TRACE Report of Indian Unit 2 Reactor

Nuc E 470 Final Project

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Abstract

Safety and smooth operation of a nuclear plant are of the most importance to nuclear engineers. One of the biggest areas of study in nuclear engineering is simulating nuclear plants to run thermo-hydraulic analysis on a reactor based on real data. These simulations allow engineers to study different scenarios of a reactor and the effect it has on the reactor system. For this PWR simulation, Symbolic Nuclear Analysis Package (SNAP) software and TRACE code are used to model the reactor core and secondary components of the core. When modeling a PWR, the main goal is to model the reactor at a steady state and transient state above normal power to provide an accurate analysis that can help to prevent certain accident scenarios from happening. For this analysis, the Indian Point Unit 2 was used to provide real plant specifications for the model. The model included major components of a PWR reactor for analysis including the reactor core, hot and cold legs of the coolant system, pressurizer, pumps and steam generators. All components were modeled separately and combined to simulate a full PWR reactor model. A control system was implemented to simulate a reactor system feed water inlet and turbine power output. The model was ran to find where the model reached steady state and conditions for the plant were then compared to actual plant specifications for Indian Point Unit 2 from a Final Safety Analysis Report (FSAR). Our model produced 605.111 MW from a 4.25 GW reactor yielding an efficiency of 14%.

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Introduction

Reactor safety is one of the most important aspects of nuclear engineering. Current nuclear reactor systems are so complex that an understanding of basic reactor theory and reasoning is not enough to fully understand a system's response to different scenarios. To be cost-effective with reactor safety, reactors are modeled with sophisticated thermo-hydraulics codes in order to observe the behavior of different components of a reactor. This allows engineers to view aspects of the reactor in real and transient scenarios, which helps engineers to make decisions for safety and operation of a plant. Thermo-hydraulics of a reactor is really important to reactor safety and operation as the flow of coolant through a reactor plays a pivotal role in creating steam for power to the turbine and cooling of the reactor system. Changes in pressure, temperature and mass flow can cause major problems for a reactor system, so these parameters are heavily monitored during reactor simulations.

Pressurized water reactors (PWR) is a type of nuclear reactor that uses a highly pressurized primary coolant to transfer heat from the core of the plant to create steam, which is then converted into mechanical energy via a turbine. A pressurizer keeps the primary coolant at a high pressure to prevent the coolant from boiling because it's circulated at very high temperatures. The primary coolant is circulated through the core using pumps to collect heat from the core. The primary coolant is at a heightened temperature and travels through the "hot leg" of the plant to reach the steam generators. The steam generators create steam by transferring the heat from the primary coolant to a secondary fluid, where it is converted into mechanical energy via a turbine that is connected to an electric generator. The primary coolant is then circulated back through the "cold legs" of the plant and repeats the previous cycle.

In order to simulate a PWR, nuclear analysis software is needed to model both the neutronics and thermo-hydraulics of the reactor system. For this particular simulation, Symbolic Nuclear Analysis Package software was used for the input geometry and initial conditions for the reactor that can be used for starting input for TRACE. The data for this scenario is obtained from actual specifications of the Indian Point Unit 2 reactor. This data was used as a starting point for modeling all components of the reactor. In order to calculate the power for the turbine output, a control system was implemented. The goal of this simulation was to model a PWR reactor, bring it to a steady state and create a transient of 5% change in thermal power. The data obtained from the model's baseline steady state and transient are compared to actual plant specifications of the Indian Point Unit 2 reactor from its FSAR. A Richard extrapolation analysis is performed on an isolated steam generator system for the error using steam mass flow rate.

