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Title: Proof of Concept of the use of RAVEN and RELAP5-3D for Risk Informed Safety Margin Characterization

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This project was a proof of concept of the use of the RAVEN software, a tool developed for the Risk Informed Safety Margin Characterization (RISMC) approach, with RELAP5-3D. This novel approach combines older probabilistic and mechanistic approaches to look at how and why the complex systems of a nuclear power plant might fail in an accident scenario in greater detail than the older approaches allow. This is done by combining the mechanistic outcomes of RELAP5-3D with random sampling of stochastic parameters to account for the probabilistic elements of an accident scenario. Proof of concept was provided by modeling Station Blackout scenarios for both a generic Mark 1 Boiling Water Reactor (BWR) and for the Multi-Application Small Light Water Reactor (MASLWR) design. The research shows that RAVEN and RELAP5-3D can be used to collectively generate a reasonable and realistic failure space for numerous stochastic parameters, and in doing so allows for greater examination of which safety systems might fail, how they fail, and what can be done to most effectively improve them. The intent is
that these insights will be used to more effectively examine power plant accident scenarios and allow for more informed decision making with regards to power plant safety margins.
Proof of Concept of the use of RAVEN and RELAP5-3D for Risk Informed Safety Margin Characterization

by

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APPROVED:

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I understand that my thesis will become part of the permanent collection of Oregon State University libraries. My signature below authorizes release of my thesis to any reader upon request.

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Thomas H. Riley, Author
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Proof of Concept of the use of RAVEN and RELAP5-3D for Risk Informed Safety Margin Characterization
1 Introduction

Idaho National Laboratory (INL) has developed a novel approach to Probabilistic Risk Assessment (PRA) analysis, using a Monte Carlo (MC) informed approach. RAVEN is a probability distribution agnostic, code agnostic platform for using MC style sampling of input parameters for Thermal Hydraulics (TH) codes for the purposes of a Risk Informed Safety Margin Characterized (RISMC) approach to PRA analysis. As well as the standard deterministic safety margins – often characterized as a ratio of the stress on a safety mechanism to its ability to withstand stress – RISMC allows for probabilistic safety margins – the probability that a safety mechanism will be overwhelmed. This allows for new, more detailed analysis of transients.

The goal of this project was to provide the proof of concept of RAVEN as a tool for examining the probabilistic strength of safety margins and discovering and mapping paths to failure in Nuclear Power Plant (NPP) accident scenarios. As part of this probabilistic analysis, a TH code was needed for systems level computational modeling, for which RELAP5-3D was used. Two proof of concept case studies were completed for this thesis: an analysis of a Station Blackout (SBO) transient for a generic Boiling Water Reactor (BWR) and an analysis of a SBO transient for a Multi-Application Small Light Water Reactor (MASLWR). In both cases, numerous initial conditions of the transient and basic physical parameters of the reactor were sampled using MC methods and known probability distributions. For each set of sampled parameters, a TH model was run to determine whether the reactor would reach core damage conditions or if safety systems would keep the reactor safe.
2 Literature Review

2.1 PRA development

Probabilistic Risk Assessment has existed since the dawn of commercial Nuclear Power in the United States. In the early days of Nuclear Power generation (1957-1975), US Nuclear Safety Regulations were guided by WASH-740, ‘Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants,’ or ‘The Brookhaven Report’ for short (USAEC, 1957). The Brookhaven Report is an analysis of what is, in its writers’ opinion, the maximum credible accident. They decided this was a major meltdown at a plant with no containment building, unfavorable weather conditions, and half of the reactor released into the atmosphere as a fine dust. Original estimates from these analyses and assumptions were 3400 deaths, 43,000 injuries, and property damage of roughly $60 billion (2013 dollars) of property damage. This was later increased to 45,000 deaths, 100,000 injuries, and roughly $130 billion in property damage when the Brookhaven Report was revised to account for newer, larger reactors (WASH-740). Assuming the worst case conditions for all variables, however, was unrealistic and later revised through further, more realistic analysis.

WASH-1400 (Rasmussen et al, 1975), titled “The Reactor Safety Study” but also called the “Rasmussen Report,” after its primary author, changed the approach to accident analysis from assuming the worst conditions in all cases to using fault trees and event trees that describe the evolution of an accident scenario and assess the probability of an accident progressing to core damage and the release of radionuclides to the environment. It also considered the consequences of such an accident. WASH-1400 concluded that the probability of a full core meltdown in a modern (at the time) LWR was roughly 1 per 20,000 years of reactor operation. The American Physical Society later criticized the report for only accounting for deaths in the first 24 hours after
an accident, ignoring the potential of a high radiation dose to cause cancer deaths many years after the initiating event (APS, 1984), but WASH-1400 remains as the first attempt to apply modern fault tree methods to nuclear power plant accident analysis.

In response to the accident at Three Mile Island (TMI), NUREG-880 was published (USNRC, 1983), setting new public safety goals for Nuclear Power – mostly qualitative goals, but with one quantitative goal of having no more than one core melt per 10,000 reactor operating years. These goals were set to provide “an explicit policy statement on safety philosophy and the role of safety-cost tradeoffs” (NRC, 1983), give industry safety discussions a series of guidelines to direct safety related decisions, state the NRC’s views on the acceptable level of risk to public health, and to address increasing public concern in the wake of TMI.

Three years later the NRC revised 10CFR50 with 51FR30028, containing two qualitative goals and two quantitative goals (USNRC, 2002a). The Quantitative Goals contained in 51FR30028 are:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 0.1% of the sum of prompt fatality risks from other accidents to which members of the U.S. population are generally exposed; and
- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 0.1% of the sum of cancer fatality risk resulting from all other causes.

The Qualitative Goals contained in 51FR30028 are:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health; and
Societal risk to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should bear no significant addition to other societal risks.

The Quantitative and Qualitative Goals of NUREG-880 are two ways to express the same goal – Nuclear Power must be safe. NUREG-880 also recommended that “the overall mean frequency of a large release of radioactive materials to the environment should be less than 1 in 1,000,000 years of reacter operation,” a guideline later codified into 10CFR50.109 (USNRC, 2003b) and used when evaluating facility changes and updates.

Since NUREG-880 was published, many studies of the consequences of Nuclear Power have been performed, including NUREG-1150 (USNRC, 1990) in 1990 and State-of-the-Art Reactor Consequence Analysis (SOARCA) in 2012 (USNRC, 2012). NUREG-1150 was an analysis of five nuclear power plants, using Accident Progression Event Trees, to quantify the progression of power plant accidents and the likelihood of safety systems being unable to properly withstand the accident. The plants analyzed were the Peach Bottom Atomic Power Station in Pennsylvania, Surry Power Station in Virginia, Grand Gulf Nuclear Generating Station in Mississippi, Zion Nuclear Power Station in Illinois, and Sequoyah Nuclear Generating Station in Tennessee. These plants were chosen to give a broad survey of both Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs) and of multiple models of BWR and PWR, and analyzed using the Accident Progression Event Tree approach.

An Accident Progression Event Tree (APET) looks at the way containment can fail or be bypassed, as well as the way severe accidents affect the mode and timing of containment failure and magnitude of radionuclide release to the environment (Hakobyan et al, 2008). In the older WASH-1400 event trees, accident progression was examined solely on the basis of whether a particular safety system succeeds or fails on demand, where the new Accident Progression Event Tree approach asks non-binary questions like ‘how long does battery power last’ and ‘how fast
does the vessel leak water in a LOCA’ and so forth. The questions all have at least two answers, creating multiple branches to follow after each branch point in the sequence of events. Generally, fault tree analysis is not used to generate branch probabilities for branch points in an APET. Branching probabilities are calculated using physical conditions from the severe accident scenario in comparison to the criteria for the different branches of a branch point (Hakobyan, 2006). Unfortunately, APETs do not give a deterministic pass/fail outcome for a scenario – they are inherently probabilistic, and uncertainty analysis is used to determine failure probability by performing accident progression calculations with different input assumptions.

SOARCA later built on NUREG-1150 by incorporating onsite and offsite actions that may prevent or mitigate accident consequences, as well as more detailed computer modeling to look at how severe accident conditions affect a reactor and how a radionuclide release could affect the public. This analysis was done using MELCOR and MACCS2, two severe accident scenario modeling codes, to look at Peach Bottom Atomic Power Station and the Surry Power Station (USNRC, 2012).

Modern PRA efforts are divided into three levels that correspond to important transition points in the progression of an accident scenario. Level 1 starts with an initiating event and ends at Core Damage, Level 2 starts with Core Damage and ends with Radionuclide release, and Level 3 starts with Radionuclide release and examines the consequences. The advantage of higher level PRA is that it provides more in depth and detailed analysis of the risks and repercussions of accident scenarios than lower level PRA. The disadvantage is that higher level PRA costs a great deal more than lower level PRA.

A Level 1 PRA is solely a calculation of the core damage frequency. It looks at accident progression in terms of accidents that lead to core damage to estimate the Core Damage
Frequency (CDF), starting from an initiating event and branching out through safety system success or failure until core damage is reached. This is represented graphically with fault trees. Each of these fault trees is analyzed to provide a core damage frequency for that particular accident, then all the frequencies are added together to get a total CDF.

A Level 2 PRA begins at the end of a Level 1 PRA by examining the plant’s response to the Level 1 events that lead to core damage, and analyzing how the plant responds to this state. Incidents that lead to core damage are typically called severe accidents. Level 2 PRA is analysis of the plant’s severe accident response, and whether it is capable of keeping the severe accident consequences sealed within the containment building. This uses further fault trees and, rather than primarily looking at safety systems success/failure, looks at phenomenological events like “Steam Generator tube rupture” or “hydrogen explosions.” Because different severe accident paths lead to different Plant Damage States (PDS) when Core Damage occurs, severe accident progression analysis is necessary for each PDS, making Level 2 PRA drastically more expensive and lengthy than Level 1 PRA.

A Level 3 PRA begins with loss of secondary containment. It estimates the consequences of a radionuclide release, and combined with levels 1 and 2, presents an overall estimate of the effect on the people living near the plant, and the potential for the plant to contaminate the surrounding environment with radioactive material. The consequences, both in terms of the health of the public and the quality of land, depend on multiple factors, though all of these factors affect both the health of the public and the land quality. For example, population density and evacuation readiness and conditions affect only the health effects white others, like weather conditions, geography, and the size of the radionuclide release affect both. A Level 3 PRA estimates the final measure of risk by combining the consequences and likelihood of a radionuclide release. However, it is rarely done because it requires a great deal of computational
power and is very expensive, as the various paths to radionuclide release in an accident scenario affect the nature of the radionuclide release, and these differences need to be accounted for.

