Advanced Thermal Hydraulic Simulations for Human Reliability Assessment of Nuclear Power Plants

by

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Submitted to the Department of Nuclear Science and Engineering in partial fulfillment of the requirements for the degree of Masters of Science in Nuclear Science and Engineering at the MASSACHUSETTS INSTITUTE OF TECHNOLOGY

February 2017

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Abstract

Human Risk Assessment (HRA) in the nuclear power industry has advanced in the last two decades. However, there is a lack of understanding of the magnitude of the effect of thermal hydraulic (TH) uncertainties upon the failure probabilities of the operator actions of nuclear units. I demonstrate in this work that there is an effect of TH uncertainties on the operating crew’s probability of recognizing errors during a loss of coolant accident (LOCA) initiating event. The magnitude of the effect of the TH uncertainty on the operator’s ability to recognize errors is dependent upon the size of the break, the operating state of the plant (in operation or shutting down), and the error that is committed. I utilized an uncertainty software, Dakota, coupled with an advanced TH software, MAAP4, to perform a Monte Carlo analysis to propagate selected TH uncertainties through a LOCA initiating event in which the automatic safety coolant injection system fails to automatically actuate. The operator mission is to manually actuate the safety coolant injection system. Two errors that the operating crew could make are 1) entering fire procedures and 2) testing for saturation of the primary system before the saturation occurs. I calculate the operator failure probabilities using the MERMOS HRA methodology (used by the French electric utility company Electricité de France, EdF). My results show a reduction in scenario failure probability from the values reported by EdF in its published MERMOS Catalogue of more than 80% for the operator recognizing the the error in entering fire procedures. For the error in testing for saturation of the primary system before saturation occurs, I calculated a scenario failure probability in Mode B of 0.0033, while the MERMOS Catalogue listed the scenario failure probability as negligible. My results show that there is an effect from TH uncertainties on operator failure probabilities. This research provides a method of improving the accuracy of failure probabilities in established HRA methodologies using TH simulations.

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Acknowledgments

This work was only possible with the resources and support of Electricité de France (EdF), the electric utility company in France, and the Massachusetts Institute of Technology (MIT). In addition, this research would not be possible without the support of many people.

First, I would also like to thank my advisor, Dr. Michael Golay, for his guidance throughout the project. His vast experience and knowledge in the field of probabilistic risk assessment has been invaluable to me. He always offers a relevant perspective and has improved my critical thinking considerably.

Second, I would like to thank Dr. Valentin Rychkov from EdF. He gave me an opportunity to work at the EdF office in Paris, France which was an incredible experience. He have given me great guidance and I am very grateful for him taking the time to help me.

Third, I would not have started working on this project if it were not for Dr. Brittany Guyer. She was a large part of the reason I chose to enroll at MIT. She introduced me to this project and her passion for probabilistic risk assessment encouraged me to enter this field.

Finally, I express gratitude for my family. I am thankful for my parents for not only supporting my ambitions but for also encouraging me to pursue a career in engineering. I would also like to thank my older siblings, Debbie, Doug, and Lizzie, for their inspiration and motivating me to achieve all that I can.
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Chapter 1

Introduction

Nuclear power plants were designed using deterministic safety assessments (DSA). In this type of risk assessment, design basis events are formulated. Safety is determined by demonstration that the nuclear power plant can adequately respond to these events according to specified acceptance criteria [1]. However, with time, use of probabilistic risk assessment (PRA), sometimes also referred to as probabilistic safety assessment (PSA), became a design requirement along with the DSA. In PRA, safety is determined from regarding a large span of system scenario initiating events and tracing the effects through event and fault trees [2]. The progressions of the system scenarios are influenced by the nuclear unit’s operators’ actions. The study of examining these human actions is referred to as human reliability analysis (HRA). In a PRA, the operators’ actions can be modeled within the event tree or within the fault tree, depending on if an action is scenario specific (the former) or a requirement of a system (the latter).

