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Light Water Reactor Sustainability Program

NPP Simulators for Coupled Thermal and Electric Power Dispatch

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ABSTRACT

Within the Light Water Reactor Sustainability (LWRS) Program, the Flexible Plant Operation and Generation (FPOG) Pathway works to diversify the revenue streams of light water reactors (LWRs) by opening opportunities for the co-generation of non-electric products in addition to supplying electrical power to the electricity grid. Producing hydrogen with maximum efficiency using nuclear power requires dispatching both electrical and thermal power from the nuclear plant to the hydrogen plant; therefore, developing and proving concepts of operations for combined electrical and thermal power dispatch (TPD) from a nuclear power plant (NNP) to a hydrogen plant is of interest. Coupling the power generation deck of a NPP to a hydrogen production facility introduces new risks, especially during operational transients. Consequently, it is important to evaluate how these transients can be observed and managed by the NPP operators.

This report summarizes progress in developing and testing full-scope NPP simulators at the Idaho Nation Laboratory (INL) Human Systems Simulation Lab (HSSL) that are being used to test operating concepts, address human factors, and prove the NPP operators can reliably and safely dispatch thermal and electrical power to a hydrogen plant. Development and testing of NPP simulators addresses two principal LWRS needs. First, testing of simulators with human operators in real-time provides validation of the concept-of-operations to ensure the modifications to the nuclear power plant enable achieving the intended objectives of rapid dispatch of both thermal and electric power while not compromising safety, including human factors considerations. Second, testing the simulators in real-time with human operators and physical hardware-in-the-loop verifies the functionality and safeguards in the proposed control systems.

Generic NPP simulators for a pressurized water reactor (PWR) and a boiling water reactor (BWR) have been modified by GSE Systems to support these R&D efforts. Additionally, a PWR simulator for a specifical three-loop Westinghouse NPP is being modified by Westinghouse to support this work. Coordination with lead utilities to develop plant-specific simulators for use in the utility training simulator is also being pursued.

Two versions of the GSE Systems coupled BWR/hydrogen plant simulator have been developed. The first version uses synthetic oil as a heat transfer fluid in a closed delivery heat loop to generated steam at the hydrogen plant. The second version uses steam as the heat transfer fluid to directly provide steam to the hydrogen plant. The estimated thermal power delivery distance is approximately one kilometer. The amount of thermal power dispatched in the simulators is 15% of the total reactor thermal power.

A Cooperative Research and Development Agreement (CRADA) between Battelle Energy Alliance (BEA) and Westinghouse Electric Corporation (Westinghouse) was approved by DOE to support the LWRS program efforts in using LWR simulators to develop operating concepts and control systems for dispatching nuclear energy to the industrial applications. The CRADA will first focus on modification of a three-loop Westinghouse PWR reactor. The three-loop simulator will provide close representation of the Westinghouse two-loop and four-loop PWR reactors.

The modified PWR and BWR simulators by GSE were verified by evaluating the NPP thermal flows and core heat rate response as the hydrogen plant transitioned from hot standby to full capacity in under 2 hours. As steam flow increases in the extraction steam line (XSL), flow through the turbine and feedwater heater systems decrease as expected, which causes the feedwater temperature entering the reactor to decrease from 423 °F to 409 °F. In this simulation, as the flow in the XSL increased to 390 lbm/s, the reactor power increased from 100% to nearly 102%. As the steam flow in the XSL decreased, the reactor power returned to 100%. Maintaining the reactor power constant while increasing flow in the XSL would require decreasing the feed water flow, which would further decrease the turbine power production. This

work confirms the need to address operational transitions as part of the LWRS effort to develop human factors, operating concepts, and control systems logic and programming for implementation and practice.

The details of this design and results from the simulations were described in a manuscript that was submitted to the peer-reviewed journal "Progress in Nuclear Energy." The manuscript is included in this report as Appendix A. The modified simulators by GSE, Westinghouse, and other Utilities will continue to be used in FY2023 to support the technical objective of the FPOG Pathway.

CONTENTS

FIGURES

TABLES

ACRONYMS

A/E	Architectural/engineering
API	Application programming interface
BWR	Boiling water reactor
CRADA	Cooperative Research and Development Agreement
DI	Deionized or demineralized water
DOE	Department of Energy
DRTS	Digital real time simulation
EIL	Energy Innovation Laboratory (Human Systems Simulation Laboratory)
ESL	Energy Systems Laboratory (Physical Testbeds and Real Time Data Simulator)
FORCE	Framework for Optimization of ResourCes and Economics
FPOG	Flexible Plant Operations and Generation
GONUKE	Guideline on Nuclear Usability Knowledge Elicitation
GPWR	Generic pressurized water reactor
H_2	Hydrogen
HMI	Human-machine interface
HOIL	Human-operator-in-the-loop
HSI	Human-system interface
HSSL	Human Systems Simulation Laboratory
HTE	High temperature electrolysis
HTF	Heat transfer fluid
HTSE	High temperature steam electrolysis
HWIL	Hardware-in-the-loop
I&C	Instrumentation and Control
INL	Idaho National Laboratory
LDRD	Laboratory directed research and development
LWR	Light Water Reactor
LWRS	Light Water Reactor Sustainability Program
MCR	Main control room
MVA	Megavolt-amperes
MWe	Megawatt electric
MWt	Megawatt thermal
NPP	Nuclear power plant
NRC	Nuclear Regulatory Commission

OMCF	Open Meta Calculation Framework
PEPSE	Thermal-hydraulics modeling software
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
R&D	Research and development
R/O	Reduced-order
R/O-TPD-PWR	Reduced-order model for thermal power dispatch from pressurized water reactor
RTDS	Real time digital simulation
SSH	Secure socket shell
SOEC	Solid oxide electrolysis cell
TEMS	Thermal Energy Management System
TPD	Thermal power dispatch
TCP	Transmission control protocol
VLAN	Virtual local area network
VM	Virtual machines

INSTALLATION AND TESTING OF VENDOR NPP SIMULATORS AT HSSL FOR COUPLED THERMAL AND ELECTRIC POWER DISPATCH

1. INTRODUCTION

Within the Light Water Reactor Sustainability Program (LWRS), the Flexible Plant Operation and Generation (FPOG) Pathway works to diversify the revenue streams of light water reactors (LWRs) by opening opportunities for the co-generation of non-electric products in addition to supplying electrical power to the grid. Specific research objectives of the FPOG Pathway include:

1. Developing design and cost estimates for thermal and electric power dispatch from a representative pressurized water reactor (PWR) and representative boiling water reactor (BWR) to tertiary industrial loads at different levels ranging from 10–70% of the total rated reactor power.
2. Developing concepts of operation for dispatching thermal and electric power from representative LWRs to the electric grid and tertiary industrial loads.
3. Developing automated control systems for these operations. Different control systems will be developed independently that can be used to rigorously meet the requirements of the United States (U.S.) Nuclear Regulatory Commission (NRC) for specific plants and for sharing with stakeholders to assist in hardware integration. Nuclear power plants (NPPs) are licensed by the U.S. Nuclear Regulatory Commission (NRC). This license is based on the Final Safety Analysis Report (FSAR), which specifies the operating conditions of the NPP.
4. Testing proposed concepts of operation and integrated system performance using human operator-in-the-loop (HOIL) and hardware-in-the-loop (HWIL) tests. These tests will employ reduced-order (R/O) and full-scope simulators, as needed, to demonstrate the feasibility of dynamic operations in normal and off-normal events.
    5. Recent events have added greater motivation to these efforts. For example, the recent Inflation Reduction Act (IRA) passed by the U.S. federal government offers substantial tax incentives for producing clean hydrogen. In addition, water-splitting electrolysis can produce both clean and pure hydrogen. One advantage of electrolysis is the ability to ramp production up and down in a short period of time. This feature allows a nuclear power plant (NPP) to quickly switch the electricity supply between the grid and the electrolysis plant. Fortunately, the technology readiness level (TRL) of dispatchable and high-efficiency hydrogen production has dramatically increased in a short amount of time.

This report documents progress toward accomplishing the first two objectives to design thermal and electric power dispatch from a representative PWR and a representative BWR to tertiary industrial loads at different power levels as well as test the concepts of operations using simulators. Specifically, for this work, GSE Power Systems provided a full-scope generic boiling water reactor (GBWR) simulator to INL and then partnered with the University of Florida to modify that simulator for dispatching thermal and electric power to a high temperature electrolysis (HTE) hydrogen production plant. Two different versions of the modified full-scope simulator were developed for dispatching thermal power. Both versions employed a thermal power extraction (TPE) subsystem for removing steam from the BWR and separate subsystems to deliver heat to the HTE plant. The difference is that the first version employed synthetic oil as the heat delivery fluid (HTF), while the second employed steam for that purpose. Simulations were performed using both versions of the simulator to test the transient response of a BWR plant due to the dispatch of electric and thermal power. The details of the GSE GBWR simulator and the two versions that were modified for TPD operations are presented in Section 2. Ongoing and future work involving a PWR simulator from an architectural/engineering (A/E) firm is summarized in Section 3. The report concludes with Conclusions in Section 4.

2. GSE GBWR SIMULATORS

2.1 GSE GBWR Original (Unmodified) Full-scope Simulator

The GSE GBWR simulator is an adaptation of an actual plant simulator, currently in use by the United States Nuclear Regulatory Commission (NRC) in operations examiner training. This simulator was developed previously by GSE Systems and replicates a certain GE BWR/4. It has been maintained and modified by both the NRC and GSE, and modifications are ongoing. The primary thermohydraulic model has been upgraded to the GSE real time RELAP-5 package and the reactor neutronics are represented by the GSE REMARK model. Many secondary systems are implemented in a non-graphical modeling technology (legacy) code; however, the Primary Containment and Reactor Building models have been replaced with the GSE JTopmeret modeling package. Similarly, the Instructor Station uses the current GSE Java Instructor Station (JIS).

The GBWR simulator was originally developed as a “hard-panel” simulator but has since been modified to replace the original control room controls (switches, lights, meters) with an emulated digital control (DCS) scheme. This adaptation permits the GBWR to be conveniently operated in a classroom setting or on individual personal computers. This emulated DCS representation replaces only the control room physical hardware; the simulated operational and control logic remains as originally designed. The original (unmodified) GBWR was provided to INL as well as two versions that were modified for TPD operations. The GBWR/TPD project was staged on a GSE-operated virtual machine (VM), and a Mantis Bug Tracker project was created and used for version control and to documents resolution of issues that were encountered.