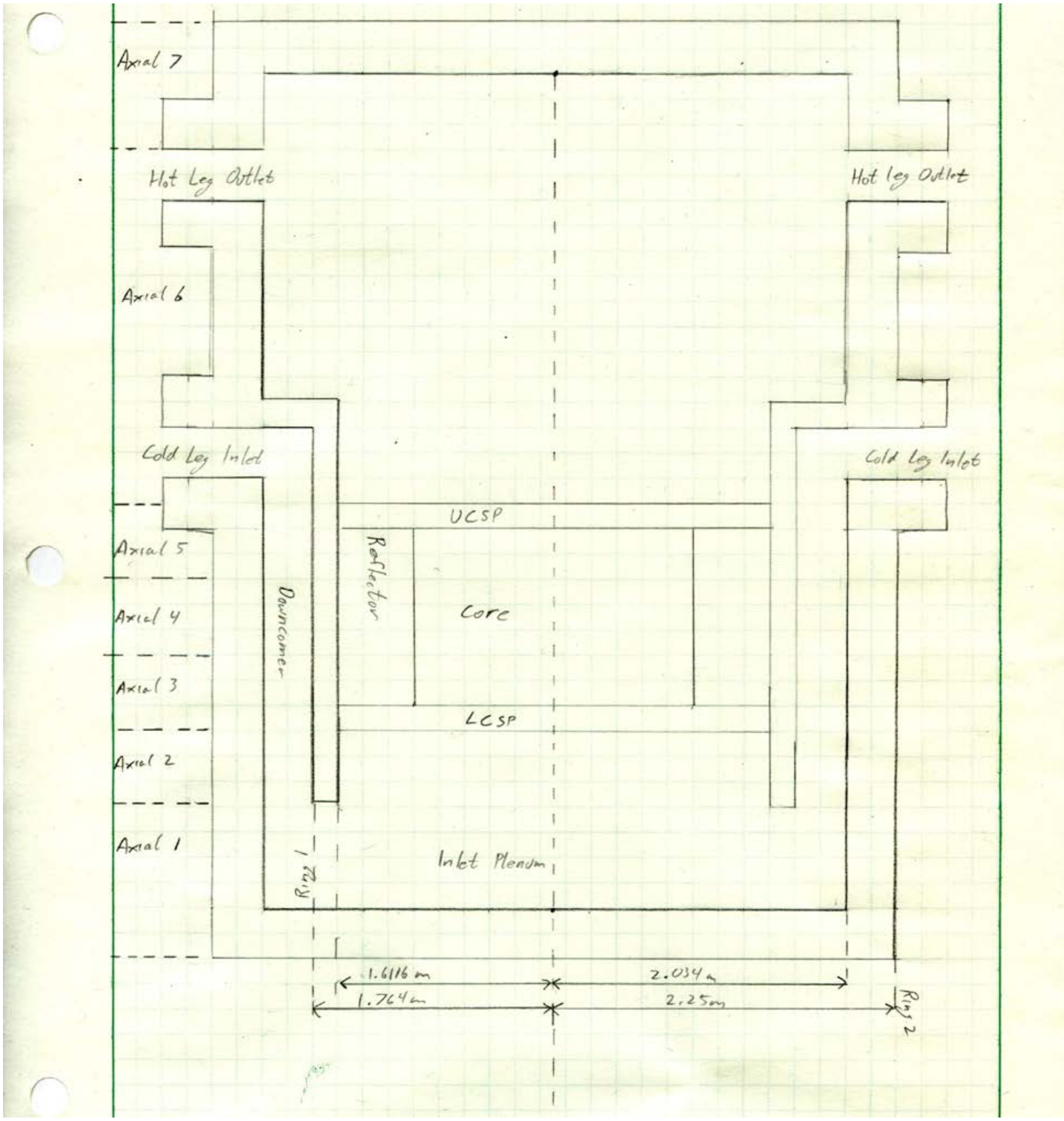

Figure 1. Reactor Sketch

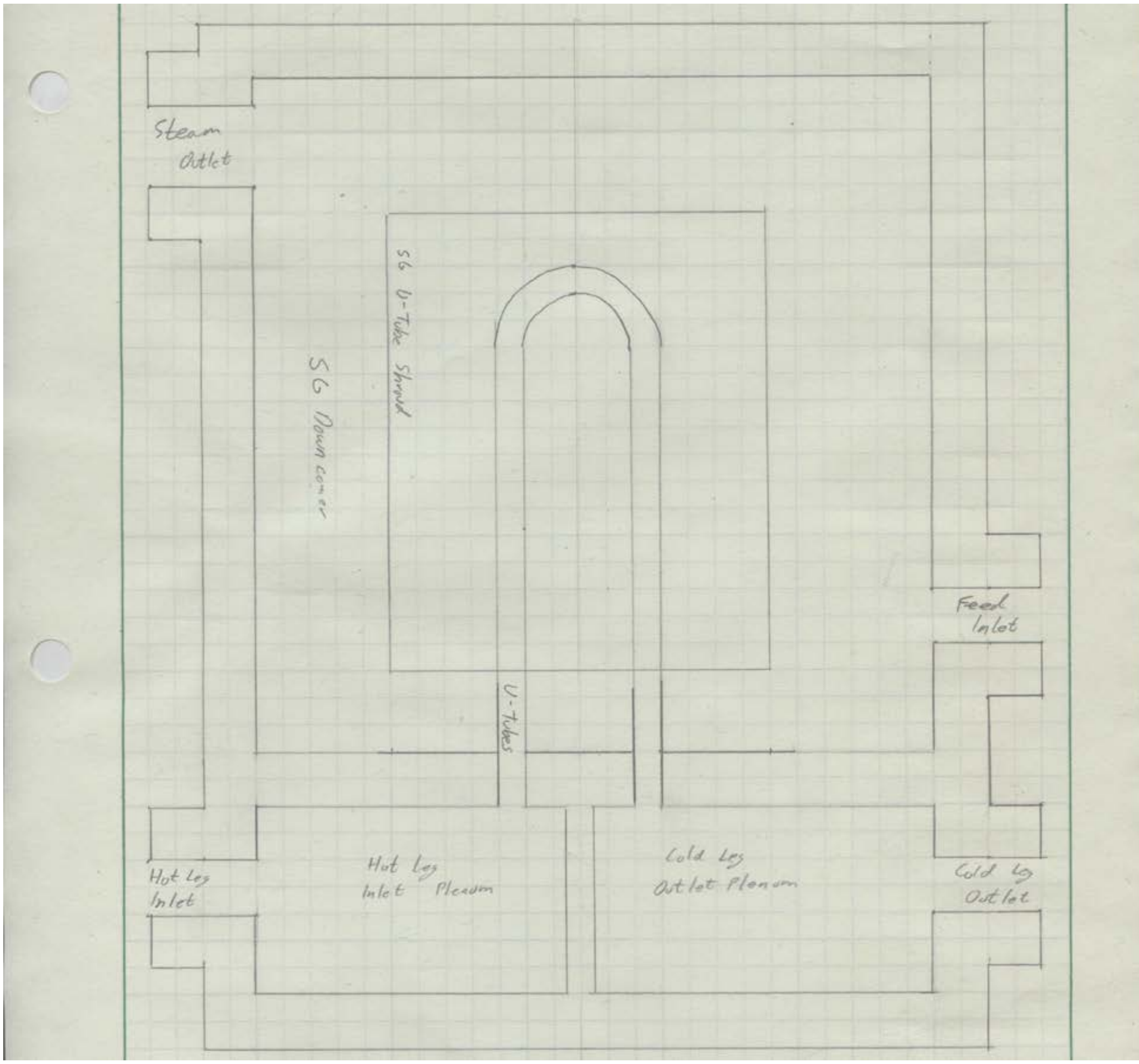

Figure 2. Steam Generator Sketch

Development of Core Model

Steam Generator

Modeling of the Reactor started with the steam generators. Using specifications of the Indian Point Unit 2 Reactor and an input file was used as a base for the steam generator. In this plant model, 4 U-tube style steam generators were used. Each steam generator is assumed identical to each other with respect to initial geometries and input values. The input file used for the basic model of the steam generator is stgenSS3.inp. The components considered in the steam generators are the primary U-tubes, secondary side boiler and steam dome, downcomer and the inlet connection for feedwater. In order to simplify calculations for heat transfer and fluid flow, changes to the geometry were made to create a simpler model.

Parameter	Given		Range	As Built	
Core Thermal Power	4,250	MW	0.0175-.0222	4,250	MW
Net Electrical Power	1,450	MW	.00105-.0127	1,450	MW
Efficiency	34	%	9-11%		
Hot Leg Temperature	603	K	.68-.87	603	K
Cold Leg Temperature	565	K	.68-.78	557.7	K
RCS Mass Flow per Loop	4780	kg/s	3000-4000	4551	kg/s
Primary System Pressure	15513203	Pa		15542569	Pa
Steam Pressure	7101600	Pa		5092596	Pa
Steam Flow Rate per Loop	618	kg/s		588.4	kg/s
Pressurizer Volume	75	m^3		75	m^3
Number of Fuel Assemblies	205			205	
Fuel Lattice	17 x 17			17 x 17	
Active Fuel Length	4.27	m		4.27	m
Rods Per Assembly	264			264	
Average Linear Heat Rate	17.913	kW/m		17.913	kW/m
Number of Control Rods	73			73	

Table 1. Key Reactor Design Parameters

The table below displays values that were selected to be within the standard measurements of the parameters listed in the FSAR for Indian Point Unit 2.

Selected Values For Primary Side			
Tube Outer Diameter	$D_{tubeouter}$	0.0222	m
Tube Wall Thickness	$t_{tubewall}$	0.001	m
Height of Tube Bundle	H_{tube}	11	m
Hot Leg Inner Diameter	$D_{innerhotleg}$	0.7	m
Cold Leg Inner Diameter	$D_{innercoldleg}$	0.7	m
Number of SG Tubes	N_{tubes}	3999	
P.S. Inlet Temperature	T_{hot}	603	K
P.S. Outlet Temperature	T_{cold}	565.7	K
P.S. Pressure	$P_{primary}$	15513203	Pa
SG P.S. Flow Rate	$\dot{M}_{primary}$	4780	kg/s

Table 2. Selected Values for S/G Primary.

Here are equations used to find the selected values above for the primary side of the steam generator.

$$D_{tubeinner} = D_{tubeouter} - 2 * t_{tubewall} = .0202m$$

$$FA_{inletplenum} = \pi * \frac{D_{innerhotleg}^2}{4} = .384845\ m^2$$

$$FA_{exitplenum} = \pi * \frac{D_{innercoldleg}^2}{4} = .384845\ m^2$$

$$VAFA_{plenum} = 3 * FA_{inletplenum} = 1.154535\ m^2$$

$$V_{plenumtotal} = VAFA_{plenum} * H_{plenum} = 1..154535\ m^3$$

$$L_{avgtube} = 2 * H_{tube} + 1 = 23\ m$$

$$D_H(primary) = \pi * \frac{D_{innerhotleg}^2}{4} = .384845\ m^2$$

$$P_w = \pi * D_{inner-tube} * N_{tubes} = 253.777\ m$$

$$FA_{SG-innertubes} = \pi * \frac{D_{inner-tube}^2}{4} = .00032\ m^2$$

$$A_{SG-innersurface} = \pi * D_{tubeinner} * L_{avgtube} = 1.459584\ m^2$$

$$V_{tube-total} = N_{tubes} * FA_{SG-innertubes} * L_{avgtube} = 29.47622\ m^3$$

Selected Values For Secondary Side			
SG Overall Height	H_{SG}	20	m

Feedwater Inlet Diameter	$D_{feed-inlet}$	0.364	m
Height of Downcomer	$H_{downcomer}$	11	m
Lower Shell OD	$D_{lowershell-outer}$	3	m
Lower Shell Thickness	$t_{lowershell}$	0.0668	m
Upper Shell OD	$D_{uppershell-outer}$	4	m
Upper Shell Thickness	$t_{uppershell}$	0.0889	m
Feedwater Temperature	T_{feed}	503.15	kg/s
Barrier Thickness between downcomer	$t_{barrier}$	0.0668	m
S.S. Pressure	$P_{secondary}$	7101600	Pa
SG S.S. Flow Rate	$\dot{M}_{SG}$	618	kg/s
Downcomer Thickness	$t_{downcomer}$	0.7	m
Distance Between tubes	Pitch	0.001	m

Table 3. Selected Values for S/G Secondary

Here are equations used to find the selected values above for the secondary side of the steam generator.

$$D_{lowershell-inner} = D_{lowershell-outer} - (2 * t_{lowershell}) = 2.8664\ m$$

$$D_{uppershell-inner} = D_{upper-outer} - \left(2 * t_{uppershell}\right) = 3.8222\ m$$

$$D_{boiler-outer} = D_{lowershell-outer} - 2 * t_{lowershell} - 2 * t_{downcomer} = 1.4664\ m$$

$$D_{boiler-inner} = D_{boiler-outer} - 2 * t_{barrier} = 1.3328\ m$$

$$P_{w-SS} = 2 * N_{tubes} * \pi * D_{tube-outer} = 557.8074\ m$$

$$D_{H-boiler} = 4 * \frac{FA_{boiler}}{P_{w-SS}} = .022747m$$

$$D_{H-downcomer} = \frac{\pi * (D_{boiler-outer} - D_{lowershell-inner})^2}{\pi(D_{boiler-outer} + D_{boiler-inner})} = .452363\ m$$

$$D_{H-feed} = D_{feed-inlet} = .364\ m$$

$$P_{w-downcomer} = \pi(D_{boiler-outer} + D_{lowershell-inner}) = 13.61189\ m$$

$$A_{SG-outersurface} = \pi * D_{tubeouter} = .069743\ m^2$$

$$A_{boilerregion} = 2 * D_{tubeouter}^2 * N_{tubes} = 3.941734\ m^2$$

$$FA_{boiler-region} = A_{boilerregion} - 2 * FA_{exitplenum} = 3.172044\ m^2$$

$$A_{lowershell} = \pi * \frac{D_{lowershell-outer}^2}{4} = 7.068583\ m^2$$

$$FA_{downcomer} = A_{lowershell} - A_{boilerregion} = 3.126849\ m^2$$

Reactor Core/Vessel

The reactor vessel houses the core of the reactor. Since the reactor vessel is such a complex structure, it is modeled in three dimensions. The reactor vessel is modeled with axial, radial and azimuthal dimensions considered. The input file used for the base of this component was reactorCore.inp. The core has many structures, such as fuel rods and structural components that can obstruct flow through the vessel, so an effective flow area was used to account for these. HEAT STRUCTURE component was used to model components in the core including the rods, thermal shield, vessel wall and core basket. In order to model this component separately from the steam generators and other pipes, fills and breaks were used to simulate flow in and out of the vessel.