Probabilistic risk assessment (PRA) methodology includes determining undesirable consequences, identifying ways that these consequences could happen, estimating the likelihood of such ways, and providing the analysis to decision makers to be used in choosing strategies to reduce risk levels. PRAs have been used in the nuclear power industry since the creation of the method in the Reactor Safety Study (also known as WASH-1400) in 1975 [3]. In the 1990s, each licensed nuclear power unit in the US performed a plant-specific search for vulnerabilities using PRA methodologies, as per
requirements of the Nuclear Regulatory Commission (NRC) [4].

Conducting a PRA of a nuclear power plant has many benefits in safety management. In general, in the United States, a PRA will be conducted by the licensee of a nuclear power plant to demonstrate compliance with regulations. In addition, a PRA can be used as a tool to highlight actions from which the nuclear unit can receive the most safety benefits. It does this through use of several measures of risk such as Fussell-Vesely (FV) or Risk Achievement Worth (RAW) classifications of systems and components [2]. This allows the licensee to focus resources on systems that will make the plant the safest, rather than focusing many resources on systems that will have little effect on safety.

The core of HRA is the understanding of human responses during accidents. There exist major limitations in the current PRA methodology surrounding the treatment of human interactions with the nuclear unit system. One such limitation is absence of thermal hydraulic (TH) uncertainty input into the failure probabilities associated with operator actions. Because of this, it is not known to what extent these TH uncertainties affect the operator failure probability. If there is evidence that there is an effect from the TH uncertainties, it is important to know which thermal hydraulic factors (ie. primary system temperature) play the largest role in influencing operator failure probabilities.

To examine the effect that TH uncertainties have on operator actions, I utilized an industry HRA methodology coupled with an advanced TH simulation. The HRA methodology selected was MERMOS, used by the French electric utility company, Electricité de France (EDF). Due to the proprietary nature of the MERMOS results, I utilized a generic publication, MERMOS Catalogue. The TH simulation software selected was the Modular Accident Analysis Program, Version 4, (MAAP4), which is owned and licensed by the Electric Power Research Institute (EPRI). To couple the HRA methodology with the TH software, I utilized an uncertainty software, Dakota, which is an open-source toolkit developed by Sandia National Laboratories.

I performed a sensitivity analysis on TH inputs inputs into MAAP4 for a scenario listed in a MERMOS analysis. The scenario is a small break loss of coolant accident
(SBLOCA) with failure of automatic safety injection. It is the operator’s mission to determine that a SBLOCA is occurring and to manual start the safety injection. Failure to do so could result in damage to the nuclear unit’s core. I looked specifically at two paths to operator failure to manually start the safety injection, labelled Case 1 and Case 2:

**Case 1**: The first path to failure occurs when the steam that escapes the primary system causes a fire alarm to activate. The operators misdiagnose the transient as a fire and enter fire procedures. The operator will only succeed if he recognizes his misdiagnosis and initiates the manual safety injection before core damage occurs.

**Case 2**: The second path to failure occurs when the operator tests for saturation of the coolant in the primary system of the nuclear unit before it occurs. The saturation test consists of the operating crew checking the readings of an ebulliometer, which shows the margin to the boiling point at the current temperature and pressure in the system. If the operator does not see saturation of the coolant, then he will not be directed to the SBLOCA emergency procedures, and will not manually initiate the safety injection. The operator will only succeed if he recognizes the SBLOCA through a second saturation test, and manually initiates the safety injection system before core damage occurs.

I determined through the sensitivity analysis of the two cases that the failure probabilities reported in the MERMOS Catalog are dependent upon the TH inputs. The uncertainties in the selected TH parameters are propagated through the transient to determine the uncertainty in the operator failure probability.
Chapter 2

