density differences in the fluid are produced by heat removal on one side and heat supply on the other side of the circuit. The density differences result in a pressure difference in the system, which is also referred to as the driving pressure. In the following equation 10.1, the pressure difference generated by the different densities is shown.

$$\Delta p = h \cdot g \cdot (\rho_2 - \rho_1) \tag{10.1}$$

As can be seen very clearly from the equation, the pressure difference depends only on the height to be overcome, as well as the temperature at heat source and heat sink, which leads to the density difference in fluid. By this pressure difference, the heated fluid flows from the heat source to the heat sink. When the natural circulation comes to a closed circuit, there is talk of gravity circulation. With gravity circulation, it should be noted that the heat source must be located lower than the heat sink in the circuit. This is one of the prerequisites that the cycle needs to work. As the fluid enters the heat source its density decreases. As a result, it rises by static buoyancy in a gravitational field up to the heat sink. At the heat sink heat is removed from the fluid, which leads to a decrease in temperature and thus to higher density. The denser fluid now sinks from the heat sink to the heat source and the cycle begins again. Figure 10.1 illustrates again the mode of operation of the natural circulation.

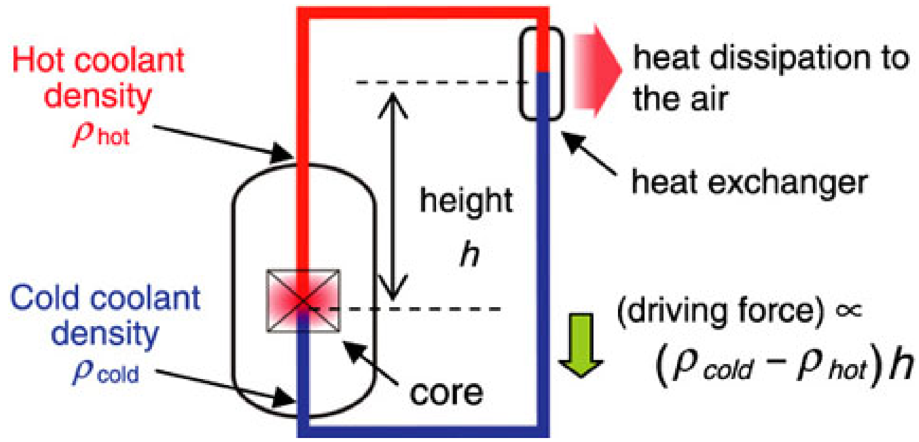

Figure 10.1: Operation of Natural Circulation [39]

It is both possible to create a naturally circulating circuit in which the fluid undergoes a phase change, as well as a cycle without phase change. In addition to that, natural circulating circuits can be generated with different materials (e.g. water, liquid metal or gas). Therefore, natural circulation is used in various applications, for example: solar water heaters or furnaces [40][41][42].

10.2 Application in the SMR

As an application example, a small modular reactor with natural water circulation has been selected. In this reactor, the cooling of the core is guaranteed exclusively by natural circulation, that means no pumps are present in the primary circuit. Suppose the circulation starts at the core of the SMR. The fluid, in this case water, that is under high pressure, is heated by the core. The heat is transferred from the fuel rods to the coolant water. Due to the increase of the temperature the density of the fluid decreases and because it is under a high pressure, no phase change is caused. Due to the lower density, the water experiences buoyancy in the primary cycle of the SMR and rises in the direction of the steam generator. At the steam generator, the primary cycle fluid releases its thermal energy to the secondary cycle fluid, which then undergoes a phase change. By exchanging the energy of the cycles, the temperature of water in primary cycle decreases again, and this results in a density increase of the fluid. With the help of gravity the denser water sinks again towards the reactor core where the procedure can start anew. This entire cycle runs completely without input of external energy and is kept running only by the arrangement of the core and the steam generator [19][36][37][38].

11 RELAP 5 Model of the NuScale SMR

This chapter discusses the exact structure and specific parameters to model the NuScale SMR in RELAP 5. All the important components and parameters needed for the operation of the NuScale SMR will be explained in this chapter. These include:

- Core
- Primary system parameters
- Turbine generator
- Steam generator

The exact tasks of the individual components have already been explained in the previous chapters. Thus, this chapter will only deal with the values and specific parameters of the components required by RELAP 5. The parameters in the following tables were taken from the Final Safety Analysis Report from NuScale. This report is an application to the Nuclear Regulatory Commission (NRC) for licensing of a nuclear power plant.

11.1 Core

Table 11.1: Design Data of the SMR core [43] [44] [45]

Parameter	Value (SI)	Value (Britsh)
Diameter of active core	1.505 m	4.94 ft
Height of active core	2.0 m	6.57 ft
Height-to-diameter ratio of active core	1.33	1.33
Core thermal output	160 MW_t	160 MW_t
System pressure	12.75 MPa	1850 psia
Inlet temperature	531.5 K	497°F
Core average temperature	557 K	543°F
Average temperature rise in core	55.44 K	100°F
Best estimate flow	587.15 kg/s	4.66E+6 lb/hr
Core bypass flow (best estimate)	7.33 %	7.33 %
Heat transfer area on fuel surface	583 m^2	6275.6 ft^2
Core flow area	0.91 m^2	9.79 ft^2
Core average coolant velocity	0.82 m/s	2.7 ft/s
Number of fuel assemblies	37	37
Rods per fuel assemblie	264	37

Table 11.2: Design Data of Rods and Assemblies [43] [44] [45]

Parameter	Value (SI)	Value (Britsh)
Fuel Assemblie		
Effective fuel length	243.56 cm	95.89 in
Fuel weight per assemblie	249.47 kg	550 lbm
Fuel assemblie pitch	21.5 cm	8.466 in
Fuel rod pitch	1.26 cm	0.496 in
Fuel Rod		
Fuel pellet diameter	0.81 cm	0.3195 in
Fuel colum length	200 cm	78.74 in
Cladding outside diameter	0.81 cm	0.3174 in

11.2 Primary System Parameters

Table 11.3: Design Data of the SMR primary system [43] [44] [45]

Reactor power		Primary Flow		Primary Coolant Temperature			
%	Power	%	best estimate	Core ΔT	T_{cold}	T_{avg}	T_{hot}
15	24 MW_t	47.7	280.2 kg/s	31.7°F	528.5°F	543.3°F	558.2°F
50	80 MW_t	75.6	443.7 kg/s	66.5°F	512.2°F	543.3°F	574.4°F
75	120 MW_t	88.9	521.6 kg/s	84.6°F	503.8°F	543.3°F	582.9°F
100	160 MW_t	100	587 kg/s	99.8°F	496.6°F	543.3°F	590.1°F

11.3 Turbine Generator

Table 11.4: Design Data of the SMR turbine generator [43] [44] [45]

Type: Single, 10-stage condensing		
Parameter	Value (SI)	Value (Britsh)
Operating speed	3600 1/min	3600 rpm
Phase/frequency	60 Hz	60 Hz
Estimated power	57.5 MVA	57.5 MVA
Power factor	0.85	0.85
Generator power output (electric)	50 MW_e	50 MW_e
Pipe diameter (sec. system)	30.48 cm	12 in

11.4 Primary System Geometries

Table 11.5: Geometry Data of the SMR primary system components [43] [44] [45]

Component	tot. Volume	Sub-Component	Ave. Flow Area	Length
Riser	635 ft^3	Lower riser	24.9 ft^2	9.4 ft
		Upper riser	15.4 ft^2	26.0 ft
Downcomer	1199 ft^3	DC and SG	25.7 ft^2	46.0 ft
Core	89 ft^3	Fuel assemblies	10.3 ft^2	7.9 ft
		Reflector	0.9ft^2	7.9 ft
Pressurizer	578 ft^3	Main steam plenum	36.1 ft^2	1.7 ft
		Cylin. pressurizer	61.4 ft^2	6.9 ft
		Pressure vessel head	41.2 ft^2	2.2 ft

Table 11.6: Volume Data of the SMR primary system components [43] [44] [45]

RCS Region	Volume
Hot Leg (lower riser, riser transition, upper riser, riser supports)	635.3 ft^3
Cold Leg [feedwater plenums, downcomer transition, downcomer (lower riser), core barrel, RPV bottom head, flow diverter]	578.1 ft^3
Core Region (fuel assembly region and reflector cooling channels)	88.74 ft^3
SG Region	620.7 ft^3
PZR Region (main steam plenums, PZR, RPV top head)	577.5 ft^3
PZR Region, cylindrical (main steam plenums and PZR)	487.3 ft^3

11.5 Steam Generator

Table 11.7: Design Data of the SMR steam generator [43] [44] [45]

Type: Helical, once-through		
Parameter	Value (SI)	Value (Britsh)
Total heat transfer	159,13 MW_t	159,13 MW_t
Total number of helical tubes per NPM	1380	1380
Number of helical tube columns per NPM	21	21
Internal pressure - secondary	14.5MPa	2100 psia
External pressure - primary	14.5MPa	2100 psia
Internal temperature - secondary	616.5 K	650°F
External temperature - primary	616.5 K	650°F
Steam flow (full power)	67 kg/s	532100 lbm/hr
Tube wall outer diameter	15.875 mm	0.625 inches
Tube wall thickness	1.27 mm	0.050 inches
Total heat transfer area	1665.57 m^2	17928 ft^2
Normal steam pressure	3.45 MPa	500 psia
Normal steam temperature	574.8 K	575°F
Design pressure upstream / downstream	14.5MPa / 6.9 MPa	2100 psia / 1000 psia
Normal feedwater temperature	422 K	300°F

11.6 Development of the Model

With these geometric and thermodynamic values from the tables, it is now possible to develop a numerical model of the NuScale small modular reactor. For the later simulation of the model, as already mentioned RELAP 5 is used. The modeled components in RELAP 5 can be divided into three different categories [27][28][29][30][31].

- hydrodynamic components
- heat structures
- junctions

Most of the components used in RELAP 5 are hydrodynamic components. These simulate all parameters used in the two-fluid model and show how a reactor would behave under certain boundary conditions. Generally, all hydrodynamic components are simulated with a combination of single-volumes and junctions. In RELAP 5 various hydrodynamic components are preprogrammed and can be called and used very easily. In this chapter some of these preprogrammed components will be discussed. However, only components that were used in the developed model are discussed. The hydrodynamic components of the model used in RELAP 5 are:

- Branch

- Pipe
- Annulus
- Time-Dependent Volume

11.6.1 Branch

The branch component is a single volume that can be modeled with up to nine junctions. It may be used, on the one hand, for the branching of piping systems and, on the other hand, as an upper or lower plenum in a reactor vessel. In general, it can be said that it is responsible for diverting and distributing a mass flow in the system if several mass flows coincide at one point. The branch component was used as steam generator core and upper/lower plenum in the designed model. The input text at the end of this section shows a modeled branch component. As already described RELAP 5 uses so-called cards to model the different components.

The first card to model a branch is the card CCC0000. The CCC stands for the component number and can be chosen variably. The card requires two inputs. First, the name of the component, which is also variable, and the word branch, which defines the component type. The card CCC0001 defines the number of junctions and the unit of the initial conditions for these junctions. To select fluid velocities, a zero must be entered and for mass flows, a one. The next card has the number

CCC010N. This defines the geometrical properties, such as area length and volume, the volume orientation and the hydrodynamic diameter of the branch. In this order, the different values must be entered into the card. The card CCC0200 defines the initial values for the thermodynamic properties of the fluid and also what kind of fluid it is. The control word ebt consists of three numbers. The first two numbers define what kind of fluid it is. To select water without boron, zero must be selected for the first two numbers (e and b). The third number defines which initial thermodynamic values this fluid has. To set initial values for pressure and temperature, three must be selected as the third number (t). There are other variations for this control word and thus it is possible that in addition to pressure and temperature other values can be put into the card. But if t=3 only two values are expected and every other value gives an error. Card CCC110N is about the junctions of the branch. The first value of this card defines where a certain mass flow comes from. The second value defines where a certain mass flow goes to. This is achieved by the value CCCXX000F. CCC determines the component number XX the volume number and F (1 inlet, 2 outlet) with which surface the junction is connected. The third value of this card defines the junction area. The last card of a branch is the card CCC120N. This card specifies the mass flows or fluid velocities of the fluid phases in junctions at the beginning of the simulation. The values for the liquid and vapour phases are defined seperately. The first value of the card establishes the initial condition of a liquid and the second value the initial condition of vapour. The third value of this card is always zero [28][31].

RELAP 5 Branch Input

```
*crdno component name component type

CCC0000 "Name" branch

*crdno number of junctions vel/flw

CCC0001 X  M

*crdno flowarea length volume

CCC0101 A  L  V

*crdno horz.orient vert.orient deltaZ

CCC0102 0.0  phi  H \newline

*crdno roughness hyd. diameter fe

CCC0103 r  D_h  00

*crdno vol.ic pressure temperature

CCC0200 ebt  p  T

*crdno from  to  area

CCC110N CCCXX000F CCCXX000F A_j

*crdno f.flowrate g.flowrate j.flowrate
```

```
CCC120N mf_L mf_V 0.0
```

11.6.2 Pipe/Annulus

Pipes and annulus are a combination of single-volume and single-junction components. These components are used to simulate piping systems in RELAP 5 models but they can also be modeled as the components of a reactor vessel riser or down comer. The input text at the end of this section shows a modeled pipe/annulus component.

The first card to model a pipe/annulus is again the card CCC0000. The CCC stands for the component number and can be chosen variably. The card requires two inputs. First, the name of the component, which is again also variable, and the word pipe/annulus, which defines the component type. The card CCC0001 defines the number of volumes or better said in how many part the pipe/annulus is divided. The cards CCC010N CCC030N CCC040N CCC060N and CCC070N define the area the length the volume the angle and the height of the respective partial volume of the pipe/annulus. If every partial volume has the same conditions the volume number which is the secound value of each of this cards equal to the number of pipe/annulus volumes. Card CCC080N determines the roughness and hydraulic diameter of each pipe/annulus volume. The next card is the card CCC090N. This card defines the flow energy loss coefficients for forward losses (word 1) and reverse losses. (word 2). The third word of this card is the junction number and shows in which junction the losses occur. The card CCC120N defines the initial values for pressure and temperature and also what kind of fluid it is. Again the control word ebt consists of three numbers. The first two numbers define what kind of fluid it is. To select water without boron, zero must be selected for the first two numbers (e and b). The third number defines which initial values this fluid has. To set initial values for pressure and temperature, three must be selected as the third number (t). After the value for the temperature there are three more values, which are not used in the model. Any value would lead to an error. Therefore these are set to zero. The last value determines the partial pipe/annulus volume. There are other variations for this control word and thus it is possible that in addition to pressure and temperature other values have to be put into the card. But if t=3 only two values are expected and every other value, except zero, gives an error. The card CCC1300 determines the unit of the initial conditions for the pipe/annulus junctions. To select fluid velocities, a zero must be entered and for mass flows, a one. The last card of a pipe/annulus is the card CCC130N. This card specifies the mass flows or fluid velocities of the fluid phases in junctions at the beginning of the simulation. The values for the liquid and vapour phases are defined seperately. The first value of the card establishes initial condition of a liquid and the second value the initial condition of vapour. The third value of this card is always zero. The last value of this card specifices the junction number. Note the number of junctions is one less than the number of volumes [28][31].

RELAP 5 Pipe/Annulus Input

```
*crdno component name component type

CCC0000 "Name" pipe

*crdno number of volumes

CCC0001 N

*crdno vol.area vol.no.

CCC0101 A  N

*crdno length vol.no.

CCC0301 L  N

*crdno volume vol.no.

CCC0401 V  N

*crdno vert.angle vol.no.

CCC0601 phi  N

*crdno elev.change vol.no.

CCC0701 H  N
```

```
*crdno roughness hyd.diameter vol.no.

CCC0801 r  D_h  N

*crdno floss rloss junNr

CCC0901 Fl  RL  JN

*crdno vol.ic pressure temperature vol.no.

CCC1201 ebt  T  p  N

*crdno vel/flw

CCC1300 1

*crdno f.flowrate g.flowrate j.flowrate jun.no.

CCC1301 mf_L mf_V 0.0 JN
```

11.6.3 Time-Dependent Volume

Time-dependent volumes are used wherever the fluid enters or leaves the simulated system. With these components it is possible to model volume-related boundary conditions on the system. Volume-related boundary conditions are for example pressure, temperature, void fraction, and quality. For example time-dependent volumes are used to simulate mass flows of a cycle or to simulate a pressurizer part on top of a reactor pressure vessel. The input text at the end of this section shows a modeled time-dependent volume component. Card CCC0000 it the first card which is used to model a time-dependent volume. The CCC stands again for the component number and can be chosen variablyand the card requires two inputs. First, the name of the component, which is again variable, and the word tmdpvol, which defines the component type. The next card has the number CCC010N. This card defines the geometrical properties, such as area length and volume, the volume orientation and the hydrodynamic diameter of the time-dependent volume. In this order, the different values must be entered into the card. The card CCC0200 defines the initial values for the thermodynamic properties of the fluid and also what kind of fluid it is. The control word ebt consists of three numbers. The first two numbers define what kind of fluid it is. To select water without boron, zero must be selected for the first two numbers (e and b). The third number defines which initial thermodynamic values this fluid has. To set initial values for pressure and temperature, three must be selected as the third number (t). There are other variations for this control word and thus it is possible that in addition to pressure and temperature other values can be put into the card CCC020N. But if t=3, only two values are expected and every other value gives an error. The card CCC020N defines the thermodynamic values of the time-dependent volume depending on a control value (tdigit). The control value can be chosen in card CCC0200 but usually time is chosen as the value. Now for the simulation in RELAP 5 the time-dependent volume changes its conditions depending on the control value. For example pressure and temperature are depending on time in a time-dependent volume [28][31].