2.2 Monte-Carlo Applications

A key piece of RAVEN’s functionality is Monte Carlo sampling of input parameters for deterministic code calculations. At a simplistic level, Monte Carlo sampling is the idea that if a person throws a sufficient number of darts at a dartboard, one can establish, with good confidence, the probability of a thrown dart landing in any designated area of the dartboard. It is an extremely well established technique for finding answers to problems with uncertain input parameters and problems too complex to be solved analytically or computationally. In this project, the former was of greater concern, as current Thermal-Hydraulics codes are more than capable of modeling and predicting plant behavior mechanistically. The greatest pitfall of TH codes is that one cannot precisely predict the initial conditions of a plant when a transient begins (for example, the temperature outside affects the state of the plant, and is nigh impossible to predict precisely). To circumvent this issue, the Monte Carlo technique is used to repeatedly run a TH simulation of choice with a variety of initial conditions randomly sampled and imposed for each run. Using Monte Carlo techniques to randomly sample initial conditions for a mechanistic simulation is done in other fields to establish a good prediction of what conditions will cause a system to fail.

Monte Carlo techniques have been used to account for stochastic parameters in a variety of fields, including radiation transport, robotics, aerospace, microelectronics, and telecommunications applications. In Aerospace applications, Monte Carlo techniques were used to account for component failure in models of long term manned spaceflight operations, with particular interest in a manned mission to Mars (Bavuso, 1997). By sampling for component
failure of both active components breaking down during operations, as well as on demand failures from backup components, the researchers examined the overall reliability of a spaceflight system for a manned mission to Mars. In the past, Monte Carlo techniques were not used in aerospace applications for reasons similar to those why Monte Carlo techniques have not been previously used for systems level thermal hydraulics modelling of NPPs – it was excessively expensive due to the cost of the computer resources involved with using Monte Carlo techniques. However, advances in both supercomputing and in Monte Carlo techniques themselves have reduced the costs of Monte Carlo techniques greatly.

In microelectronics, Monte Carlo techniques are used to aid in modelling what are called ‘single event effects’ (Weller et al, 2010). These are when the fine semiconductors in microelectronics are disrupted by radiation. Due to the nature of radiation, the timing and location of a radiation event is inherently stochastic and cannot be known a priori. Previously, other methods worked sufficiently well to model single event rates and effects, but the increasingly minute size of computer semiconductors has necessitated the use of Monte Carlo techniques to model the effects of radiation on microelectronics.

In robotics, Monte Carlo techniques are used for autonomous robot localization (Wu et al, 2006). For the purposes of mapping out a robot’s surroundings, it is impossible to know in advance where obstacles will be, or to evaluate whether an apparent path is viable. According to Wu et al, other, previous techniques for an autonomous robot mapping its surroundings are not suitable for unstructured outdoor environments. Using Monte Carlo techniques, these environments can be navigated. According to Wu et al, Monte Carlo techniques are also more computationally efficient than older methods of localization, on top of being more versatile.
In telecommunications, Monte Carlo techniques are used for the analysis of error rate performance in communications systems (Bononi et al, 2009). Bononi et al lay out a methodology for the use of Monte Carlo techniques in analyzing telecommunications systems. With their methodology, Monte Carlo techniques are adaptively used to improve Importance Sampling techniques – an older method for estimating the occurrence of events in the system – by making them self-adapting. This greatly lowers the prior knowledge one needs of the system, and shortens the planning and construction phase of the model. Similarly, in NPP modeling, by using a reactive model and Monte Carlo techniques, it is possible to analyze the transient event sequence of a reactor without knowing a great deal, in advance, about how the plant will respond to the transient.

Given these prior uses in a variety of situations where the researchers lack prior knowledge of the exact circumstances of a situation to be investigated, the Monte Carlo method employed by RAVEN appears to be an appropriate way to account for the uncertainty in the initial conditions of a plant transient scenario.

2.3 RELAP5 validation and modeling of LWRs and Natural Circulation (NC) conditions

RELAP5-3D was the model used with RAVEN for this study. Originally developed at Idaho National Laboratory (INL), it is a transient simulation code modeled to examine light water cooled reactors under accident scenarios (RELAP5-3D code manual, Vol.1, 2005). It is one of the most used codes for thermal hydraulics analysis and is an industry standard in nuclear power and research. It has been validated and assessed extensively on LWRs (Soares et al, 2011, Lyria et al, 2013) and is a primary benchmark for LWR safety analysis. Because of this extensive use, it is known that RELAP5 is suitable for modeling LWRs – the only concern is whether it can be used for natural circulation conditions.
RELAP5 has also been validated for NC conditions (Guzhi, Xinrong, and Xingwi, 2013) and found to be a viable TH code for NC conditions (Kozmenkov, Rohde, and Manera, 2012). One study (Mangal, Vikas, and Nayak, 2012) found that there were issues using RELAP5 to model NC conditions if the spatial nodalization of the problem is too large or too small – if the grid is too coarse, it will fail to capture spatially fine NC phenomena, and if the grid is too small, the simulation will reach the Courant limit (a condition required for a stable solution in which the fluid velocity multiplied by the timestep and divided by the spatial node length must be less than a value specific to the method used to solve the discretized equation) and not achieve a stable solution (Courant, Friedrichs, and Lewy, 1967). However, these are standard concerns with RELAP5 and not particular to NC conditions, as the Courant limit applies to any and all situations where partial differential equations are solved numerically with the method of finite differences. Additionally, Reis et al used RELAP5 to model a TRIGA reactor (Reis et al, 2010). The TRIGA research reactor is a common model of research reactor that can be built in NC or forced convection configurations, and Reis et al evaluated a NC TRIGA reactor configuration based off the work of Marcum et al (Marcum et al 2010). Given that the problems facing the use of RELAP5 to model NC conditions apply to effectively all uses of RELAP5 and that other studies found RELAP5 to produce acceptably accurate, it was deemed an appropriate code for use in modeling MASLWR.

2.4 Prior MASLWR modeling

To date, only one computational thermal-hydraulics modeling study of MASLWR has been published (Mascari et al, 2010). The researchers used TRACE – TRAC/RELAP Advanced Computational Engine – to evaluate the code’s ability to predict heat exchange between the MASLWR primary and secondary loops for the MASLWR helical coil steam generator under natural circulation (NC). One of the goals of this project was to further explore RELAP’s
capabilities in regards to MASLWR – considering its extensive use in both LWRs and in NC conditions, it was believed that RELAP would make an appropriate code for modeling MASLWR. For core modeling, a study using SIMULATE-3 was performed (Soldatov and Palmer, 2010) to find a more optimal core design for MASLWR, but that is somewhat outside the scope of this project, as the work presented here is focused on thermal-hydraulics, transient analysis, and risk analysis. In terms of physical testing and data to use for benchmarking, the MASLWR test facility at Oregon State University (OSU) has been used to perform tests of the passive safety systems of MASLWR and the data generated can be used to benchmark thermal-hydraulics simulation performance (Reyes et al, 2007).

3 Research Question

The goal of this project was to examine and, if possible, validate the RAVEN code as a way to perform RISMC approach PRA analysis. Station Blackout transients were examined for two reactor designs until they had been examined sufficiently to provide proof of concept. This was determined to be when sufficient data had been produced to establish a predicted failure space for the stochastic parameters with regards to plant safety, as this would allow for the predictions of RAVEN to be compared to experimental data. Additionally, Oregon State University (OSU) and NuScale use the MASLWR design for design certification studies in the ongoing development of the NuScale reactor design.

For RAVEN to be considered a successful system for analyzing transients under the RISMC approach, it needs to produce a prediction for the failure space of stochastic parameters for an accident scenario and for this failure space to match, within reason, established data on paths to failure in an accident scenario. Failure to predict known paths to failure in an accident scenario (i.e. false negatives) or predicting unreasonable paths to failure (i.e. false positives) are
both reasons to discard a set of RAVEN predictions. Where prior data on paths to failure is not available, engineering judgement must be used to evaluate and examine RAVEN’s predictions to see if they are reasonable and realistic.

4 Methodology

4.1 The RISMC Approach

The RISMC (Smith, Rabiti, Martineau, 2012) approach not only allows the frequency of undesirable transient outcomes to be determined, but also how close the system comes to those outcomes when a transient occurs, and how probable these outcomes are. To accomplish this, the RISMC approach uses both mechanistic and probabilistic analyses. The interaction between these approaches is described in Figure 1. The Plant Description directs both RAVEN and RELAP5-3D in that it is the guideline by which both the RELAP5-3D mechanistic model and the RAVEN probabilistic model are developed. RAVEN is used to handle elements like stochastic parameter probability distribution functions, which in turn represent operational rules and failure models, as well as being used to generate combinations of stochastic parameters to account for a variety of scenarios. RELAP5-3D is used to model all of the mechanistic, physical elements of the plant – primarily thermal-hydraulics, but also heat transfer and material properties to a lesser extent. RAVEN and RELAP5-3D inform each other in that physical parameters from RELAP5-3D are accounted for when selecting stochastic parameters and probability density functions in RAVEN, and the scenarios generated by RAVEN dictate the progression of the physical model within RELAP5-3D. Both RAVEN and RELAP5-3D inform margin and uncertainty quantification in that both the
outcomes of physical scenarios and the probability of those scenarios occurring are used to support decisions regarding safety margins.

The probabilistic analysis is performed with standard risk assessment techniques, and the mechanistic approach is accomplished through plant physics calculations. By melding the two, uncertainties can be quantified and safety margins characterized. This interaction between the two approaches is accomplished through alterations in the plant physics model parameters and accident scenarios. In the case studies presented here, a single accident scenario was run in each case study and was chosen to be a Station Blackout (SBO) accident, in which a Loss of Offsite Power (LOOP) accident occurs, and the Emergency Diesel Generators (EDGs) fail to compensate and provide AC power for the plant.

Figure 1: Probabilistic and Mechanistic Approach Interactions (Mandelli et al, 2013)
4.2 BWR Case Study

In a BWR, during normal operations water is boiled directly in the core, and the resulting steam is directly utilized to drive a turbine, then is cooled in a condenser, turned back into liquid water, and pumped back into the core for a complete loop of the water. A high level schematic of the BWR components involved in this process is shown in Figure 2.

Upon exiting the core of the BWR, the steam is passed through a series of heaters and steam dryers in the Upper Plenum of the Reactor Vessel to ensure that there is no liquid water vapor carried along by the steam. Because of the high rotational velocity and fine precision of a turbine blade, liquid water present in the steam can cause tremendous damage to the turbine and plant. After being heated and dried, the steam is piped out of the Reactor Pressure Vessel (RPV) and into the two stage turbine. It first enters the high pressure stage of the turbine then enters the low pressure stage of the turbine. These provide mechanical energy to a generator, which converts it to electricity. A two-stage turbine is not required for a plant to operate, but significantly boosts the efficiency with which the turbine and generator can produce electricity from the heat the core produces.
After leaving the turbines, the steam enters the condenser, where it exchanges heat with a secondary loop of water that acts as an ultimate heat sink. This generally takes the form of either cooling towers that release the heated steam from the secondary side into the atmosphere, or a large body of water that acts as a nearly infinite heatsink. The steam on the primary side of the condenser is converted back into liquid water and pumped through water heaters and back into the core, where it is heated back into steam and the cycle begins again.