Vessel and Core Table Data

Vessel OD	4.5	m
Vessel Wall Thickness	0.216	m
Vessel ID	4.068	m
Down comer Width	0.27	m
Core Barrel Thickness	0.1524	m
Reflector Thickness	0.381	m
Fuel Rod Diameter	0.0094996	m
Fuel Rod Cladding Thickness	0.0005588	m
Fuel Rod Gas Gap Thickness	0.0001905	m
Control Rod Diameter	0.0096774	m
Holes in LCSP	80	-
LCSP Hole Diameter	0.3048	m
Holes in UCSP	80	-
UCSP Hole Diameter	0.3048	m
Active Fuel Length	4.27	m
Inlet Temperature	557.7	K
Outlet Temperature	603	K
Primary Side Pressure	15513203	Pa
Flow Rate per Hot Leg	4780	Kg/s
Number of Fuel Rods per Assembly	264	-
Number of Fuel Assemblies	205	-
Loss Coeff. For LCSP	0.9999967	-
Loss Coeff. For UCSP	0.9999967	-
Surface area of Control Rods	0.1298185	m^2
Fuel Pellet Diameter	0.008001	m
Core Area Diameter	3.2232	m
Area of Core Region	8.159515845	m^2
Vessel ID	4.068	m
Vessel Wall Thickness	0.27	m
Vessel ID	4.068	m
Dh of Core Barrel	0.013	m
Dh of Core	0.013	m

Dh of Downcomer	0.30483109	m
Downcomer Pw	23.863538	m
Core Pw	7.068583471	m
Flow Area of LCSP	5.8372702	m^2
Core Flow Area	1.1517717	m^2
Flow Area Downcomer	1.18158706	m^2
Diameter of Core Basket Outer	2.52	m
Diameter of Core Basket Inner	2.25	m

Table 4. Vessel and Core Data

Here are equations used to find the selected values above for the core vessel.

$$Vessel\ ID = Vessel\ OD - (2\ x\ Vessel\ Wall\ Thickness)$$

$$SA_{Control\ Rods} = \pi\ D_{CR} L_{CR}$$

$$D_{Pellet} = D_{FuelRod} - \big(2 * (Cladding\ Thickness)\big) - (2 * (Gas\ Gap\ Thickness)))$$

$$A_{Core} = \frac{\pi}{4} D_{Core}^2$$

$$D_{Core} = \left(\pi \left(\frac{A_{Core}}{4}\right)\right)^{1/2} \text{*Area was found using model}$$

$$PW_{Down} = \left(\pi D_{ID,Vessel}\right) + (D_{ID,Vessel} - (2W_{Down})$$

$$FA_{LCSP} = \left(N_{Holes,LCSP} * D_{Hole}^2 * \frac{\pi}{4}\right)$$

$$FA_{Core} = \left(\frac{\pi}{4} * \left(D_{CB,Outer}^2\right) - \left(D_{Fuel}^2 - (N_{Fuel} * N_{Assemblies})\right)\right)$$

$$D_{H,Down} = \left(4 \frac{FA_{Down}}{PW_{Down}}\right)$$

$$A_{USCP} = \frac{\pi}{4} N_{Holes,USCP} D_{USCP}^2$$

$$Loss\ Coefficient\ for\ LCSP = 0.4 \left(1 - \left(\frac{A_{Core}}{A_{USCP}}\right)^2\right)$$

$$Loss\ Coefficient\ for\ UCSP = \left(1 - \left(\frac{A_{USCP}}{A_{Rod}}\right)^2\right)$$

Pressurizer

The pressurizer was modeled using the PRIZER component. The hot and cold legs were modeled using PIPE components as tests for temperatures coming into and out of the reactor core. For the pipe connected to the pressurizer, a tee pipe was created to connect to both the core and the pressurizer.

Reactor Coolant System Piping Data		
	Limitations	As built
Cold Leg ID	.6858-.762 m	0.7087 m
Hot Leg ID	.6858-.762 m	0.7087 m
Crossover Leg ID	.6858-.762 m	0.7087 m
Length of Cold Leg	10 diameter	7.087 m
Length of Hot Leg	10 diameter	7.087 m
Length of Crossover Leg	20 diameter	14.174 m
Pressurizer ID	2-3 m	3.09 m
Pressurizer Heater Power		1.80E+06 W
Surge Line Length	10 m	10 m
Surge Line ID	0.3556 m	0.3556 m
Pressurizer Volume		75 m^3
Pressurizer Height		10 m
Reactor Coolant Pump Flow Area		1.5708 m^2
Reactor Coolant Pump Dh		0.7 m

Table 5. Reactor Coolant Piping Data

Parameter	Given Parameter	Units
Hot Leg Cross Section Area	0.3945089	m^2
Hot Leg inner diameter	0.7087343	m
Hot Leg Length	7.0873434	m
Cold Leg Cross Section Area	0.3945089	m^2
Cold Leg inner diameter	0.7087343	m
Cold Leg Length	7.0873434	m
Crossover Leg ID	0.7087343	m
Crossover Leg Length	14.174687	m

Surge Line Length	10	m
Surge Line ID	0.3556	m
Surge Line Flow Area	0.0993147	m^2
PZR Volume	75	m^3
PZR Height	10	m
PZR ID	3.0901936	m
PZR Inner Radius	1.5450968	m
PZR Heater Power	1.80E+06	W
PZR Pressure Setpoint	15531203	Pa
Delta P	20	Pa

Table 6. Combined Global Data

Pump

In order for the model to be complete, pumps were needed to drive the primary coolant through the system at a certain pressure. The steam generators have coolant exiting the primary side and flowing through a cross over leg into the pump. The pumps are connected to cold legs where the coolant is pushed through the core. The model includes four pumps that are identical in certain rated values shown in the table.

The pumps were tested separately from the main model in a separate model to ensure that no problems arose during their operation. Fills and breaks were connected to the pumps via cross over legs where the fill used parameters of coolant coming out of steam generator and the break used conditions of fluid going into cold leg/core.

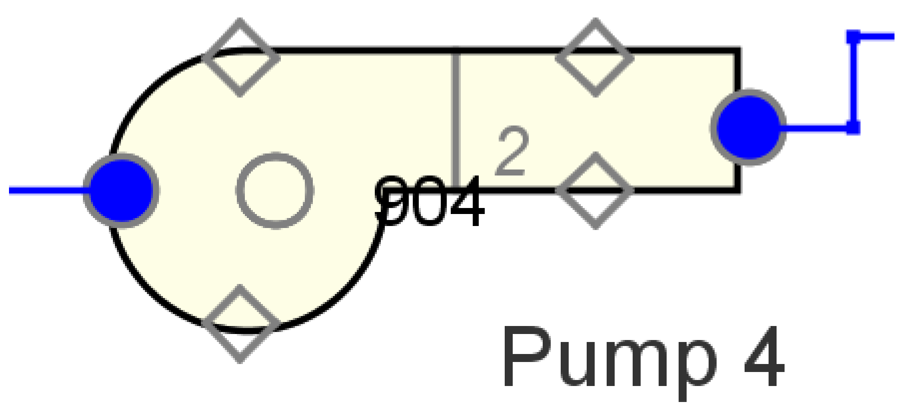

Figure 3. Pump Detail

RCP MOI	3455 $kg * m^2$
RCP H_r	831 m^2/s^2
RCP T_r	35949 N-m
RCP Q'''	5.651 m^3/s
RCP ρ_r	745 kg/m^3
RCP ω_r	124.5 rad/s

Table 7. Reactor Coolant Pump Data

Turbine

In order to model the turbine, a control system was created to calculate the work of the turbine output because the process in the actual turbine is not of significant importance for this simulation. In order to calculate work produced by the equation below.

$$\dot{W}_{turbine} = \dot{m}\,\Delta h$$

The work of the turbine is calculated using the inlet and outlet streams energy traveling through the turbine. The equation reads as the total mass flow multiplied by the change in enthalpy.

Results of Reactor Components

Steam Generators

In the steady state model, the outlet temperature of our steam generator fluctuates before finally converging to approximately 557 Kelvin at 400 seconds as shown in figure 2.

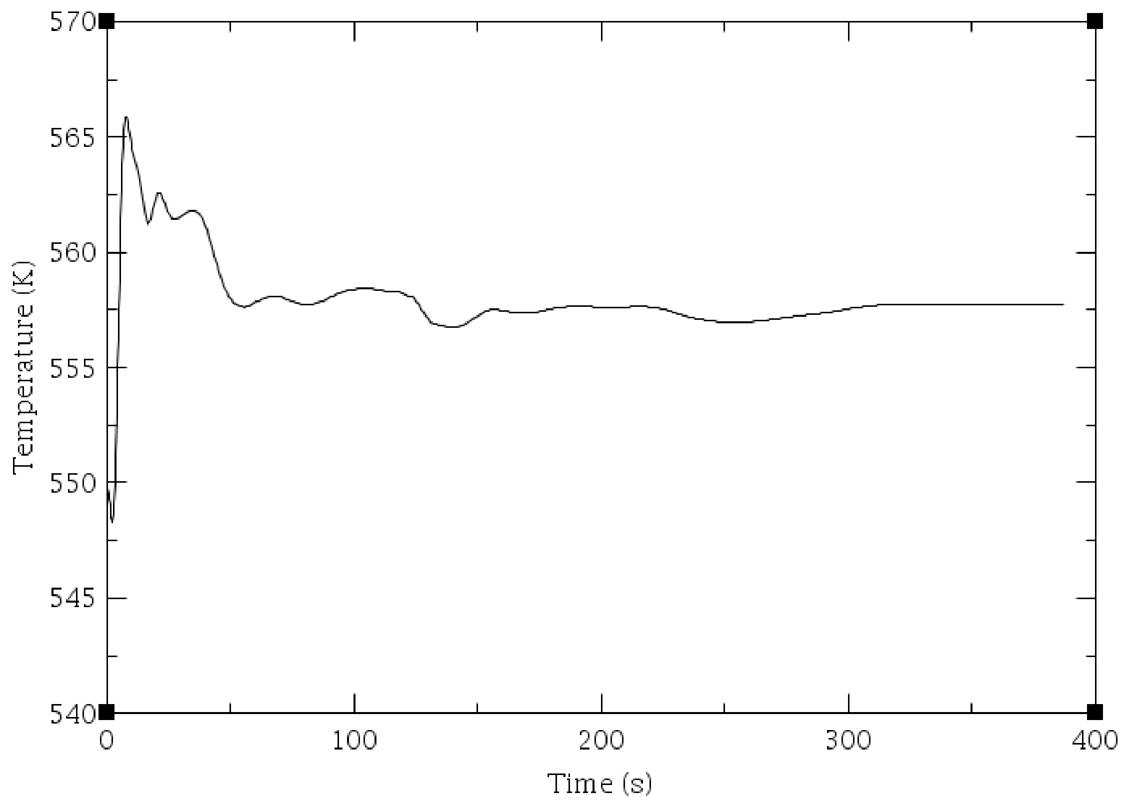

Figure 4. Cold Leg Temperature

The mass flow rate of our steam generator also varies greatly before reaching a consistent value of 557.7 kg/s as shown in figure 4.