RELAP 5 Time-Dependent Volume Input

```
*hydro component name component type

CCC0000 "Name" tmdpvol

*hydro area length volume

CCC0101 A  L  V

*hydro horz.angle vert.angle deltaZ

CCC0102 0.0  phi  H

*hydron roughness hyd.diam fe

CCC0103 r  D_h  0.0

*hydro ic

CCC0200 ebt

*hydro tdigit pressure temperature

CCC0201 t  T  p
```

11.6.4 Heat Structure

In RELAP 5, heat structures are considered to be the fixed components of the thermal hydrodynamic system. Metal structures such as vessel walls, steam generator tubes, fuel rods and reactor vessel installations are simulated with these components. Each heat structure is simulated with a left and a right side. According to the convention, it is only possible that each side of a heat structure is connected to at most one hydrodynamic volume. However, it is possible that more than one heat structure may be associated with the same hydrodynamic volume. The overall system response depends on the heat transferred between the heat structures and the liquid. In addition, the temperature distributions in the heat structures are often important simulation requirements. The input text at the end of this section shows an example for a modeled heat structure. The first card of a heat structure component is the card 1CCCG000. The CCC stands for the heat structure number. This number is variable. The G stands for the geometry number which distinguishes between different heat structures which are connected to the same volume. The first value of this card determinates the number of axial heat structures. The second value is the number of radial mesh points for this heat structure. The third value inputs the geometry type of the heat structure. Enter 1 or -1 for rectangular type , 2 or -2 for cylindrical type, or 3 or -3 for spherical type. The sign of the number tells the program if the heat structure is coupled to its hydrodynamic component or not. Enter a positive number for coupled and a negative number for decoupled. The fourth value of this card is the steady-state initialization flag. Use zero if the initial condition temperatures are entered on input Cards 1CCCG40N or one if the code calculates the initial condition temperatures itself. The last value of the card is the position of the left boundary coordinate of the heat structure. The next card is the card 1CCCG100. This card enters the formats of the mesh point locations and how the intervals are specified. The card 1CCCG10N specifies how many intervals there are between the left and right boundary condition are and also which position the right boundary condition has. In the card 1CCCG20N, the material parameters for heat structure intervals are determinated. It is possible to simulate a heat structure which is a combination of different layers of materials. The material values have to be defined in tables at the end of the code. The first value of this map is the material parameter and the second value defines how many mesh intervals are made of this material. The card 1CCCG30N specifies the source value for every interval of the heat structure. The next card is card 1CCCG40N. The first value of this card defines the initial temperature of the mesh points of the heat structure and the second value defines the mesh points. The next cards connect the heat structure to the hydrodynamic volumes. Card 1CCCG501 is the left connection and card 1CCCG601 is the connection to the right hydrodynamic volume of the heat structure. The first value of these card is the connection to the volume with the word CCCXX0000. The CCC is the component number and the XX the volume number. The next value is the increment word. This value is used if the next heat structure parts are connected to the next volumes of the hydrodynamic component. For one dimensional volumes the increment value is 10000 or -10000 depending from which side of the hydrodynamic component the volumes are counted. The third value defines the boundary condition type. The fourth and fifth values are used together. If the fourth value is zero the fifth value is the area of the left or right side of the heat structure. The last value is the heat structure number. The card 1CCCG701 determines whether there is a source term in the heat structure or not. The first value is the source type term. A source term is necessary if for example a fuel rod is simulated because in the rod heat is generated due to nuclear fission. If the heat structure simulates a steam generator or a vessel wall no source term is necessary and the value is zero. If the heat structure has a source, the first value is a special table which determinates how much energy over time is generated from the heat structure. The second value of the card is the multiplication factor for every heat structure part. If the generated energy in the whole heat structure has an heterogeneous distribution, with this value it is possible to simulate it. The third and fourth value define which sides (left and right) are heated by the source term. These are usually zero. The last value is the heat structure number. The last cards of the heat structure are the cards 1CCCG800, 1CCCG900, 1CCCG801 and 1CCCG901. These cards define the boundary conditions for the two sides of the heat structure. The cards 1CCCG800 and 1CCCG900 determine the complexity of the boundary conditions for each side of the heat structure. Both cards have a control value which defines how many boundary conditions have to be entered in the cards 1CCCG801 and 1CCCG901. The cards of the heat structure are 1CCCG801 and 1CCCG901. These cards determine the boundary conditions for each side (left and right) of the heat structure [28][31].

RELAP 5 Heat Structure Input

```
*crdno heat.str mesh.pts geometry init.flag left.coord

1CCCG000 N  MN  Geo  0  left_Coor

*crdno mesh.locn.flag mesh.fmt.flag

1CCCG100 0  1

*crdno no.intervals rt.coord

1CCCG101 NoInter  rt_Coor

*crdno comp.no. interval.no.

1CCCG201 Comp  N.Inter

*crdno source.value mesh.int.no.

1CCCG301 S.V  Mesh.Int

*crdno temperature mesh.pt.no.
```

```
1CCCG401 T  MeshNo

*crdno left.volume incr b.cond sa.code area/factor ht.str.no.

1CCCG501 CCCXX0000  Inkr  1  0  A  N

*crdno rt.volume incr b.cond sa.code area/factor ht.str.no.

1CCCG601 CCCXX0000  Inkr  1  0  A  N

*crdno s.type s.mult left.heat right.heat ht.str.no.

1CCCG701 0  0.0  0.0  0.0  N

*crdno con.val

1CCCG800 Con.V

*ADD LEFT BOUNDARY CONDITION

*crdno equiv.diam heated.len's grid.len's grid.cf's bf ht.str.no.

1CCCG801 DHd  40.0  40.0  0.0  0.0  0.0  0.0  1.0  N

*crdno con.val

1CCCG900 0

*ADD RIGHT BOUNDARY CONDITION

*crdno equiv.diam heated.len's grid.len's grid.cf's bf ht.str.no.

18201901 DHd  40.0  40.0  0.0  0.0  0.0  0.0  1.0  N
```

11.6.5 Single-Junction

The single-junction component is used to connect other components, for example pipe or time-dependent volumes, to each other. Initial junction conditions of this component are fluid velocities or mass flow rates. The input text at the end of this section shows a modeled single-junction. Card CCC0000 is the first card which is used to model a single-junction. The CCC stands again for the junction number and can be chosen variably. The card requires two inputs. The first junction input is the name of the junction, which is like the component names variable, and the word sngljun, which defines the junction type. The card CCC0101 is about the connected components. The first value of this card defines from which component the certain mass flow comes from. The second value defines to which component the certain mass flow goes to. This is achieved by the value CCCXX000F. CCC determines the component number XX the volume number and F (1 inlet, 2 outlet) with which surface the junction is connected. The third value of this card defines the area of the junction. The card CCC0201 determines the unit of the initial conditions for the single-junction and specifies the mass flows or fluid velocities of the fluid phases in single-junction at the beginning of the simulation. To select fluid velocities the first value of this card has to be a zero. To select mass flows a one has to be entered. The values for the liquid and vapour phases are defined seperately. The second value of the card establishes initial condition of a liquid and the third value the initial condition of vapour. The fourth value of this card is always zero [28][31].

RELAP 5 Single-Junction Input

```
*hydro component name component type
CCC0000 "Name" sngljun
*hydro from to area f.loss r.loss fvcahs
CCC0101  CCCXX000F  CCCXX000F  A_j  0.0  0.0  0
*hydro f.flowrate g.flowrate j.flowrate
CCC0201  1  mf_L  mf_Vb  0.0
```

11.6.6 Time-Dependent Junction

Time-dependent junction components are used when fluid velocities or mass flow rates are a function of time or another time-advanced quantity. With these components it is possible to model flow boundary conditions on the simulated system. The boundary conditions can be specified as either a volumetric or mass flow rate. Time-dependent junctions can connect like single-junctions, two different system volumes or time-dependent volumes. The input text at the end of this section shows a modeled time-dependent junction. The first card of a time-dependent junction component is the card CCC0000. Again the CCC stands for the junction number and can be chosen variably. The card requires two inputs. The first junction input is the name of the junction, which is also variable, and the word tmdpjun, which defines the junction type. The card CCC0101 is about the connected components. The first value of this card defines from which component the certain mass flow comes from. The second value defines to which component the certain mass flow goes to. This is achieved by the value CCCXX000F. CCC determines the component number XX the volume number and F (1 inlet, 2 outlet) with which surface the junction is connected. The third value of this card defines the area of the junction. The first value of card CCC0200 determines the unit of the initial conditions for the time-dependent junctions. A zero must be entered to select fluid velocities and a one for mass flows. The three other values are optional trip and control values. The card CCC020N defines the fluid velocity or mass flow rate of the different phases of the fluid depending on a control value (tdigit). For example the mass flow rate is changed over time. The control value can be chosen in card CCC0200 but usually time is chosen as the value. Now for the simulation in RELAP 5, the time-dependent junction changes its conditions depending on the control value [28][31].

RELAP 5 Time-Dependent Junction Input

```
*hydro component name component type
```

```
CCC0000 "Name" tmdpjun
*hydro from to area
CCC0101  CCCXX000F  CCCXX000F  A_j
*hydro vel/flw trip.no.
CCC0200  1
*hydro tdigit f.flowrate g.flowrate j.flowrate
CCC020N  t  mf_L  mf_V  0.0
```

11.6.7 Valve Junction

In general valve junction components have the capability to vary the flow area. Each valve type has its own input rules, which will not be discussed here. Eight valves out of six types can be chosen in RELAP 5. The types are:

- check valves (CHKVLV)
- trip valves (TRPVLV)
- inertial swing check valves (TRPVLV)
- motor valves (MTRVLV)
- servo valves (SRVVLV)
- relief valves (RLFVLV)

The input text at the end of this section shows an example for a modeled trip valve. The first card of a valve junction component is the card CCC0000. The CCC stands again for the junction number and can be chosen variably. The card requires two inputs. The first junction input is the name of the junction, which is also variable, and the word valve, which defines the junction type. The card CCC0101 deals with the connected components. The first value of this card defines from which component the certain mass flow comes from. The second value defines to which component the certain mass flow goes to. This is achieved by the value CCCXX000F. CCC determines the component number XX the volume number and F (1 inlet, 2 outlet) with which surface the junction is connected. The third value of this card defines the area of the junction. The next card has the number CCC0200. The first value of this card determines the unit of the initial conditions for the valve junction. To select fluid velocities, a zero must be entered and for mass flow rates, a one. The next values specify the mass flow rates or fluid velocities of the fluid phases in junctions at the beginning of the simulation. The values for the liquid and vapour phases are defined seperately. The second value of the card establishes the initial condition of the liquid and the third value the initial condition of the vapour. The fourth value of this card is always zero. With the card CCC0300, the valve type is determinted. In this example, a trip valve was chosen thus enter the control word trpvlv. The other control words can be found in the list at the beginning of the section. The input of the next card CCC0301, is dependent on the type of valve which was chosen. In this case the values for a trip valve have to be entered. For trip valves only the trip number is necessary. This value makes a link to the trip which controls the valve [28][31].

RELAP 5 Valve Junction Input

```
*crdno Name Type
CCC0000 "Name" valve
*hydro from to area
CCC0101  CCCXX000F  CCCXX000F  A_v
*crdno vel/flw f.flowrate g.flowrate j.flowrate
CCC0201  1  mf_L  mf_V  0.0
*crdno Valvetype
CCC0300 trpvlv
*crdno TripNumber
CCC0301 trip
```

11.7 The Model

The next step was to summarize the information in the tables as well as the information about the various RELAP 5 components in a thermal hydraulic model. The input of the model can be seen in Appendix 16. The next two figures show a schematic representation and the nodalization diagram of the simulated NuScale reactor. The numbering in the figures is the same in the model in the appendix.

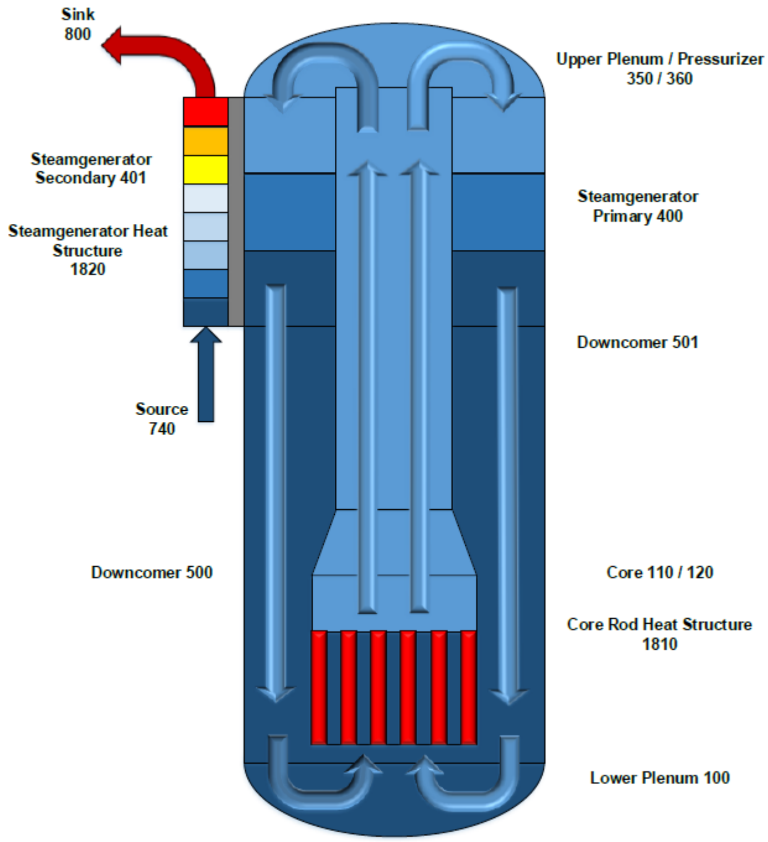

Figure 11.1: Schematic representation of the SMR RELAP 5 Model.