Background

In order to examine HRA estimates of operator crew failure probabilities for the nuclear industry, it is important to know the aspects of HRA that are unique to the nuclear industry. While some aspects of HRA methodologies in the nuclear industry may be similar to those of other industries that require operators (i.e., chemical processing plants, wastewater treatment facilities, etc.), several aspects are unique to an operator at a nuclear power plant. First, the operator has a lot of information in front of him due to the large number of indicators and instrumentation in the control room. This presents a benefit as the information is available, but it is also a detriment as the operator needs to filter the information in order to find and use the important parts. In addition, the actions that an operator must take are largely guided by pre-established procedures upon which they are trained. These procedures and the training are often regulated. In the United States these standards are established by the Nuclear Regulatory Commission. Often, the actions that an operator must undertake have a time constraint associated with them that is affected by the TH state of the unit [5]. Finally, it is important to note the composition of the operators in the control room: generally there are two licensed reactor operators, a senior nuclear operator, a shift manager, and several non-licensed auxiliary operators [6].
2.1 Human Reliability History in the Nuclear Industry

There have been two generations of human reliability analysis methods in the nuclear industry. The first generation was born out of human factors engineering, and had a main focus of finding potential pathways for human errors. The second generation has not yet been implemented at a wide-scale, and was begun in the 1990s as the weaknesses of the first generation methodologies were deemed undesirable for inclusion in PRAs [6].

2.1.1 First Generation Methods (Used in Industry Today)

The birth of HRA methodologies occurred within human factors engineering, a field created after World War II. Human factors was a field of research focused upon reducing the frequencies of unwanted outcomes that would arise from human operator actions in complex systems. It constituted a switch from using a "part should fit into the system" model of efficiency and effectiveness to a recognition that even with perfect training, a human operator can make important mistakes. It focused on finding potential for human errors early in the design process so that needed design changes could be made. This focus was carried over into the beginning of HRA, which was defined as a "method employed to quantitatively assess the impact of potential human errors on the proper functioning of some system composed of equipment and people [7]. It is by nature both qualitative and quantitative.

Human reliability analysis was used in the WASH-1400 Reactor Safety Study [3] to assess the reliability of nuclear power plant operators. The methodology employed in that report was THERP (Technique for Human Error Rate Prediction) [8] [9]. About fifteen HRA methods were used in the PRAs that followed the WASH-1400 report. There were weaknesses in the HRA methodologies used in the PRAs. It was hard to compare results from one PRA to another because there were too many methodologies. In addition, there was not enough adequate data for use in the models.
and so there was a large reliance on the use of expert judgement [7].

The first generation HRA for nuclear power plant PRAs has many drawbacks, as summarized by [10] in the list below:

1. Consideration of only errors of omission (failure to perform a required action) and lack of consideration of errors of commission (performance of an undesired action)
2. Little theoretical foundation for error probabilities
3. Lack of a causal model for operator errors
4. Lack of structure leading to large variability between different analyses

2.1.2 Second Generation Methods (Current Research)

Research on second generation HRA methods began in the 1990s. It was motivated by the need for easier use of HRAs in PRAs, as well as a desire for increased accuracy. Most second generation HRA models are not used in industry applications as they are still in development. There are two different approaches to improving the first generation HRA methods. One approach improves the quality of the HRA for use in current static PRA methods. The other approach improves the quality of HRA methods for use in dynamic PRA methods [10].

HRA research has made progress in error identification for errors of commission. Macwan and Mosleh (1994) describe a methodology to include errors of commission in the PRA of a nuclear unit by combining information from the nuclear unit (PRA, operating procedures, etc.) and performance influencing factors. The combination of information is used to create an "initial condition set" that is used to determine sequences of human actions, which can include the errors [11]. Julius et al. (1995) shows a framework for identifying errors of commission through analyzing situations in which the operating crew is required to intervene to bring the unit to a safe state following an initiating event [12]. It has become standard practice in development of HRA models to incorporate errors of commission. There is also progress being made
in developing causal models for human failure probabilities, such as ATHEANA (A Technique for Human Error Analysis) [13] and IDA (Identify, Decide, Act) [6].