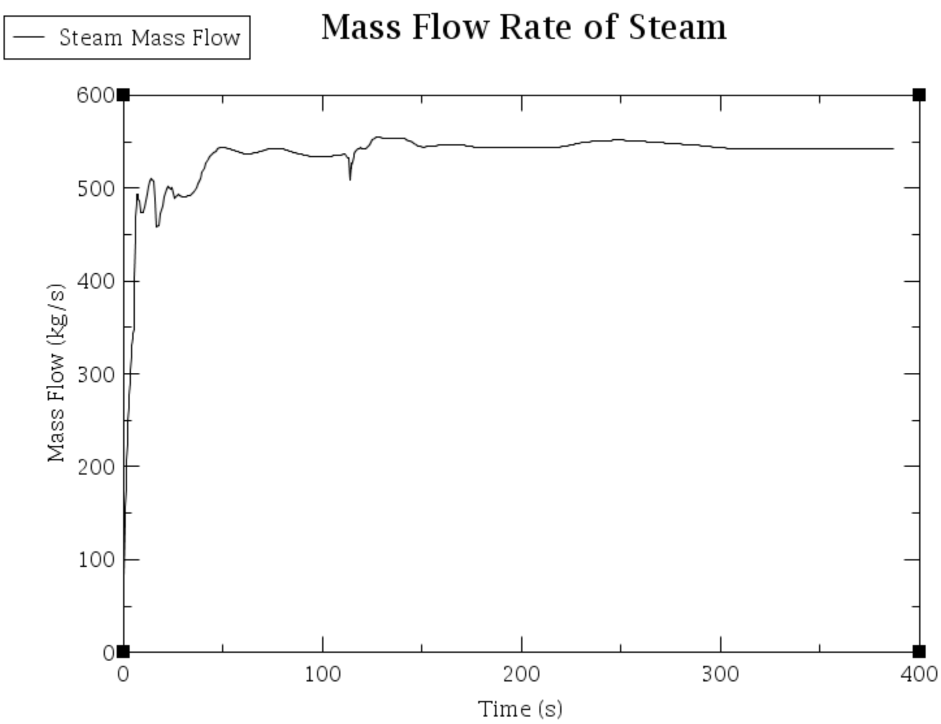

Figure 5. Mass Flow Rate of Steam in Secondary Plant.

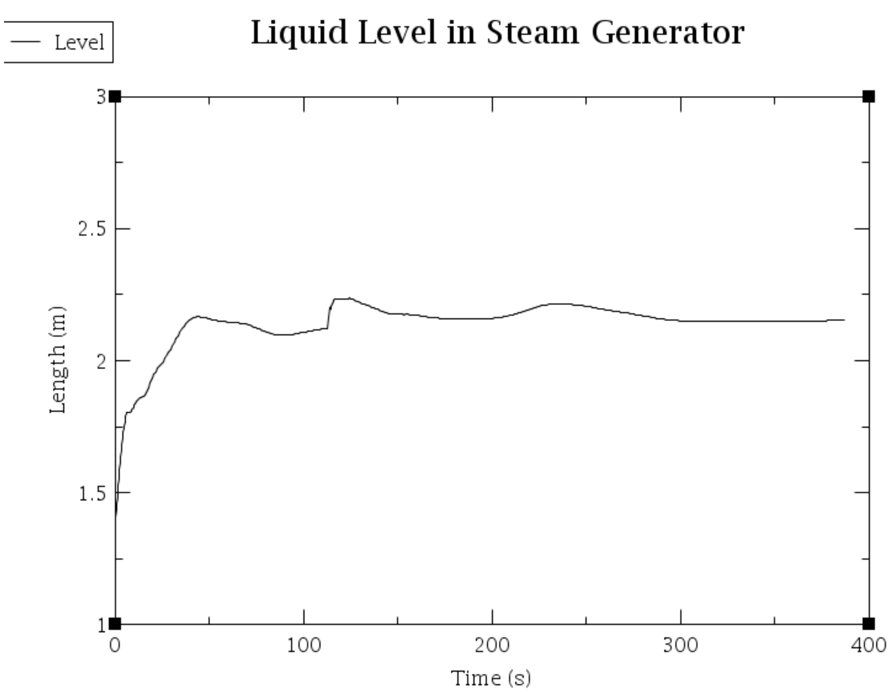

Figure 6. Steam Generator Liquid Level.

Reactor Core/Vessel

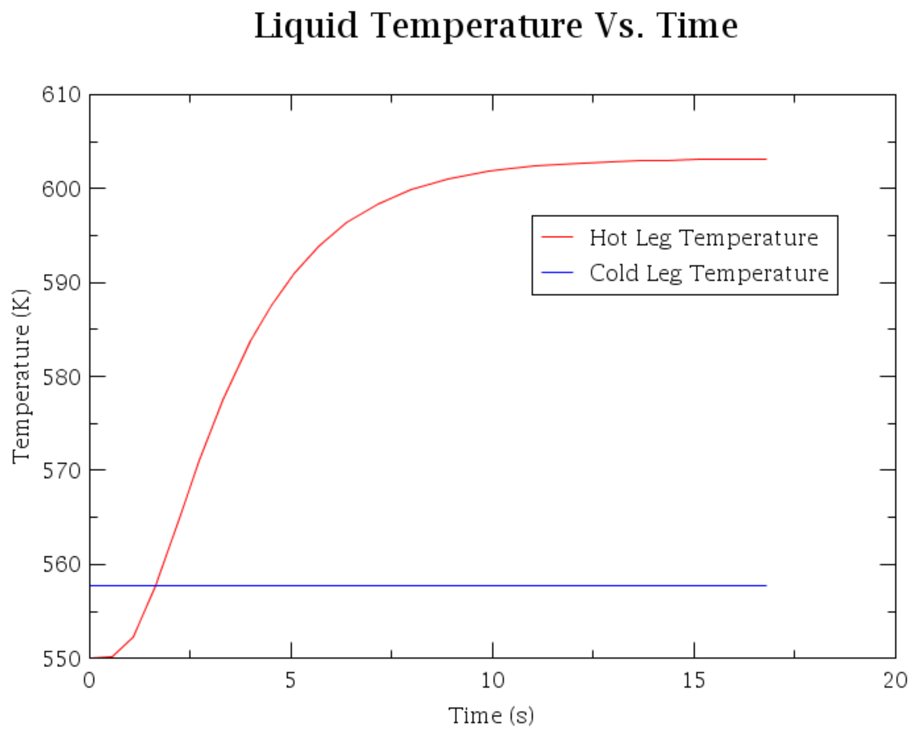

Figure 7. Coolant Temperature before Transient.

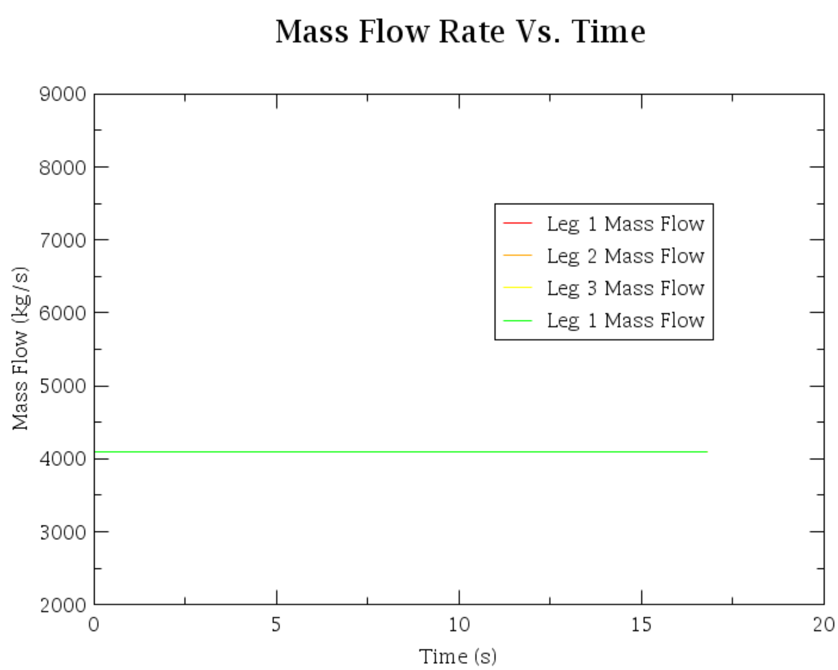

Figure 8. Coolant Mass Flow Rate before Transient.

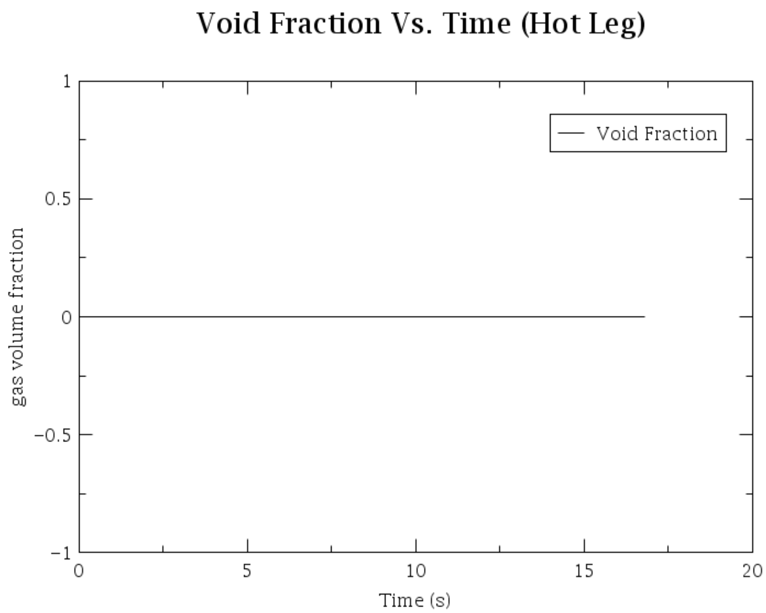

Figure 9. Reactor Coolant Void Fraction before Transient.