BWRs have a variety of redundant, diverse, and independent safety systems. These safety systems include the Reactor Protection System (RPS), the various components of the Emergency Core-Cooling System (ECCS), the Standby Liquid Control System (SLCS), and the Containment Building. The RPS is a computerized
system designed to quickly and completely shut down the reactor and render it into a safe and stable configuration without human interference. The RPS will SCRAM the reactor, which is to quickly insert all of the control rods into the reactor, killing the nuclear chain reaction. Additionally, the RPS will activate ECCS subsystems as needed to provide emergency core cooling. SLCS is a safety system that injects a neutron poison into the reactor to shut down the chain reaction, and acts as a backup to the RPS in case of failure to SCRAM the reactor.

The ECCS consists of numerous subsystems to provide makeup water to the core to keep the fuel covered and cooled, including the Automatic Depressurization System (ADS), the High-Pressure Coolant Injection system (HPCI), the Reactor Core Isolation Cooling system (HPCI), the Low-Pressure Core Spray system (LPCS), and the Low-Pressure Coolant Injection system (LPCI). With the exception of the ADS, all of these systems, at a basic level, add water to the core. The ADS is a system to vent much of the contents of the core, reducing the pressure and allowing the high volume, low pressure systems to be used. HPCI and RCIC are high pressure systems that can be used to add lesser amounts of water to the core at high pressure, while LPCS and LPCI are low pressure systems that can be used to inject huge amounts of water into the core once it is depressurized. In a SBO transient, LPCI and LPCS cannot be used because they require AC power, while HPCI, RCIC, and the ADS can be operated while battery power is available. Figures of the SLSC and all discussed ECCS subsystems can be found in Appendix A.

In the first case study, a generic BWR power plant with a Mark I containment was modeled. The Plant Physics Calculations were performed using the RELAP5-3D
For the thermal-hydraulics simulations, the main structures examined were the Reactor Pressure Vessel (RPV) and primary containment – the Drywell, Wetwell (also called the Pressure Suppression Pool, or PSP), and Reactor circulation pumps. These are pictured in Figure 3. Additionally, the High Pressure Core Injection System (HPCI), Reactor Core Isolation Cooling System (RCIC), Safety Relief Valves (SRVs), and Automatic Depressurization System (ADS) were considered.

Figure 3: Cutaway View of the BWR Mk. I Primary Containment (Mandelli et al. 2013)

The HPCI injects water from the Condensate Storage Tank (CST) into the RPV at high pressure and at a high rate (5000 gallons per minute). The HPCI is powered by a
turbine that siphons steam from the main steam line and vents it into the wetwell, providing for both HPCI and RCIC. The valves controlling the flow of steam to the turbine are powered by onsite batteries, and HPCI and RCIC lose power and shut off when the batteries are no longer able to control the valves that feed the turbine. RCIC is similar to HPCI, but injects water at a lower rate (600 gpm). The SRVs are battery powered valves that control the RPV pressure and vent excess steam into the wetwell. The ADS is a separate set of battery operated relief valves that are capable of rapidly depressurizing the RPV. This rapid depressurization is known as an RPV blowdown. Further discussion and diagrams of these safety systems can be found in Appendix A.

In addition to the actual physical specifications of the BWR, extensive control logic was implemented to accurately reflect when and how various safety systems of the BWR would engage. These included, among other things, a series of four graphs (Figures 4a, 4b, 4c, 4d) from the BWR technical specifications that dictated boundaries for the RPV pressure, wetwell temperature, pressure, and level, and the drywell temperature and level.
Figure 4a – Boundary Limit for PSP level vs. RPV pressure (Mandelli et al, 2013)

When the first of these bounds (Figure 4a) was exceeded, HPCI and RCIC would stop drawing water from the Condensate Storage Tank and begin pulling water from the PSP. In Figure 4a, the boundary being exceeded meant the PSP water level being too high or the RPV pressure being too high. When the other three of these bounds (Figures 4b, 4c, 4d) were exceeded, the plant would, if capable, blow down. Blowdown was only prevented by an inability to operate the battery powered ADS valves (ie only if battery power had been exhausted).
In Figure 4b, the Drywell temperature and pressure needed to correspond to a point below and to the right of the line, or the ADS would activate. As a Station Blackout incident does not involve steam being vented into the drywell – what one would expect to cause the drywell temperature and pressure to rise – this curve was never observed to have triggered an ADS activation.
For the PSP limits shown in Figure 4c, if the PSP temperature grew too great for a given PSP water level and RPV pressure, the ADS was initiated and the reactor depressurized. For the purposes of analysis, only the blue >900 psig line was used, as the RPV pressure was held between 900 and 1100 psig until the ADS was activated.
Figure 4d – Boundary Limit for PSP Level vs. PSP Pressure (Mandelli et al, 2013)

For Figure 4d, if the PSP Pressure grew too great, the ADS would trigger. Here, the PSP water level was given no upper limit, as the curve in Figure 4a accounts for that contingency. Figures 4c and 4d were the limitations of the high pressure safety systems that were responsible for all observed ADS activations in the simulations, as the Drywell temperature and pressure were never observed to be problematic. It is expected that the curve in 4b would be used more in a Loss of Coolant Accident (LOCA) scenario where coolant would potentially be vented directly into the drywell.

Along with the wetwell and drywell heat capacity curves, logic for the SRVs, HPCI, and RCIC was implemented. The SRVs were set to open when the RPV pressure
reached 1100 psia and close once the RPV pressure dropped to 900 psia, and HPCI and RCIC were set to engage when the core is below 39.67 feet of water in the RPV and to shut off when the core reaches 48.5 feet of water. These set-points were designed to keep the core cooled and at a safe pressure.

To accurately reflect the varying nature of transients, and to capture as many of the nuances as possible, numerous stochastic parameters were chosen to sample. These parameters were sampled using a Monte-Carlo method approach, using previously established boundaries and distributions for the Monte-Carlo sampling, and the numbers reflecting these stochastic parameters within the RELAP5-3D model were altered accordingly. The stochastic parameters chosen were as follows in Table 1.

<table>
<thead>
<tr>
<th>Stochastic Parameter</th>
<th>Parameter Range Considered</th>
</tr>
</thead>
<tbody>
<tr>
<td>EDG Failure Time</td>
<td>0 hours - 4 hours</td>
</tr>
<tr>
<td>EDG Recovery Time</td>
<td>0 hours - 12 hours</td>
</tr>
<tr>
<td>Offsite AC Power Recovery Time</td>
<td>0 hours - 12 hours</td>
</tr>
<tr>
<td>Battery Power Lifetime</td>
<td>5 hours – 8 hours</td>
</tr>
<tr>
<td>Cycles Until an SRV fails Open</td>
<td>0 cycles - 30 cycles</td>
</tr>
<tr>
<td>Time Until HPCI Fails to Run</td>
<td>10 minutes - 8 hours</td>
</tr>
<tr>
<td>Time Until RCIC Fails to Run</td>
<td>10 minutes - hours</td>
</tr>
<tr>
<td>Firewater Injection Alignment Time</td>
<td>0 hours - 4 hours</td>
</tr>
<tr>
<td>Wetwell Failure Pressure</td>
<td>74 psi - 114 psi</td>
</tr>
<tr>
<td>Cladding Failure Temperature</td>
<td>1800F - 2600F</td>
</tr>
<tr>
<td>Reactor Power</td>
<td>100% - 120%</td>
</tr>
</tbody>
</table>

Table 1: BWR SBO Stochastic Parameters and Considered Ranges

These parameters were chosen partly in anticipation of the expected possible progressions of a BWR SBO transient. The divergent nature of the analysis is caused by the randomly sampled stochastic parameters. Because of the different initial conditions and input parameters, some randomly sampled scenarios will unfold one way, others will evolve differently, and yet others will end quickly before reaching a potential branching
point at all. At any point during the event sequence progression, if the EDGs come back online, grid power is restored, or if firewater alignment is successful, the plant is considered to have achieved a safe and stable state.

The expected progression of the transient is that when Loss of Offsite Power (LOOP) condition occurs, numerous things will immediately happen. In no particular order, the Reactor Protection System will SCRAM the reactor, inserting the control rods and halting the chain reaction, the Main Steam Isolation Valves (MSIVs) will close, isolating the turbine from the reactor, and the EDGs will engage and maintain plant AC power. Using the plant AC power, the Residual Heat Removal (RHR) system will remove core decay heat and keep the core at a stable and safe configuration. Meanwhile, it is expected that the grid owner will begin attempts to repair offsite AC power.

If the EDGs do not fail before offsite AC power is repaired, then the LOOP condition ends without ever transitioning into a Station Blackout (SBO) conditions and, barring other malfunctions, plant safety will be maintained. However, it is assumed that at some point, the EDGs will fail and the plant will enter SBO conditions. At this point, AC power is lost and the plant will switch over to HPCI and RCIC to maintain core cooling, as these are operable using only battery power and steam generated by the core. The Safety Relief Valves are also operable using battery power and will be used to maintain RPV pressure at a safe level. It is also expected that plant operators will immediately begin efforts to recover EDG function.

From SBO conditions, the scenario is expected to progress in one of two ways, depending on whether battery power is depleted before the PSP heat limits are reached or not. If battery power depletes before the heat limits are reached, then HPCI, RCIC, and
the ADS all stop functioning. The core contents will boil away, the fuel will uncover, and core damage will ensue. However, if AC power resumes after battery power fails, the ADS can be used to depressurize the RPV and low pressure systems can be used to replenish the water level of the core, maintaining plant safety.

If the heat limits of the PSP are reached before battery power depletes, the ADS will be activated and blowdown will occur. Because of the low pressure in the RPV, HPCI and RCIC become inoperable. To provide core cooling, it is expected that plant operators will begin to align firewater lines to inject water into the RPV. If they align the firewater sufficiently quickly, core cooling is reestablished. Otherwise, the core contents boil, the fuel uncovers, and core damage ensues. For the purposes of this test case, aligning the firewater was considered to be a success scenario, and was one of the end conditions of the simulation.

To model the BWR, the plant was divided up into many small nodes. The nodalization of the plant consisted of several dozen RELAP5-3D components, many of which were themselves divided into many smaller volumes to increase the spatial resolution of the model for particularly important or complex pieces of the plant. The nodalization is shown in Figure 6. Within the RPV itself, the reactor core, core bypass channel, steam dome, lower and upper plena, recirculation loop, jetpump mixer, downcomer, SRVs, ADS, and recirculation pumps were all modeled. Outside of the RPV, the drywell, wetwell, drywell downcomer, HPCI and RCIC lines, firewater injection line, ADS vent lines, and Main Steam line (including the MSIV) were all modeled.
To model the decay heat curve of the reactor, RELAP5-3D’s built-in Point Kinetics heat mode – which includes a highly realistic decay heat curve based on the American National Standards Institute (ANSI) decay heat curve - was used to generate a decay heat power vs. time curve, which was then built into a table and used to dictate the core power. This was done because of the varying start times of the simulation – sometimes the simulation starts hours after the original reactor SCRAM, and RELAP5-3D was originally incapable of properly reflecting this reality in the decay heat. Specifically, in the RELAP5-3D model, the reactor would not SCRAM until the onset of SBO conditions, rather than at the onset of LOOP conditions, when the reactor actually SCRAMs in a real life SBO accident scenario. Additionally, this alternate implementation of the decay heat allowed for the reactor power to more easily be sampled and adjusted, as the original Point Kinetics method required a more detailed change to the model, and the method of using a table to dictate the decay heat allowed for a simple factor of 1.0-1.2 to be entered into the model to dictate the reactor power.
Figure 5: BWR SBO Accident Scenario Model Nodalization (Mandelli et al, 2013)
4.3 **MASLWR Case Study**

In the second case study, the MASLWR test facility at OSU was modeled. Unlike the BWR design discussed previously, the MASLWR design (Figure 7) uses a two loop system to generate power. Highly pressurized water (roughly double the pressure of a BWR) is heated and driven by natural circulation convection currents past a helical coil steam generator, where it heats and boils less pressurized water in a secondary loop. The cooled water is driven by convection currents to the bottom of the RPV, where it circulates from the lower plenum back into the core. On the secondary side of the loop, water passes through the helical coil steam generator where it is boiled and heated into superheated steam. This steam then drives the turbine, generating power, and passes into a condenser where it is cooled back to liquid form and pumped back to the steam generator again.