To support human-in-the-loop testing required for the planned research activities to develop, evaluate and demonstrate a TPD concept of operations, the GBWR was installed at the Idaho National Laboratories (INL) Human System Simulation Laboratory (HSSL; see Figure 1). Images of the simulator running in the HSSL are displayed in Panel a of Figure 2. Unlike the GSE pressurized water reactor simulator (GPWR), which represents a largely analog control room, the GBWR uses windowed displays to represent all the plant systems, as featured in Panels b, c, and d of Figure 2. This change represents a fundamental shift in control room operations and provides a valuable capability for the planned operator studies.

Figure 1. Human Systems Simulation Laboratory showing the GSE GPWR analog control room panels represented virtually on touch screen displays.

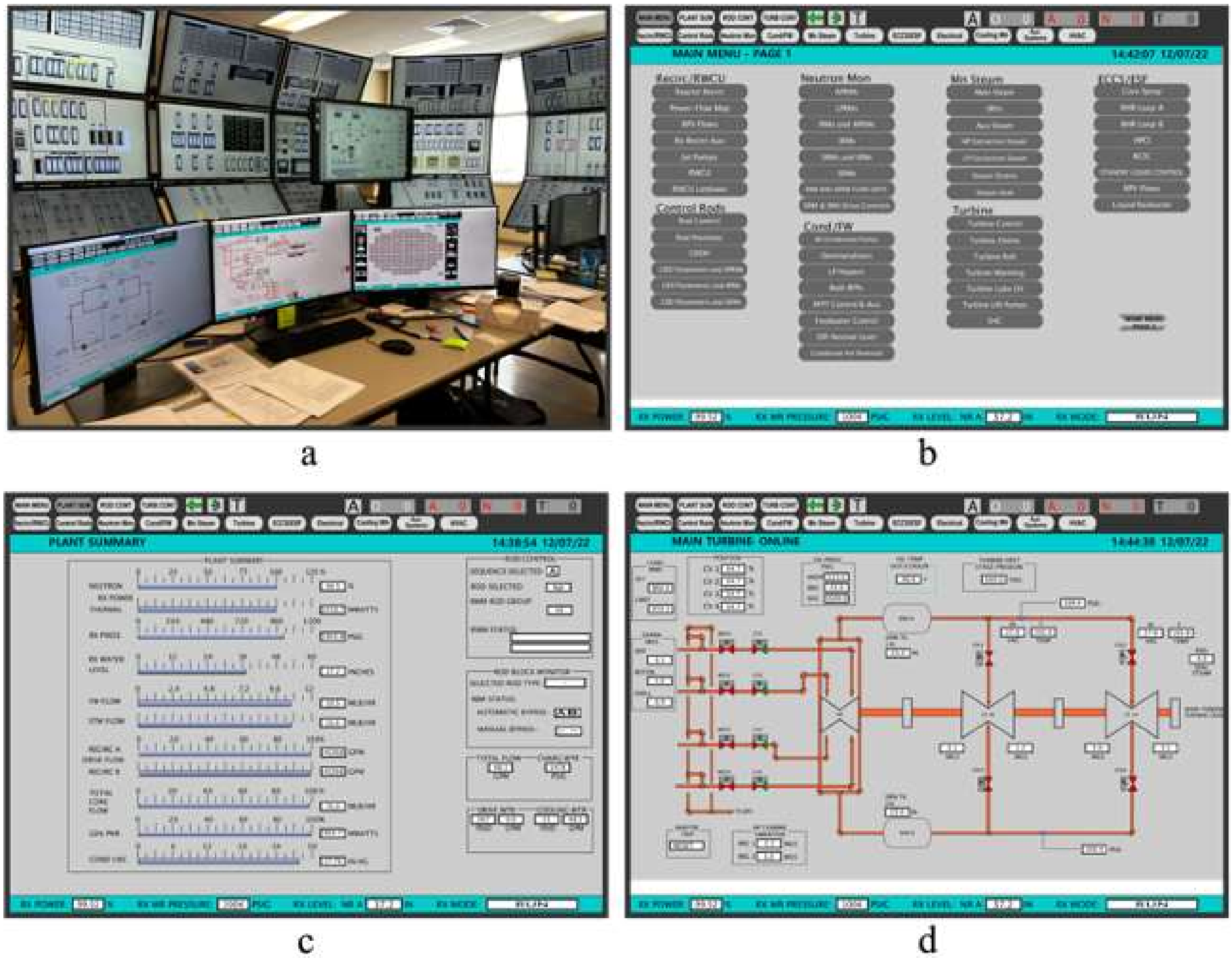

Figure 2. Images of the GBWR simulator. Panel a shows the GBWR simulator on an operator workstation in the HSSL. The remaining three panels show the digital displays depicting the main menu navigation (b), overview display containing plant summary (c), and the main turbine system (d).

2.2 GSE TPD GBWR Simulator Version 1

As noted above, two different versions of the modified full-scope GBWR simulator were developed for dispatching thermal power. Simplified process flow diagrams (PDFs) of the two modified versions are shown in Figure 3. The first modified simulation version employed synthetic oil as the heat delivery fluid (HTF), as shown in Figure 3a. The synthetic oil is heated using steam that is extracted from the Main Steam system at the common steam header outside the Primary Containment. The Steam Bypass valves in the GE BWR plant design are connected to the same location in the main steam header, so there is already precedence and basis for extracting large amounts of steam from this location. The line that carries the steam extracted from the main steam header is referred to as the Extraction Steam Line (XSL). A heat exchanger transfers heat from the extracted steam to a closed loop of synthetic oil, referred to as the Delivery Heat Loop (DHL) loop, which results in condensing the steam. This heat exchanger is located near the BWR to minimize the volume and latency of the steam flow in the XSL. The product condensate is returned to the Main Condenser. The DHL includes a second heat exchanger that transfers heat from the oil to deionized (DI) or demineralized (DM) water to generate steam that is fed to the hydrogen plant. The second heat exchanger is located near the hydrogen plant, which may be several hundred meters or even kilometers, distant from the BWR.

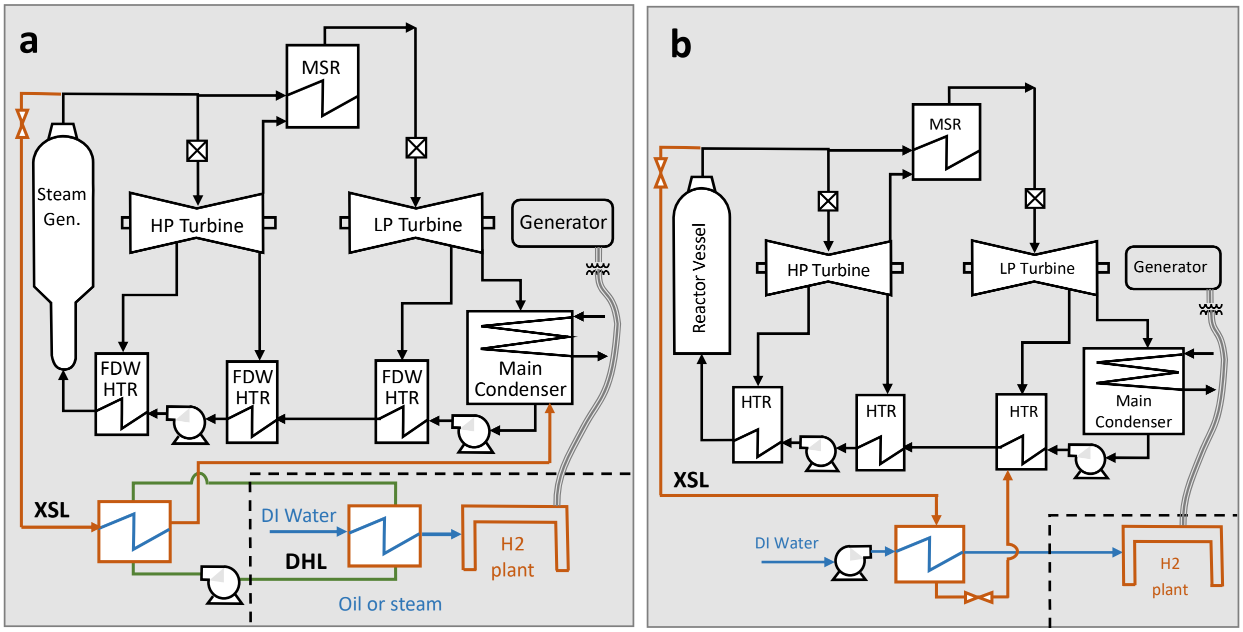

Figure 3. Simplified process flow diagrams for the GSE TPD GBWR Simulators. Panels (a) and (b) show the versions that use synthetic oil and steam as the HFT, respectively. *(XSL- Extraction Steam Line; HP- High Pressure; LP- Low Pressure; MSR- Main Steam Reboiler; Feedwater- FDW; Heater – HTR; DI- Deionized Water)*

The design of this system closely follows a design developed by INL as documented in (Hancock, Westover, Luo; 2021). A potential issue with this design is that radionuclide contaminants, including tritium and mixed fission and activation products, could be transferred from steam in the XSL to the synthetic oil in the DHL through small leaks in the XSL heat exchanger. Methods must be developed and approved to measure radioactive contaminants in synthetic oil before this design can be implemented because the synthetic oil in the DHL crosses the protected area boundary.

The TPD subsystems simulation also includes modeling of several control loops to control the extraction steam, various tank levels, and the DHL flow and return temperature. These loops include a simplified human/machine interface (HMI) placed directly on the network diagrams, as well as additional control actions (remote function) realized through the GBWR Java Instructor Station operator interface. The initial design of this TPD subsystem was capable of delivering approximately 7% of the GBWR rated reactor thermal power (175 MW_{th}) to the hydrogen plant. Later this design was uprated to deliver about 17% of the rated reactor power (420 MW_{th}) to the hydrogen plant. The University of Florida (UF) carried out testing of this updated Version 1 of the GSE TPD GBWR simulator.

2.3 GSE TPD GBWR Simulator Version 2

As noted above, the second version of the GSE TPD BBWR simulator uses steam as the HTF. A simplified process flow diagram is shown in Figure 3b. The use of steam allows improved performance as the steam is directly useable by the hydrogen plant, so that the DHL filled with synthetic oil is completely eliminated. Using steam as the HTF has other advantages as well. For example, the pumping power requirement for using steam as the HFT is approximately 10 times lower than that required for synthetic oil because steam can transfer much more heat in the form of latent heat (heat released when steam is condensed to water). Water is also a less expensive HTF than synthetic oil, which becomes more important as the separation between the BWR and hydrogen plant increases. Finally, methods have been developed and approved to measure concentrations of radioactive contaminants in steam, so that potential releases of contaminants outside the protected area can be monitored. Protecting against unintentional releases of contaminants may require a more sophisticated TPD design than is shown in Figure 3b. For example, an additional steam loop may be required, similar to the DHL, shown in Figure 3a.