MERMOS is an early second generation HRA method developed by the French utility, EdF. It is one of the only second generation method that is in regular use in the nuclear energy industry. It is solely used by EdF. MERMOS replaced EdF’s previous approach to HRA, Probabilistic Human Reliability Assessment (PHRA) [14]. The replacement was motivated by a new view of the operating team. Instead of viewing failures as the result of one operator’s actions, failure is considered to be a function of the entire operating system (the operating system includes both the human operators as well as the technical elements of the system and organization of the system). In other words, if one operator makes an error, that error is considered to be an error of the operating system. The move from PHRA to MERMOS was done to reproduce more accurately results from simulator experiments where communication and coordination of the crew played an important role in operation during emergencies. It was not enough to model each operator’s actions individually because there were interactions between the individual operators. EdF developed a new definition of the entire operating system, "Emergency Operating System." EdF also developed a model to describe the system called Strategy, Actions, Diagnosis (SAD). In this system, an individual operator error is considered to be component within the a situation. Another new feature of MERMOS is using the amount of time an operating system has to handle a transient as a factor in determining failure probability [15].

2.2 Motivation

There is much room for improvement in the area of error quantification in HRA methodologies. This is because most methods for HRA lack an underlying theoretical basis relating the key factors in the failure probability determination. In most cases the failure probabilities are generated by expert elicitation. It is speculated by Mosleh and Chang (2004) that the reason for the lack of these strong theoretical bases are the tendencies of model developers to try to capture the entire operator interaction
picture. In doing this, they are able to generate a strong conceptual framework for their models, but the models have typically poor and/or unspecified details and data. In addition, this approach often extends a specific experimental result to the overall interaction model, even to places where the experimental results are not relevant [10].

There is a specific limitation in HRA methods, both first and second generation, in that the uncertainties of the nuclear unit’s thermal hydraulics (TH) are not captured in the HRA analysis. The extent to which the TH uncertainties affect the operator failure probabilities is unknown. Determining the extent of this effect is important because it can provide a case to either include or exclude TH in the underlying theoretical basis relating the key factors in the failure probability determination. If the TH uncertainties do not affect the failure probabilities of the HRA analysis, then further research into including the uncertainties in the HRA analysis may not be necessary. However, if the TH uncertainties do show an effect of the failure probabilities of the HRA analysis, then there will be a need to research a method of including the TH uncertainties into the HRA analysis.

I have developed an interface between second generation HRA methodology, MERMOS, and a dynamic thermal hydraulic model, MAAP, utilizing an uncertainty software, Dakota. In general, HRA methods have some sort of TH "boundary condition" upon which the human action is in some way dependent (see Figure 2-1).

The sophistication of this dependence varies between the first generation methods. The MERMOS treatment of this is described below. To link the HRA method to the TH simulation, I tracked this boundary condition for each run of the simulation.
2.3 Tools Used

2.3.1 MERMOS

Electricité de France (EdF), the electric utility company in France, utilizes an HRA methodology named MERMOS. I selected this method for use in this work in collaboration with EdF researchers. In addition, MERMOS operator failure probabilities have a clear and simple dependence upon boundary conditions. For example, many failure probabilities of an operator action are functions of the time that an operator has available to complete that action. This "mission time" is a boundary condition: it is determined by the TH conditions of the unit and affects the failure probabilities of an HRA.

The MERMOS method has an analysis in which every known path leading to the failure of an operator mission is considered. Each path constitutes a "failure scenario." Each path is assumed to be mutually independent, and so the sum of the failure probabilities for the respective failure scenarios plus a residual probability is the total mission failure. The residual probability is an addition to the failure probability in order to account for paths that are unknown and therefore not modeled. The residual probability captures the failure scenarios that are either small enough to be considered negligible or failure scenarios that were not thought of by experts. This relationship is stated in Equation 2.1 as follows

\[
P_{\text{mission failure}} = \sum_{i=1}^{\text{all failure scenarios}} P_{\text{failure scenario}} + P_{\text{residual}}. \tag{2.1}
\]

Each failure scenario has three properties: situation characteristics (PS for Probability of a Situation), context (CICA for Caracteristiques Importantes de la Conduite Accidentelles), and non-reconfiguration (NR for Non-Reconfiguration). Utilizing the three properties, one can find the probability of a failure scenario occurring using Equation 2.2

\[
P_{\text{failure scenario}} = \prod_{i=1}^{n_1} P_{PS,i} \times \prod_{j=1}^{n_2} P_{CICA|PS,j} \times \prod_{k=1}^{n_3} P_{NR|CICA,k}. \tag{2.2}
\]
The description of each of these properties follows:

- **Situation Characteristics, PS**: These are the characteristics of the nuclear unit and of the transient itself. These characteristics constitute information that the operator receives in the context. These are determined by experts. The total number of characteristics is $n_1$.