![Figure 6: MASLWR Conceptual Design Layout (Modro et al., 2003; Reyes et al., 2007)](image-url)
MASLWR’s safety systems are very simple. An implicit safety aspect of MASLWR is the lack of pumps in the primary loop, and the primary loop coolant is driven solely through natural convection. Because of this, coolant flow through the core is maintained even during LOOP and SBO transients. As far as explicit safety systems relevant to Station Blackout accidents go, the RPV is encapsulated inside a secondary pressure vessel, the High Pressure Containment (HPC). This secondary barrier is filled with water, and can serve a similar function to the RPV in case of severe accidents – if a jet of molten corium melts through the RPV, it will then be impeded by a body of water and another steel wall. The HPC itself is then submerged in a large tank of water, providing yet more potential barriers against radionuclide release, and potential heatsink material in case of a severe accident. To protect against severe accidents occurring, MASLWR has a series of valves in the walls of the RPV. There are steam vent valves at the top of the RPV (labeled Vent valve (High ADS valves) in Figure 7) as well as lower depressurization valves on the sides of the RPV (labeled Depressurization Valve (Middle ADS valves) in Figure 7) and sump water makeup valves also on the sides of the RPV (labeled Sump makeup valve in Figure 7).

MASLWR is much simpler than older designs, taking a ‘less is more’ approach, as a simplified set of safety systems means that fewer components are required to keep the plant safe, resulting in a safer and more reliable reactor. The basic procedure to respond to nearly any transient is to blow down the reactor, then open the Sump makeup valve to provide makeup water to the core. If an incident occurs that could credibly threaten core safety, the ADS activates, opening the depressurization valves and equalizing the pressure between the RPV and HPC, at which point the sump makeup
valves open, allowing the contents of the HPC to be used as additional core cooling water. As the core boils the core contents, the steam rises out of the steam valve, and more water flows in through the sump makeup valves, keeping the core covered in water for very long periods of time due to the lower power density of MASLWR and the immense amount of water in the HPC. The large quantity of water in the HPC can provide emergency cooling for long periods of time. The intended outcome of this procedure is that the heat capacity of the High Pressure Containment is great enough to mitigate the decay heat of the core for a long enough period of time to put the core in a safe configuration.

As with the BWR case study, the Plant Physics Calculations were performed using RELAP5-3D and RAVEN. For the TH simulations, the primary components modeled were the Core, Hot Leg Chimney, Cold Leg Downcomer, Helical Coil Steam Generator, Reactor Pressure Vessel, High Pressure Containment, and Automatic Depressurization System (ADS). The secondary loop was not modeled in detail, as it ceases to function during SBO conditions.

Similar to the simplified physical design of MASLWR, the control logic for the plant is significantly simpler than it was for the older BWR design. In the model, the only control logic needed were trips to open the vent and sump valves and trips to turn the pressurizer on and off to maintain an appropriate operating pressure within the core during normal operations. During an SBO transient, however, there is no AC power with which to operate the pressurizer, so it was turned off permanently. The vent valves were set to open when the RPV pressure exceeds 9.0E+6 MPa (normal operating pressure is
After the vent valves open, the RPV begins to blowdown, venting steam into the HPC. Once the two equalize in pressure, the sump valves open to let makeup water from the HPC flow into the RPV.

To model the various ways transients can unfold, numerous stochastic parameters were chosen, as seen in Tables 2a and 2b. Given the MASLWR’s simplified design, there were fewer stochastic parameters to model. As with the BWR SBO analysis, the parameters were sampled using a Monte-Carlo method approach and appropriate parameters within the model altered to appropriately represent the sampled value of the stochastic parameters.

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</tr>
<tr>
<td>Offsite AC Power Recovery Time</td>
<td>0 hours - 12 hours</td>
</tr>
<tr>
<td>HPC Failure Pressure</td>
<td>7.0 MPa – 8.0 MPa</td>
</tr>
<tr>
<td>Steam Vent Valve Failure Start Time</td>
<td>0 hours - 8 hours</td>
</tr>
<tr>
<td>Steam Vent Valve Failure End Time</td>
<td>0 hours - 8 hours</td>
</tr>
<tr>
<td>Water Sump Valve Failure Start Time</td>
<td>0 hours - 8 hours</td>
</tr>
<tr>
<td>Water Sump Valve Failure End Time</td>
<td>0 hours - 8 hours</td>
</tr>
<tr>
<td>Reactor Power</td>
<td>100% - 120%</td>
</tr>
</tbody>
</table>

Table 2a: MASLWR SBO Stochastic Parameters, Nominal Power Test Case

The expected progression of a MASLWR SBO transient is that when Loss of Offsite Power (LOOP) conditions occur, multiple systems will respond to protect the integrity of plant safety. The Reactor Protection System will SCRAM the reactor to halt the nuclear chain reaction, Emergency Diesel Generators (EDGs) will engage to maintain plant AC power, and core decay heat will be removed through the secondary loop using
the Residual Heat Removal system. Additionally, it is expected that the grid owner will quickly (ie immediately) begin efforts to restore offsite AC power.

Should EDG functionality continue until the grid is repaired and offsite AC power restored, then the plant is considered to have reached a safe and stable configuration without ever evolving into a SBO transient. However, for the purposes of analysis, EDG functionality fails at some point and the scenario evolves into a SBO transient. When the Station Blackout begins, core decay heat will stop being removed by the RHRs and the core will begin to heat up and pressurize. When the core pressure reaches the vent valve trip point, barring valve failure, the vent valves will open, depressurizing the Reactor Pressure Vessel (RPV) and venting steam into the High Pressure Containment (HPC). Once the HPC and RPV have equalized in pressure, the water makeup sump valves will, barring valve failure, open and allow makeup water from the HPC to flow into the core. From this point, whether plant safety is maintained (ie whether core damaged occurs or not) is simply a matter of whether AC power can be restored before enough of the core contents boil away to uncover the fuel and lead to core damage. Alternatively, it is possible that, given a large enough body of water in the HPC to act as a heatsink, eventually the core decay heat would be low enough that ambient heat loss to the pool of water the HPC sits in would be enough to keep the core cooled.

As well as the already discussed test case analysis of MASLWR using RAVEN and RELAP5-3D, a second test was performed with a much higher core power. The reasons will be discussed more thoroughly in the section presenting the MASLWR test case results, but, in short, it proved difficult to establish any failure space for MASLWR for a Station Blackout scenario. It was decided that this was, well, a boring result in that
the capabilities of RAVEN could not be tested, and that further investigation into the repercussions of a dramatic power uprate of MASLWR would be more interesting. This indeed enabled a way to further display RAVEN’s capabilities, unlike the reference MASLWR test. Except for the power change, the two tests were otherwise identical.

<table>
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<td>HPC Failure Pressure</td>
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</tr>
<tr>
<td>Steam Vent Valve Failure Start Time</td>
<td>0 hours - 8 hours</td>
</tr>
<tr>
<td>Steam Vent Valve Failure End Time</td>
<td>0 hours - 8 hours</td>
</tr>
<tr>
<td>Water Sump Valve Failure Start Time</td>
<td>0 hours - 8 hours</td>
</tr>
<tr>
<td>Water Sump Valve Failure End Time</td>
<td>0 hours - 8 hours</td>
</tr>
<tr>
<td>Reactor Power</td>
<td>300% - 360%</td>
</tr>
</tbody>
</table>

Table 2b: MASLWR SBO Stochastic Parameters, High Power Test Case

4.4 Stochastic Parameter Analysis

The probabilistic analysis of the stochastic parameters was done using the RAVEN code being developed by INL. RAVEN was coupled to RELAP5-3D in a manner where RAVEN ‘decides’ values for the stochastic parameters, plugs them into the RELAP5-3D model, runs the model, and processes the RELAP5-3D output file into a more user friendly format. It does this for multiple RELAP5-3D runs and their output files. This stochastic parameter analysis is performed by first determining probability distributions for the stochastic parameters using PRA techniques and justifiable engineering judgement decisions to determine both the shape of the distribution and the values encompassed. The link between these stochastic parameters and the parameters coded into the RELAP5-3D input deck is considered, the nature of how these stochastic parameters are reflected in the input deck is determined, and the RAVEN input file is
coded such that RAVEN will adjust these coded parameters as appropriate to reflect the randomly determined values for the stochastic parameters.

RAVEN takes the base RELAP5-3D and generates numerous copies of it, each of which is altered to reflect a generated set of stochastic parameter values individual to that copy of the input. Using the high performance computing clusters at INL, the RELAP5-3D runs are spread across hundreds of server nodes to be processed quickly. The immense processing power of modern high performance computing clusters is what makes this new stochastic parameter analysis possible – it can take hundreds of thousands of simulations to establish a highly confident view of the interactions between various parameters in the model. In years past, the processing power needed for this kind of analysis was prohibitively expensive, but this has ceased to be an obstacle. After the simulations have all been completed, RAVEN processes each RELAP5-3D output file into a more user friendly .csv file that can then be examined with more advanced data analysis methods than the original RELAP5-3D files readily allow.

To benchmark the model and ensure reasonable realism, numerous tests of the BWR model were run before the full 20,000 simulation data bloc was executed. In these smaller test cases only two or three parameters were adjusted – generally reactor power and one or two other parameters – to give data sets that were easy to generate and visualize, requiring no advanced data analysis techniques to process them. In the MASLWR test case, a 20,000 simulation data bloc was not feasible due to manpower concerns for data analysis – where the BWR test case had nearly a dozen contributors, the
MASLWR test case was largely a solo effort. In the MASLWR test case, the exploration of RAVEN’s capabilities was limited to the smaller, more manageable tests.

Reactor power was commonly used as power uprates are greatly sought after, but have the potential to significantly erode safety margins. By evaluating the safety margin impact of power uprates, it can be judged whether an uprate poses a significant risk to plant safety. License extensions from 40 to 60 and even 80 years are also greatly sought after and can erode the safety of nuclear power, but it is much more difficult to simply adjust a dial and flip a switch to represent the impacts of plant age on the safety margins. Rather, by sampling the stochastic parameters involved, it is possible to represent, at least in part, the consequences of an aging plant on safety margins.