The initial design of Version 2 of the GSE TPD GBWR simulator also followed a design previously developed and tested at INL for TPD from a PWR (Hancock, Westover, Luo; 2021). In this design, a "kettle reboiler" was used to condense the steam in the XSL and boil the water going to the hydrogen plant. Later, GSE modified the design to use a "thermosiphon reboiler" rather than a "kettle reboiler." The newer design involves a separator above a closed circulation loop containing a vertical heat exchanger. The extracted steam is conducted to one side of this heat exchanger as heating fluid, and the density difference between the downcomer from the separator and the riser/heat exchanger drives the natural circulation flow. This heat exchanger employs two-phase flow on both sides of the heat exchanger, so that heating steam condenses as the secondary circulating fluid boils. This design could be scaled quite easily by adding additional thermosyphon loops to the single common separator, each with its own heating steam admission valve. In this case, the feedwater connection would be moved to the common separator. Level in the separator is controlled similar to a natural circulation boiler by adjusting feedwater flow. The details of this design and results from the simulations were described in a manuscript that was submitted to the peer-reviewed journal "Progress in Nuclear Energy." The manuscript is included in this report as Appendix A.

2.4 GSE TPD GBWR Simulator Results

Simulations were performed to test the transient response of the BWR plant as the hydrogen plant transitions from hot standby to hydrogen production at full rated capacity, which corresponds to dispatching approximately 15% of the steam from the main steam header or 390 lb of steam per second. For simplicity, it is assumed that the steam extraction during hot standby is negligible and that the steam flow rate through the XSL increases at a rate of 10 lb/s per minute. Once a steam flow of 390 lb/s was achieved in the XSL, the steam flow was held steady at that value for 15 minutes and then ramped down at 10 lb/s per minute until 0% flow was achieved. In addition to the steam flow rate in the XSL, the other parameter that was controlled was the temperature of the oil that was delivered to the hydrogen production facility. That temperature was held constant at 510 °F by adjusting the flow rate of oil in the DHL. It was further assumed that the hydrogen plant adjusted electric power and heat consumption to match the heat delivered by the DHL.

Figure 4 shows the flow in the main steam line, the steam flow to turbine, the steam flow to XSL, and the turbine electric power during the simulation. As steam flow increases in the XSL, flow through the turbine and feedwater heater systems decrease as expected, which causes the feedwater temperature entering the reactor to decrease from 423 °F to 409 °F, as shown in Figure 5. The change in feedwater temperature causes inverse changes in reactor power because at constant flow rate the reactor void fraction changes with feedwater temperature. Lower reactor void fraction corresponding to lower feedwater temperature increases neutron moderation and thus increases BWR reactor power. In this simulation, as the flow in the XSL increased to 390 lb/s, the reactor power increased from 100% to nearly 102%. As the steam flow in the XSL decreased, the reactor power returned to 100%. The decrease in reactor void, main steam pressure and reactor dome pressure are all shown in Figure 6. Maintaining the reactor power constant while increasing flow in the XSL would require decreasing the feed water flow, which would further decrease the turbine power production.

As noted above, the results from the GSE TPD GBWR Simulator Version 2 are included in Appendix A. A manuscript describing the details of that simulator and results were submitted to the peer-reviewed journal "Progress in Nuclear Energy."

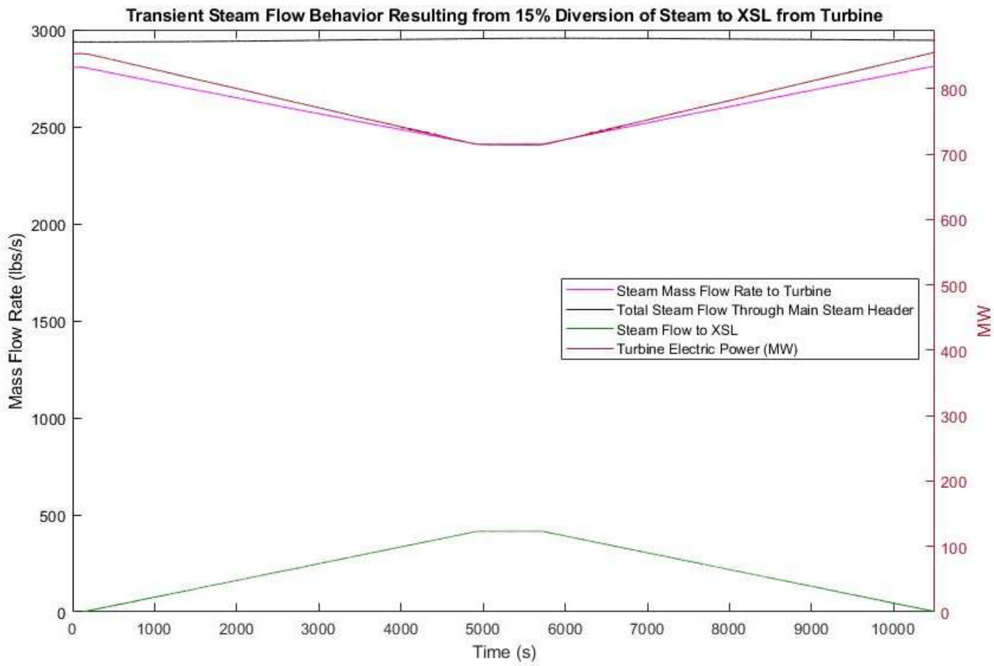

Figure 4. Steam flow rates and turbine electric power for the transition to 15% TPD using oil as the heat transfer fluid.

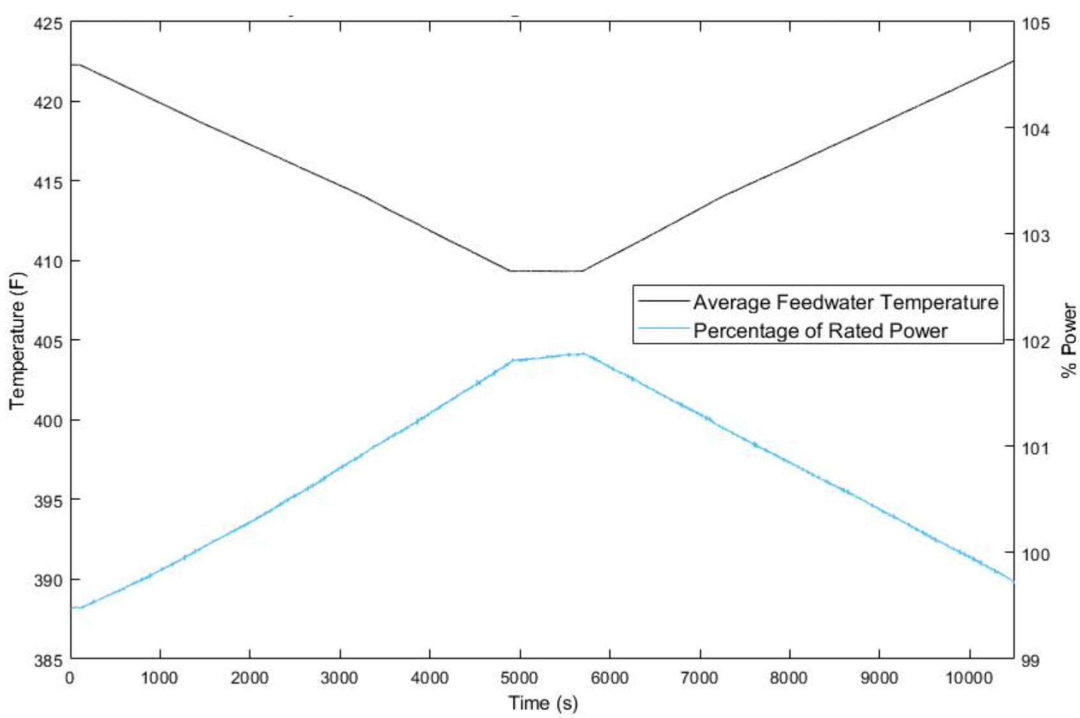

Figure 5. Steam flow rates and turbine electric power for the transition to 15% TPD using oil as the heat transfer fluid.

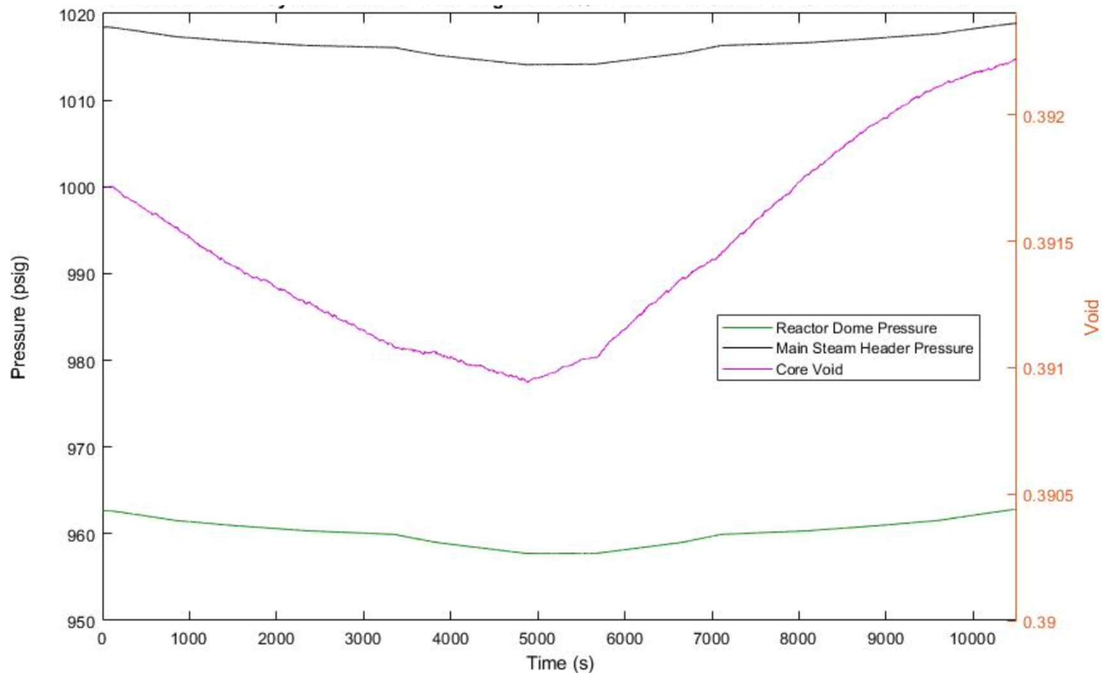

Figure 6. Steam flow rates and turbine electric power for the transition to 15% TPD using oil as the heat transfer fluid.