- **Context, CICA**: This is the environment surrounding the operator capable of influencing his behavior. It includes the personnel present in the operator room, the organization of the crew, and the information presented to the crew. The elements of the environment are determined by experts. The total number of elements is $n_2$.

- **Non-Reconfiguration, NR**: The model MERMOS allows for the operator to realize mistakes, and to go back to fix them. However, there is a chance that the operator will not realize his mistake in time for correction. This relationship is represented as non-reconfiguration. These failure probabilities are determined by experts. The total number of mistakes to correct is $n_3$.

Since 2011, EdF has been working on an addition to MERMOS, named the "MERMOS Catalogue," which is a generic HRA analysis for many different scenarios [5]. In principle, this generic analysis incorporates the information of all existing prior analyses performed for similar scenarios. Since the MERMOS analyses are proprietary, I utilized the more generic MERMOS Catalogue scenarios in my work. Specifically, I utilized two case studies from the MERMOS Catalogue resulting from a SBLOCA initiating event with failure of automatic safety injection. These two case studies were determined to be risk-dominant failure scenarios in the MERMOS Catalogue analysis of this transient [5].

### 2.3.2 Modular Accident Analysis Program (MAAP)

The code MAAP is a licensed software owned by the Electric Power Research Institute (EPRI). It models the transients in nuclear reactor systems (both light water-cooled
and heavy water-cooled). Operator actions can be simulated in the input file processor [16]. It was selected because of my familiarity in running the code. In addition, MAAP4 was selected because it uses a relatively small amount of computational time.

2.3.3 DAKOTA

The code Dakota is an open-source software developed by Sandia National Laboratories. It is a toolkit that allows for iterative uncertainty analysis. The sixth version of the software was utilized in this work [17]. Our work could also have been successful with other similar software, such as OpenTURNS [18], however Dakota was selected due to ease of learning the programming and to time limitations for completing the work. Dakota also has convenient features, such as an algorithm to develop an interface between the uncertainty inputs and a software code (in this case MAAP) [17].
Chapter 3

Methodology

The transient examined in my work is a SBLOCA initiating event in both in operation ("on-power," Mode A) and during planned shutdown of the unit ("off-power," Modes B and C). While a nuclear unit is being taken offline (shutdown), it follows a specific path. Modes B and C are defined by the primary system pressure and temperature. In general, Mode B occurs during the first 6.5 hours after the scram event occurs, and Mode C occurs during the 6.5 hours to 13.5 hours after the scram event occurs. The progression of temperature and pressure with time in Modes B and C is shown in Figure 3-1.

Figure 3-1: Approximate Shutdown Temperature and Pressure Profile
The definitions of the operational modes and submodes based on pressure and temperature are summarized in Table 3.1.

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<thead>
<tr>
<th>Time Since Scram</th>
<th>Mode B</th>
<th>Mode C1</th>
<th>Mode C2</th>
<th>Mode C3</th>
<th>Mode C4</th>
</tr>
</thead>
<tbody>
<tr>
<td>0 to 6.5h</td>
<td>6.5 to 7.5h</td>
<td>7.5 to 9.5h</td>
<td>9.5 to 11.5h</td>
<td>11.5 to 13.5h</td>
<td></td>
</tr>
<tr>
<td>Pressure</td>
<td>2.8 to 15.5MPa</td>
<td>3.0MPa</td>
<td>3.0MPa</td>
<td>3.0MPa</td>
<td>3.0MPa</td>
</tr>
<tr>
<td>Temperature</td>
<td>180 to 286°C</td>
<td>180°C</td>
<td>170°C</td>
<td>120 to 170°C</td>
<td>70 to 120°C</td>
</tr>
</tbody>
</table>

Table 3.1: Off-Power Mode Conditions

In Mode B, the temperature of the reactor primary system is between 180 and 286°C. The reduction in temperature occurs fairly linearly with time. The pressure of the reactor primary system is between 2.8 and 15.5 MPa. The reduction in pressure also occurs fairly linearly with time.