5 Results

5.1 BWR Test Case Results

The first test case of the model was to examine the effects of reactor power on the time at which the wetwell would reach the heat curve limits and trigger the ADS. Using values of 100%, 105%, 110%, 115%, and 120% of rated power, simulations were run for each reactor power value. This was done to find the time between the loss of diesel power and the ADS activation for different values of reactor power. This is seen in Figure 7. When the ADS is activated, all of the Safety Relief Valves (SRVs) open, allowing high pressure steam from the RPV to vent into the wetwell. The activation of the ADS can be seen visually – when the temperature of the wetwell reaches the temperature limit prescribed in the technical specifications, the temperature shoots up because the ADS is
introducing hot steam into the wetwell in large amounts, rapidly increasing the temperature.

As expected, increasing reactor power decreases the time between SBO conditions and ADS activation. With each increase in reactor power, the time from the loss of diesels to the activation of the ADS decreases. It can be seen that the wetwell temperature increases in small increments, corresponding to when the SRVs open and steam flows to the wetwell, then remain flat for extended periods of time, corresponding to when HPCI and/or RCIC are injecting large amounts of cold water into the core, suppressing the pressure (and thus the need to vent steam to lower the pressure) for large periods of time. It was theorized, but never tested, that if the power were lowered to 95% or 90%, HPCI/RCIC would have engaged one more time before the blowdown, with the wetwell temperature just below the blowdown set point. This would cause another long period of time with no increases in wetwell temperature and significantly increase the amount of time to ADS activation. For the 100% data, the line ends prematurely because the simulation proved to be unstable during depressurization, but this was deemed irrelevant, as the focus of the test was on the timing of the depressurization, not the events after depressurization.
The second test case was to examine the direct impact of diesel generator failure time and reactor power on the time to reach ADS activation. Running simulations for power equal to 100%, 110%, and 120%, time to diesel generator failure was set to be 0.5 hours, 1 hour, 2 hours, and 4 hours. It was predicted that a longer time to diesel failure would result in more time between SBO conditions and ADS activation, and a higher reactor power would have the opposite effect, reducing the time from SBO condition start to ADS activation.

Figure 7 – Pressure Suppression Pool Temperature vs. Time vs. Reactor Power

(Mandelli et al, 2013)
As predicted, a power increase means reaching the Heat Capacity Temperature Limit (HCTL) and activating the ADS faster, while increasing the time to DG failure makes this happen more slowly.

Though they were not included in the stochastic parameters for the full 20,000 simulation analysis, the initial conditions of the wetwell were deemed relevant to the SBO transient and were investigated. Specifically, the repercussions of variations in the initial wetwell temperature and level were examined. For the main set of simulations, the

Figure 8 – EDG Failure Time vs. Time to ADS Activation

vs. Reactor Power (Mandelli et al, 2013)
wetwell initiation temperature and level were set as 90 degrees Fahrenheit and 15’ of water, respectively. In the previous test cases, it became clear that increasing the core power reduces the time between blackout conditions and the triggering of the ADS, giving plant operators less time to align the firewater or recover the EDGs. The next logical step was to examine if there were any simple ways to compensate for a power uprate to allow utilities to cheaply and safely justify a power uprate, to see if the increased power can be offset by increasing the heat capacity of the wetwell – in this case by keeping the wetwell cold or by adding water to it. Either would, in theory, increase the heat capacity of the wetwell and forestall the activation of the ADS.

Figure 9 – Time to ADS Activation vs. Pressure Suppression Pool Initial Temperature vs. Reactor Power (Mandelli et al, 2013)
As can be seen in Figure 9, an increase in PSP initial temperature is associated with a decrease in time to ADS activation. Data points were tested with the PSP initial temperature at 60, 70, 80, and 90 degrees Fahrenheit. Additionally, an increase in reactor power is associated with a decrease in time to ADS activation. This was to be expected, as the PSP can only absorb so much thermal energy from the core (in the form of vented steam from the RPV) before the PSP temperature climbs too high and the ADS has to be activated. A higher initial PSP temperature decreases the amount of thermal energy the PSP can absorb before the ADS must be activated, and increasing the reactor power increases the rate at which thermal energy is passed to the PSP.

Interestingly, the shape of the data for the three different reactor power levels is not the same, with 120% power diverging from the other two power levels when the PSP has an initial temperature of 70 degrees Fahrenheit. It is believed that this is due to the nature of HPCI and RCIC preventing the need to vent steam to the PSP for large periods of time, as seen in Figure 7. For the 120% power and 70 degrees Fahrenheit simulation, it is believed that the ADS activated just before HPCI and RCIC would have engaged, which would have forestalled the ADS activation significantly. Additionally, it is believed that if the temperature increments were smaller, the flat lines seen in Figure 7 would be seen here, as well.
Figure 10 – Time to ADS Activation vs. Pressure Suppression Pool Initial Water Level

vs. Reactor Power (Mandelli et al, 2013)

For the next test, the time to ADS activation was benchmarked against the PSP initial water level inside the wetwell. The results can be seen in Figure 10. As expected, increasing the wetwell level or decreasing the initial temperature both increase the time between SBO conditions and ADS activations, while increasing reactor power decreases the time from the onset of SBO conditions until the ADS activates. As with the previous test, it can be viewed as a matter of how much heat the PSP can absorb before the ADS activates. More water inside the wetwell means that more heat can be sunk into the PSP before the ADS activates.
More simple test cases were performed to study the effects of a power uprate on the timings of various events in an accident scenario. Specifically, the timing of the ADS activation and the time to core damage, as well as the time from ADS activation to core damage were examined. To generate the data, simulations were executed for several values for the DG lifetime (ie time between LOOP and SBO conditions) and reactor power level. A marker inside RELAP5-3D was used to note the time at which the ADS was activated. Additionally, the simulation was set up such that core damage was inevitable. This was done by permanently disabling Firewater Alignment and preventing

Figure 11 – Time to ADS Activation vs. Emergency Diesel Generator Failure

Time vs. Reactor Power (Mandelli et al, 2013)
AC Power Recovery. Figure 11 shows that a power uprate noticeably decreases the time to activate the ADS, while increasing the availability of AC power at the beginning of a transient dramatically increases the time to ADS activation.

Figure 12 – Time to Core Damage vs. Emergency Diesel Failure Time vs. Reactor Power (Mandelli et al, 2013)

Figure 12 shows similar conclusions – a power uprate speeds up the transient, causing core damage to occur earlier, while a longer DG lifetime greatly delays the occurrence of core damage. Increasing the reliability of DGs or finding an alternate way to supply AC power to the plant for a period of time would easily offset the risk of a power uprate in these areas.
Figure 13 – Time from ADS Activation to Core Damage vs. Emergency Diesel Failure Time vs. Reactor Power (Mandelli et al, 2013)

Figure 13 continues to reinforce these ideas, despite the data being messy and difficult to fit to a trend line. Due to quirks in the model that will be discussed shortly (Figure 14, Figure 15, and accompanying discussion), the data for both the time to ADS activation and the time to Core Damage tended to fluctuate, and the difference between the timing of these two events inherited the fluctuations of both sets of data, causing the results to become somewhat erratic. Fortunately, the oddities in the data were not enough to overshadow the trends in the relationship between core power, DG lifetime, and time.
from ADS activation to core damage. In general, the higher power simulations had less
time between the activation of the ADS and the occurrence of core damage.

The sometimes erratic behavior in the model is believed to be due to the cyclical
nature of the reactor simulation. To examine this behavior, a series of tests were run with
HPCI and RCIC set to fail at various times, with all other stochastic parameters held
constant, and examined the time to ADS activation. DG lifetime was held constant at one
hour, and the failure of HPCI and RCIC was measured from the start of SBO conditions.
The results are presented in Figures 14, 15, and 16 are discussed below.

Figure 14 – Time to ADS Activation vs. Time to HPCI/RCIC Failure vs. Reactor Power
(Mandelli et al, 2013)
As can be seen in Figure 14, there were periods of time where the precise timing of HPCI/RCIC does not correlate to a change in the time to ADS activation. This is observed in the flat parts of the graph as there are times where the failure time of HPCI and RCIC can vary by up to 3000 seconds – nearly a full hour – without any impact on the sequence of events. This is due to the nature of HPCI and RCIC. HPCI turns on only when a leak is too great for RCIC to cope with, and RCIC turns on only when the RPV level becomes low, refilling the core contents by injecting cold water into the RPV. When RCIC refills the RPV, there is a period of time where the SRVs slowly vent steam to the wetwell, thus lowering the RPV water level. The SRVs cycle several times before the RPV water level becomes low enough again to trigger RCIC for another injection of cold water. From a plant safety perspective HPCI/RCIC failure on demand and HPCI/RCIC failure prior to demand are functionally the same.

Interestingly, because of the cyclical nature of HPCI, RCIC, and the SRVs, there were times where a higher core power would perform slightly better than a lower core power. This is caused by the higher core power speeding up the progression of the transient. The higher power generates steam faster, causing the SRVs to cycle faster, causing RCIC to cycle faster, which would cause RCIC to trigger and refill the core with cold water just before RCIC would fail, where in the lower power scenario RCIC would fail just before it would be triggered. However, these windows are small and generally insignificant, with core safety continuing to favor, as expected, the lower power scenario.
In Figure 15, the cyclical nature of the SRVs and of RCIC is shown clearly. When the RPV pressure oscillates from 900 psi to 1100 psi repeatedly, it is due to the SRVs venting steam to relieve pressure in the RPV. It can be seen that when RCIC engages and the RCIC flowrate increases, the RPV pressure is suppressed due to the large injection of cold water into the core contents. This negates the need for the use of the SRVs for a relatively large period of time, explaining the windows of time seen in Figure 14 where the higher power scenario triggered ADS later than the lower power scenario. Additionally, these oscillations in the RPV pressure and water level significantly alter the
properties of the contents of the RPV when the ADS is activated, contributing to the erratic data seen in Figure 13.

In addition to these tests, the specific repercussions of the clad failure temperature were examined. This was accomplished by holding all stochastic parameters constant, with the exception of the clad failure temperature. The clad was, for numerous simulations, set to fail at 1800, 1900, etc. up to 2600 degrees Fahrenheit. Additionally, the possibilities of firewater alignment or AC power recovery were ignored. Each simulation was run, without possibility of success, to see exactly how long it would take for the cladding to fail and how much the clad failure temperature would impact this timing.
It can be seen in Figure 16 that there is a strong linear correlation between the time to core damage and the cladding failure temperature. This is because the time between the core uncovering and the clad failing is simply a matter of the clad failure temperature, the heat capacity of the cladding and the rate at which the core produces heat. Increasing the clad failure temperature increases the amount of heat that needs to be added to the cladding before it will fail, and the heat production rate is essentially the same from 2.06E4 seconds to 2.2E4 seconds. For a sufficiently high clad failure temperature, the exponential drop in decay heat would be reflected in the time to reach

Figure 16 – Time to Core Damage vs. Fuel Cladding Failure Temperature (Mandelli et al, 2013)
core damage and the relationship between time to core damage and clad failure temperature would look more exponential than linear.