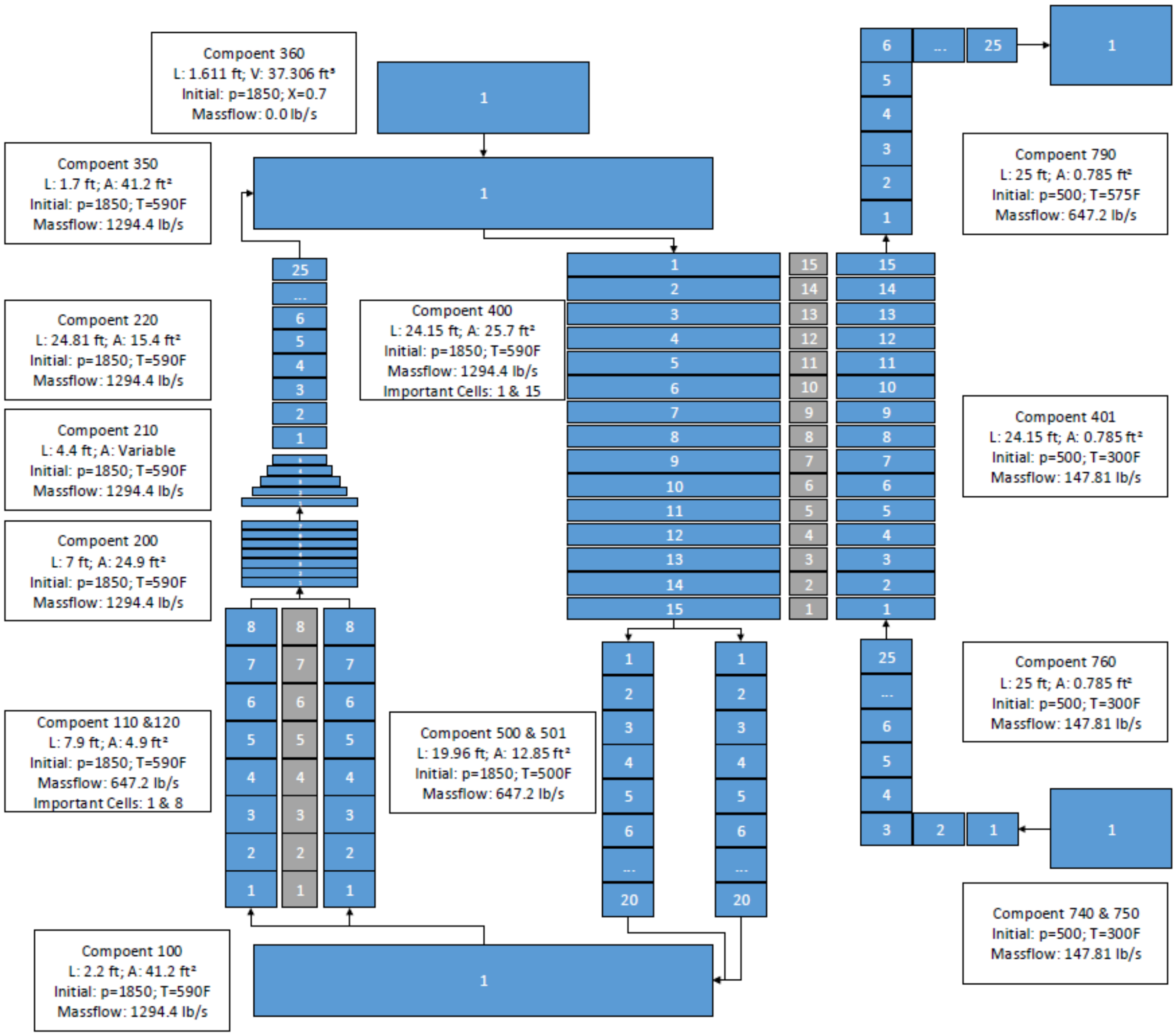

Figure 11.2: Nodalization diagram of a SMR RELAP 5 Model.

12 Steady State Model

The next step in the development of the SMR model is to reach steady state conditions. This means the calculated values of the model do not change over time. To reach these conditions, RELAP 5 has to run the model for a long time. The time to reach steady state was 5000 seconds. The most important calculated values are pressure, temperature, void fraction, and mass flow at the different volumes and junctions of the SMR model. Especially the parameter at the inlet and outlet of the core and the steam generator primary and secondary are very important. At the heat structures of the core, heat is generated. This heat is then transported by the liquid water to the steam generator in the primary loop. The heat structures of the steam generator transfer the heat to the secondary side which is in steady state from the beginning of the simuation, because it is an open cycle. The simulated reactor has a thermal power of 160 MW. Therefore the fuel rods have to produce 160 MW and the heat structures of the steam generator have to move this amount of power from the primary side to the secondary side. The heat transfer can be calculated by the following equation.

$$Q_{th} = \dot{m} \cdot (h_{out} - h_{in}) \tag{12.1}$$

The enthalpies in this eqaution depend on the temperature and the pressure of the fluid in the volume. With the help of thermodynamic steam tables, it is possible to interpolate the enthalpy for the calculated parameters. The next figure shows a shematic of the reactor model and where the different components are located. This shematic figure will help the reader to better understand what processes occur in the simulated model.

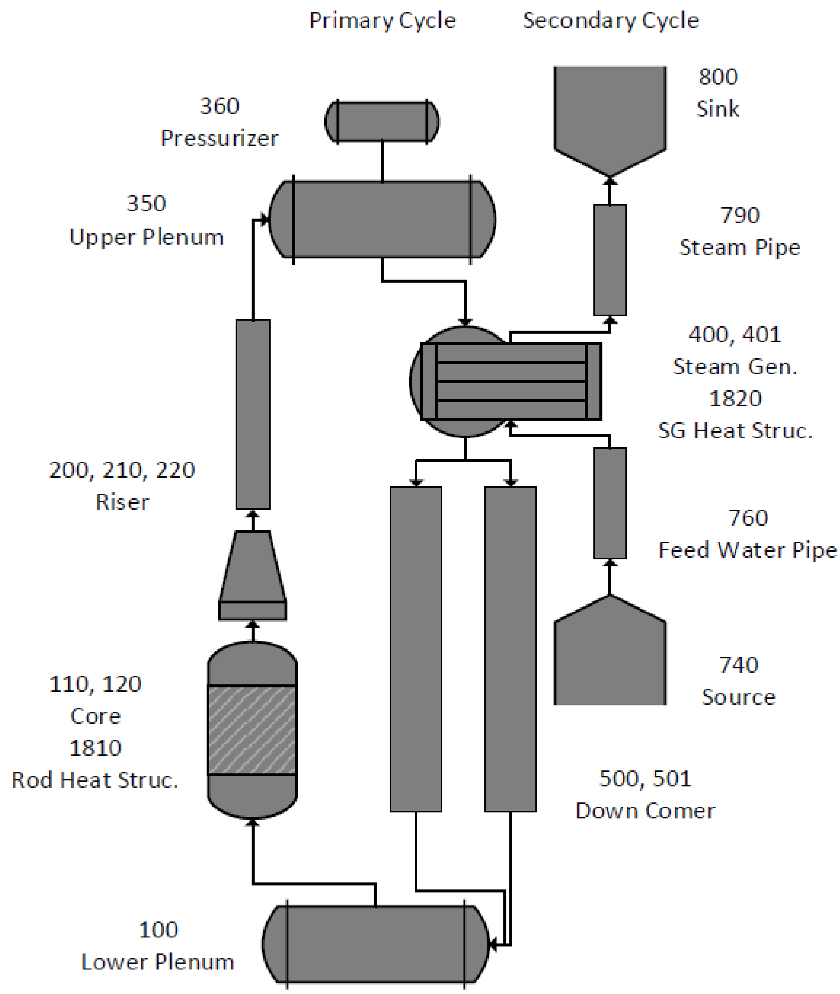

Figure 12.1: Schematic representation of the model.

Table 12.1: Model Component Data

Component number	Name	Type	Volumes
100	Lower Plenum	Branch	1
110	Core	Pipe	8
120	Core	Pipe	8
200	Lower Riser	Pipe	7
210	Middle Riser	Pipe	5
220	Upper Riser	Pipe	25
350	Upper Plenum	Branch	1
360	Pressurizer	Branch	1
400	Steam Generator	Pipe	15
401	Steam Generator	Pipe	15
500	Down comer	Pipe	20
501	Down comer	Pipe	20
740	Source	Tim.Dep.Volume	1
760	Feed Water Pipe	Pipe	25
790	Steam Pipe	Pipe	25
800	Sink	Tim.Dep.Volume	1
1801	Fuel Rods	Heat Structure	8
1802	SG-Tubes	Heat Structure	15

The next sections show the measured values at the specific volumes over the simulation time.

12.1 Core

The core in the model is located between component 100 (lower plenum) and component 200 cell 1 (lower riser). At these specific volumes the measurements are made. The next figures show the pressure and temperature over time in both volumes.

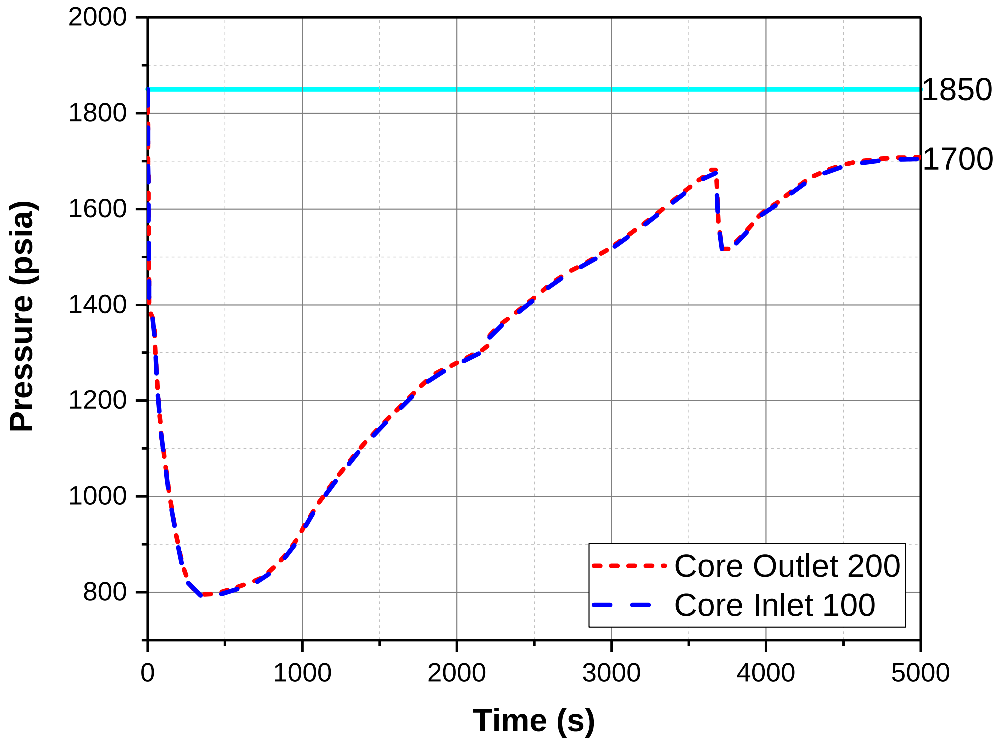

Figure 12.2: Steady State Pressure of the Core Inlet and Outlet.

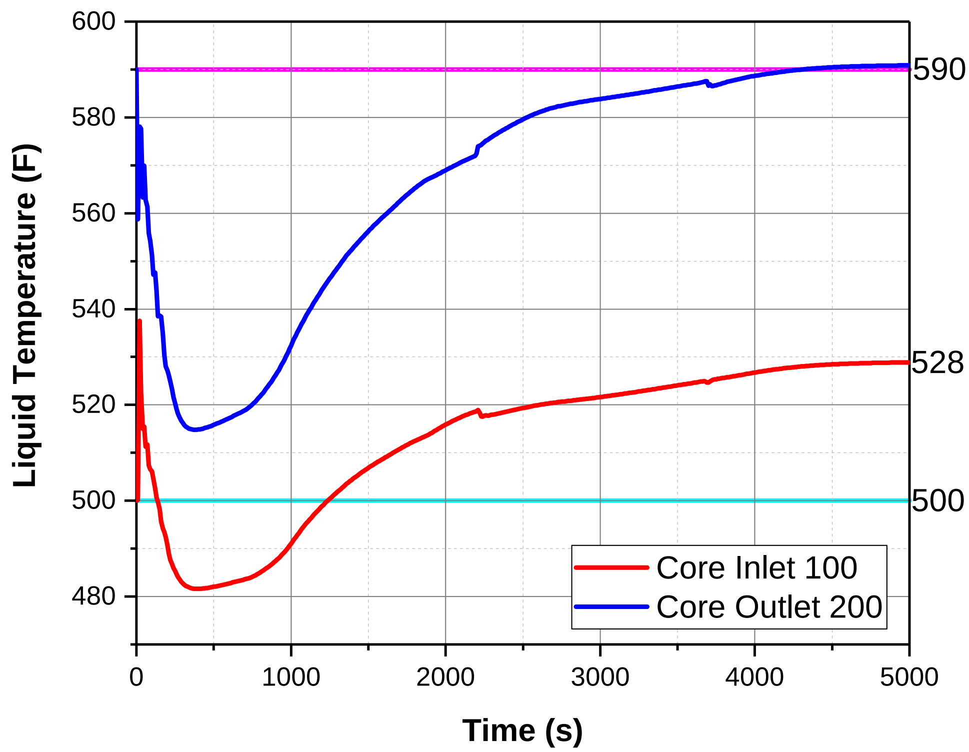

Figure 12.3: Steady State Temperature of the Core Inlet and Outlet.

At the beginning of the simulation it is clearly seen that RELAP 5 calculates to reach steady state conditions. After 4500 seconds, pressure and temperaure are constant. At this point the model is in steady state. The steady state pressure and the steady state temperatures have the values of 1700 psi, 590°F and 528°F. Figure 12.2, it is seen that the pressures from the core inlet and core outlet are slightly different. The reason for this is hydrostatics because the inlet is lower than the outlet. But the difference is very small and thus it can be neglected. The pressure is a little bit lower than expected but the difference is only 150 psi. The biggest problem with a lower pressure is the corresponding lower boiling temperature. But in this simulation this is not a problem because the boiling temperature for a pressure of 1850 psi is 624.9°F and for 1700 psi 613.9°F. This is a small difference and can be neglected due to the fact that the highest temperature in the simulation is approximatly 25°F away from this point. The temperature difference is approximatly 62°F compared to expected 90°F. This can be explained by the frictional losses in the system. Frictional losses in the primary loop are very important in determining the actual pressure drop and therefore, mass flow rate, heat transfer, pressures, and temperatures. In the simulation all components are simulated as smooth, thus fictional losses are small. Only in the core and steam generator components are values implemented which slow down the circulation. The result of this lower temperature difference is a higher mass flow rate in the system to compensate for this condition. When RELAP 5 has reached steady state conditions, it is possible to show how much energy the core has input into the system. For this purpose, thermodynamic steam tables can be used to calculate the enthalpy of the fluid inside the system before entering and after leaving the core. The entalpy values are h_{outC} = 1398.9 kJ/kg and h_{inC} = 1211.4 kJ/kg. To calcutale the power of the core it is necessary to look up the steady state mass flow rate of the core. RELAP 5 measures mass flow rates in the junctions and not in the volumes of its models. To see the right mass flow rates, the junctions of the core inlet and core outlet have to be checked. Because the core in this model is divided into two parts, both mass flows have to be added into one single mass flow to show the total mass flow rate in the core components. The mass flow rates can be seen in the next figure.

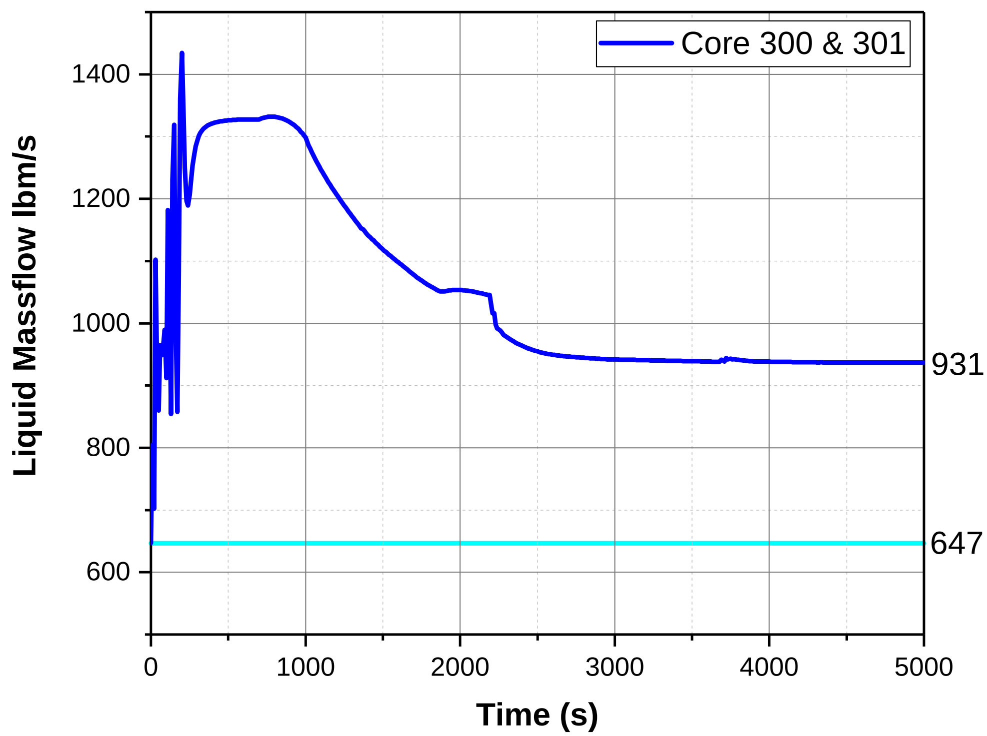

Figure 12.4: Steady State Massflow in the Core.