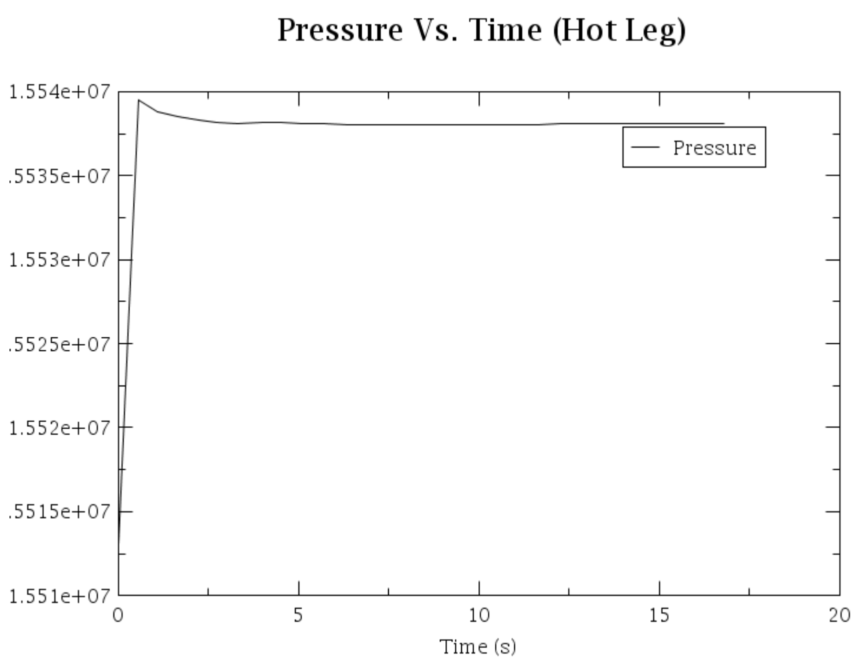

Figure 10. Primary Pressure before Transient.

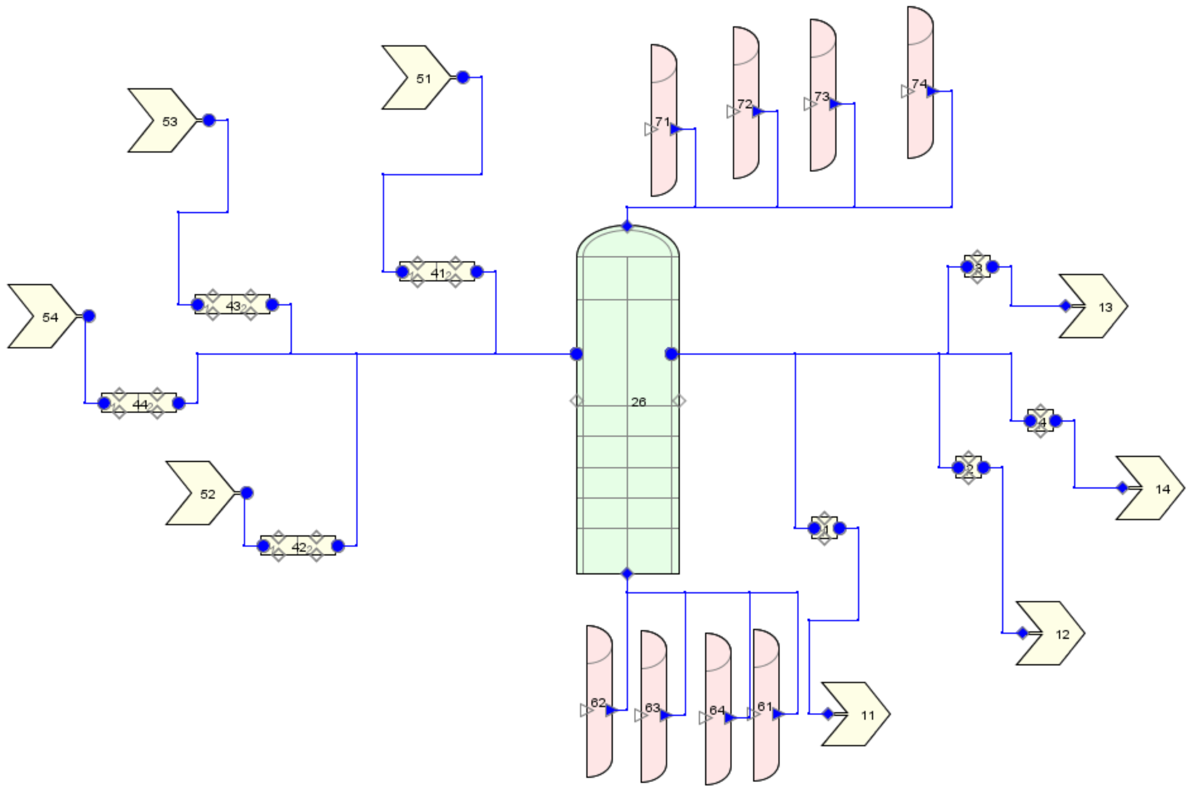

Figure 11. Steady State Core Model.

Pressurizer

The pressurizer (PZR) is designed to maintain constant pressure throughout the entire primary side of the reactor. This is achieved using heater elements to heat the water in the pressurizer when pressure drops, and initiates spray flow when pressure is too high (the latter is not modeled in TRACE as it was beyond the scope of this project).

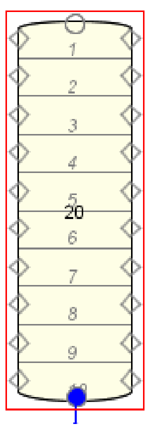

Figure 12. Pressurizer Model.

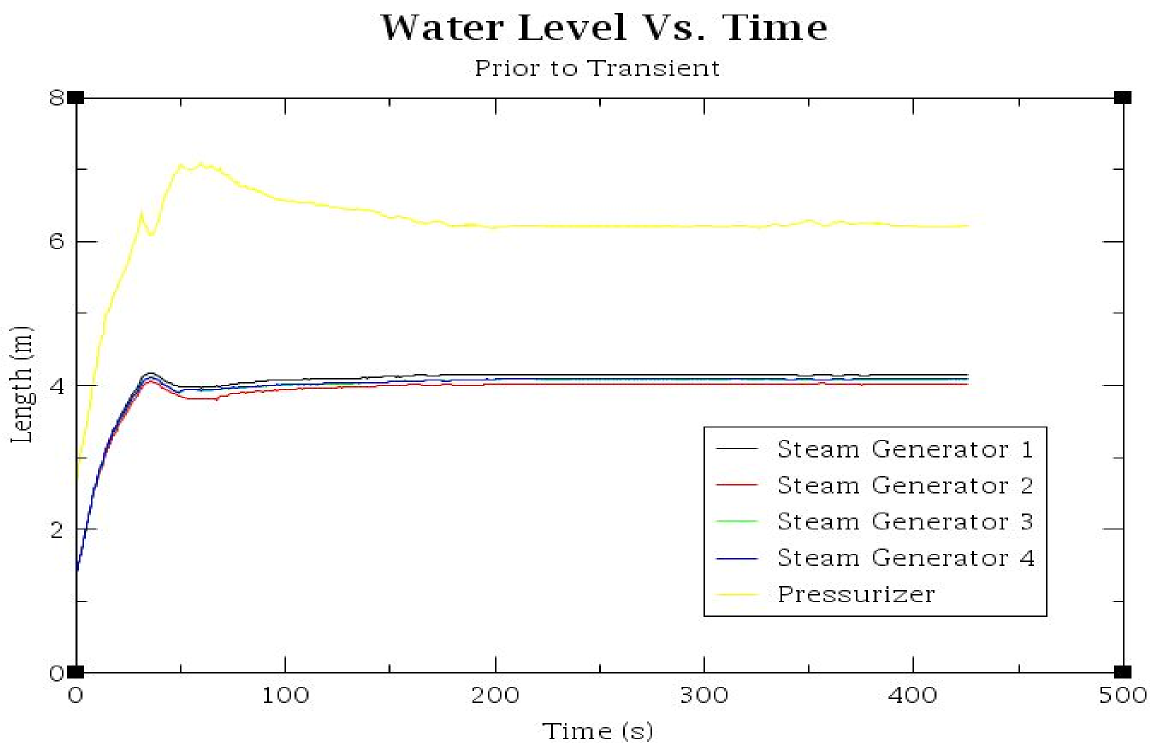

Figure 13. System Water Levels.