In addition to these relatively basic test cases, more advanced, but still preliminary test cases were conducted. In these test cases, two stochastic parameters were chosen to be randomly sampled over the course of several hundred simulations. Additionally, within each test case, the simulations were conducted with varying reactor power at 100%, 110%, and 120%. These test cases were run to provide proof that we could use Monte Carlo sampling to establish realistic failure spaces for the chosen stochastic parameters. In one test case, the firewater alignment time and battery life were chosen for the stochastic parameters to be sampled, in the other test case the AC power recovery time and EDG failure time were sampled.

The results of the first of the two advanced test cases, where the firewater alignment time and battery life were sampled, are presented in Figures 17a, 17b, 17c, and 17d. All three values of reactor power – 100%, 110%, and 120% - are shown together. For this test case, successfully aligning and injecting firewater to the core at any point before core damage was considered a success, while the occurrence of core damage, inevitable without core cooling, was considered a failure. For this test case, the recovery of AC power was disabled, forcing the simulation to end by either core damage or successful firewater alignment.

From the outset, it was expected that a longer battery life and shorter firewater alignment time would lead to success. For a given reactor power, the amount of time
from station blackout conditions to ADS activation is a fixed amount of time, as is the amount of time it takes for the ADS to depressurize the RPV to the point that firewater can be injected to the core. The ADS requires battery power to operate, and the simulation assumed no human intervention, thus loss of battery power leads directly to ADS failure. Additionally, once the ADS depressurizes the core, there is only so much time until the water level drops to the point that the core uncovers and core damage occurs. If firewater can be aligned before battery power depletes or the core uncovers and melts, then the simulation will be a success.
In Figure 17a, it can be seen that if the Battery Power does not last long enough, core damage is guaranteed. This is because the battery power fails before blowdown can occur, it is impossible to align and engage the firewater lines and provide cooling to the core. Additionally, if the Firewater Alignment takes too long, the ADS activates, the core contents boil away, and the fuel uncovers and melts before Firewater Alignment can be achieved. For Firewater Alignment Times of less than 1.5 hours and a Battery Power Lifetime between four and seven hours, plant safety is dependent on the specific values
of the scenario, as a very quick Firewater Alignment Time allows for a shorter Battery Power Lifetime. For these values, any increase in Battery Power Lifetime allows for a longer Firewater Alignment Time without compromising plant safety.

In Figure 17b, it can be seen that a shorter Battery Power Lifetime is acceptable for core safety, as long as the Firewater Alignment Time is extremely quick. For 110% Reactor Power, core damage is not guaranteed unless the Battery Power Lifetime is less than $1.4 \times 10^4$ seconds, roughly 20 minutes shorter than the minimum Battery Power
Lifetime for 100% Reactor Power. However, the maximum allowable Firewater Alignment Time is shorter for 110% Reactor Power than for 100% Reactor Power, because the core will uncover and melt more rapidly with increasing power.

Figure 17c – Failure Space for Firewater Alignment Time vs. Battery Power

Lifetime at 120% Reactor Power (Mandelli et al, 2013)

The trends from Figure 17a and Figure 17b continue. With 120% Reactor Power the minimum battery life required for plant safety is even shorter than the minimum battery life required for plant safety for 110% Reactor Power. Similarly, the maximum Firewater Alignment Time that can achieve plant safety for 120% Reactor Power is even
shorter than the maximum Firewater Alignment Time that can achieve plant safety for
110% Reactor Power.

The lack of human intervention in the simulations leads to interesting results. In
Figure 17a, Figure 17b, and Figure 17c, the downward slope at the left end of the limit
surface represents scenarios where battery power lasts long enough for the core to
depressurize, but not so long that the core simply uncovers and melts – in these low
battery life situations, the battery power runs out, the ADS valves shut without battery
power to keep them open, and the core repressurizes before firewater injection can occur.
Eventually, even under the higher pressure, the core uncovers and melts. Additionally,
for certain slices of firewater alignment time and battery life, a higher reactor power will
actually help maintain plant safety where it would have failed at a lower power. This is
because the higher power leads to an earlier ADS activation – if the battery power is just
barely too short for 100% reactor power, it is possible that the earlier ADS activation in a
120% simulation will lead to successful firewater alignment just before the battery power
would fail.

For the second advanced test case, DG failure time and AC power recovery time
were used as the stochastic parameters to be sampled. It was expected that a very long
DG lifetime would assure plant safety, regardless of the AC power recovery time. If the
DGs operate for a very long time, the amount of decay heat produced by the core at the
time of DG failure will be much less than if the DGs had failed immediately.
Mathematically speaking, one hour after the reactor SCRAMs, the core produces roughly
one fifth as much decay heat as it did at the moment of reactor SCRAM. The reduced
decay heat exhausts the plant safety systems’ finite ability to remove heat from the core much more slowly.

Similarly, it was expected that very early recovery of AC power would lead to plant safety regardless of DG lifetime. When AC power comes back online, SBO conditions end and the numerous AC powered safety systems of the BWR Mk.1 come online again and ensure core safety. If this happens quickly enough, it will occur before battery power fails or the wetwell’s heat limits can be exhausted.

Figure 18a – Failure Space for AC Power Recover Time vs. EDG Failure Time for 100% Reactor Power (Mandelli et al, 2013)
In Figure 18a, it can be seen that if the power is recovered before the EDGs fail, the plant will always be safe. Even if the EDGs fail before power is recovered, if the EDGs last at least six hours, the plant will always be safe, due to the greatly reduced decay heat that must be mitigated. However, it is unknown why the line delineating the failure space vs. the success space suddenly changes slope at the six hour mark for EDG failure time. While it is understood that this is related to the reduced decay heat that is associated with a long EDG failure time, the exact mechanisms within the simulation that lead to this marked change are unknown. Initial expectations were that it would simply continue to trend slowly upward with respect to the maximum safe AC power recovery time and instead the maximum safe AC power recovery time increases indefinitely once the EDGs last more than six hours.

Looking to the AC power recovery time, if the AC power is recovered before seven hours, the plant will be safe. Battery power was assumed to last indefinitely, and these results indicate that if HPCI, RCIC, and the ADS all function, core integrity will be maintained for roughly seven hours before failing. Eventually, the ADS will activate as the Pressure Suppression Pool temperature rises, and the resulting loss of pressure in the RPV will prevent HPCI and RCIC from operating, and the loss of these systems’ functionality will lead to core damage as the contents of the core boil off and cannot be replaced with more water.

Interestingly, when the AC Power Recovery time is roughly eight hours and twenty minutes, and the EDG Failure Time is approximately two hours and fifteen minutes, core integrity is maintained, when the trends in the surrounding data would
indicate that core integrity would be lost if the AC Power Recovery time is longer than seven hours and forty five minutes for an EDG Failure Time of two hours and fifteen minutes.

Figure 18b – Failure Space for AC Power Recover Time vs. EDG Failure Time for 110% Reactor Power (Mandelli et al, 2013)

For 110% Reactor Power, the maximum AC recovery time for which the plant will remain safe, regardless of EDG Failure Time, decreases to six hours. This is because of the increased decay heat production in the core that accompanies an increase in power production. The increased decay heat more rapidly exhausts the plant safety system’s
ability to remove heat from the core. Because of the increased initial decay heat, the minimum EDG Failure Time for which the plant will remain safe, regardless of AC Power Recovery time, increases to seven hours.

Figure 18c – Failure Space for AC Power Recover Time vs. EDG Failure Time for 120% Reactor Power (Mandelli et al, 2013)

Interestingly, the maximum AC Power Recovery Time for which the plant will remain safe, regardless of EDG Failure Time, remains the same for 120% Reactor Power as it was for 110% Reactor Power, holding steady at roughly six hours between the two sets of data. Similarly, the minimum EDG Failure Time for which the plant will remain
safe, regardless of AC Power Recovery time, also remains the same between 110% Reactor Power and 120% Reactor Power. The primary difference is in the slope of the line delineating the Failure Space as the EDG Failure Time increases from zero to seven hours. For 110% Reactor Power, the boundary slopes upward strongly as the EDG Failure Time increases from zero to 1.5 hours, then remains fairly level until seven hours, at which point it is fairly vertical. For 120% Reactor Power, the boundary simply slopes upward steadily from zero to seven hours.

After the limited scope test scenarios, a full workup of 20,000 simulations was run sampling most of the initially theorized parameters. Wetwell pressure capacity and Heroic Actions to increase the CST capacity were proven to be inconsequential, as the wetwell pressure never rose enough to threaten the integrity of the wetwell and the CST is large enough that its capacity was never an issue. Battery Power lifetime and Heroic Actions to extend Battery Power lifetime were merged into one ‘Total Battery Lifetime’ parameter due to some limitations in RAVEN’s compatibility with RELAP5-3D. Similarly, Recovery time of EDGs and Offsite AC Power Recovery Time were merged into AC Power recovery time.

To process the immense amount of data generated, the maximum core temperature was modeled as a function in an n-dimensional space, where n was the number of parameters used. As part of doing this, all of the data was normalized. To normalize the data, the mean and standard deviation of each parameter was calculated. For each particular dimension of each data point, we subtracted the mean value of that dimension across all of the data from the value of that dimension for that particular data
point, then divided the result by the standard deviation of that dimension across all of the data. For example, to normalize the DG failure time dimension, the mean and standard deviation DG failure time were calculated using all of the sampled DG failure times. Then, for any particular data point, the DG failure time was normalized by subtracting the mean DG failure time from that particular data point’s DG failure time, then dividing this difference by the standard deviation of the DG failure time data.

In order to simplify the analysis, the data was normalized and modeled according to each stochastic parameter as a function of core temperature and (n) parameters used. This produces weighted data points that follow clear trends. From the multidimensional surface that was procured, local minima and maxima and gradient characteristics could be analyzed and used to discover trends in the data. One local minimum and three local maxima were discovered, corresponding to three different paths to failure. The paths from the minima to these maxima can be seen below in Figure 19, a visual summary of the topology of the data.
Due to the highly abstract nature of a many-dimensional set of data, it is difficult to visually represent the data with all of its details, but a simplified view is more feasible. Figure 19 represents trends in the three paths to failure that were discovered. In Figure 19, principal component analysis was used to compress the many-dimensional data into two dependent variables and a mapping function, allowing it to be represented with only three dimensions.