The next figure shows the densities in the core inlet and core outllet. The density difference is the driving force in natural circulation and leads to the movement of the liquid water in the simulated SMR reactor. It is clearly seen in figure 12.5 that a density difference occurs between core inlet and core outlet. In addition to that it is seen in figure 12.4 that mass moves through the system. This result is the proof for the existence of natural circulation in the simulated SMR reactor.

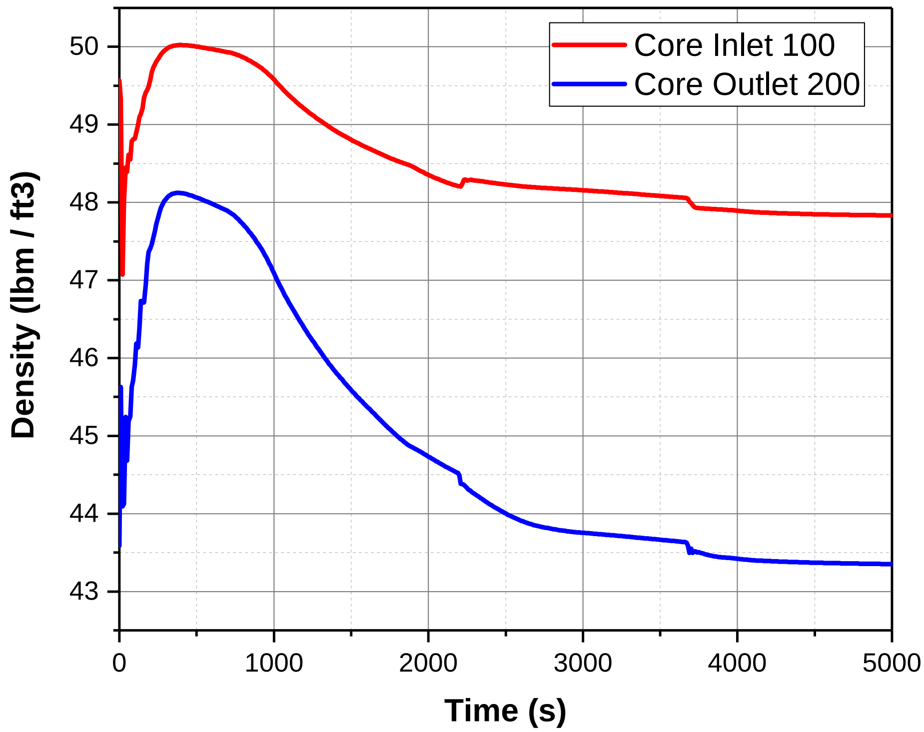

Figure 12.5: Steady State Density at the Core Inlet and Outlet.

It is seen at the beginning of the simulation that the steady state mass flow rates in figure 12.4 has to be reached like the pressure and temperatures. After this time, the mass flow rates are constant and have a value of 931 lbm/s. Again this is a little bit higher because of the lower temperature difference. The primary loop is liquid water only because of the relativly high pressure of the system. That is shown by the calculation of the enthalpies and in the next figure which shows the liquid fraction of the appropriate volumes.

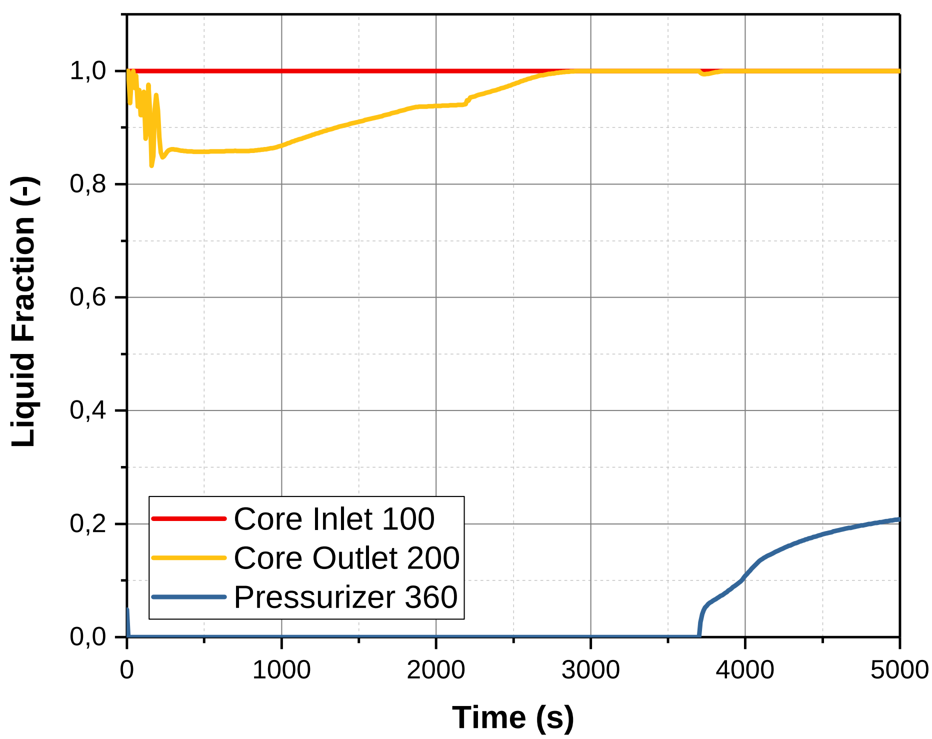

Figure 12.6: Steady State Liquid Fraction at the Core Inlet and Outlet.

With the steady state volumes for enthalpy and mass flow it is now possible to calculate the power of the core. The power can be calculated with equation 12.1.

The result is 158.69 MW. This shows the model works and the correct power level is reached.

12.2 Steam Generator Primary

The primary steam generator in the model is located between component 350 (upper plenum), component 500 (down comer) and component 501 (down comer). For all components the first cell is used. At these specific volumes the measurements are made. It is necessary to check temperatures, pressures and mass flow rates at the inlet and outlet of the steam generator. The next figures show the pressure and temperature over time in the measured volumes.

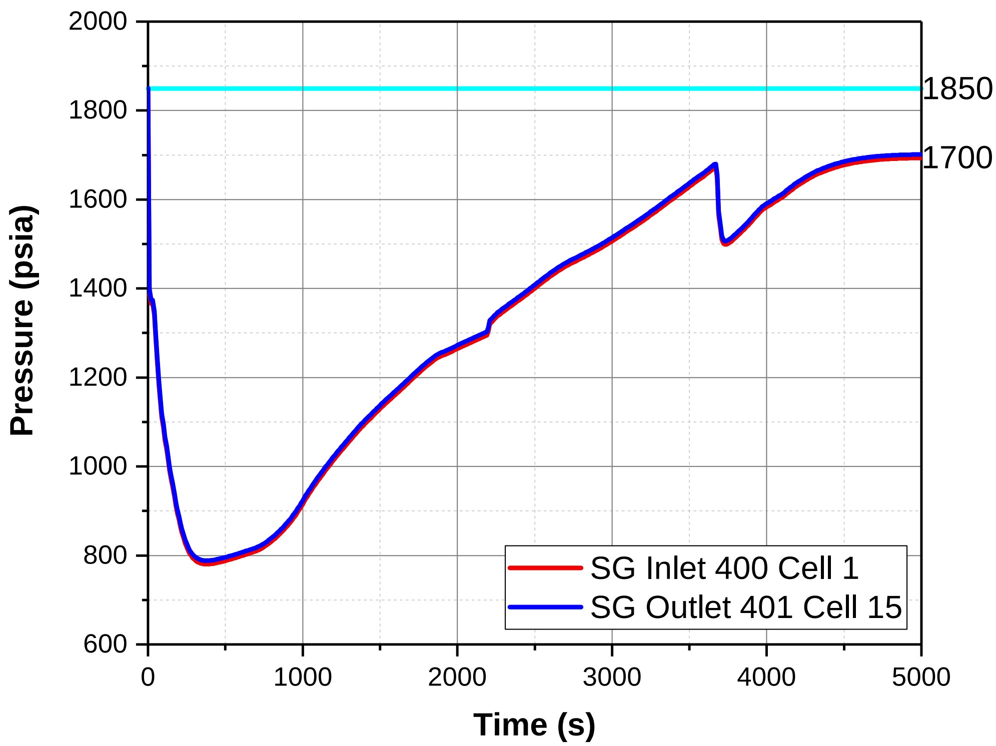

Figure 12.7: Steady State Pressure at the Steam Generator Inlet and Outlet.

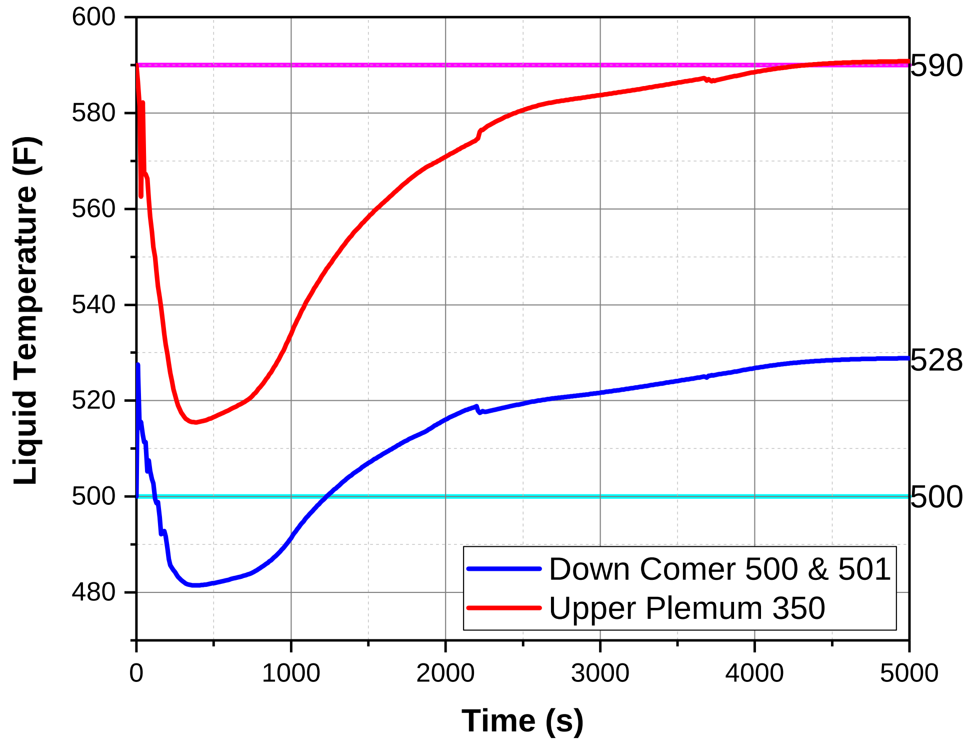

Figure 12.8: Steady State Temperature at the Steam Generator Inlet and Outlet.

Again it is clearly seen that RELAP 5 at the beginning of the simulation tries to reach steady state. After 4500 seconds the steady state temperatures are T_H=590°F and T_C=528°F and the steady state pressure is 1700 psi. The steam generator is located between the hot leg and the cold leg of the reactor high pressure cycle. It has the function to remove all the heat energy which is produced in the core to the secondary side. As already calculated the power of the core is 158.69 MW. The same power has to be removed by the steam generator. The reasons that the parameters are not exactly the parameters from the tables from the previous chapter are the same as already described in the previous subchapter. With these parameters for pressure and temperature, it is again possbile to look up the enthalpies for the hot leg and the cold leg of the reactor. From the steam tables, the enthalpies are h_H=1398.9 kJ/kg and h_C=1211.4 kJ/kg. To calculate the removed heat in the steam generator the mass flow rate is needed. The next figure shows the mass flow through the primary side of the steam generator.

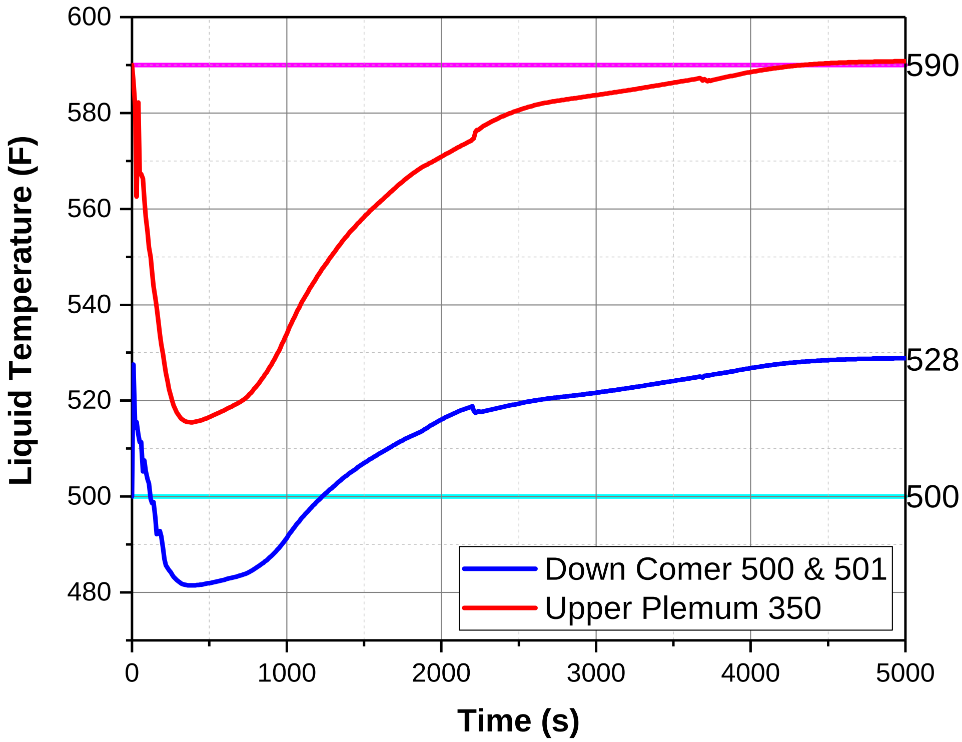

Figure 12.9: Steady State Mass flow though the Steam Generator.

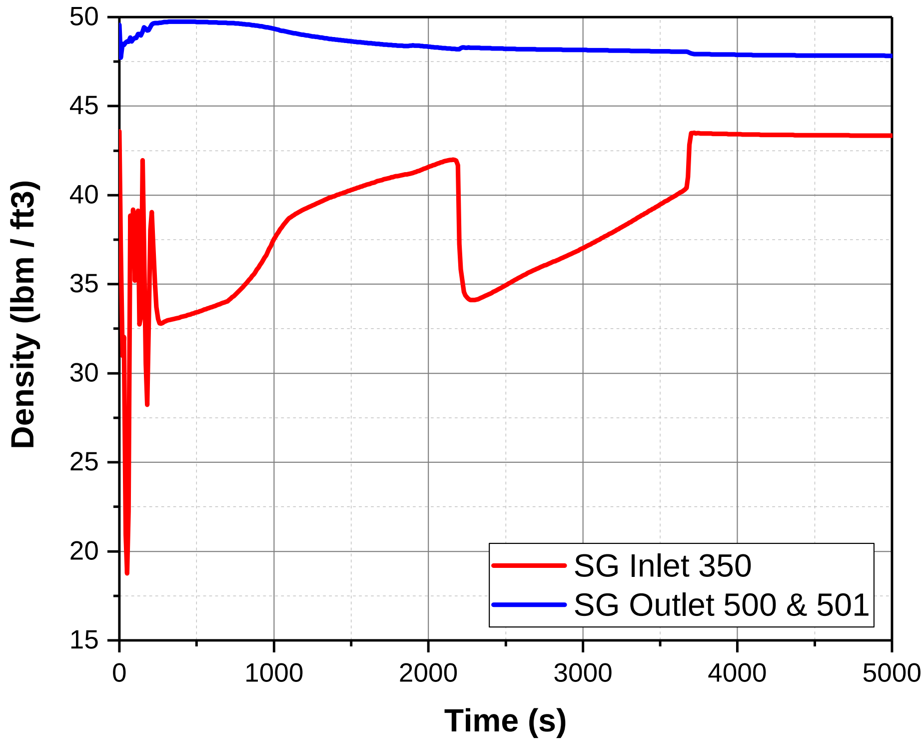

Figure 12.10: Steady State Density at the Steam Generator Inlet and Outlet.

As seen in the figure, the steady state mass flow is 1865 lbm/s. The reason that the mass flow in the steam generator is double the core mass flow in figure 12.4, is the core is divided into two parts. The mass flow of the two parts can be added to full system mass flow of 1865 lbm/s. To illustrate natural circulation, the densities at the steam generator inlet and outlet are shown in figure 12.10. It is clearly seen that the inlet density is lower than the oulet density. This illustrates the natural circulation in the simulated SMR. Figure 12.11 shows the liquid fraction at steam generator inlet and outlet. Also the calculations of the enthalpies have shown that the whole system is liquid water only at steady state conditions..