Plant Base Steady State

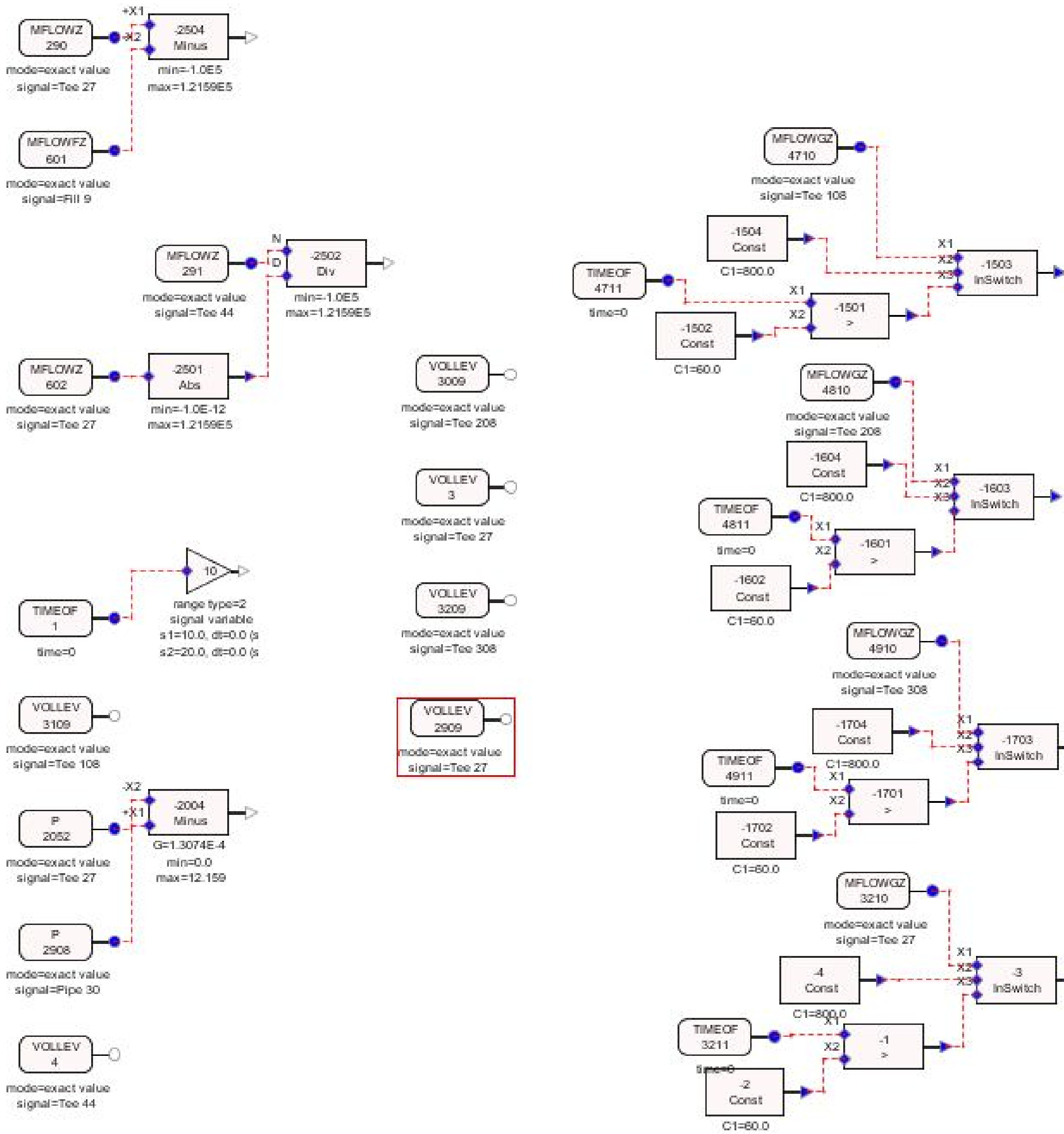

Figure 14. Control System for Steady State Calculation

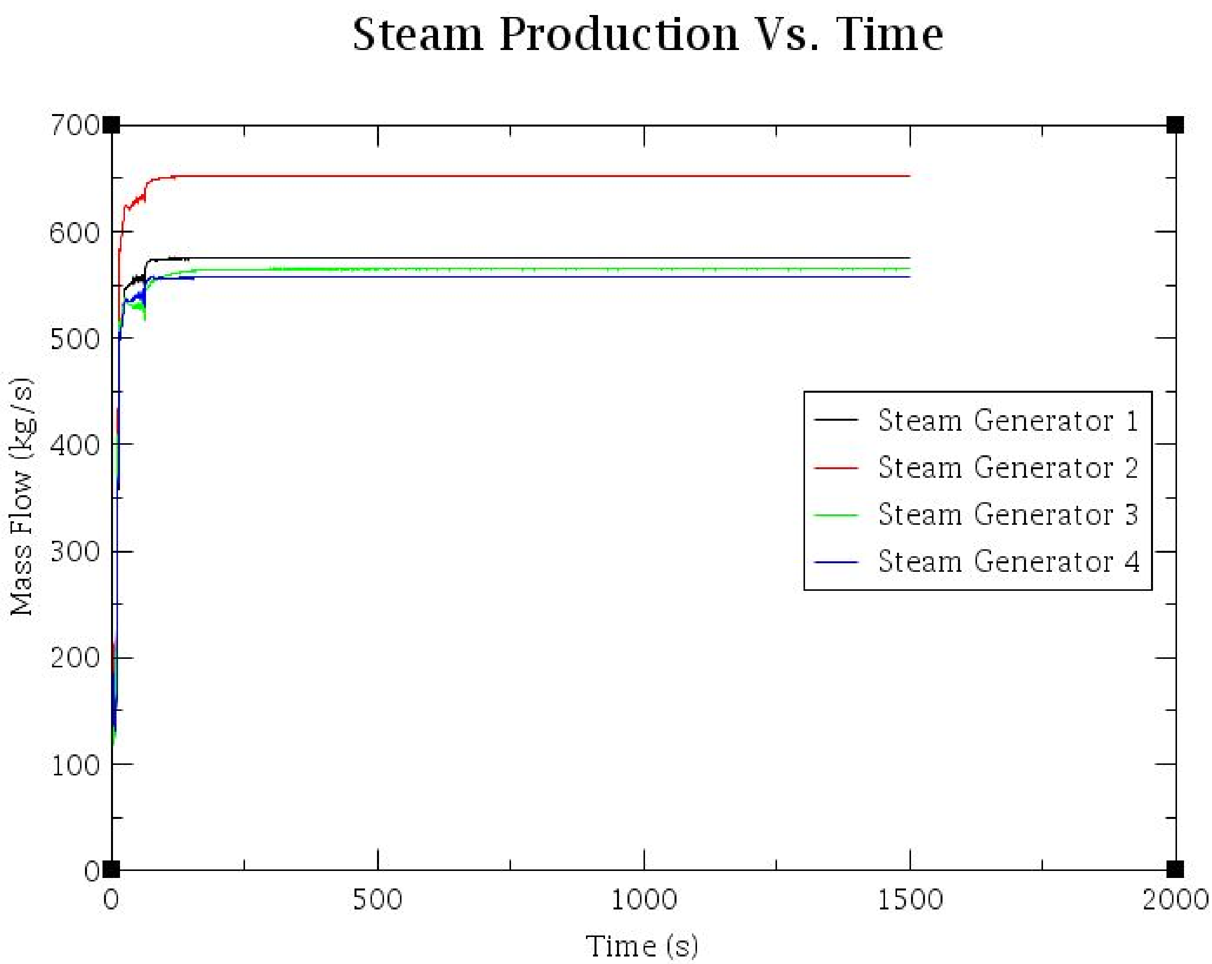

Figure 15. Steam production vs. time for steam generators at steady state

Plant Transient

From a roughly steady state condition, we initiated a 5% drop in reactor power over a period of 15 seconds. This resulted in a rise in S/G water level, a drop in PZR water level, and a drop in reactor coolant temperature. This is demonstrated in the following graphs:

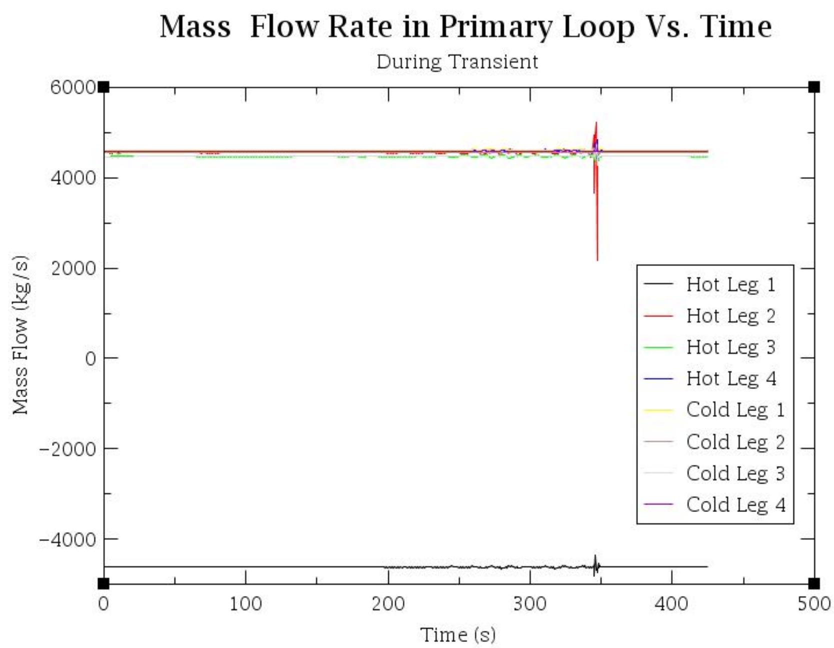

Figure 16. Mass flow rate in primary loop vs. time during the transient

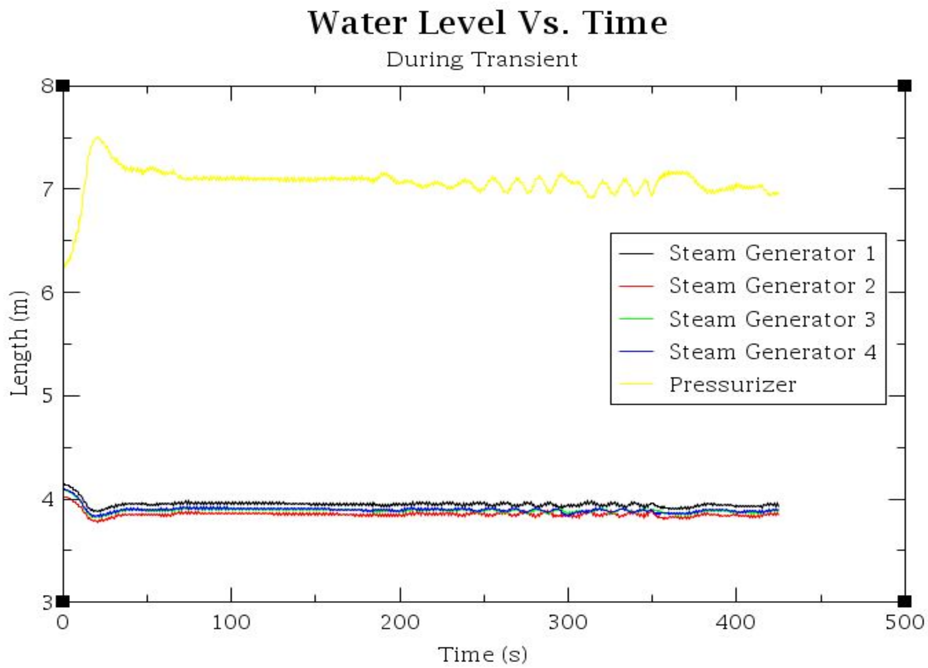

Figure 17. System water levels during transient analysis

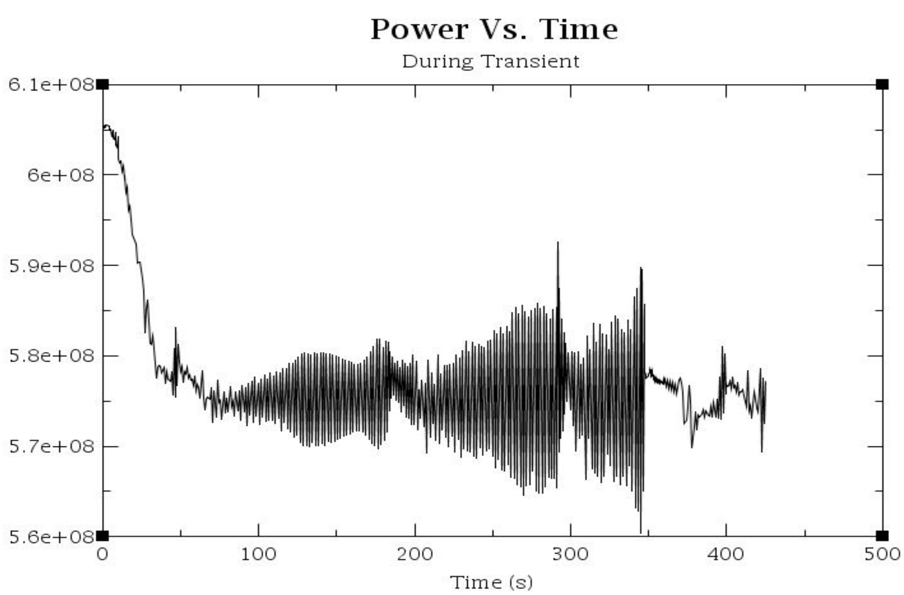

Figure 18. System water levels during transient analysis

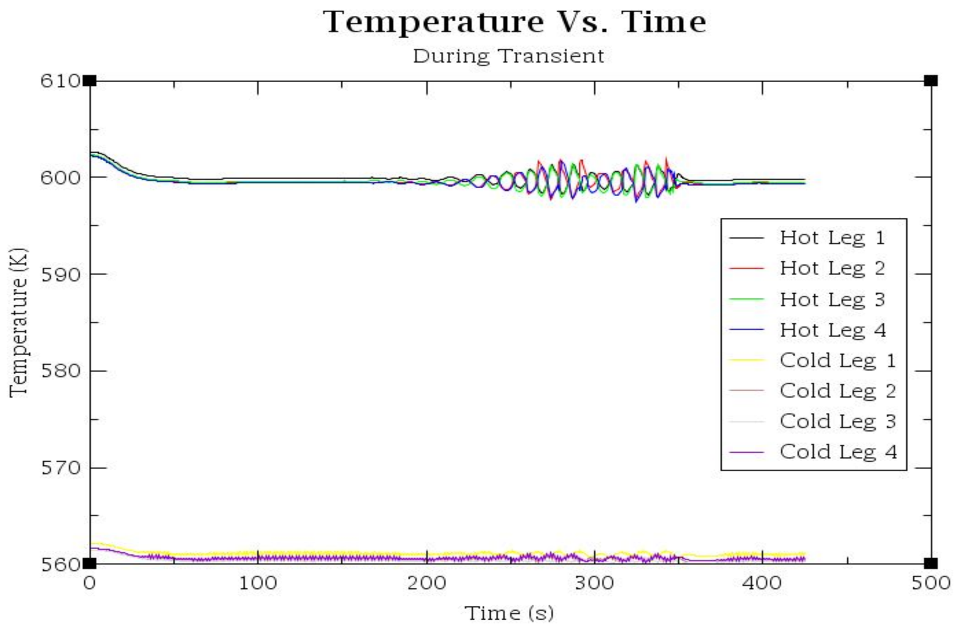

Figure 19. Temperature rate vs. time for both hot and cold legs during transient

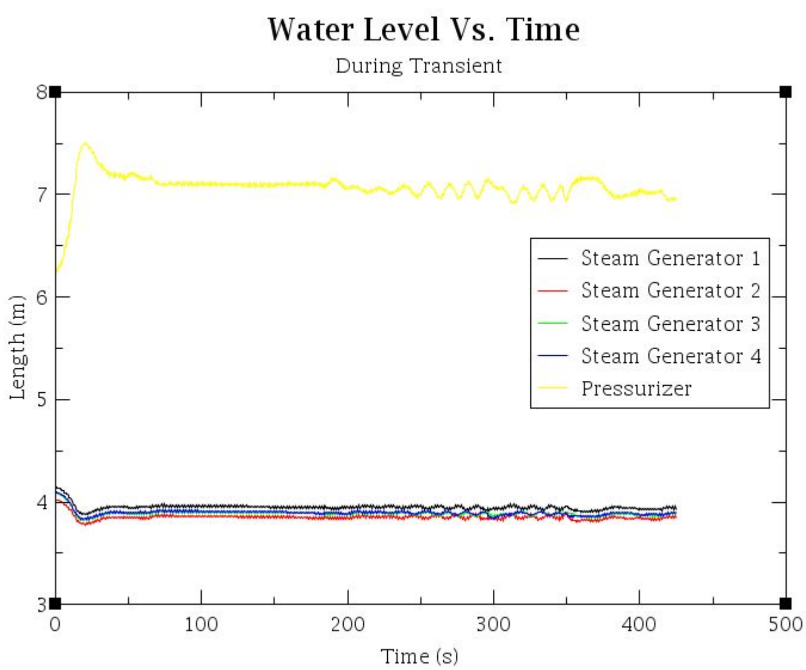

Figure 18. System water levels during transient analysis

Steam Generator Richardson Error Analysis

A Richardson error analysis was performed on out reactor model. Figure 19 shows the three different nodilizations that were used, one with 5 nodes, one with 8 nodes and the final with 14 nodes. The mass flow rates were 2285.7515, 2289.53, and 2259.6025 repectivly. The values for the error analysis were taken at 25 seconds as it was at the highest rate of change.

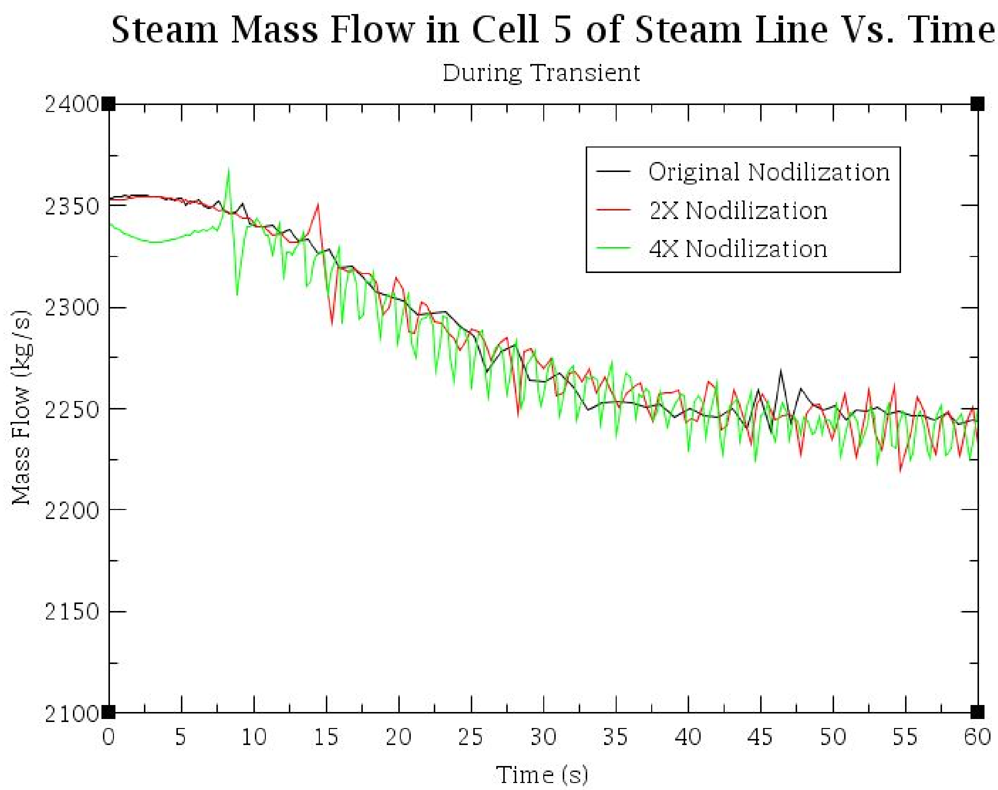

Figure 19. Steam mass flow rate in cell 5 of stream line for base, 2x, 4x nodalization

$$P = \frac{LN\left[\frac{(f_3 - f_2)}{f_2 - f_1}\right]}{LN(r)}$$

$$P = \frac{LN\left[\frac{(f_3 - f_2)}{f_2 - f_1}\right]}{LN(r)}$$

$$P = \frac{LN\left[\frac{2259.6025\ -\ 2289.53}{2289.53\ - 2285.7515}\right]}{ln(2)} = -2.98559$$

$$\varepsilon(error) = f_1 - f_{exact} = ah_1^P = \frac{(f_2 - f_1)}{(r^P - 1)}$$

$$\varepsilon(error) = \frac{(f_2 - f_1)}{(r^P - 1)}$$

$$\varepsilon(error) = \frac{2289.53\ - 2285.7515}{2^{0.547611} - 1} = -34.252$$

The final error of 34.252% was found. This needs to be further investigated however it is beyond the scope of this project.