3. FUTURE WORK

3.1 Overview

Development and testing of NPP simulators addresses two principal LWRS needs. First, as noted above, testing of simulators with human operators in real-time provides validation of the concept-of-operations to ensure the modifications to the nuclear power plant enable achieving the intended objectives of rapid dispatch of both thermal and electric power while not compromising safety, including human factors considerations. Second, testing the simulators in real-time with human operators and physical hardware-in-the-loop (HWIL) verifies the functionality and safeguards in the proposed control systems.

Three types of control systems will be developed in this work. The control systems differ in their fidelity and the associated level of public accessibility. Models with high-fidelity and complexity are proprietary and are much more difficult to use, while reduced order (R/O) models with lower fidelity can be made publicly available and have much lower barriers to learn and modify to explore new consequences of new control architectures. The three types of control systems and future project plans have been described in a previous report (Westover et al., 2022) and therefore are only briefly summarized here.

The first type of control system is generalized systems that apply to R/O simulators. This type of control system is developed and implemented in python or a similar universal programming language and will be shared with partners to support collaborations to advance coupling NPPs to tertiary industrial loads. This is the principal type of control system that will be used in HWIL tests that couple industrial equipment, such as high temperature electrolysis hydrogen production systems, to NPP simulators to verify compatibility of control architectures with physical equipment. This project has an internal milestone in July to develop and test an R/O TPD PWR simulator with automated controls. The testing of that control system will include HWIL tests using a 25+kW SOEC system to verify inter-operability of controls and communications between the SOEC physical hardware and simulator.

The second control system is provided by full-scope NPP simulator vendors and will include dynamic thermal power dispatch to tertiary industrial loads, such as hydrogen plants. This type of control system is embedded in modified full-scope NPP simulators and has a relatively simplified automated controls with complete descriptions published for public dissemination and unrestricted use. The modified TPD GBWR simulators described above fit in this category. The control systems developed by full-scope nuclear power plant simulator vendors will focus on relatively low levels of TPD (less than approximately 10% of rated reactor thermal power). The actual design and function of the control system will be made available to partner institutions, such as universities, that have a license to the original full-scope NPP simulator.

This project has an internal milestone in June to develop and test a full-scope TPD PWR simulator with automated controls. That test will verify that standard operations, such as transitioning a coupled hydrogen plant between operating modes, can be performed automatically while maintaining the PWR in normal operating condition. A contract has been set up with GSE Power Systems and work is proceeding to accomplish that milestone.

The third control system will be provided by one or more architectural/engineering (A/E) firms and will be tested by human operators using a modified full-scope NPP simulator also provided by the A/E firms through Corporative Research and Development Agreements (CRADAs). The operators will interact with the simulator and control system through a realistic HMI in a control room environment. The control system provided by the A/E firm will be designed to meet all applicable requirements as determined by the A/E working with partner groups, such as Sargent and Lundy (S&L) and the Hydrogen Regulatory Research and Review Group (H3RG). The design requirements and control system results will be openly published, but the actual control system will be proprietary and will not be shared.

3.2 Developing, Installing and Testing A Full-Scope Simulator Provided by an A/E Firm

A CRADA (22CRA18) was established with Westinghouse Electric Corporation in October 2022 to develop and test a TPD simulator provided by Westinghouse. The intent is that PWR and BWR TPD simulators will be ultimately developed with iteratively refined models that support various levels of TPD. INL and Westinghouse reviewed the available simulators Westinghouse could provide and mutually selected a suitable simulator that is representative of existing light water reactors operating in the U.S. The first TPD simulator that will be developed and tested will be based on a PWR plant with two three-loop Westinghouse reactors in Ascó, Catalonia, Spain. Westinghouse PWRs are sufficiently similar that a simulator of a three-loop reactor is an appropriate representation for two-loop and four-loop reactors. The three-loop simulator will initially be modified for coupling to a 100 MW HTE hydrogen production plant that will require approximately 25 MW of thermal power while operating at its maximum rated capacity. In fiscal year 2024, this simulator will be further modified for coupling to a 500 MW HTE hydrogen production plant, which would require approximately 120 MW of thermal power, and will later be modified for coupling to other industry loads that require greater amounts of heat. The simulator developed by Westinghouse is similar to the GPWR simulator that INL has already obtained from GSE Systems but has a few important added benefits. First, the Westinghouse simulator is based on digital controls and has additional screens that can be called up to show parameter trends to assist operators in decision-making. The Westinghouse simulator also has upgrades to the controls and hardware representations, such as valve actuators, that make it more realistic and flexible in terms of accurately simulating and controlling plant response from operator inputs. Even with these improvements, the Westinghouse simulator is sufficiently similar to the GPWR simulator already available at the HSSL that the procedures that INL has already developed for TPD operation can be used with the Westinghouse simulator with little modification. Both simulators are for Westinghouse three-loop PWR plants of the same vintage and U.S. design. Figure 7 compares panel representations for the GPWR, Westinghouse

(Ascó,) simulators. The differences in the visual elements are due to the physical control room panels virtually represented by the simulators in addition to the graphical environments used by each of the vendors to build the software for the simulation. Specifically, aesthetic differences in the panels and variations in the layout of the individual elements are apparent across the different panels. The suite of instruments and controls are system dependent and therefore these simulators have largely the same instruments and controls implemented with specific plant variations. In terms of layout, the analog panels are quite standard with the typical arrangement of alarms positioned along the top, indication in the middle, and controls arranged along the bottom apron of the panel.

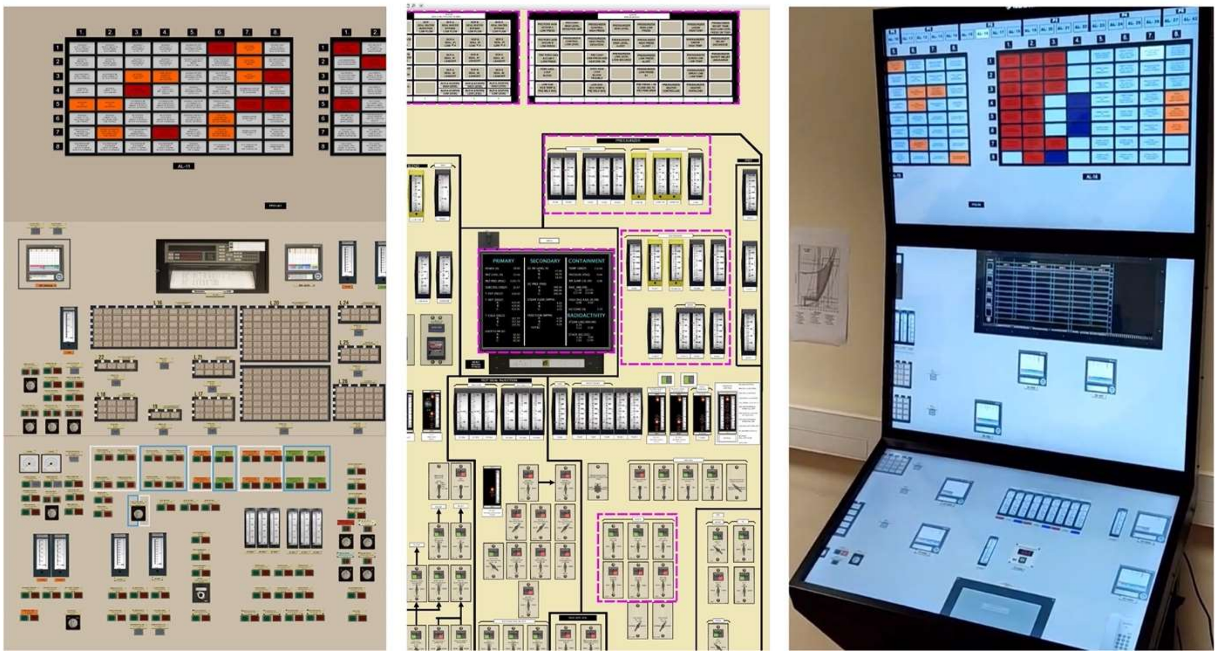

Figure 7. Comparison of simulator panels depicting similar general layouts of the instrumentation and controls for the Westinghouse (left) and GSE Systems GPWR (center and right) simulators.

4. CONCLUSIONS

Coupling the power generation deck of a nuclear power plant to a hydrogen production facility introduces new possibilities for operational transients that must be addressed. In particular, the startup and shutdown procedures of the hydrogen production facility need to be evaluated to ensure there are no adverse effects on the operation of the existing NPP. The concept of operations involving the NPP, the hydrogen plant, and the electric power grid must be tested using NPP simulators and operating procedures that have been modified for TPD operations. These tests must also include dynamic simulations of coupled tertiary thermal and electric loads as well as coordinated activities with NPP operators, tertiary load operators and grid power coordinators. This report summarizes progress in developing and testing full-scope NPP simulators at the HSSL, including a generic BWR simulator from GSE Systems, Inc. and generic PWR simulator from Westinghouse.

This effort lays the foundation to proceed in developing human factors and testing operating concepts preparatory to implementation at an actual nuclear power plant. Support from commercial companies such as GSE, Westinghouse, and NPP owners is important and signals the intent of the NPP industry to continue the pursuit of hydrogen production to bolster plant revenues.

A discussion of technical conclusions is found in the Discissionsand Conclusion Section of the research article attached. LWRS supported and CRADA research will continue in FY2023 and FY2024 using these simulator capabilities to develop and evaluate human factors, operating concepts, and control systems that can be implemented at NPPs to dispatch thermal and electrical power to hydrogen plants, as well as other industrial users.

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6. APPENDIX A: MANUSCRIPT SUBMITTED TO "PROGRESS IN NUCLEAR ENERGY"

Integrated Thermohydraulic Transient Testing of Steam Extraction Loop System to Support Joint Electricity-Hydrogen Production Using Generic Boiling Water Reactor

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Abstract
The economic competitiveness of the current fleet of light water reactors may be improved by coupling nuclear reactors to industrial facilities. This concept allows using high-quality steam and electric power from the reactor when the cost of electricity is low compared to the generation cost. One promising application is hydrogen production. Understanding the plant response to steam offtake is an important to maintaining the safe operation of the reactor facility. Analysis has been done for pressurized water reactors however, no previous work has been done for boiling water reactors. Boiling water reactors make up 30% of the US operating fleet. This paper demonstrates the feasibility of coupling a generic boiling water reactor to a hydrogen production facility. The system response is analyzed for a 15% reduction in the electrical power output. This work shows that when steam is extracted from the main steam header, reactor power will increase beyond 100% rated power. We present addition analysis to maintain power below rated conditions by varying feedwater recirculation pump speeds.