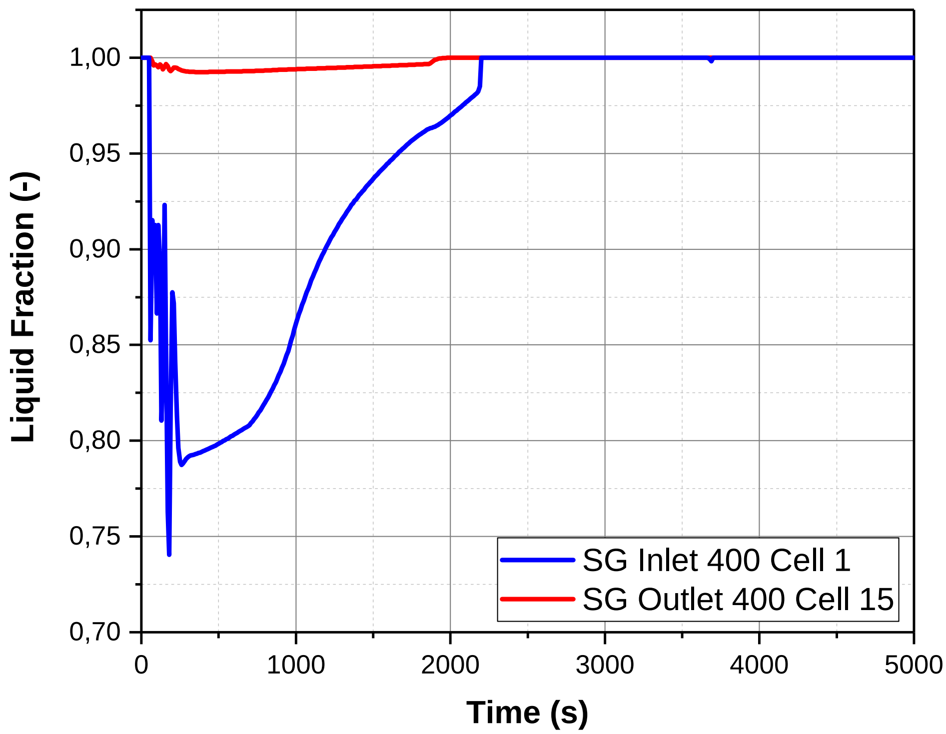

Figure 12.11: Steady State Liquid Fraction in the Steam Generator primary Inlet and Outlet.

Figure 12.11 shows that at the beginning of the simulation, boiling happens in the steam generator. This is because of the pressure drop at the beginning of the simulation which is seen in figure 12.7. This pressure drop is caused by the inconsistency of the choosen initial conditions of the simulation model. The removed heat can be calculated with equation 12.1. The results is 158.69 MW. This result is the same as in the produced power in the core. Therefore the reactor has reached steady state.

12.3 Steam Generator Secondary

The steam generator secondary in the model is located between component 760 (Watpip) cell 25 and component 790 (Stepi) cell 1. In the steam generator secondary side, the fluid makes a phase change from liquid to vapor. In cell 25 of component 760 (inlet), the fluid state is liquid and in cell 1, of component 790 (outlet) the fluid state is vapor. The following figures show the liquid and vapor fractions at the inlet and outlet of the secondary steam generator.

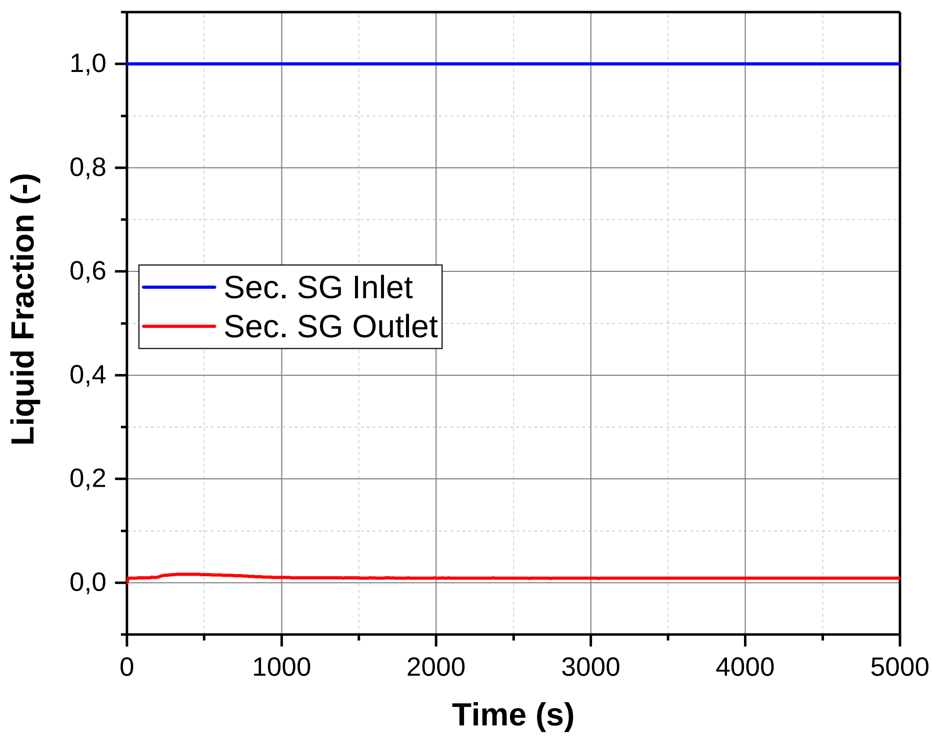

Figure 12.12: Liquid Fraction of the Steam Generator sec. side Inlet and Outlet.

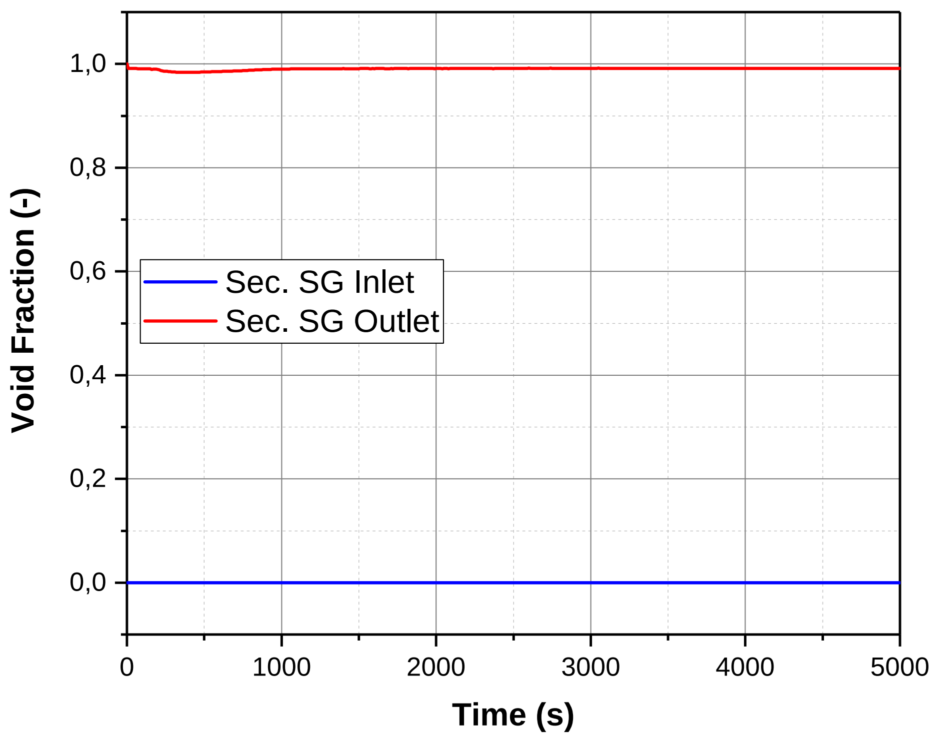

Figure 12.13: Vapor Fraction at the Steam Generator sec. side Inlet and Outlet.

As clearly seen in the two figures, the fluid before the steam generator is solid liquid and after leaving it the fluid is fully vapor. Therefore, it is verified that the entering liquid makes a phase change to vapor inside the steam generator. The next figure shows boiling development in the steam generator cells over time.

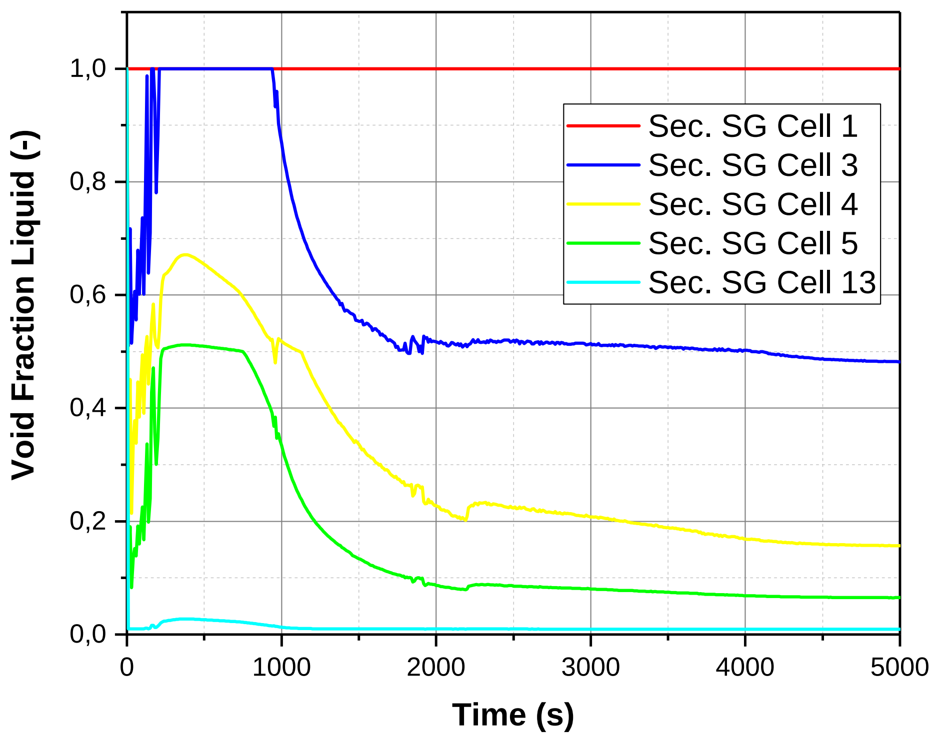

Figure 12.14: Liquid Fraction in the Steam Generator sec. Cells.

It is seen in figure 12.14 that the fluid state in the first cell of the secondary steam generator is solid liquid. At the third cell the fluid changed about 50 percent from liquid to vapor. It is also seen that the rate of conversion drops over time. While at the beginning (cell 1 to cell 3) 50 percent of the liquid made a phase change, only 10 percent of the liquid made a phase change at the end of the steam generator (cell 5 to cell 13). The reason for this is that some of the heat energy is taken to superheat the steam and is not used to phase change the rest of the liquid. To see what condtions the fluid has before entering and after exiting the steam generator it is necassary to look up the liquid and vapor temperatures for component 760 and 790. Because RELAP 5 expects to have liquid and vapor in every volume, it always calculates the temperatures for both fractions in every volume. The next figures show the temperatures of the fluid.

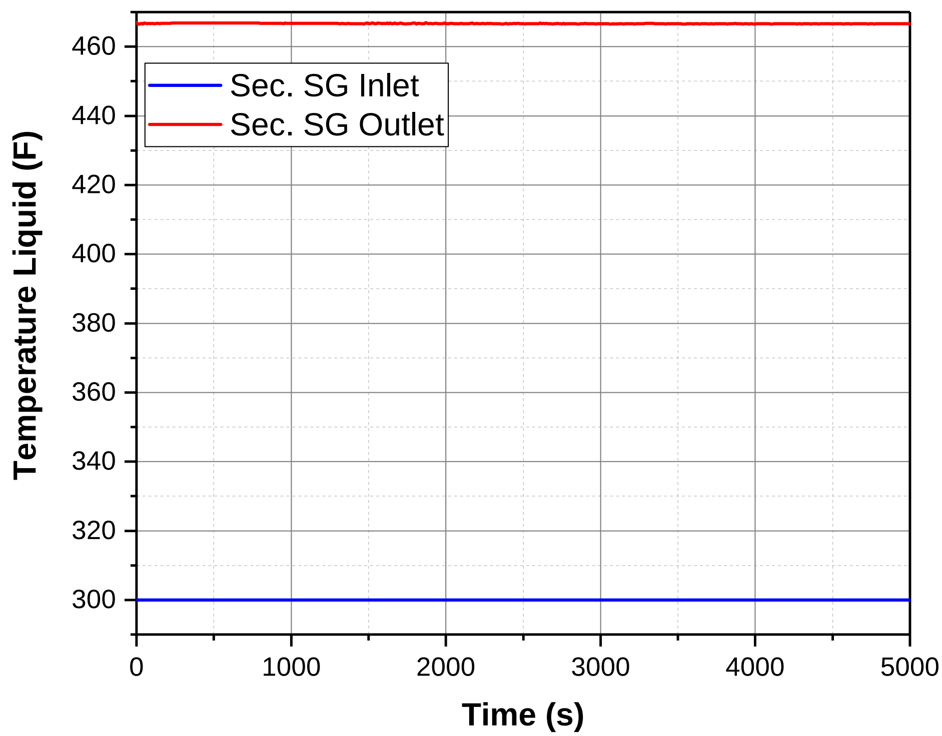

Figure 12.15: Liquid Temperature at the Steam Generator sec. Inlet and Outlet.

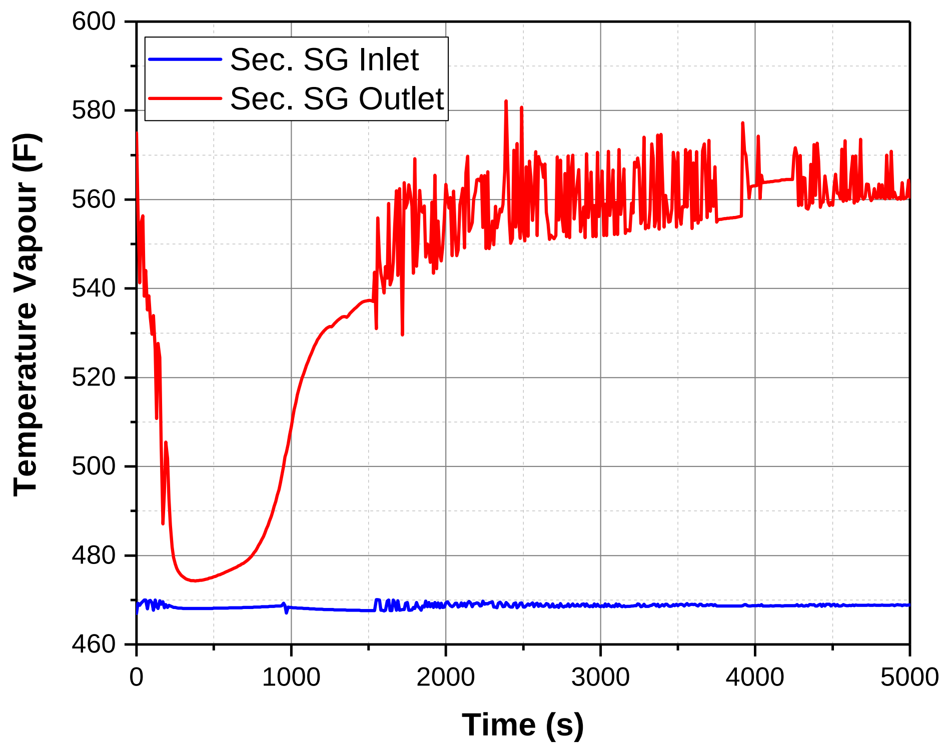

Figure 12.16: Vapor Temperature at the Steam Generator sec. Inlet and Outlet.

It is seen in the figures that the liquid before entering the steam generator has a temperature of 300°F specified by fixed boundary conditions. The vapor after leaving the steam generator has a temperature of 565°F. With the help of steam tables, it is observed that the outlet vapor has superheated conditions and the inlet liquid has subcooled conditions. It is also possible to look up the enthalpies for the liquid and vapor. The liquid inlet enthalpy is h_{in}=629.34 kJ/kg and the vapor outlet enthalpy is h_{out}=2970.4 kJ/kg. The pressure is the second variable to calculate the enthalpies and thus the conditions of the fluid. The next figures show the pressures of the inlet and outlet of the secondary steam generator.