In Mode C, the temperature of the reactor primary system is between 70 and 180°C. The reduction in temperature does not occur linearly with time (see Figure 3-1). The pressure of the reactor primary system is 3.0 MPa. It is constant with time. Due to the change in progression of temperature with time in Mode C (uniform and then linearly decreasing), it was necessary to break it into smaller submodes. Each submode represents a linear section of the temperature reduction as a function of time. This is illustrated in Figure 3-1. Each operational mode or submode modeled is separated by the vertical lines.

I chose a defining size for a "small" break to be a diameter under 0.0508m. This is the threshold between small and medium LOCAs in the South Texas Project PRA [19]. Following the occurrence of break, the safety coolant injection system fails to actuate automatically. In the case of on-power status, this occurs because the automatic coolant injection system does not properly receive a signal to actuate. In the case of an off-power condition, this occurs because the automatic safety coolant injection signal is turned off as the system is reducing pressure and temperature [20].

A generic Westinghouse 4-Loop PWR unit was selected as the example system for analysis of the transient. This is because the generic specifications for the unit are
provided in the MAAP4 default input code deck [16]. The safety injection (SI) function is needed in order to avoid core damage and thus the operator needs manually to initiate the injection. This is the operator’s mission. As indicated in Chapter 2, there are two dominant failure scenarios for this action [5] as outlined in Case Study 1 and Case Study 2, respectively. The cases depend upon two boundary conditions, Saturation Time (time to reach saturation of the primary system coolant, $\Delta t_{st}$, see Equation 3.1) and Mission Time (time allowed for operator success, $\Delta t_{mt}$, see Equation 3.2).

Saturation Time is the time from the reactor scram until the primary system coolant reaches the thermodynamic saturation point. There is a possibility of misdiagnosing the transient if this time is so large that the operating crew tests for saturation \(^1\) before the primary system coolant reaches the thermodynamic saturation point. In this case, the test will indicate that the system is not in a state of saturation and so the operating crew will not realize that the transient was initiated by a LOCA event.

Saturation Time is calculated by

$$\Delta t_{st} = t_{saturation} - t_{scram}, \quad (3.1)$$

where $t_{saturation}$ is the time of the primary system saturation and $t_{scram}$ is the time of scram for on on-power operating mode (Mode A) or the low pressurizer signal for the off-power modes (Modes B and C).

I define Mission Time as the time extending from the condition of primary system coolant saturation, which would be the first indication of occurrence of a LOCA event, until the reactor core is uncovered from the coolant. I assume that this condition corresponds to one of permanent damage.

Mission Time is calculated by

$$\Delta t_{mt} = t_{core\ uncover} - t_{saturation}, \quad (3.2)$$

\(^1\)The saturation test consists of the operating crew checking the readings of an eboulliometer, which shows the margin to the boiling point at the current temperature and pressure in the system.
where $t_{\text{core uncover}}$ is the time at which the nuclear fuel begins to uncover and $t_{\text{saturation}}$ is defined previously.

### 3.1 Case 1 Description

In Case 1, a spurious fire alarm signal near the coolant line break location is triggered by the steam flashing upon exit from the break. The operators are assumed to misinterpret the transient as a fire, and to follow fire procedures rather than test for a different initiating event. Doing this delays the manual initiation of the safety coolant injection. The fire procedures consist of locating the fire, identifying the functions that are likely to be affected by the fire, and conducting electricity cut-offs (isolating actions) [5]. Therefore, the Mission Times obtained from my simulations indicates how much time the operator has in order to follow fire procedures, and then to realize that the true situation is in fact a LOCA occurrence. This is a reconfiguration, and the failure to achieve it successfully is the probability of non-reconfiguration. The probability of non-reconfiguration is a function of Mission Time as stated in Equation 3.3.