1 Introduction

Nuclear energy has been a staple of long-term, consistent base load energy generation since the first reactors came online for commercial energy generation in the 1950's. The onset of the diversification of the energy portfolio of the United States has called for a change in the energy generation behaviors of traditionally stable and consistent energy output. The transient energy production profile of renewable energy sources, such as wind and solar, has created a need for nuclear reactors to dial down core thermal power to facilitate the purchasing of renewables when their energy production is the most economical[1]. This often means that nuclear reactors would operate at power levels significantly lower than the name plate power rating, resulting in less efficient and economical energy production[2]. This percussive inefficiency, when carrying the effect of coupling renewable energy with nuclear, resonates on a daily cycle, leaving nuclear energy in a disadvantageous state[3]. The injection of renewable energy production into the energy portfolio of the United States is sought after and inevitable, which gives the nuclear energy sector an opportunity to make nuclear reactors more diverse in their utility. The goal for both the long-term health and economic viability of the nuclear facility is to be able to maintain maximum power output for as long as possible[1].

Hydrogen, both as a fuel and ingredient in industrial processes, is widely used in the global market. Currently, the worldwide need for hydrogen is roughly 500 billion standard cubic meters per year, with the most notable consumer being fertilizer production using hydrogen to synthesize ammonia[4]. Hydrogen is widely used in several other industry sectors as well, such as refineries, bulk chemical production, fuel cell electric vehicles, and end-use sectors utilizing natural gas as a fuel[5]. The drive to reduce carbon emissions has created an opportunity for the capitalization of hydrogen in traditionally carbon fueled sectors. The advantages of using hydrogen as a fuel are the significantly higher energy density, 140 MJ/kg versus 50 MJ/kg in traditional solid fuels, and the only emission being water[6]. The main environmental disadvantage of using hydrogen is that 96% of the stock is produced via non-renewable fossil fuels, with the most notable being steam methane reforming[4]. This method of hydrogen production relies on the decomposition of methane into carbon monoxide and hydrogen which requires temperatures in excess of 750°C. To reach these temperatures, the most common fuel that is burned is natural gas[4]. It is clear that this process to produce hydrogen cannot be an alternative to fossil fuels because it is based on fossil fuels.

The simultaneous need for a carbon-free hydrogen production method and the need for a flexible tertiary load for nuclear reactors offers new opportunities to use nuclear energy for dynamic hydrogen production. For example, a hydrogen production facility coupled to a nuclear power plant can rapidly ramp down hydrogen production (and power demand) when available renewable power drops due to changing solar or wind conditions. Similarly, the hydrogen facility can ramp hydrogen production back up (and increase power consumption) when renewable energy is sufficient to meet grid needs. In this way, the combined nuclear/hydrogen system is able to support greater penetration of renewable energy in the transmission grid.

The optimized pairing of nuclear reactors to advanced hydrogen production methods have been studied extensively. The common conclusion for the most optimal pairing comes in the form of a very high temperature reactor paired with an advanced thermochemical process, such as the sulfur-iodine cycle[7], [8]. A process developed by Argonne National Laboratory called the Cu-Cl cycle shows promise for the coupling of hydrogen production and nuclear energy[3], [7], [9], [10]. With the highest temperature demand to complete the cycle being 500°C, this method of hydrogen production is among the lowest of the thermochemical methods. Though 500°C is the lowest target to produce hydrogen, it is nearly double the average core outlet temperature for LWR's. Without the use of a chemical heat pump to upgrade the temperature of the steam, that puts the most efficient method for hydrogen generation out of the scope of possibilities[11]. All the hydrogen production methods that do not require electrical current as a driving force, e.g., electrolysis, require temperatures in excess of the average core outlet temperatures by more than double, which is why very high temperature reactors (VHTR) and reactors with outlet temperatures higher than 650°C are preferred[10].

The onset of a shared power grid is much closer to the present time than finishing construction of a coupled VHTR and hydrogen production facility, therefore, analyzing the thermodynamic behavior of the existing reactor fleet when paired with current technologies is of utmost importance. As the most optimal pairings of hydrogen production method and LWR has not been officially established, though high temperature hydrolysis is a promising option[5], this analysis will focus on the thermodynamic response of transferring heat from the power loop of the reactor to a heat sink that represents the hydrogen production facility. This analysis will show the response of the LWR to ensure that the removal of heat from the power loop of the reactor will not have deleterious effects on steady state normal operation or violate the design basis of the reactor license. The heat that is removed could be used to support hydrogen production or other industrial processes that require heat, such as iron reduction, cement production, petrochemical refining or district heating. Previous work performed a similar analysis using a generic pressurized water reactor (PWR) simulator as the reactor type supplying the heat[5]. The previous work simulating a generic PWR's systemic response to the diversion of steam to a similar system yielded promising results, though slightly erratic[5]. To demonstrate the ability of both types of LWRs, and the possibility of a more stable system, this analysis will focus on simulating the response of boiling water reactors (BWRs) when heat is removed from the power system.

In this paper, a generic BWR (GBWR) is simulated to demonstrate that it can support what would be equivalent to a 15% MWe diversion of the steam from its primary steam supply loop during steady state operation to feed the heat necessary to support a coupled hydrogen production facility. A 15% MWe diversion of steam during steady state operation for this simulated BWR equates a drop from 845 MWe to 720 MWe generated by the turbine.

2 Methodology and Design

2.1 Description of Software Used

GSE's Generic Boiling Water Reactor (GBWR) simulation software is a faithful representation of a BWR-5 plant, which allows the user to control all aspects of the plant, make changes, and test the results of the changes. The GBWR allows the user to design specific scenarios that are used to stress the plant in ways that cannot be done in the real world, allowing researchers to suggest changes to the systems and evaluate alternative solutions. GSE's simulators provides holistic environments to test a variety of engineering changes and understand the total impact on integrated plant performance. They are used to train nuclear, fossil, or process plant operators in real-time power plant scenarios. Therefore, when a simulator model is complete, it must accurately represent plant data for any given operational procedure or accident scenario. GSE currently has a BWR model available for use for this research. The GBWR is based on a General Electric Type 5 boiling water reactor with Mark II containment[13]. During the development of this simulator, real plant data from the reference plant was used to validate the models. This package includes RELAP, as well as legacy code that is available for immediate use. The GBWR simulator is based on hundreds of FORTRAN source code files that invokes various programs used by GSE Systems, Inc. These include RELAP5-HD, a modified version of RELAP5-3D for real-time simulation as well as JADE, a GSE owned software for generating thermodynamic and logical flowsheets and source codes[12]. RELAP5 relies on the solving of the six basic field equations for two-fluid nonequilibrium flow[14]. The model consists of two phasic continuity equations, two phasic momentum equations, and two phasic energy equations. The equations are recorded in differential stream tube form with time and one space dimension as independent variables and in terms of time and volume-average dependent variables[14]–[16].

Figure 3 is a simplified diagram of the process that is used to extract heat from the BWR and includes an extraction steam line (XSL) that removes steam from main steam line and a heat exchanger that condenses steam in the XSL before the condensate is sent to the main condenser. Heat from condensation of steam in the heat exchanger is used to vaporize deionized (DI) or demineralized water that is sent to the hydrogen or other industrial plant in a delivery heat line (DHL).

The GBWR simulator uses a graphical user interface (GUI) with screen drawings that are used to modify the underlying model, so the XSL that extracts heat from the BWR is modeled as block items in the GUI, as shown in Figure 4. Although for simplicity only a single heat exchanger is shown in FIgure 1, two heat exchangers are modeled in practice, as shown in Figure 4. The first heat exchanger is a condenser for the steam, and the second heat exchanger is a subcooling heat exchanger that increases enthalpy that can be removed from the reactor side of the heat transfer system. JADE is used to interface the new items with the RELAP model of the primary system. Pressure reliefs and high-level drains needed for the initial analysis are included in the drawing and model [13].

2.2 Description of XSL Design

The total steam flow rate is 10.57 MPPH with a 100% turbine power output of approximately 845 MWe. A thermal power delivery system was developed to syphon steam directly from the main steam line. As noted above, an extraction steam line (XSL) removes steam from the main steam line of the NPP and delivers that steam to extraction heat exchangers. The steam extracted from the main steam line enters a series of heat exchangers where demineralized water is heated and vaporized to produce high quality steam, as shown in Figure 3. This steam then travels along a 1 km pipe to the theoretical hydrogen plant where it may be used directly for hydrogen production or as a heat source. Any residual condensate is pumped back to the extraction heat exchangers. A 1 km pipe is used in the simulation to demonstrate the ability to deliver steam to industrial heat users a significant distance away without a significant reduction in steam quality.

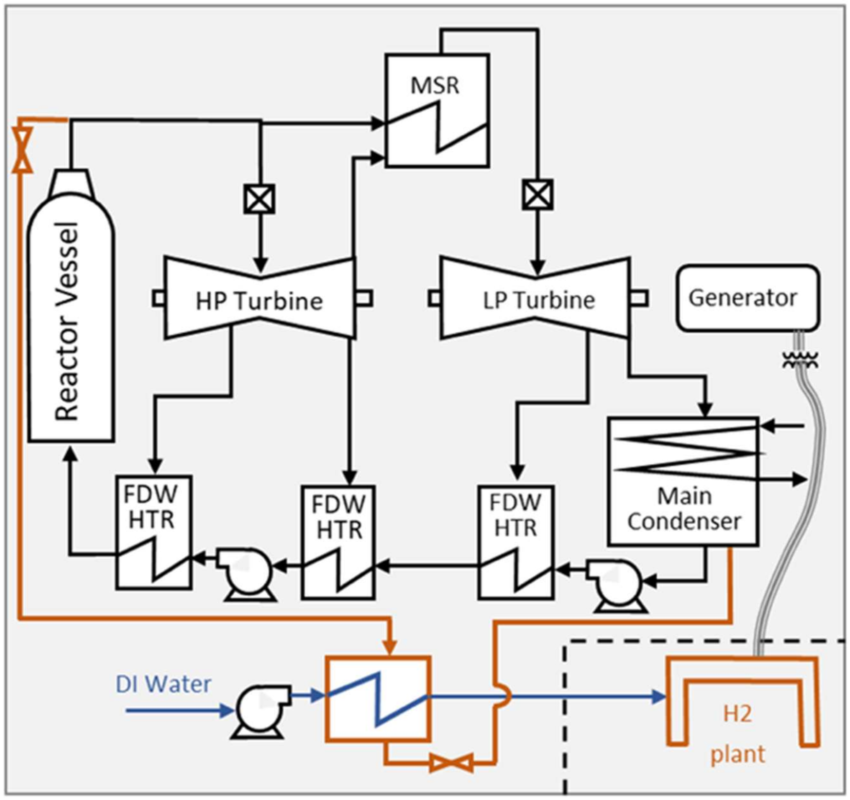

Figure 3: Simplified diagram of the Thermal Power Dispatch GBWR Simulator. The dashed line indicates the boundary of the nuclear power plant (NPP).