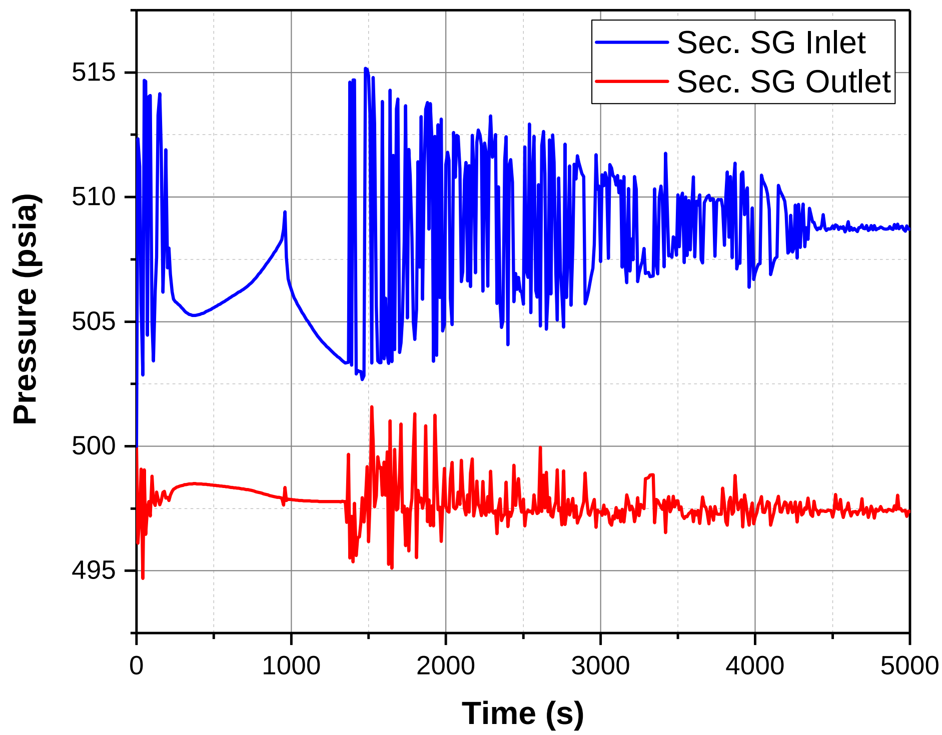

Figure 12.17: Pressure of the Steam Generator sec. Inlet and Outlet.

The mass flow rate of the two phase fluid though the secoandary steam generator can be seen in figure 12.18. The steady state mass flow rate is used to calculate a heat transfer of 158.69 MW, which is considered with the core power output.

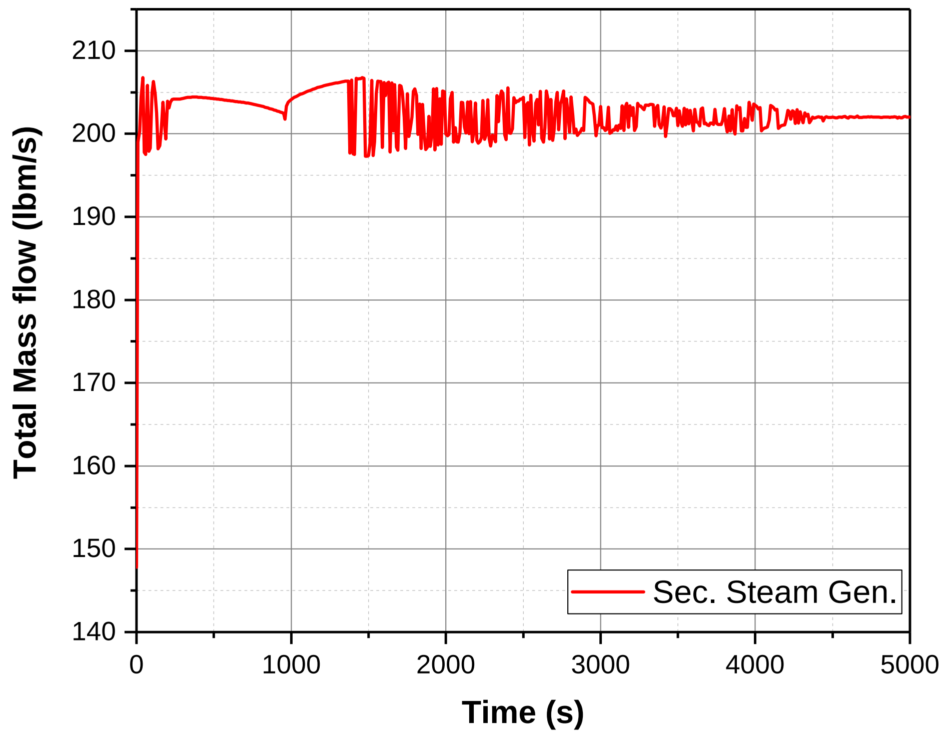

Figure 12.18: Mass flow in the Steam Generator secondary.

12.4 Optimization

The important parameters for the model are the parameters of the primary cycle. These are the hot leg temperature, the cold leg temperature, the primary mass flow rate and the primary system pressure. In the model, the secondary mass flow rate, the firctional losses and the steam generator surface had the most impact on the important values. In this section, a variation of the input parametes and their influence on the important output parameters will be presented.

As seen in table 12.2, the manipulation of the input parameters lead to an improvement of the output parameters. The most significant discovery was the relationship between primary system mass flow rate and frictional losses in the primay cycle. The NuScale system runs via natural circulation in its primary cycle. The only forces witch can slow down the mass flow rate at natural circulation are frictional losses. If there were no frictional losses in the model the mass flow would increase to a too high level. But the heat transfer would not change because of the mass flow rate, enthalpy difference tradeoff from equation 12.1. If there were frictional losses in the model the mass flow would decrease and the enthalpy difference would rise, because the enthalpy is dependent on pressure and temperature. The coolant would have more time to heat up and cool down in the core and in the steam generator.

One big change, which is not seen in the table, is the rise of the roughness in the riser and in the downcomer of the model. This can be seen in the appendix where the input model is located. The calculated pressure and the hot leg temperature at the last case are exactly the same as in the final safety analysis report of NuScale. The calculated cold leg temperature and the primary mass flow rate are still higher than expected, especially the mass flow rate is higher than in the final safety analysis report of NuScale.

In addtion to that, it is possible to look up the enthalpies in the thermodynamic steam tables for the exact pressure and temperatures from the NuScale FSAR. Together with the exact mass flow rate from the FSAR, it is possible to calculate the transferred heat in the system. This value is only approximatly 154 MW, expacted was 160 MW. This is a reason why the mass flow rate in the developed model is higher than expected. In RELAP 5 there are plenty of optional cards which could be used to reach the final and expected steady state of the system, but this would be too complicated for this simple model and for this reason the eighth case is the final steady state of the model for the futher development of the model.

Table 12.2: Parametric variation of the important parameters.

Intput Variables			Output Variables				Error			
Exact Parameter			590°F	500°F	1850 psia	1294.11 lbm/s	-	-	-	-
Loss-Co.	$Area_{SG}$	$\dot{m}_{se}$ (lb/s)	T_H	T_C	p (psi)	$\dot{m}_{pri}$ (lb/s)	ΔT_H	ΔT_C	Δp (psi)	$\Delta \dot{m}_{pr}$ (lb/s)
0.975	1250.2 ft^2	202.0	590°F	528°F	1700.0	1865	0°F	28°F	150.0	570.89
0.9	1250.2 ft^2	250.0	481°F	456°F	900.0	3203	590°F	44°F	950.0	1908.89
0.9	1500.2 ft^2	202.0	567°F	512°F	1255.9	2107.5	109°F	12°F	594.1	813.39
1.25	1250.2 ft^2	202.0	588°F	522°F	1650.0	1771.4	23°F	22°F	200.0	477.29
1.5	1250.2 ft^2	197.0	624°F	556°F	3928.8	1738.2	2°F	56°F	2078.8	444.09
1.5	1250.2 ft^2	199.0	614°F	542°F	2960.8	1645.2	34°F	42°F	1110.8	351.09
1.5	1350.2 ft^2	199.0	606°F	536°F	2500.0	1645.2	24°F	36°F	650.0	351.09
1.5	1350.2 ft^2	201.0	595°F	524°F	1833.2	1660.2	5°F	24°F	17.0	366.09

13 Further Development

The RELAP 5 model of the NuScale small modular reactor presented in this thesis has much potential. At the moment only a closed primary loop and an open secondary loop is modeled and tested. The NuScale SMR reactor has two passive safety systems which are called emergency core cooling system (ECCS) and decay heat removal system (DHRS). The ECCS and DHRS system were discussed in the previous chapters. These two systems can be added to the NuScale model in the future. Figure 13.1 shows schematic the enhanced model. The differences can be cleary seen in this figure when compared to Figure 12.1. To model these passive safety systems, pipes, junctions, and valves would be used in RELAP 5. After the development of the new model, it is possible to test and evaluate several accident scenarios with the new modeled passive safety systems. Examples for accident scenarios are:

- Steam generator tube rupture
- Failure of secondary cooland pump

- Accidental opening of a vent valve

- Small break in the primary system

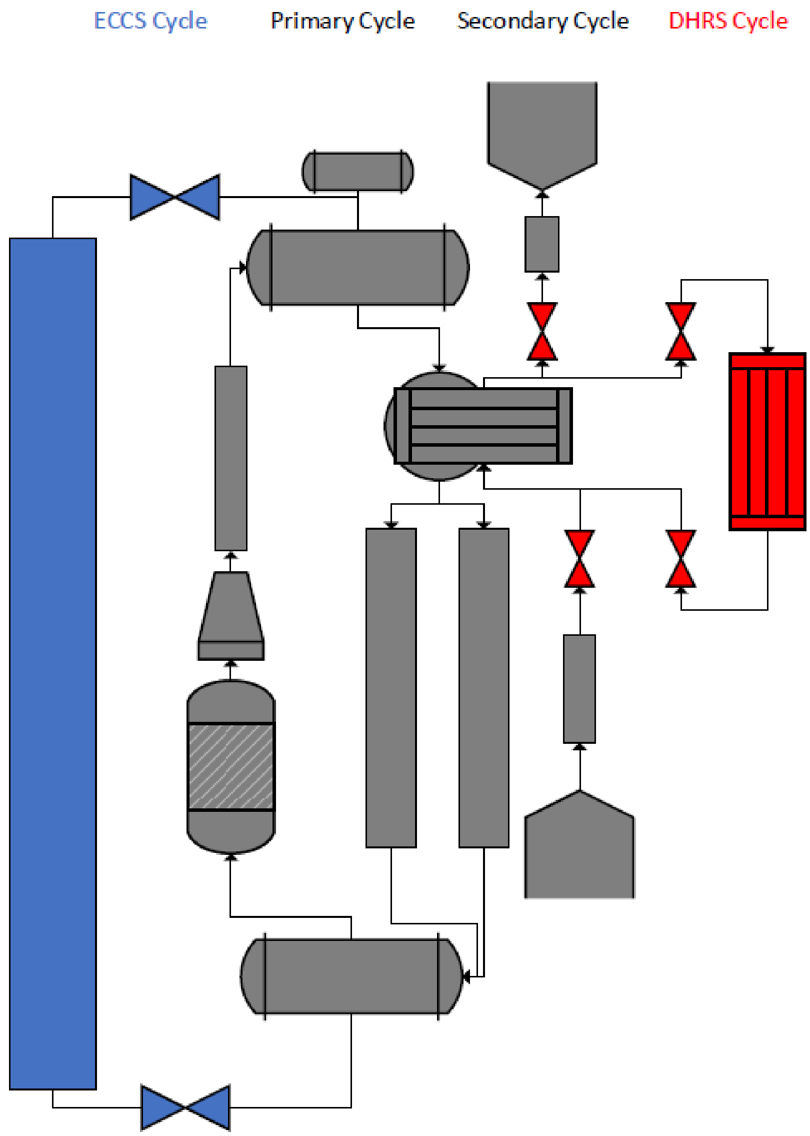

Figure 13.1: Schematic representation of the future SMR model.

14 Conclusions

The goal of this research was to develop a thermal hydraulic model to simulate the NuScale design small modular reactor with RELAP 5. The first step for this work was to survey design of small modular reactors (SMRs) and technology aspects. Furthermore, since the simulated NuScale SMR is an integral PWR, the operation of pressurized water reactors needed to be exammed to develop the model. The NuScale design as descibed in the final safety analysis report formed the basis of this model. A detailed of the NuScale small modular reactor was developed for achieving a steady state. The model has two main parts, a closed primary cycle and an open secondary cycle. It was achieved to model all components of the NuScale SMR and run a simulation where the steady state parameters of the FSAR and the calculated RELAP 5 parameters apporximatly matched. The significance of this verification was that it was proven that this reactor design works, also when it is only viewed from a physical-mathematical point. The steady state is the basis for all other following accident simulations. In the steady state simulation, it could be verified that the NuScale SMR design as described in the final safety analysis report is consistent. However, it was also discvered that frictional losses play a significant role in the NuScale system balance of mass flow and heat transfer. Also, it was shown that natural circulation only is enough to operate this kind of reactor. These basis simulations showed the great potencial of the NuScale design and the corresponding developed RELAP 5 model. For future work it will be necessary to improve the RELAP 5 model and program more parts of the whole NuScale SMR system. An example could be the two passive safety systems, DHRS and ECCS. The addition of these system would give the model a greater complexity and would improve the scope of application. At the end the model could be used to predict and analyse several operation and accident scenarios, which is significant because it would be not necessary anymore to wait for experimental data from hardware reactor models. This would lead to a fast development and would safe time and money. This work will be part of further development of the RELAP 5 Nuscale SMR model.

15 Appendix Nomenclature

15.1 Latin letters

t time (s)

x spatial variable (m)

υ fluid velocity (ms^{-1})

υ_i interphase velocity (ms^{-1})

A cross-sectional area (m^2)

V volume(m^3)

m mass (Kg)

W mass flow (Kgs^{-1})

P pressure (Pa)

B_x projection of gravity onto the pipe direction (ms^{-2})

FW wall friction coefficient (s^{-1})

FI interphase friction coefficient (s^{-1})

C virtual mass coefficient (non-dimensional) compressibility (m^3)

K local pressure drop coefficient (non-dimensional)

$HLOSS$ form loss coefficient (ms^{-1})

u specific internal energy (JKg^{-1}).

H heat transmission coefficient per unit volumen ($Js^{-1}\ m^{-3}K^{-1}$), control volume height (m)

Q heat power per unit volume ($Js^{-1}m^{-3}$)

h specific enthalpy (JKg^{-1})

T temperature (K)

s specific entropy ($JKg^{-1}K^{-1}$)

$DISS$ dissipation power per unit volume ($Js^{-1}m^{-3}$).