The probability of non-reconfiguration is

$$P(NR|CICA) = \begin{cases} 
1 & \Delta t_{mt} \leq 25\text{min.} \\
0.9 & 25\text{min.} < \Delta t_{mt} \leq 35\text{min.} \\
0.3 & 35\text{min.} < \Delta t_{mt} \leq 45\text{min.} \\
0.1 & 45\text{min.} < \Delta t_{mt} \leq 90\text{min.} \\
0.01 & 90\text{min.} \leq \Delta t_{mt}, 
\end{cases} \quad (3.3)$$

where $\Delta t_{mt}$ is the Mission Time from Equation 3.2. The lowest possible probability of failure is assumed to be 0.01. The timeline of Case 1 is illustrated in Figure 3-2.
3.2 Case 2 Description

In case 2, when the primary loop reaches saturation after a low pressure trip, emergency operator procedures (EOPs) state that the operator is to either confirm automatic coolant injection or manually to initiate the injection. However, if the system does not reach steam saturation between the scram and the testing for saturation, then the operator will miss the steam saturation and move onto the next step in the EOPs. The test for saturation occurs right away, within a few minutes. The operator must realize that saturation has occurred and return to test for saturation. The failure to do so is represented as a non-reconfiguration failure event [5]. The probability of this occurring is a function of Saturation Time (Equation 3.1). I assume the situation will occur (probability of 1) if the Saturation Time is greater than 3 minutes. The probability of failure to recognize the mistake is a function of Mission Time as stated in Equation 3.4.

The probability of non-reconfiguration is

\[
P(NR|CICA) = \begin{cases} 
1 & \Delta t_{mt} \leq 25\text{min.} \\
0.3 & 25\text{min.} < \Delta t_{mt} \leq 35\text{min.} \\
0.1 & 35\text{min.} < \Delta t_{mt} \leq 50\text{min.} \\
0.01 & 50\text{min.} \leq \Delta t_{mt}, 
\end{cases} \tag{3.4}
\]

where \( \Delta t_{mt} \) is the Mission Time from Equation 3.2. The lowest possible probability of failure is assumed to be 0.01. The timeline of Case 2 is illustrated in Figure 3-3.
Figure 3-3: Timeline of Case 2

### 3.3 Uncertainty Parameters

The distributions of the TH parameters used in the simulation were selected through studying previous work. The selection of the TH parameters is outlined in Appendix A. The distributions are summarized in Tables 3.2, 3.3, and 3.4 for Modes A, B, and C.

<table>
<thead>
<tr>
<th>MAAP4 Variable</th>
<th>Distribution</th>
<th>Characteristics</th>
</tr>
</thead>
<tbody>
<tr>
<td>PPSL: Pressurizer Low Pressure Trip Point</td>
<td>Lognormal</td>
<td>$\mu = 12.5$ MPa, $\sigma = 0.1$ MPa</td>
</tr>
<tr>
<td>FCDBRK: Break Discharge Coefficient</td>
<td>Triangular</td>
<td>mode = 0.75, upper = 1.0, lower = 0.6</td>
</tr>
<tr>
<td>VFSEP: Void Fraction Threshold (above this void fraction the primary system is no longer a homogenous two-phase mixture)</td>
<td>Triangular</td>
<td>mode = 0.5, upper = 0.8, lower = 0.2</td>
</tr>
<tr>
<td>QCR0: Initial Core Thermal Power</td>
<td>Discrete Binary</td>
<td>$p(3236$ MW) = 0.70, $p(2265$ MW) = 0.30</td>
</tr>
<tr>
<td>ABB: Break Size Diameter</td>
<td>Histogram</td>
<td>$p(0.00635m \leq D &lt; 0.0127m) = 0.57$, $p(0.0127m &lt; D \leq 0.0381m) = 0.25$, $p(0.0381m &lt; D \leq 0.0508m) = 0.18$</td>
</tr>
</tbody>
</table>