Conclusions

When modeling a PWR, all major components of the reactor must be modeled individually to ensure a smooth integration and simple troubleshooting of errors. We began by modeling the steam generator, keeping the physical parameters within the limitations given by the project requirements and maintaining the specified inputs and outputs such as steam flow rate, feed water flow and temperature and primary coolant inlet/outlet temperatures. The next component modeled was the reactor core/vessel, where again we kept the physical parameters within given ranges and monitored outputs that were directly correlated to inputs into the steam generators. At this point we added additional components such as a pressurizer, coolant pumps and coolant piping with appropriate length and bends; which required minor alterations to the existing components to achieve steady state conditions.

In order to properly analyze the model after reaching steady state, a transient of 5% decrease in power over 15 seconds was run. After running the 15 second transient, the model continued to run for 400 seconds, in which it never reached convergence. It is possible that a slower transient of the same magnitude would have allowed more stability, however this condition is outside the scope of this project and was never run.

A Richardson extrapolation error estimate was performed using the full core model. The boiler region of the steam generator and U-tubes were first renodalized with twice as many nodes and then four times as many nodes. The U-tubes were only renodalized in the straight sections. This analysis demonstrated a spatial error of 34.25% in our model. The reasoning for this error was beyond the scope of this project and would need further study.

References

1. Indian Point Unit 2 FSAR Ch 3 & 4 : Provided as course material in NucE 470, The Pennsylvania State University, 2012.
2. Watson, Justin. NucE 470 Powerplant Simulation Course Notes. State College: The Pennsylvania State Univeristy, 2012.

Appendix

Final Design Model

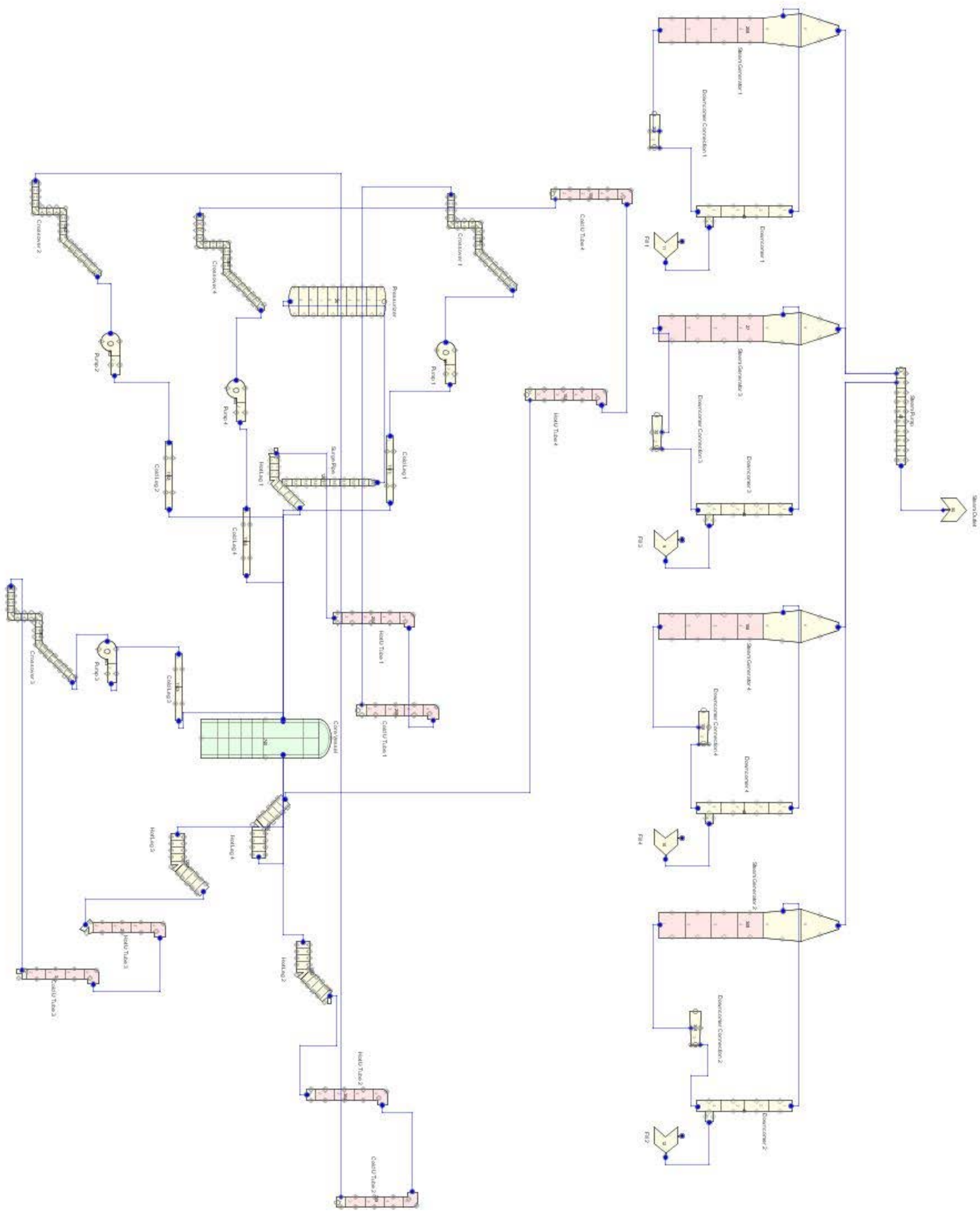

Figure 20. Final Reactor Model

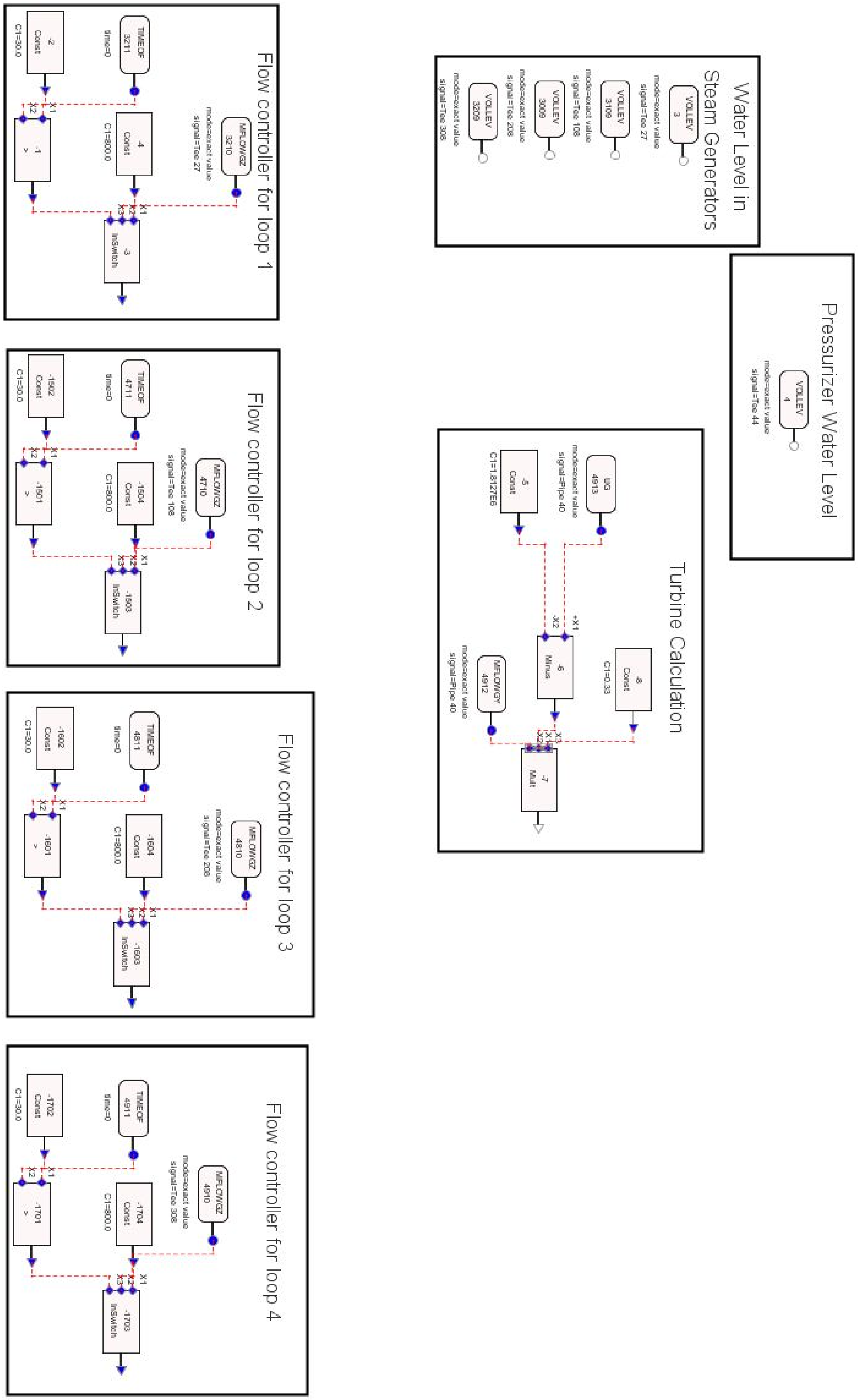

Figure 21. Final Control System Layout

Enclosures

1. Crosby_Frister_Geleskie_Core.inp
2. Crosby_Frister_Geleskie_Full_Syster.inp
3. Crosby_Frister_Geleskie_Full_Sysem.med
4. Crosby_Frister_Geleskie_Full_Syster_2X.inp
5. Crosby_Frister_Geleskie_Full_Syster_4X.inp
6. Crosby_Frister_Geleskie_Pipes.inp
7. Crosby_Frister_Geleskie_Steam_Generator.inp