Breaking the data into a more complex, but informative form allows the specific conditions that lead to core failure to be assessed. This is done using inverse coordinate plots that display each sampled parameter as a dependent variable of the maximum clad temperature reached. In Figure 20, on each of the ten inverse coordinate plots, the normalized stochastic parameter specific to that plot is plotted on the Y-Axis, while the
maximum cladding temperature reached is plotted on the X-axis from 1008.8° F to 2600° F. It can be seen in Figure 20 that the green and blue paths to failure are nearly identical with regards to seven of the ten variables, only diverging significantly for the amount of time it takes for the operators to trigger the ADS once the pressure suppression pool limits have been reached, the time at which a Safety Relief Valve becomes stuck open, and the failure time of the Emergency Diesel Generators.
From Figure 20 it can readily be seen that a long AC Power Recovery time is necessary for core damage to occur – if the AC Power is recovered quickly, the AC
powered safety systems will engage and bring the core to a stable state before it can be endangered. Interestingly, it can be seen in the green path to failure that a long Emergency Diesel Failure Time was not sufficient to guarantee core safety – if an SRV inadvertently becomes stuck open and HPCI/RCIC fail to run, core damage would occur. HPCI and RCIC failing to run was also a major factor in the blue path to failure – with an early EDG Failure time and HPCI and RCIC quickly failing to run, the core was damaged. The red path to failure further reinforces the necessity of HPCI and RCIC, as Battery Power failing early shuts down HPCI, RCIC, and the ADS. For a sufficiently long AC Power Recovery Time and a sufficiently short Battery Power Lifetime, little else matters with regards to the safety of the plant – the core will be damaged.

5.2 MASLWR Nominal Power Test Case Results

Prior to using RAVEN, a series of tests were run using RELAP5-3D alone using a grid spacing for several stochastic parameters. These were done to ensure that the results from the model were reasonable and would allow RAVEN to establish a reasonable failure space for a MASLWR SBO transient. Additionally, these preliminary tests were done to establish a rough idea of the failure space of MASLWR for an SBO transient to make sure that, for the later tests, RAVEN is “aimed” at the right stochastic parameter values to identify any possible failure space in the analysis.

The first test done was to examine how long it took for the steam vent valves to trigger for various values of Emergency Diesel Generator (EDG) lifetime and several values of Reactor Power. The results can be seen in Figure 21.
As was expected, the time from SBO conditions to steam valve activation (and thus blowdown) increased as EDG lifetime increased and decreased as Reactor Power increased. The times ranged from about four minutes at the lowest, with 120% Reactor Power and instant EDG failure, to about ten minutes at the highest, with 100% Reactor Power and four hours until the EDGs failed. Interestingly, increasing the EDG lifetime from half an hour to an hour did not seem to significantly impact the time from onset of SBO conditions to blowdown, except for 120% Reactor Power.

After examining the consequences of EDG lifetime and Reactor Power on blowdown timing, the effects of blowdown on RPV and HPC pressure was examined. Additionally, the test was run to examine potential concerns regarding the HPC
maximum pressure. The test was run with the EDGs failing immediately on demand and at 120% Reactor Power to maximize the potential stress on the HPC. The results are presented below in Figure 22, a short term, ten minute plot of the HPC and RPV Pressure vs. Time. The results are also plotted, on a longer term basis, in Figure 23 an eleven hour plot of the HPC and RPV Pressure vs. Time.

Figure 22 – Ten Minute Plot of the HPC and RPV Pressure vs. Time

Figure 22 shows a more detailed view of the start of the HPC and RPV Pressure vs. Time data. Blowdown was triggered roughly four minutes into the onset of SBO conditions, and the RPV is depressurized by five minutes into the onset of SBO conditions. Blowdown occurs extremely quickly, with the RPV and HPC equalizing in pressure less than a minute after blowdown begins. From there, the pressure drops off, as can also be seen in Figure 22. The HPC pressure rises and the RPV pressure drops as the steam vent valves blow down the RPV into the HPC, venting hot, pressurized steam into
the HPC. The cold water in the HPC condenses the steam, reducing the pressure in both the HPC and the RPV over time.

![Figure 23 – Eleven Hour Plot of the HPC and RPV Pressure vs. Time](image)

As can be seen in Figure 23, the RPV and HPC pressures equalize extremely quickly after the steam vent valves open and remain in equilibrium the rest of the simulation. Considering that there are a set of open valves connecting the two, this is what was expected. Additionally, it can be seen that the pressure steadily drops for both the HPC and the RPV over the course of the simulation. The HPC acts as a pressure suppression pool during blowdown, condensing a great deal of steam and lowering the pressure in both vessels over time. It is expected that eventually, without heat loss to the ambient environment, the water in the HPC would reach saturation temperature and eventually repressurize as the RPV and HPC contents boil.

After testing the blowdown characteristics of the MASLWR model, a series of tests examining the repercussions of a blowdown on the core were executed. The
repercussions of a failure to blowdown, or a partial blowdown, were examined as well. The first test performed was to examine when, if at all, core damage will occur when the reactor successfully blows down and the water makeup sump valves are opened. To vary the potential timing of core damage Reactor Power and EDG Lifetime were varied. It was found, as might be expected, that when the plant’s safety measures respond successfully, plant safety is maintained. Even the harshest test, with the EDGs failing on demand and 120% Reactor Power, failed to damage the fuel or even strain the safety margins of the MASLWR design within twelve hours of the onset of SBO conditions.

After testing the margins of the fully functioning safety systems, more tests were performed to examine the effects of a partially disabled set of safety systems. In one case, both the water makeup sump valves and the steam vent valves were disabled. Additionally, both Reactor Power and EDG Lifetime were varied. The initial set of tests crashed, unfortunately. Due to an oversight in the model, when the steam vent valves failed shut, the RPV pressurized to the point that the model crashed, at roughly ten times the operating pressure of the plant. To rectify this error, secondary vent valves were added, safety relief valves (SRVs). These SRVs will only release steam at pressures significantly greater than the set point for the steam vent valves and will not stay open if the pressure drops below their setpoint. Once these SRVs were added and the simulations successfully run, it was found that even if the vent valves failed shut and the core was significantly over-pressurized, core damage did not occur for any variations of the test.
A test was run with only the water makeup sump valves broken. The idea was that without functional water makeup sump valves, the plant would be unable to replenish the core coolant. If the core coolant cannot be replenished, the core will eventually be uncovered and fuel will melt. However, even at 120% power and the EDGs failing on initial demand, core damage did not occur within twelve hours of the onset of SBO conditions. Core coolant levels had dropped, and it was obvious that core damage would occur eventually, but not within the anticipated repair time for the EDGs or offsite AC power. To test this idea further, the AC Power Recovery time was extended from twelve to 24 hours. In spite of the increased AC Power Recovery time, though, core damage still did not occur. After 24 hours the core was partially uncovered and the upper parts of the fuel rods had begun to heat up, but core damage still had not occurred. Due to MASLWR’s immense resilience to SBO transients, it was decided that RAVEN’s capabilities would not be able to be fully explored, as it appeared that there would be no failure space at all. To remedy this, a second set of tests were run, with the Reactor Power ranging from 300% to 360% Reactor Power.

5.3 MASLWR High Power Test Case

For the High Power Test Case, a series of two-variable tests were run with the reactor power set to 300% and to 360% to test the MASLWR design’s ability to remain safe under extreme conditions in a MASLWR-SBO transient. These entirely unrealistic scenarios were chosen only to determine the functionality of RAVEN (the goal of this study) and not to suggest any potential operational capabilities of the MASLWR design. The two variable tests were as follows in Table 3. In each test, two stochastic parameters
were sampled repeatedly to establish the failure space for variations of those two parameters while all others were held constant.

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Sampled Parameter 1</th>
<th>Sampled Parameter 2</th>
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<tbody>
<tr>
<td>1</td>
<td>Offsite Power Recovery</td>
<td>HPC Failure Pressure</td>
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<tr>
<td>2</td>
<td>Offsite Power Recovery</td>
<td>EDG Failure Time</td>
</tr>
<tr>
<td>3</td>
<td>Sump Valve Recovery Time</td>
<td>EDG Failure Time</td>
</tr>
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<td>4</td>
<td>Sump Valve Failure Time</td>
<td>EDG Failure Time</td>
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<tr>
<td>5</td>
<td>Vent Valve Recovery Time</td>
<td>EDG Failure Time</td>
</tr>
<tr>
<td>6</td>
<td>Vent Valve Failure Time</td>
<td>EDG Failure Time</td>
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<td>7</td>
<td>Sump Valve Restoration Time</td>
<td>Sump Valve Failure Time</td>
</tr>
<tr>
<td>8</td>
<td>Vent Valve Restoration Time</td>
<td>Vent Valve Failure Time</td>
</tr>
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Table 3: High Power Test Case Simulation Variables

For ease of reading, Table 2b is repeated here and shows the Stochastic Parameters used in this test:

<table>
<thead>
<tr>
<th>Stochastic Parameter</th>
<th>Parameter Range Considered</th>
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<tbody>
<tr>
<td>EDG Failure Time</td>
<td>0 hours – 4 hours</td>
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<tr>
<td>EDG Recovery Time</td>
<td>0 hours – 12 hours</td>
</tr>
<tr>
<td>Offsite AC Power Recovery Time</td>
<td>0 hours – 12 hours</td>
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<td>HPC Failure Pressure</td>
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<td>Steam Vent Valve Failure Start Time</td>
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<tr>
<td>Steam Vent Valve Failure End Time</td>
<td>0 hours – 8 hours</td>
</tr>
<tr>
<td>Water Sump Valve Failure Start Time</td>
<td>0 hours – 8 hours</td>
</tr>
<tr>
<td>Water Sump Valve Failure End Time</td>
<td>0 hours – 8 hours</td>
</tr>
<tr>
<td>Reactor Power</td>
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</tr>
</tbody>
</table>

Table 2b: MASLWR SBO Stochastic Parameters, High Power Test Case

Finally, Table 4 shows the nominal values for each stochastic parameter. Whenever the parameters were not being specifically scrutinized, these values were used, where appropriate, as a default value for each parameter.
<table>
<thead>
<tr>
<th>Stochastic Parameter</th>
<th>Nominal Value</th>
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</thead>
<tbody>
<tr>
<td>EDG Failure Time</td>
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</tr>
<tr>
<td>EDG Recovery Time</td>
<td>12 hours</td>
</tr>
<tr>
<td>Offsite AC Power Recovery Time</td>
<td>12 hours</td>
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<tr>
<td>HPC Failure Pressure</td>
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<tr>
<td>Steam Vent Valve Failure Start Time</td>
<td>8 hours</td>
</tr>
<tr>
<td>Steam Vent Valve Failure End Time</td>
<td>0 hours</td>
</tr>
<tr>
<td>Water Sump Valve Failure Start Time</td>
<td>8 hours</td>
</tr>
<tr>
<td>Water Sump Valve Failure End Time</td>
<td>0 hours</td>
</tr>
<tr>
<td>Reactor Power</td>
<td>300%/360%</td>
</tr>
</tbody>
</table>

Table 4: MASLWR Stochastic Parameter Nominal Values

Exceptions to Table 4 are that for Test 3, EDG Failure Time vs. Sump Valve Failure End Time, the Sump Valve Failure Start Time was set to 0 seconds to explore the potential failure space as thoroughly as possible. Similarly, for Test 4, EDG Failure Time vs. Sump Valve Failure Start Time, the Sump Valve Failure End Time was set to 8 hours. Similarly for Tests 5 and 6, which instead examined the failure of the MASLWR Vent Valves. Finally, each test was run at both 300% rated power and 360% rated power, using two thousand test cases of RELAP5 for each test to thoroughly explore the stochastic parameter space.