15.2 Greek letters

α fraction of phase (non-dimensional)

ρ density (Kgm^{-3})

Γ phase generation rate per unit volume ($Kgm^{-3}s^{-1}$)

δ Dirac's delta function, centered around the point x_0 (m^{-1})

15.3 Subindices

k phase (liquid or vapor)

k' phase opposite to phase k

i relative to the interphase

w relative to the wall

b relative to the fluid bulk

m relative to the mixture

g relative to vapor

f relative to liquid

16 Appendix RELAP 5 Model

```
=SMR Model


*crdno problem type option
0000100 new transnt
*----------------------------------------------------------------
*crdno inp-chk/run option
0000101 run
*----------------------------------------------------------------
*crdno input units output units
0000102 british british
*
*crdno end time min dt max dt ssd00 minor ed major ed restart
0000201 600.0 1.0-6 0.01 0003 500 10000 10000
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                                 Primary Loop
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
```

```
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 100- Lower Plenum
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
1000000 "LowPle" branch
*
*crdno number of junctions vel/flw
1000001 4 1
*
*crdno flowarea length volume
1000101 41.2 2.2 0.0
*
*crdno horz orient vert orient delta z
1000102 0.0 -90.0 -2.2
*
*till here geometry
*
*crdno roughness hyd. diameter fe
1000103 0.00015 41.2 00
*
*crdno vol ic pressure eq temp
1000200 003 1850.0 500.0
*
* Junktions lower Plenum
*
```

```
*crdno from to area f loss r loss cahs
1001101 100010001 110010001 4.9 0.0 0.0 0000
1002101 100010001 120010001 4.9 0.0 0.0 0000
1003101 500200002 100010001 12.85 0.0 0.0 0000
1004101 501200002 100010001 12.85 0.0 0.0 0000
*
*crdno f flowrate g flowrate j flowrate
1001201 647.2 0.0 0.0
1002201 647.2 0.0 0.0
1003201 647.2 0.0 0.0
1004201 647.2 0.0 0.0
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 110- Core Left
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
1100000 "Core" pipe
*
*crdno number of volumes
1100001 8
*
*crdno vol area vol.no.
1100101 4.9 8
*
*crdno length vol.no.
1100301 1.0 7
```

```
1100302 0.9 8
*
*crdno volume vol.no.
1100401 0.0 8
*
*crdno vert angle vol.no.
1100601 90.0 8
*
*crdno elev change vol.no.
1100701 1.0 7
1100702 0.9 8
*
*till here geometry
*
*crdno roughness hyd. diameter vol.no.
1100801 0.00015 4.9 8
*
*crdno Floss Revloss Jun.No
1100901 1.5 1.5 7
*
*crdno fe vol.no.
1101001 0 8
*
*crdno fvcahs jun.no.
1101101 0 7
*
*crdno vol ic pressure eq temp vol.no.
1101201 003 1850.0 590.0 0.0 0.0 0.0 8
*
```

```
*crdno makes lbm/s
1101300 1
*
*crdno f flowrate g flowrate j flowrate jun.no.
1101301 647.2 0.0 0.0 7
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 120- Core Right
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
1200000 "Core" pipe
*
*crdno number of volumes
1200001 8
*
*crdno vol area vol.no.
1200101 4.9 8
*
*crdno length vol.no.
1200301 1.0 7
1200302 0.9 8
*
*crdno volume vol.no.
1200401 0.0 8
*
*crdno vert angle vol.no.
```

```
1200601 90.0 8
*
*crdno elev change vol.no.
1200701 1.0 7
1200702 0.9 8
*
*till here geometry
*
*crdno roughness hyd. diameter vol.no.
1200801 0.00015 4.9 8
*
*crdno Floss Revloss Jun.No
1200901 1.5 1.5 7
*
*crdno fe vol.no.
1201001 0 8
*
*crdno fvcahs jun.no.
1201101 0 7
*
*crdno vol ic pressure eq temp vol.no.
1201201 003 1850.0 590.0 0.0 0.0 0.0 8
*
*crdno makes lbm/s
1201300 1
*
*crdno f flowrate g flowrate j flowrate jun.no.
1201301 647.2 0.0 0.0 7
*
```

```
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 130- Core Left Riser Conection
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
**hydro component name component type
1300000 "LeCoRiC" sngljun
*
*hydro from to area f loss r loss fvcahs
1300101 110080002 200010001 4.9 0.0 0.0 0000
*
*crdno f flowrate g flowrate j flowrate
1300201 1 647.2 0.0 0.0
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 140- Core Right Riser Conection
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*hydro component name component type
1400000 "RiCoRiC" sngljun
*
*hydro from to area f loss r loss fvcahs
1400101 120080002 200010001 4.9 0.0 0.0 0000
*
*crdno f flowrate g flowrate j flowrate
1400201 1 647.2 0.0 0.0
```

```
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 200- Lower Riser
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
2000000 "lowRise" pipe
*
*crdno number of volumes
2000001 7
*
*crdno vol area vol.no.
2000101 24.9 7
*
*crdno length vol.no.
2000301 1.0 7
*
*crdno volume vol.no.
2000401 0.0 7
*
*crdno vert angle vol.no.
2000601 90.0 7
*
*crdno elev change vol.no.
2000701 1.0 7
*
*till here geometry
```

```
*
*crdno roughness hyd. diameter vol.no.
2000801 3.0 24.9 7
*
*finish till here
*
*crdno fe vol.no.
2001001 0 7
*
*crdno fvcahs jun.no.
2001101 0 6
*
*crdno vol ic pressure eq temp vol.no.
2001201 003 1850.0 590.0 0.0 0.0 0.0 07
*
*crdno makes lbm/s
2001300 1
*
*crdno f flowrate g flowrate j flowrate jun.no.
2001301 1294.4 0.0 0.0 6
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 205- Lower Middle Riser Connection
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
**hydro component name component type
2050000 "LoMiRiC" sngljun
```

```
*
*hydro from to area f loss r loss fvcahs
2050101 200070002 210010001 24.9 0.0 0.0 030000
*
*crdno f flowrate g flowrate j flowrate
2050201 1 1294.4 0.0 0.0
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 210- Middle Riser
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
2100000 "MidRise" pipe
*
*crdno number of volumes
2100001 5
*
*crdno vol area vol.no.
2100101 24.9   1
2100102 22.525 2
2100103 20.15  3
2100104 17.775 4
2100105 15.4   5
*
*crdno length vol.no.
2100301 1.0 1
2100302 1.0 2
```

```
2100303 1.0 3
2100304 1.0 4
2100305 0.4 5
*
*crdno volume vol.no.
2100401 0.0 5
*
*crdno vert angle vol.no.
2100601 90.0 5
*
*crdno elev change vol.no.
2100701 1.0 4
2100702 0.4 5
*
*till here geometry
*
*crdno roughness hyd. diameter vol.no.
2100801 0.00015 24.9    1
2100802 0.00015 22.525  2
2100803 0.00015 20.15   3
2100804 0.00015 17.775  4
2100805 0.00015 15.4    5
*
*finish till here
*
*crdno fe vol.no.
2101001 0 5
*
*crdno fvcahs jun.no.
```

```
2101101 0 4
*
*crdno vol ic pressure eq temp vol.no.
2101201 003 1850.0 590.0 0.0 0.0 0.0 5
*
*crdno makes lbm/s
2101300 1
*
*crdno f flowrate g flowrate j flowrate jun.no.
2101301 1294.4 0.0 0.0 4
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 215- Middle Upper Riser Connection
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*hydro component name component type
2150000 "MiUpRiC" sngljun
*
*hydro from to area f loss r loss fvcahs
2150101 210050002 220010001 15.4 0.0 0.0 030000
*
*crdno f flowrate g flowrate j flowrate
2150201 1 1294.4 0.0 0.0
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 220- Upper Riser
```

```
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
2200000 "UpRise" pipe
*
*crdno number of volumes
2200001 25
*
*crdno vol area vol.no.
2200101 15.4 25
*
*crdno length vol.no.
2200301 1.0 24
2200302 0.81 25
*
*crdno volume vol.no.
2200401 0.0 25
*
*crdno vert angle vol.no.
2200601 90.0 25
*
*crdno elev change vol.no.
2200701 1.0 24
2200702 0.81 25
*
*till here geometry
*
*crdno roughness hyd. diameter vol.no.
```

```
2200801 1.5 15.4 25
*
*finish till here
*
*crdno fe vol.no.
2201001 0 25
*
*crdno fvcahs jun.no.
2201101 0 24
*
*crdno vol ic pressure eq temp vol.no.
2201201 003 1850.0 590.0 0.0 0.0 0.0 25
*
*crdno makes lbm/s
2201300 1
*
*crdno f flowrate g flowrate j flowrate jun.no.
2201301 1294.4 0.0 0.0 24
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 350- Upper Plenum
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
3500000 "UppPlen" branch
*
*crdno number of junctions vel/flw
```

```
3500001 4 1
*
*crdno flowarea length volume
3500101 41.2 1.7 0.0
*
*crdno horz orient vert orient delta z
3500102 0.0 90.0 1.7
*
*till here geometry
*
*crdno roughness hyd. diameter fe
3500103 0.00015 41.2 00
*
*crdno volic pressure eq temp
3500200 003 1850.0 590.0
*
* Junktions Upper Plenum
*
*crdno from to area f loss juncArea(r) loss cahs
3501101 220250002 350010001 15.4 0.0 0.0 0000
3502101 360010001 350010002 23.16 0.0 0.0 0000
3503101 350010001 400010001 12.85 0.0 0.0 0000
3504101 350010001 400010001 12.85 0.0 0.0 0000
*
*crdno f flowrate g flowrate j flowrate (flowrate)
3501201 1294.4 0.0 0.0
3502201 0.0 0.0 0.0
3503201 647.2 0.0 0.0
3504201 647.2 0.0 0.0
```

```
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 360- Pressurizer
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
3600000 "WGIC" branch
*
*crdno number of junctions vel/flw
3600001 1 1
*
*crdno flow area length volume
3600101 0.0 1.6110 37.306
*
*crdno horz orient vert orient delta z
3600102 0.0 90.0 1.6110
*
*crdno roughness hyd. diameter fe
3600103 0.00015 0.0 00
*
*crdno vol ic pressure quality
3600200 002 1850.0 0.7
*
* Junktions Water-Gas-Interface-Component
*
*crdno from to area f loss juncArea(r) loss cahs
3601101 360010001 350010002 23.16 0.0 0.0 0000
```

```
*
*crdno f flowrate g flowrate j flowrate
3601201 0.0 0.0 0.0
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 400- Steam Generator Primary
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
4000000 "SGPri" pipe
*
*crdno number of volumes
4000001 15
*
*crdno vol area vol.no.
4000101 25.7 15
*
*crdno length vol.no.
4000301 1.61 15
*
*crdno volume vol.no.
4000401 0.0 15
*
*crdno vert angle vol.no.
4000601 -90.0 15
*
*crdno elev change vol.no.
```

```
4000701 -1.61 15
*
*crdno roughness hyd. diameter vol.no.
4000801 0.00015 25.7 15
*
*crdno Floss Revloss Jun.No
4000901 0.995 0.995 14
*
*crdno fe vol.no.
4001001 0 15
*
*crdno fvcahs jun.no.
4001101 0 14
*
*crdno vol ic pressure eq temp vol.no.
4001201 003 1850.0 590.0 0.0 0.0 0.0 15
*
*crdno makes lbm/s
4001300 1
*
*crdno f flowrate g flowrate j flowrate jun.no.
4001301 1294.4 0.0 0.0 14
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 405- SG Downcomer Conncetion 1
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
```

```
*hydro component name component type
4050000 "SGDC1" sngljun
*
*hydro from to area f loss r loss fvcahs
4050101 400150002 500010001 12.85 0.0 0.0 0000
*
*crdno f flowrate g flowrate j flowrate
4050201 1 647.2 0.0 0.0
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                          COMPONENT 406- SG Downcomer Conncetion 2
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*hydro component name component type
4060000 "SGDC2" sngljun
*
*hydro from to area f loss r loss fvcahs
4060101 400150002 501010001 12.85 0.0 0.0 0000
*
*crdno f flowrate g flowrate j flowrate
4060201 1 647.2 0.0 0.0

$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                          COMPONENT 500- SG Downcomer 1
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
```

```
*
*crdno component name component type
5000000 "downcom1" annulus
*
*crdno number of volumes
5000001 20
*
*crdno vol area vol.no.
5000101 12.85 20
*
*crdno length vol.no.
5000301 1.0  19
5000302 0.96 20
*
*crdno volume vol.no.
5000401 0.0 20
*
*crdno vert angle vol.no.
5000601 -90.0 20
*
*crdno elev change vol.no.
5000701 -1.0  19
5000702 -0.96 20
*
*till here geometry
*
*crdno roughness hyd. diameter vol.no.
5000801 1.5 12.85 20
*
```

```
*finish till here
*
*crdno fe vol.no.
5001001 0 20
*
*crdno cahs jun.no.
5001101 0 19
*
*crdno vol ic pressure eq temp vol.no.
5001201 003 1850.0 500.0 0.0 0.0 0.0 20
*
*crdno makes lbm/s
5001300 1
*
*crdno f flowrate g flowrate j flowrate jun.no.
5001301 647.2 0.0 0.0 19
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 501- SG Downcomer 2
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
5010000 "downcom2" annulus
*
*crdno number of volumes
5010001 20
*
```

```
*crdno vol area vol.no.
5010101 12.85 20
*
*crdno length vol.no.
5010301 1.0 19
5010302 0.96 20
*
*crdno volume vol.no.
5010401 0.0 19
5010402 0.0 20
*
*crdno vert angle vol.no.
5010601 -90.0 20
*
*crdno elev change vol.no.
5010701 -1.0  19
5010702 -0.96 20
*
*till here geometry
*
*crdno roughness hyd. diameter vol.no.
5010801 1.5 12.85 20
*
*finish till here
*
*crdno fe vol.no.
5011001 0 20
*
*crdno cahs jun.no.
```

```
5011101 0 19
*
*crdno vol ic pressure eq temp vol.no.
5011201 003 1850.0 500.0 0.0 0.0 0.0 20
*
*crdno makes lbm/s
5011300 1
*
*crdno f flowrate g flowrate j flowrate jun.no.
5011301 647.2 0.0 0.0 19
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                                 Secondary Loop
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 740- Source Volume
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*hydro component name component type
7400000 "Source" tmdpvol
*
*hydro area length volume
```

```
7400101 1.0e6 0.0 1.0e+06
*
*hydro horz angle vert angle delta z
7400102 0.0 0.0 0.0
*
*hydro roughness hyd diam fe
7400103 0.0 1.0e6 000
*
*hydro ic
7400200 003
*
*hydro tdigit pressure temperature
7400201 0.0 500.0 300.0
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 750- Source Junction
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*hydro component name component type
7500000 "Tigflo" tmdpjun
*
*hydro from to area
7500101 740010002 760010001 0.785
*
*hydro vel/flw trip no. alpha vrc numeric vrc
7500200 1
*
```

```
*hydro t f flowrate g flowrate j flowrate
7500201 0.0 0.0 0.0 0.0
7500202 1.0 201.0 0.0 0.0
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 760- Feed Water Pipe
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
7600000 "WatPip" pipe
*
*crdno number of volumes
7600001 25
*
*crdno vol area vol.no.
7600101 0.785 25
*
*crdno length vol.no.
7600301 1.0 25
*
*crdno volume vol.no.
7600401 0.0 25
*
*crdno vert angle vol.no.
7600601 0.0  19
7600602 45.0 20
7600603 90.0 25
```

```
*
*crdno elev change vol.no.
7600701 0.0 19
7600702 0.5 20
7600703 1.0 25
*
*till here geometry
*
*crdno roughness hyd. diameter vol.no.
7600801 0.00015 0.785 25
*
*finish till here
*
*crdno fe vol.no.
7601001 0 25
*
*crdno fvcahs jun.no.
7601101 0 24
*
*crdno vol ic pressure eq temp vol.no.
7601201 003 500.0 300.0 0.0 0.0 0.0 25
*
*crdno makes lbm/s
7601300 1
*
*crdno f flowrate g flowrate j flowrate jun.no.
7601301 147.81 0.0 0.0 24
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
```

```
$
$                         COMPONENT 770- Feed Water Pipe SG Connection
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*hydro component name component type
7700000 "FWSG" sngljun
*
*hydro from to area f loss r loss fvcahs
7700101 760250002 401010001 0.785 0.0 0.0 0000
*
*crdno f flowrate g flowrate j flowrate
7700201 1 147.81 0.0 0.0
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 401- Steam Generator Secondary
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
4010000 "SGSec" pipe
*
*crdno number of volumes
4010001 15
*
*crdno vol area vol.no.
4010101 0.785 15
*
```

```
*crdno length vol.no.
4010301 1.61 15
*
*crdno volume vol.no.
4010401 0.0 15
*
*crdno vert angle vol.no.
4010601 90.0 15
*
*crdno elev change vol.no.
4010701 1.61 15
*
*crdno roughness hyd. diameter vol.no.
4010801 0.00015 0.785 15
*
*crdno fe vol.no.
4011001 0 15
*
*crdno fvcahs jun.no.
4011101 0 14
*
*crdno vol ic pressure eq temp vol.no.
4011201 003 500.0 300.0 0.0 0.0 0.0 15
*
*crdno makes lbm/s
4011300 1
*
*crdno f flowrate g flowrate j flowrate jun.no.
4011301 147.81 0.0 0.0 14
```

```
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 775- SG Steam Pipe Connection
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*hydro component name component type
7750000 "SGSP" sngljun
*
*hydro from to area f loss r loss fvcahs
7750101 401150002 780010001 0.785 0.0 0.0 0000
*
*crdno f flowrate g flowrate j flowrate
7750201 1 0.0 147.81 0.0
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 780- Steam Pipe
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno component name component type
7800000 "Steapi" pipe
*
*crdno number of volumes
7800001 25
*
*crdno vol area vol.no.
```

```
7800101 0.785 25
*
*crdno length vol.no.
7800301 1.0 25
*
*crdno volume vol.no.
7800401 0.0 25
*
*crdno vert angle vol.no.
7800601 90.0 5
7800602 45.0 6
7800603 0.0 25
*
*crdno elev change vol.no.
7800701 1.0 5
7800702 0.5 6
7800703 0.0 25
*
*till here geometry
*
*crdno roughness hyd. diameter vol.no.
7800801 0.00015 0.785 25
*
*finish till here
*
*crdno fe vol.no.
7801001 0 25
*
*crdno fvcahs jun.no.
```

```
7801101 0 24
*
*crdno vol ic pressure eq temp vol.no.
7801201 003 500.0 575.0 0.0 0.0 0.0 25
*
*crdno makes lbm/s
7801300 1
*
*crdno f flowrate g flowrate j flowrate jun.no.
7801301 0.0 147.81 0.0 24
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 790- Steam Pipe Sink Connection
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*hydro component name component type
7900000 "SPSIC" sngljun
*
*hydro from to area f loss r loss fvcahs
7900101 780250002 800010001 0.785 0.0 0.0 00000
*
*hydro f flowrate g flowrate j flowrate
7900201 1 0.0 147.81 0.0
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 800- Sink
```

```
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*hydro component name component type
8000000 "Sink" tmdpvol
*
*hydro area length volume
8000101 1.0e6 0.0 1.0e+06
*
*hydro horz angle vert angle delta z
8000102 0.0 0.0 0.0
*
*hydro roughness hyd diam fe
8000103 0.0 1128.378 000
*
*hydro ic
8000200 0002
*
*hydro tdigit pressure temp
8000201 0.0 500.0 0.999
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                                 Accident Valve
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
```

```
*crdno Name Type
*9550000 "Accvlv" valve
*
*hydro from to area f loss r loss fvcahs
*9550101 400080003 401080004 12.99 0.0 0.0 0
*
*crdno f flowrate g flowrate j flowrate
*9550201 1 0.0 0.0 0.0
*
*crdno Valvetype
*9550300 trpvlv
*
*crdno TripNumber
*9550301 401
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$                                     Trip
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
0000401 time 0 gt null 0 5001.  l
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                                 Heat Structures
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
```

```
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 1820- Steam Generator Tubes
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*crdno heat str-s mesh pts geometry init flag left coord
18201000 15 3 1 1 0.024
*
*crdno mesh locn flag mesh fmt flag
18201100 0 1

*crdno no.intervals rt.coord
18201101 2 0.026
*
*crdno comp no. interval no.
18201201 001 2
*
*crdno source value mesh int. no.
18201301 0.0 2
*
*crdno Initialtempflag
18201400 0
*
*crdno temperature mesh pt no.
18201401 560.0 3
*
*crdno left volume incr b.cond sa code area/factor ht str no.
18201501 400010000 10000 1 1 1350.2 15
```

```
*
*crdno rt volume incr b.cond sa code area/factor ht str no.
18201601 401150000 -10000 1 1 1350.2 15
*
*crdno s. type s. mult left heat right heat ht str no.
18201701 0 0.0 0.0 0.0 15
*
*finish till here
*
*crdno con.val
18201800 0
*
*ADD LEFT BOUNDARY CONDITION
*crdno equiv diam heated len's grid len's grid cf's bf ht strno
18201801 0.285 50.0 50.0 0.0 0.0 0.0 0.0 1.0 15
*
*crdno con.val
18201900 0
*
*ADD RIGHT BOUNDARY CONDITION
*crdno equiv diam heated len's grid len's grid cf's bf ht str no.
18201901 5.2e-2 50.0 50.0 0.0 0.0 0.0 0.0 1.0 15
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$
$                         COMPONENT 1810- Fuel Rods
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
```

```
*crdno heat str-s mesh pts geometry init flag left coord
18101000 8 2 1 1 0.0
*
*crdno MeshLocation MeshFormat
18101100 0 1
*
*crdno no. intervals rt. coord
18101101 1 0.0312
*
*crdno comp no. interval no.
18101201 005 1
*
*crdno source value mesh int. no.
18101301 1.0 1
*
*crdno Initialtempflag
18101400 0
*
*crdno temperature mesh pt no.
18101401 590.0 2
*
*crdno left volume incr b.cond sa code area/factor ht str no.
18101501 120010000 10000 1 0 392.14 8
*
*crdno rt volume incr b.cond sa code area/factor ht str no.
18101601 110010000 10000 1 0 392.14 8
*
*crdno s. type s. mult left heat right heat ht str no.
18101701 100 0.125 0.0 0.0 8
```

```
*
*finish till here
*
*crdno con.val
18101800 0
*
*ADD LEFT BOUNDARY CONDITION
*crdno equiv diam heated len’s grid len’s grid cf’s bf ht str no.
18101801  0.041 50.0 50.0 0.0 0.0 0.0 0.0 1.0 8
*
*crdno con.val
18101900 0
*
*ADD RIGHT BOUNDARY CONDITION
*crdno equiv diam heated len’s grid len’s grid cf’s bf ht str no.
18101901  0.041 50.0 50.0 0.0 0.0 0.0 0.0 1.0 8
*
*crdno Tabletype Trip Factors
20210000 power
*
*crdno t[s] P[MW] per Heat Struc.
20210001 0.0 0.0
20210002 1.0 125.0
20210003 160.0 125.0
20210004 700.0 140.0
20210005 1000.0 160.0
*
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
```

```
$
$                                 Tables
$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
*
*Thermal Propertiels for heat structures
*
*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=***
* thermal properties of ss-316l - composition 1 *
*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*
*
*crdno mtrl type th.con ht.cap material
20100100 tbl/fctn 1 1 * ss-316l
*
* thermal properties of ss-316l
*
*crdno temperature th.cond
20100101 32.0 0.00215
20100102 100.0 0.00215
20100103 800.0 0.00306
20100104 1600.0 0.00397
20100105 3600.0 0.00397
*
*crdno temperature vol ht cap
20100151 32.0 61.30
20100152 400.0 61.30
20100153 600.0 64.60
20100154 800.0 67.10
```

```
20100155 1000.0 69.35
20100156 4000.0 69.35
*
*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*
* thermal properties of cladding - composition 4 *
*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*
*
*crdno mtrl type th.con ht.cap material
20100400 tbl/fctn 1 1 * cladding
*
* thermal properties of cladding
*
*crdno temperature th.cond
20100401 32. 1.9267e-3
20100402 392.0 1.9267e-3
20100403 752.0 2.2478e-3
20100404 1112.0 2.7297e-3
20100405 1472.0 3.0508e-3
20100406 1832.0 3.5325e-3
20100407 2192.0 4.0142e-3
20100408 2552.0 4.8169e-3
20100409 2912.0 5.7803e-3
20100410 3272.0 7.0647e-3
20100411 3632.0 8.8311e-3
20100412 3992.0 1.0918e-2
*
*crdno temperature vol ht cap
20100451 0.0 28.392
20100452 1480.3 34.476
```

```
20100453 1675.00 85.176
20100454 1787.5 34.370
20100455 3500.0 34.476
*
*
*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*****
* thermal properties of uo2 - composition 5 *
*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*****
*
*crdno mtrl type th.con ht.cap material
20100500 tbl/fctn 1 1 * uo2
*
* thermal properties of uo2
*
*crdno temperature th.cond
20100501 32.0 1.284e-3
20100502 188.6 1.284e-3
20100503 332.6 1.1235e-3
20100504 440.6 9.951e-4
20100505 500.0 9.2806e-4
20100506 650.0 7.4194e-4
20100507 800.0 7.4361e-4
20100508 897.0 5.8390e-4
20100509 1104.0 5.2310e-4
20100510 1213.0 4.9200e-4
20100511 1326.0 4.6710e-4
20100512 1684.0 4.4700e-4
20100513 1700.0 4.4701e-4
20100514 1850.0 4.5528e-4
```

```
20100515 2000.0 4.3556e-4
20100516 2150.0 4.1861e-4
20100517 2300.0 4.0472e-4
20100518 2450.0 3.9306e-4
20100519 2600.0 3.8389e-4
20100520 3100.0 3.6750e-4
20100521 3600.0 3.7028e-4
20100522 4100.0 3.9056e-4
20100523 4600.0 4.2722e-4
20100524 5100.0 4.8056e-4
*
*crdno temperature vol ht cap
20100551 32.0 34.45
20100552 122.0 38.35
20100553 212.0 40.95
20100554 392.0 43.55
20100555 752.0 46.80
20100556 2012.0 51.35
20100557 2732.0 52.65
20100558 3092.0 56.55
20100559 3452.0 63.05
20100560 3812.0 72.80
20100561 4352.0 89.70
20100562 4532.0 94.25
20100563 4712.0 98.15
20100564 4892.0 100.10
20100565 5144.0 101.40
20100566 8000.0 101.40
*
```

```
*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=**
* thermal properties of fuel gap(average core) - composition 6 *
*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=*=**
*
*crdno mtrl type th.con ht.cap material
20100600 tbl/fctn 1 1 * fuel gap
*
* thermal properties of fuel gap
*
*crdno temperature th.cond
20100601 32.0 0.00031
20100602 5400.0 0.00031
*
*crdno temperature vol ht cap
20100651 32.0 0.000075
20100652 5400.0 0.000075
*
.  end of case
```