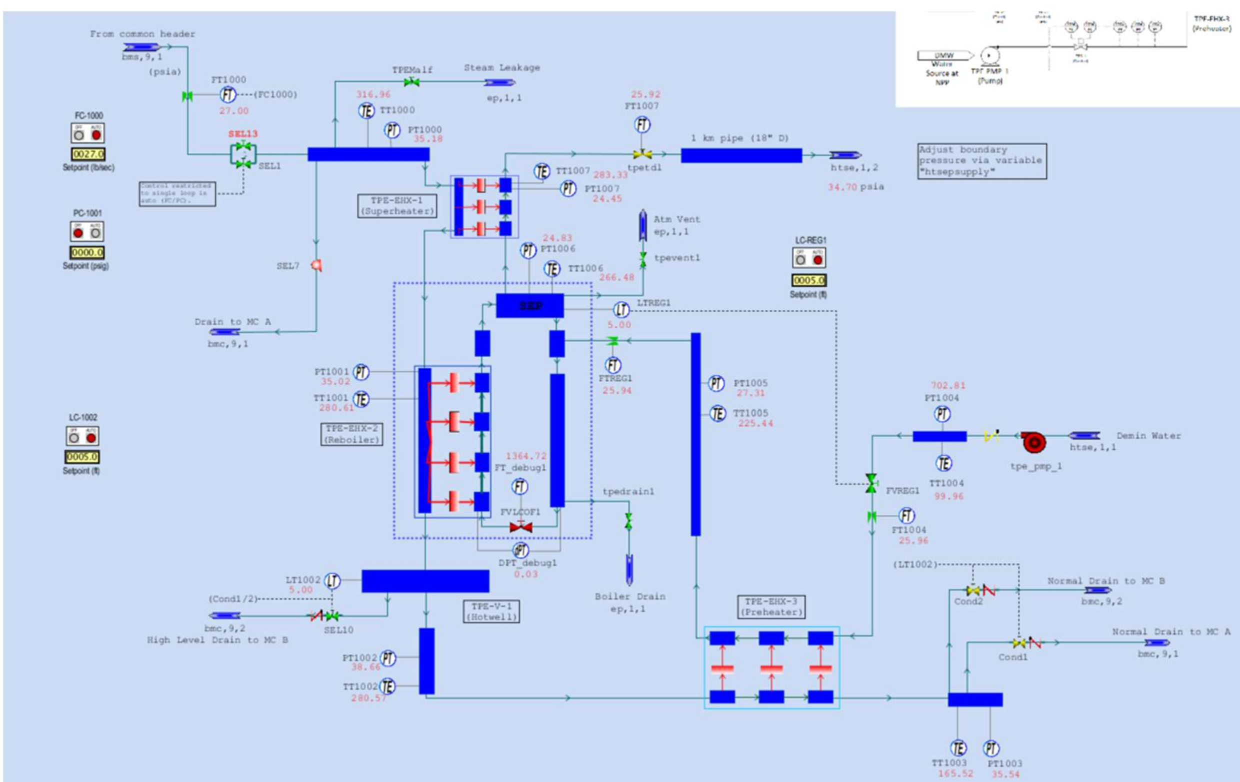

Figure 4: Schematic of the XSL. This schematic shows the flow of steam from the main steam line to the heat exchangers that provide heat to the hydrogen production facility.

Superheated steam is used in this simulation due to the low mass flow required given its high latent heat, high heat transfer coefficients, compatibility with the heat transfer fluids on both ends of the delivery line, low delivery pump power, and familiarity for NPP operations[12]. The design requirements ensure multiple purposes are accomplished, including safety of the NPP and efficient use of nuclear energy for the industrial purpose. The design requirements do not necessarily ensure that the NPP operates at maximum efficiency during thermal power dispatch operations (TPD). A leading requirement that drives the design is that the reactor power of the BWR is maintained at or slightly below the 100% steady power condition while the steam is maneuvered to allow for thermal and electrical power dispatch to the coupled industrial process.

The steam flow rate is controlled by an automatic pressure controller, which assists in maintaining a stable pressure in the main steam header. In BWR operation, the reactor power is changed by increasing or decreasing the flow of water through the BWR core. The feedwater is boiled in the reactor pressure vessel and directed to the main steam header. The reactor (steam) pressure is controlled by the steam turbine which modulates governor valves to maintain constant reactor pressure[12]. In this design, turbine power follows reactor power, which is different from a PWR in which the reactor power follows the turbine power. Since BWR reactors are maintained by holding the steam pressure constant, it makes sense to control flow in the XSL to maintain steam pressure in the main steam line and reactor pressure vessel. These controls must be modulated because simply opening the steam extraction valve to the XSL system could cause the turbine system to automatically respond and decrease turbine power[12].

Figure 4: Schematic of the XSL. This schematic shows the flow of steam from the main steam line to the heat exchangers that provide heat to the hydrogen production facility.Figure 4shows the piping system model for the delivery heat loop (DHL) in JTopmeret, a JADE program used for modeling two phase flow, typically for balance of plant systems. This version of the DHL is modeled as an open loop with an appropriate mass sink and source at the hydrogen plant to represent the heat transfer needed to create steam for hydrogen production. This approach does not capture the physics of the heat exchange process with high fidelity, especially in terms of capturing transient effects during warm up or other potential thermal power dispatch power changes. It is recognized that those precise dynamics depend on the specific thermal coupling to different industrial processes, including different levels and types of hydrogen production facilities. This approach does provide reasonable insights into the transient effects that the thermal power dispatch (TPD) system has on the BWR plant. For transient fluid dynamic simulations, a pressure versus flow rate curve is applied to the pump to provide the appropriate pressure rise as a function of desired pump flow rate[12].

The DHL contains two options for flow control, a flow controller and a temperature controller. This control scheme eliminates potential pressure instability issues in the extraction steam loop (XSL) and main steam line. When the TPD system is not operating but is expected to operate in the near future, it is beneficial to keep the lines heated and partially pressurized. This condition is referred to as Hot Standby mode, and in this mode, the steam pressure is significantly lower in the XSL than in the main steam line because the low heat transfer across the extraction heat exchangers lowers the thermal equilibrium of the steam. During the transition to full TPD, the pressure in the delivery heat loop (DHL) decreases dramatically if the flow rate is not controlled to ramp with the steam extraction flow rate. The pressure slowly recovers as the heat transfer stabilizes during the transition. Large pressure swings are undesirable not only because of increased wear on equipment but also the additional monitoring they require by the operator with increased potential for operator confusion and error. Sudden depressurization of the DHL system during power transitions is avoided by using the temperature after the first heat exchanger as the control variable for the flow rate. This approach improves the control scheme allowing the pressure to be maintained at a relatively high-level during transitions from Hot Standby to TPD operating mode.

2.3 Description of Experiment

Simulations were performed to test the transient response of the BWR plant due to the diversion of steam from the main steam header to the thermal power extraction loop until a 15% reduction of electrical power output was observed from the turbine. To accomplish this, two sets of setpoint controllers, depicted in Figure 4, were modulated until a pair of parameters were identified that reached the target of a 15% decrease in electrical power generated by the primary high-pressure turbine. In the XSL and DHL systems, two mutually exclusive setpoint controllers exist to drive the flow of steam from the main steam line.

In the XSL, PC-1000 is the setpoint controller that dictates the mass flow rate of the steam syphoned from the main steam line and PC-1001 is the controller that sets the gauge pressure just before the first heat exchanger. Both controllers feed information to the valve controller, which will regulate the governance of the valve to let in steam at the appropriate rate to match the ramp set by the setpoint controllers. PC-1000 was chosen for this experiment, as the mass flow rate of steam removed from the main steam line has a direct relationship to the amount of energy produced by the turbine and allows for the pressure within the system to vary as needed. By pairing the governor valve restrictor controller with the mass flow rate sensor, the flow leading to the XSL can be modulated instantaneously and accurately. The mass flow rate that was found to induce a 15% reduction in energy production of the turbine was 390 lbs/s steady state.

The other set of controllers, located in the DHL, control either the temperature of the steam heading to the hydrogen facility or the mass flow rate of the steam returning from the hydrogen facility. Considering that the temperature of the steam arriving at the hydrogen facility is a critical parameter for the industrial process, this variable was selected as the control parameter. Due to the nature of the thermodynamics of the heat exchanger, the highest feasible temperature was found to be 510°F (265°C). As the temperature setpoint is the variable being controlled on the hydrogen facility side of the heat exchangers, the mass flow rate of the steam is allowed to vary as necessary to balance the heat flux flowing through the system. With the two setpoints identified, 390 lbs/s steam extracted from the main steam line and 510°F steam being delivered to the hydrogen facility, a safe ramp rate is needed. A conservative ramp time of 20 minutes from no flow to max flow was chosen.

To determine if this ramp had deleterious effects on the reactor system, vital parameters were identified that would give insight to the health of the reactor core. Primary diagnostic variables, such as steam mass flow rate to the turbine, total steam flow through the main steam header, steam flow to the XSL, and turbine electrical power, were tracked to show the direct functional performance of the system. The average feedwater temperature going into the core and core power were also tracked to determine to what extent the diversion of steam would affect reactor power and temperature differential. Finally, the reactor dome pressure, main steam header pressure, and core void fraction were tracked during the ramp to give insight on the behavior of the moderator in the core and if the pressure changes seen in the core would be propagated down the line.

3 Simulator Results for Transitioning between Hot Standby and Thermal Power Dispatch

The simulation began from a hot standby with an assumed negligible steam flow diverted to the XSL and was ramped up to 390 lbs/s at a rate of 20 lbs/s per minute while maintaining a steam delivery temperature of 510°F (265°C). For operation, some amount of hydrogen would need to be diverted through the XSL to maintain hot standby for the system. As the amount of steam needed to be diverted was dependent on the needs of the industrial heat user, this simulation was performed at cold standby to determine the transient effects of standing up this system. Once the target mass flow rate was reached, the steam flow was held steady state for 10 minutes to observe behavior, then ramped down at the same rate until 0% flow was achieved. Throughout this test, the main parameters that were manipulated were the steam flow rate diverted to the XSL, which was ramped at a consistent rate, and the temperature of the steam that was delivered to the hydrogen production facility, which was held constant at 510 °F. The corresponding response to the manipulated steam flow is shown in Figure 5.