Incredibly, even at 300% power, MASLWR’s safety systems withstood the transient and no core damage occurred. For all eight tests at 300% power, MASLWR’s reactor integrity was maintained. Two of the tests did have failures, but they seemed erroneous, as the failure data was scattered seemingly randomly across both sampled stochastic parameters and did not form any kind of coherent failure space. A simulation would show as a failure, surrounded by successes on all sides, and with no other nearby failures. It is believed that these failures were simply artifacts of the idiosyncrasies of RELAP5-3D or the interaction between RELAP5-3D and RAVEN, and potentially the
result of RELAP5-3D simulations ending unexpectedly or crashing. Because these failures seemed to be simulation artifacts instead of reflective of reality, these results were discarded.

It was only when the 360% power data came through that a failure space for the MASLWR-SBO transient could be established. For five of the tests at 360% power, no recognizable failure space could be established. However, for tests 3, 4, and 7 (EDG Failure vs. Sump Failure End, EDG Failure vs. Sump Failure Start, and Sump Failure End vs. Sump Failure Start), recognizable failure spaces could be established. These are shown in Figure 24.
In Figure 24, the lighter gray diamonds mark simulations for which the simulation ended in success, where core safety was maintained, and the black diamonds mark simulations for which the simulation ended in failure, and core damage occurred. The failures are clustered in the bottom right portion of the graph, and are all simulations where the EDGs fail quickly and the Sump Valves take a very long time to restore, indicating that either a long EDG failure time or a reasonable Sump Valve function restoration time is enough to keep the plant safe in these circumstances. Specifically, if the EDGs stay online for approximately two hours, or if the Sump Valves can be brought online within four hours of the transient initiating, the plant will remain safe. It can also be seen that failure is dependent on both variables, and that the boundary of the failure space slopes up and to the right.
In Figure 25, it can be seen that, again as expected, a long EDG Failure Time contributes strongly to plant safety. Additionally, if the Sump Valves work for a period of time before failing, the plant is extremely likely to be safe – once the plant is depressurized and the core reflooded with cold water, the large thermal mass of all of the water in the core is enough to keep the plant safe for an extended period of time. If either the EDG Failure Time or the Sump Valve Failure Time are longer than about 5000 seconds, plant safety was maintained. Interestingly, unlike the results shown in Figure 24, the failure space has very little slope. Where a longer Sump Valve Restoration Time in Figure 24 allowed for failure with a significantly longer EDG Failure Time, the sampled value of the Sump Valve Failure Time appears to have little or no impact on the EDG Failure Time values for which the simulation will end in core damage.
As the results shown in Figures 24 and 25 would point to, Figure 26 shows that both a fast Sump Failure and an extremely long time to Sump Restoration are needed to cause core damage. As discussed previously, once the core depressurizes and refloods, enough coolant is available to keep the plant safe for an extremely long period of time. Figure 26 also shows the failure space’s dependence on both the Sump Valve Failure Time and the Sump Valve Restoration Time. As the Sump Valve Restoration Time increases from roughly 18000 seconds to the full 28800 seconds of 8 hours, the upper end of its sampling space, the Sump Valve Failure Time for which core damage will occur increases from a failure on demand scenario to about 4000 seconds.
6 Conclusion and Future Work

The goal of this project was to show that RAVEN can be used to explore the stochastic parameter failure space of an accident scenario and find more rigorous answers to questions regarding scenario outcomes than older PRA methods can provide. The BWR-SBO case showed the full breadth and depth of the capabilities of RAVEN to explore an accident scenario with many stochastic parameters and to explore all of them at the same time, finding multiple paths to failure in the BWR-SBO accident scenario. The MASLWR-SBO case showed both that RAVEN can be used to explore designs other than BWRs and also demonstrated RAVEN’s ability to explore a reactor’s response to extreme scenarios when the analysis of more reasonable scenarios is less than interesting.

The BWR-SBO case showed three paths to failure in the BWR-SBO accident scenario (seen in Figures 19 and 20 on pages 57 and 59). Specifically, if HPCI and RCIC fail to run and either an SRV becomes stuck open, the EDGs fail quickly, or the batteries fail and HPCI and RCIC remain inoperable, core damage occurs. These are expected paths to failure, but the ability to assign specific values to the accident parameters – the EDG failure time, the Battery Power Lifetime, the Offsite Power Recovery Time, etc – for which core damage will occur allows us to more accurately evaluate accident scenario risks and to see, in detail, what systems’ failures do the most to harm plant integrity and to have not only mechanistic values for the strength of safety systems, but also a probabilistic approach that estimates the chance that a safety system will be overwhelmed in an accident scenario.

The MASLWR-SBO case primarily underscored one of the main strengths of the MASLWR design – its passiveness. Even with its safety systems failing wholesale, at
100-120% power the MASLWR design did not fail, and seemed nigh impervious to SBO scenarios. As far as the scope of this project goes, MASLWR simply does not appear to have a failure space for SBO accidents if AC power is recovered within twelve hours after the onset of SBO conditions. The MASLWR-SBO high power test case did show that the Sump Valves, of the aspects of the MASLWR design that were modeled, are one of the most important plant systems with regards to maintaining plant safety. It is anticipated that if there is a failure space in longer duration MASLWR-SBO accident scenarios (at regular power), it will involve long duration Sump Valve failure.

As well as modeling power plant accident scenarios, RAVEN can be used with any deterministic code to allow it to account for stochastic parameters within the scenarios the deterministic code is attempting to model, both within and outside of nuclear engineering. RAVEN is built to be a ‘black box,’ where the user need only develop a short script to allow RAVEN to properly generate input commands to run the deterministic code, and an output script to convert outputs into a CSV (comma separated values) format that RAVEN can handle. The limitation to this is that RAVEN is not compatible with codes that have outputs that cannot be put into a CSV format. This can potentially change, but would require significant changes to the way RAVEN handles file output and would likely need to be done on a code by code basis.

To extend this project, it is intended to use RAVEN, as well as realistic stochastic parameter probability density curves, to use the RISMC approach to better evaluate severe accident scenarios to determine the probabilistic economic cost that risk represents. Additionally, the probabilistic economic benefit that various safety margin upgrades give in terms of risk will be examined by altering the models to reflect various
potential safety margin upgrades. The benefits of these safety margin upgrades will be compared to the costs of implementing them. The goal is that this will allow for more informed, science driven decision making in decision making with regards to plant safety systems.

To do this analysis, RAVEN will be used with RELAP-7 to model power plant accident scenarios that present a severe probabilistic risk to power plant safety using the RISMC methodology. RELAP-7 is a next generation thermal-hydraulics power plant accident scenario code that is being developed by Idaho National Lab under the MOOSE framework. RELAP-7 and RAVEN are being built to complement each other, and because of this, RELAP-7 can make use of RAVEN’s data processing capabilities in ways that other codes cannot.
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Appendix A: BWR safety system diagrams and further discussion

The Standby Liquid Control System (SLCS), seen in Figure A-1, consists of a tank of heavily borated water, a set of two pumps, a set of two explosive valves, and the necessary pipes to connect the SLCS components and to connect the SLCS to the reactor vessel. The boron in the borated water acts as a neutron poison, and will end the nuclear chain reaction in the core when injected. The SLCS never triggers automatically and must be started manually, and functions as a backup to the Reactor Protection System automatic SCRAM. When started by the operator, the SLCS explosive valves burst open and the SLCS pumps begin injecting the borated water into the core, killing the reaction.

Figure A-1: Standby Liquid Control System (nrc.gov 2012)
The High Pressure Coolant Injection (HPCI) system and Automatical Depressurization System (ADS), seen in Figure A-2, serve to maintain core inventory and core pressure during accident scenarios. Additionally, the ADS can be used to depressurize the reactor in scenarios where the HPCI system can no longer be used to maintain core inventory or scenarios where the HPCI system is insufficient to maintain core inventory. The ADS is a system of hardwired logical trips and a series of safety relief valves used to vent steam from the core into the Containment Suppression Chamber (also called the Pressure Suppression Pool). The Containment Suppression Chamber is a large tank of subcooled water that is used to cool vented steam and provide a sink to which the ADS can safely vent steam.

Aside from a supply of water, the HPCI system requires no external support from other plant systems to operate and is an independent system capable of providing make up water to the core. The HPCI system can draw water either from the Condensate Storage Tank, a backup tank of water available for use with Emergency Core Cooling System subsystems, or from the Containment Suppression Chamber. It consists of control valves, a turbine that draws steam from the Main Steam Line for power and a pump, driven by the HPCI Turbine, that injects water from the Condensate Storage Tank or Containment Suppression Chamber into the reactor vessel.
Figure A-2: High Pressure Coolant Injection and Automatic Depressurization Systems (nrc.gov 2012)
The Reactor Core Isolation Cooling (RCIC) system, seen in Figure A-3, fulfills a similar function as the HPCI system. Similar to the HPCI system, the RCIC system consists of control valves, a turbine that uses steam from the Main Steam Line for power, and a turbine driven pump that draws water from either the Condensate Storage Tank or the Containment Suppression Chamber to provide make up coolant and cooling for the core. As with the HPCI system, the RCIC system requires no external support from other plant systems aside from a supply of water. Most importantly, the HPCI and RCIC systems do not require external AC power, and can be operated for hours using DC battery power and steam from the Main Steam Line to provide core cooling during Station Blackout scenarios.
Figure A-3: Reactor Core Isolation Cooling system (nrc.gov 2012)
The low pressure Emergency Core Cooling System subsystems, seen in figure A-4, consist of two independent systems, the Low Pressure Coolant Injection (LPCI) system and the Low Pressure Core Spray (LPCS) system. The LPCI system is, in reality, a second operational mode of the Residual Heat Removal (RHR) system. During normal shut down operations, the RHR system can be used to remove decay heat from the core indefinitely, keeping it in a cooled and stable configuration. During accident scenarios, typically large break loss of coolant accident (LOCA) conditions, the RHR system can be switched to its LPCI operational mode to inject large amounts of water from the Containment Suppression Chamber into the core to provide emergency makeup water to maintain core cooling. The RHR (and thus LPCI) system consists of control valves, piping, a heat exchanger used during normal shut down operations, and a pair of pumps that draw cold water from the Containment Suppression Chamber and inject it into the core.

The LPCS system is used to condense steam generated by a major accident scenario, keeping core pressure low so the LPCI and LPCS systems can operate, as high pressure in the core prevents these systems from functioning. It accomplishes this through a pair of pumps that draw water from the Pressure Suppression Chamber and inject it into the core from above, collapsing steam voids above the core to reduce core pressure. Additionally, the LPCS system can be used in accident scenarios where the LPCI system cannot adequately maintain the core water level to spray the fuel from above to maintain core cooling despite the fuel being uncovered.
Figure A-4: Low Pressure Coolant Injection and Low Pressure Core Spray systems (nrc.gov 2012)