Bibliography

[1] Sebastian Buchholz, Anne Krüssenberg, Andreas Schaffrath, and Reinhard Zipper. *Studie zur Sicherheit und zu internationalen Entwicklungen von Small Modular Reactors (SMR): Abschlussbericht*, volume 376 of *GRS*. Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmBH, Köln, Mai 2015.

[2] International Atomic Energy Agency. Instrumentation and control systems for advanced small modular reactors: International atomic energy agency. *IAEA nuclear energy series*, 2017.

[3] International Atomic Energy Agency. Options to enhance proliferation resistance of innovative small and medium sized reactors: International atomic energy agency. *IAEA nuclear energy series*, 2014.

[4] John OHara. Human-performance issues related to the design and operation of small modular reactors. June 2012.

[5] U.S. GOVERNMENT. An examination of the safety and economics of light water small modular reactors. July 14, 2011.

[6] U.S. GOVERNMENT. Illustration of a light water small modular nuclear reactor (smr), 24. August 2015.

[7] Nuclear technology basics: Part 8 liquid metal cooled reactors: http://neurovoresnuclearnetworknews.blogspot.com/2011/07/part-8-liquid-metal-cooled-reactors.html, 2011.

[8] World Nuclear Association. Small nuclear power reactors - world nuclear association, November 2017.

[9] Molten salt reactors: https://steemit.com/science/@anarchyhasnogods/molten-salt-reactors, 2017.

[10] Hans-Dieter Berger. Nukleare energietechnik i: Umdruck und vorlesung. *Technische Universität Braunschweig*, 2016.

[11] Robert Howard Shannon and Henry C. Schwenk. *Nuclear Power Engineering*. 2012.

[12] John R. Lamarsh and Anthony John Baratta. *Introduction to Nuclear Engineering*. Pearson Education, Inc, Hoboken, NJ, fourth edition edition, 2018.

[13] CLP Group. Pressurised water reactor vs boiling water reactor: https://www.clpgroup.com/nuclearenergy/eng/power/power4_1_2.aspx, 2013.

[14] Thomas Schabbach and Viktor Wesselak. *Energie: Die Zukunft wird erneuerbar*. SpringerLink Bücher. Springer Vieweg, Berlin and Heidelberg, 2012.

[15] Dieter Smidt. *Reaktor-Sicherheitstechnik*. Springer Berlin Heidelberg, Berlin, Heidelberg, 1979.

[16] Paul Laufs. *Reaktorsicherheit für Leistungskernkraftwerke: Die Entwicklung im politischen und technischen Umfeld der Bundesrepublik Deutschland*. Springer, Berlin and Heidelberg, 2013.

[17] Günter Kessler, Anke Veser, Franz-Hermann Schlüter, Wolfgang Raskob, Claudia Landman, and Jürgen Päsler-Sauer. *Sicherheit von Leichtwasserreaktoren: Risiken der Nukleartechnologie*. Springer, Berlin and Heidelberg, 2012.

[18] The Asia Pacific Journal. Out of the shadows: Can japan's nuclear power renaissance reduce global emissions?, September 1, 2008.

[19] Jose Reyes. Nuscale plant safety in response to extreme events. *Nuclear Technology*, 2012.

[20] NuScale Power. Smr nuclear technology: http://www.nuscalepower.com/, 2007.

[21] Vundela Siva, Reddy, Subash Chndra, Kaushik, Sudhir Kumar, Tyagi, Narayanlal, and Panwar. An approach to analyse energy and exergy analysis of thermal power plants: A review, 2010.

[22] Hans Dieter Baehr and Stephan Kabelac. *Thermodynamik: Grundlagen und technische Anwendungen*. SpringerLink Bücher. Springer Vieweg, Berlin, Heidelberg, 15. auflage edition, 2012.

[23] Bernhard Weigand, Jürgen Köhler, and Jens von Wolfersdorf. *Thermodynamik kompakt*. Springer-Lehrbuch. Springer Berlin Heidelberg, Berlin, Heidelberg and s.l., 3., aktual. aufl. 2013 edition, 2013.

[24] Karl Strauß. *Wärmekraftwerke: Von den Anfängen im 19. Jahrhundert bis zur Endphase ihrer Entwicklung*. Springer Vieweg, Berlin and Heidelberg, 2016.

[25] Heinz Pitsch. Thermodynamik i rwth aachen: Kapitel 5, teil 1. 2012.

[26] Carl Pfleiderer and Hartwig Petermann. *Strömungsmaschinen*. Klassiker der Technik. Springer, Berlin, 7., unveränd. aufl. edition, 2005.

[27] The RELAP5-3D© Code Development Team. Relap5-3d© code manuel vol-

ume i: Code structure, system models, and solution methods: Volume i. *Idaho National Laboratory*, April 2005.

[28] The RELAP5-3D© Code Development Team. Relap5-3d© code manual volume ii: User's guide and input requirements: Volume ii. *Idaho National Laboratory*, April 2005.

[29] The RELAP5-3D© Code Development Team. Relap5-3d© code manual volume iv: Models and correlations: Volume iv. *Idaho National Laboratory*, February 2001.

[30] SCDAP/RELAP5-3D© Code Development Team. Scdap/relap5-3d© code manual volume iii: User's guide and input manual: Volume iii. *Idaho National Engineering and Environmental Laboratory*, October 2003.

[31] R. R. Schultz. Relap5-3d© code manual volume v: User's guidelines: Volume v. July 2002.

[32] U.S. Nuclear Regulatory Commission. Development of a computer tool for in-depth analysis and post processing of the relap5 thermal hydraulic code. April 2011.

[33] Christopher Earls Brennen. *Thermo-Hydraulics of Nuclear Reactors*. Cambridge University Press, Cambridge, 2016.

[34] Mamoru Ishii and Takashi Hibiki. *Thermo-Fluid Dynamics of Two-Phase Flow*. Smart energy systems–nanowatts to terawatts. Springer Science+Business Media Inc, Boston, MA, 2006.

[35] Hiroyasu MOCHIZUKI. Thermal-hydraulics in nuclear reactors: International graduate course tokyo institute of technology. October 2009.

[36] Mamoru Ishii. *Investigation of Natural Circulation Instability and Transients in Passively Safe Small Modular Reactors: Reactor Concepts Research Development and Demonstration*. 2016.

[37] Yoram ZVIRIN. A review of natural circulation loops in pressurized water reaktors and other systems. *Electric Power Research Institute*, 1981.

[38] International Atomic Energy Agency. Natural circulation in water cooled nuclear power plants: Phenomena, models, and methodology for system reliability assessments. November 2005.

[39] Japan Atomic Energy Agency. Residual heat removal using buoyancy force: http://jolisfukyu.tokai-sc.jaea.go.jp/fukyu/review_en/2011/1_3.html, 2011.

[40] Andreas Bott. *Synoptische Meteorologie: Methoden der Wetteranalyse und -prognose*. Springer, Berlin, 2012.

[41] Albert Ziegler and Hans-Josef Allelein. *Reaktortechnik: Physikalisch-technische Grundlagen*. Springer, Berlin, 2., beu bearb. aufl. 2013 edition, 2013.

[42] Herbert Oertel, Martin Böhle, and Ludwig Prandtl. *Prandtl - Führer durch die Strömungslehre: Grundlagen und Phänomene*. Springer Fachmedien Wiesbaden, Wiesbaden and s.l., 13., überarb. aufl. 2012 edition, 2012.

[43] NuScale Inc. Final safety analysis report - part 02 - tier 02 - chapter 05 - reactor coolant system and connecting systems. 2016.

[44] NuScale Inc. Final safety analysis report - part 02 - tier 02 - chapter 10 - steam and power conversion system. 2016.

[45] NuScale Inc. Final safety analysis report - part 02 - tier 02 - chapter 04 - reactor. 2016.