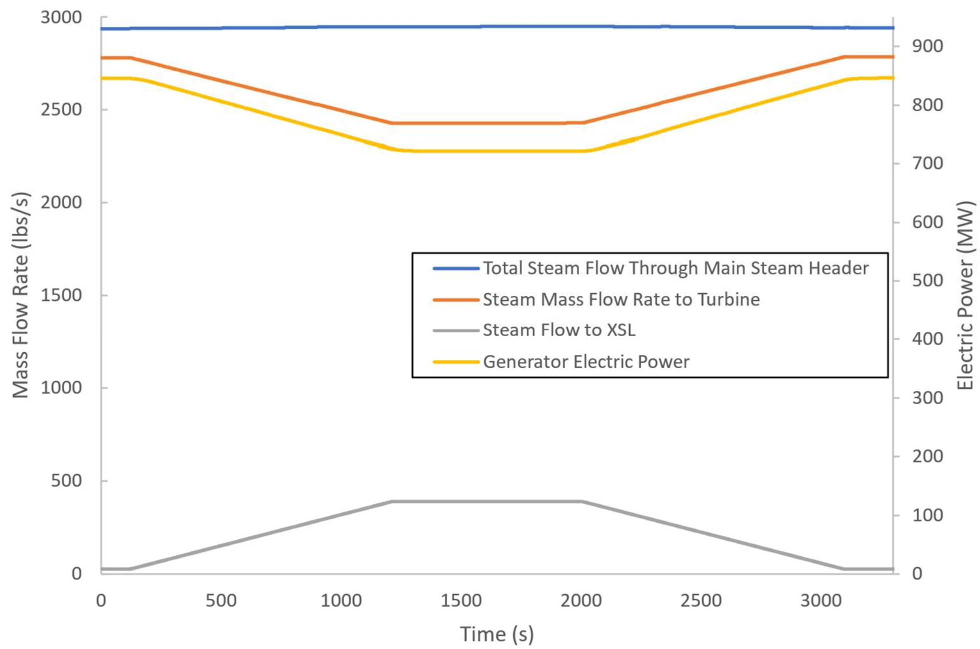

Figure 5: System steam flow rate response from ramping diverted steam to XSL from 0 lbs/s to 390 lbs/s

The general trends seen in Figure 5 demonstrates the observed response of the system. As steam was diverted from the main steam header to the XSL, the steam delivered to the turbine is reduced. The resulting reduction of steam to the turbine yields a decrease in the electric power output from the turbine. Ramping the steam delivery to the XSL to 390 lbs/s resulted in a steady state turbine electric power of 720 MW, which is a 15% drop in electrical power production from the rated ~845 MW. Though the steady state steam flow through the XSL was 390 lbs/s, it was observed that the steam flow to the turbine only decreased by 352 lbs/s, from 2780 lbs/s at hot standby to 2428 lbs/s at steady state steam extraction. This unequal drop in steam flow to the turbine can be attributed to the increase in total steam flow out of the core. The total steam flow, as can be seen in Figure 5 increases slightly, with the maximum deviation being an additional 13.3 lbs/s increase in steam flow. The trend with turbine electrical power and steam removal are both linear, which can yield a relationship between the expected change in MW generated and mass flow rate of steam. The equation below, details this relationship and the derived ratio. It is important to note that a 13.4% removal of steam flow to the turbine resulted in a 15% decrease in electrical power. This unequal drop in electrical power production due to the removal in steam indicates a drop in turbine performance as the flow of steam is reduced.

$$\frac{\Delta\, MW}{\Delta\, Mass\ Flow\ Rate\ to\ Turbine} = \frac{845.37\ MW - 720.95\ MW}{2780.64\,\frac{lbs}{s} - 2428.28\,\frac{lbs}{s}} = .3531\ \frac{MW}{\frac{lbs}{s}}$$

The direct effects on the system, noted above, also carried some important downstream effects on the overall plant as well, shown in Figure 6 and Figure 7. The average feedwater temperature to the reactor vessel trended downward with the decreased flow rate of steam to the turbine. At hot standby conditions, the average feedwater temperature was observed to be 421 °F. This temperature dropped to 410 °F at 15% electrical power removal from the turbine, which is a 3% decrease in feedwater temperature. The decrease in feedwater temperature had an almost equal but opposite effect on the power level of the reactor. The reactor's power level had an initial condition,

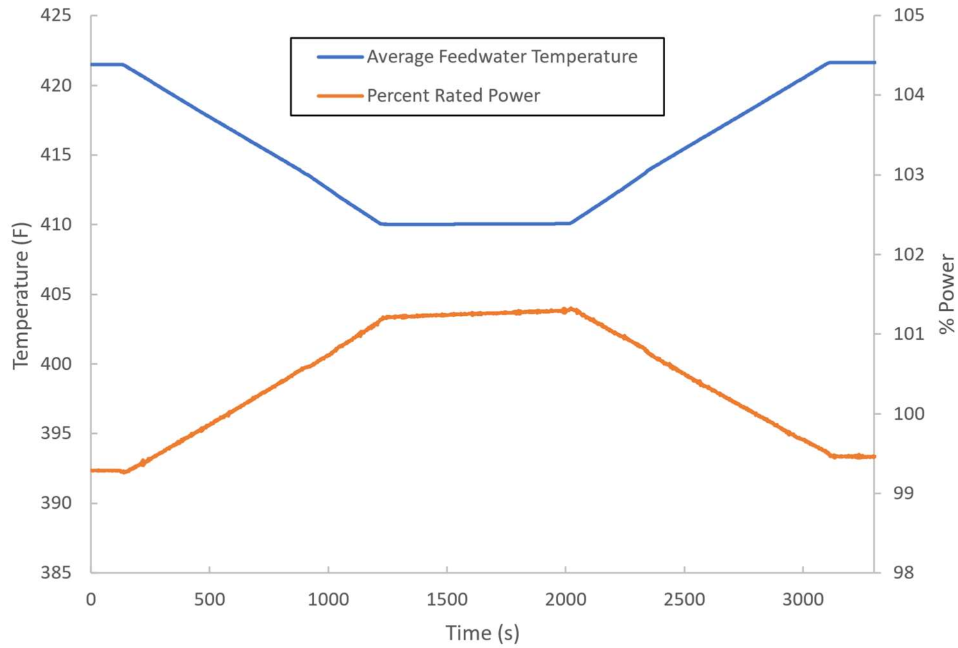

Figure 6: Transient effects of diverting steam from turbine to reduce electrical power by 15% on average feedwater temperature and reactor core power.

during hot standby, of 99.28% of rated power but rose 2.04% to 101.32% at steady state conditions. The decreased feedwater temperature decreased the void fraction of the lower third of the reactor core, resulting in an increase in moderation of the neutrons. The increased moderation yielded an increased thermal neutron population, which created more fissions to take place and thus more thermal energy to be introduced into the system.

The increase in percent power violates the safety limit that prohibits the reactor from generating more than its reported rated power. To counteract this effect, the feedwater flow rate to the reactor could be decreased by ramping down the feedwater circulating pumps, which would decrease the power of the reactor. The decrease in mass flow rate of the water entering the core would offset the decrease in feedwater temperature observed, which would counteract the observed increase in thermal power. This will balance the increased temperature rise across the core and help to maintain reactor power at or below 100%. This would most likely cause the steam flow rate through the main steam header to decrease slightly below the 100% power flow rate, rather than increase, as shown in the present example.

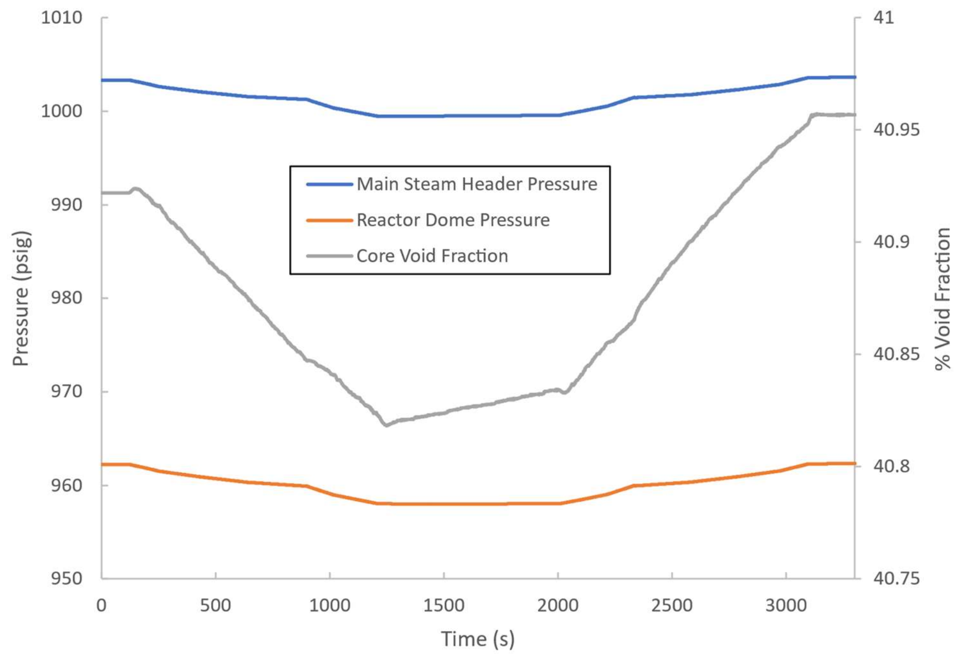

Figure 7: Transient effects on reactor dome pressure, main steam header pressure, and reactor void.

Figure 7 shows the more nuanced effects the transient had on the core. The reactor dome pressure and main steam header pressure both dropped with the decrease in steam flowing to the turbine, due to the imperfect compensation by the BWR pressure control system. The trends are not linear, with deep and gradual curves yielding local maxima at 3300s and 7000s. The pressures flatten out during steady state, which indicates that the steam supply system pressures would not continue to deviate during normal, long-term operation at this configuration. The core void is represented on this graph as well. With the decreased feedwater temperature, the voids within the core would see a noticeable drop, particularly toward the bottom of the flow channels. Due to the long ramp time, however, the positive reactivity insertion associated with a decrease in void did not yield a change in reactor power that would negatively affect the integrity of the fuel, clad, or assemblies. The change in void only takes place in the thousandths position, indicating a change that would not be detrimental to the expected flow regime in the core, indicating that the thermodynamic behavior of the moderator would remain consistent through the ramp and steady state operation.

Figure 8 through Figure 11 show the results of the same analysis with decreasing the feedwater recirculating pump speed to maintain the reactor power less than the rated power. Figure 8 shows the stepped reduction and increase of recirculation pumps A and B. Figure 9 shows similar results to Figure 7, it also shows a 2% decrease in the main steam header flow rate from 2937 lbs/s to 2879 lbs/s as the recirculation pump flow rates are reduced. The oscillations in the reactor power and void fractions are due to the step changes in feedwater recirculation pump flow rates. This analysis shows that the reactor power can be maintained under 100% rated power for a 15% reduction of the electrical power output from the turbine.

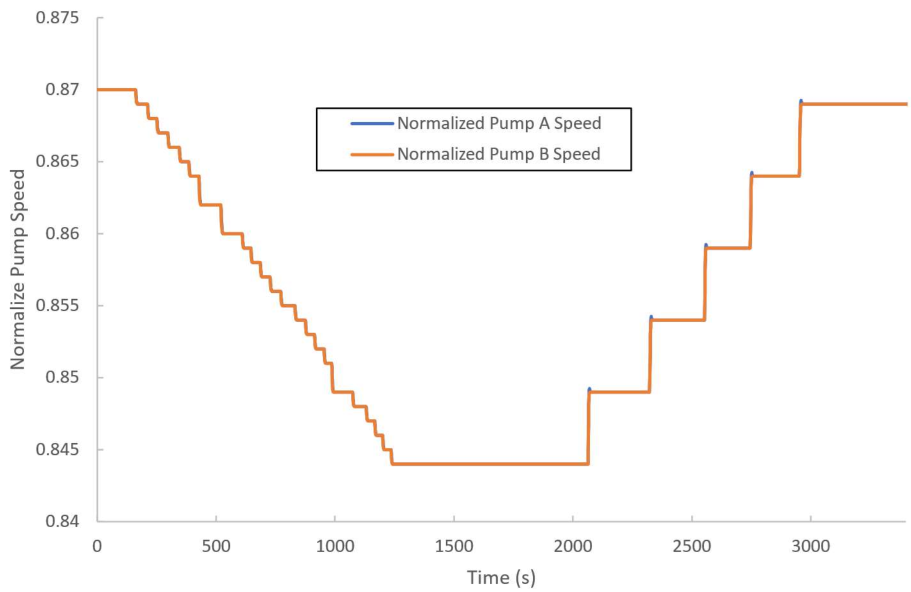

Figure 8: Normalized Recirculation Pump A and B speed.

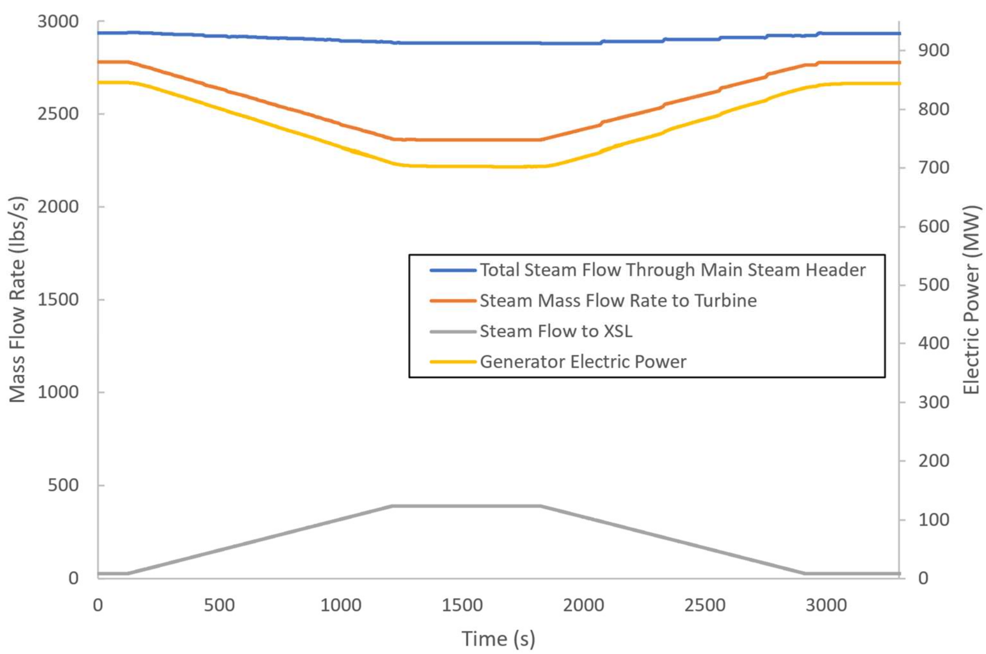

Figure 9: System steam flow rate response from ramping diverted steam to XSL from 0 lbs/s to 390 lbs/s with variable pump speed

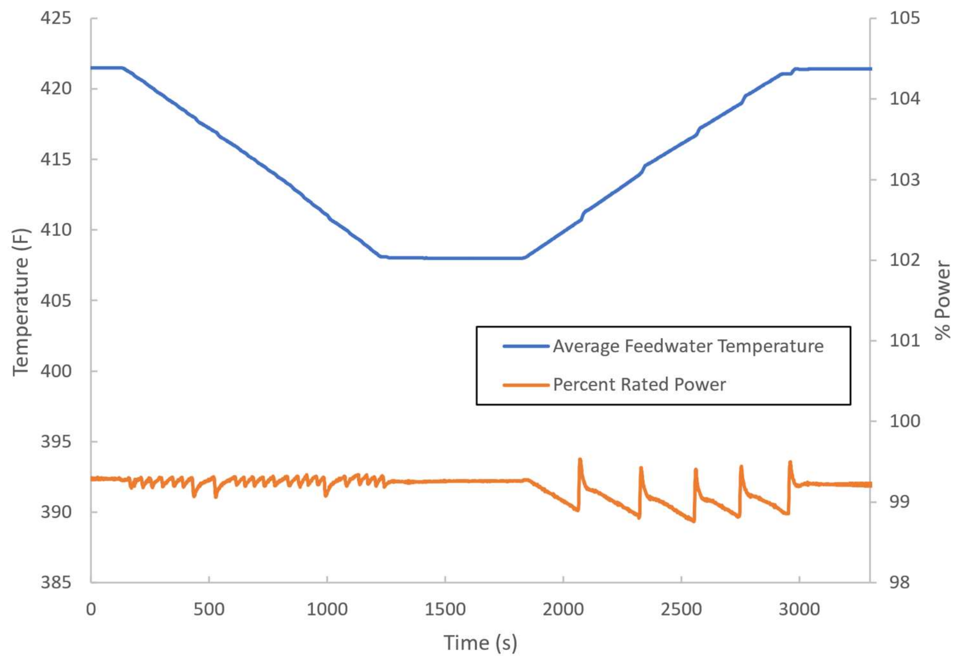

Figure 10: Transient effects of diverting steam from turbine to reduce electrical power by 15% on average feedwater temperature and reactor core power with variable pump speed.

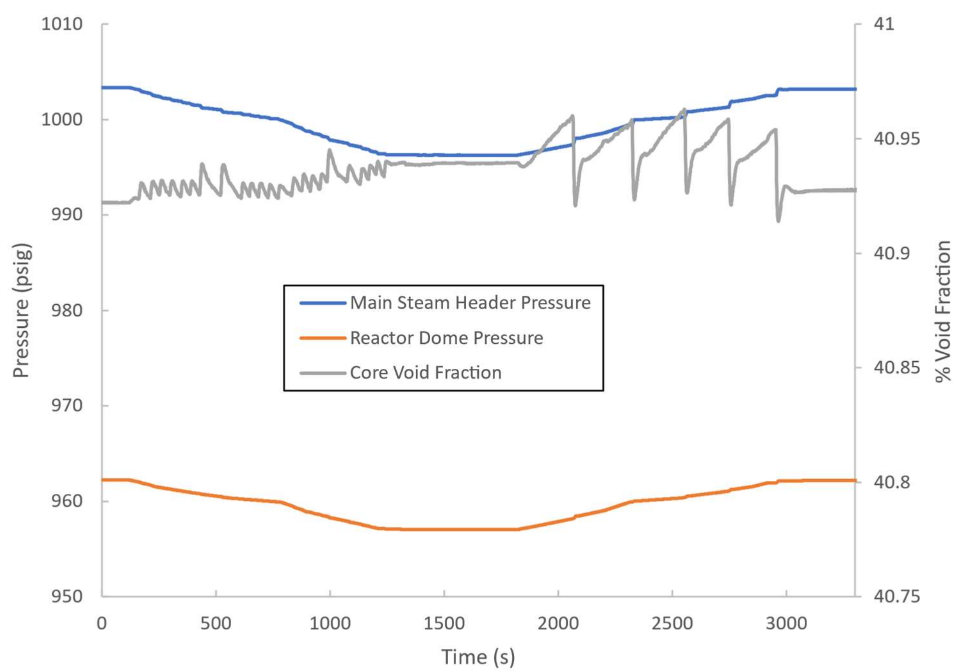

Figure 11: Transient effects on reactor dome pressure, main steam header pressure, and reactor void with variable pump speed.

4 Discussion and Conclusion

This report described the design and thermal hydraulic modeling of the coupling and testing of a Type 5 BWR and hydrogen electrolysis facility. The novel design of both a heat transfer system, which syphoned off steam from the main steam line of the primary power loop of the BWR, and the thermal power dispatch loop, which delivered the syphoned thermal power to the hydrogen facility, were built using JTopmeret, which uses RELAP5-HD for real-time simulations. The hydrogen facility in this simulation is not explicitly modeled, rather it exists as a heat sink at the end of the thermal power dispatch loop, which is modeled to be 1 km away from the nuclear reactor.

The simulation was performed using a set of automatic setpoint controllers that set either the mass flow rate or pressure of the steam on the XSL side and either the temperature or the mass flow rate of the steam being delivered to the hydrogen facility. In both cases, only one setpoint controller in each system could be used as those setpoints dictate the thermodynamics of the variables not being explicitly defined. The setpoint controller on the XSL side controlled a governor valve which controlled the flow of steam being syphoned from the main steam line. The setpoint controller on the TDL side controlled a pump on the return pipe of the loop. Choosing to use the mass flow rate as the setpoint indicator to ultimately control the heat transfer to the hydrogen production facility means that the pressure within the system would not be explicitly controlled. As the ramp speed chosen for this experiment is slow and tame, this did not manifest in any lack of control or noticeable transient pressure effects within the XSL, however, controlling the steam pressure within the XSL would allow for a smoother transition to steady state operation and prevent early maintenance on the system.

The analysis performed showed the BWR was able to support a 13.4% diversion of steam flow to an external industrial user without any detrimental effects on the power production of the core. In this case, a 13.4% removal of steam from the main power loop of the core resulted in a 15% reduction in electrical power produced by the generator. The experiment diverted 352 lbs/s of steam at a ramp of 20 lbs/s per minute to the XSL, where it heated low pressure steam to 510°F (265°C) before traveling to the hydrogen production facility. The reduction in electric power produced by the generator was linear and can be represented as a ratio of .3531 MW/lbs/s of diverted steam. There was a noticeable increase in reactor power, 2.04%, due to the decrease in feedwater temperature entering the core. To avoid power ratings larger than 100%, ramping down the feedwater circulation pumps in congruence with the decrease in feedwater temperature would offset the increase in power. Changes to the normal operation of the existing power conversion systems will require further analysis as to ensure that the plant continues to operate within licensed conditions. For hydrolysis, the temperature of steam provided to the facility is more than satisfactory, but for any of the more advanced thermochemical hydrogen production methods, a more sophisticated temperature upgrading system is needed to make the process as efficient and economical as possible.

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