Table 3.2: Uncertainty Parameters (Mode A)
<table>
<thead>
<tr>
<th>MAAP4 Variable</th>
<th>Distribution</th>
<th>Characteristics</th>
</tr>
</thead>
</table>
| FCDBRK: Break Discharge Coefficient | Triangular | mode = 0.75  
upper = 1.0  
lower = 0.6 |
| VFSEP: Void Fraction Threshold (above this void fraction the primary system is no longer a homogenous two-phase mixture) | Triangular | mode = 0.5  
upper = 0.8  
lower = 0.2 |
| ABB: Break Size Diameter | Histogram | $p(0.00635m \leq D \leq 0.0127m) = 0.57$  
$p(0.0127m < D \leq 0.0381m) = 0.25$  
$p(0.0381m < D \leq 0.0508m) = 0.18$ |
| TWPS0: Initial Primary Side Temperature | normal | $\mu = f(t)$  
$\sigma = 5^\circ C$ |
| PPS0: Initial Primary Side Pressure | normal | $\mu = f(t)$  
$\sigma = 0.25$ MPa |
| Time from Scram (t) | uniform | upper = 6.5 hours  
lower = 0 hours |

Table 3.3: Uncertainty Parameters (Mode B)
<table>
<thead>
<tr>
<th>MAAP4 Variable</th>
<th>Distribution</th>
<th>Characteristics</th>
</tr>
</thead>
<tbody>
<tr>
<td>FCDBRK: Break Discharge Coefficient</td>
<td>Triangular</td>
<td>mode = 0.75</td>
</tr>
<tr>
<td></td>
<td></td>
<td>upper = 1.0</td>
</tr>
<tr>
<td></td>
<td></td>
<td>lower = 0.6</td>
</tr>
<tr>
<td>VFSEP: Void Fraction Threshold (above this void fraction the primary system is no longer a homogenous two-phase mixture)</td>
<td>Triangular</td>
<td>mode = 0.5</td>
</tr>
<tr>
<td></td>
<td></td>
<td>upper = 0.8</td>
</tr>
<tr>
<td></td>
<td></td>
<td>lower = 0.2</td>
</tr>
<tr>
<td>ABB: Break Size Diameter</td>
<td>Histogram</td>
<td>p(0.00635m \leq D \leq 0.0127m) = 0.57</td>
</tr>
<tr>
<td></td>
<td></td>
<td>p(0.0127m &lt; D \leq 0.0381m) = 0.25</td>
</tr>
<tr>
<td></td>
<td></td>
<td>p(0.0381m &lt; D \leq 0.0508m) = 0.18</td>
</tr>
<tr>
<td>TWPS0: Initial Primary Side Temperature (C3/4)</td>
<td>normal</td>
<td>$\mu = f(t)$</td>
</tr>
<tr>
<td></td>
<td></td>
<td>$\sigma = 5^\circ C$</td>
</tr>
<tr>
<td>TWPS0: Initial Primary Side Temperature (C1/2)</td>
<td>normal</td>
<td>$\mu = f(t)$</td>
</tr>
<tr>
<td></td>
<td></td>
<td>$\sigma = 2^\circ C$</td>
</tr>
<tr>
<td>PPS0: Initial Primary Side Pressure</td>
<td>normal</td>
<td>$\mu = 3.0$ MPa</td>
</tr>
<tr>
<td></td>
<td></td>
<td>$\sigma = 0.1$ MPa</td>
</tr>
<tr>
<td>Time from Scram (t)</td>
<td>uniform</td>
<td>upper = 13.5 hours</td>
</tr>
<tr>
<td></td>
<td></td>
<td>lower = 6.5 hours</td>
</tr>
</tbody>
</table>

Table 3.4: Uncertainty Parameters (Mode C)

### 3.4 Monte Carlo Sensitivity Analysis

I coupled the MAAP4 analysis code to the Dakota code for propagating the uncertainties of the thermal hydraulic conditions from Tables 3.2, 3.3, and 3.4 through the transient. Latin hypercube sampling was used as the uncertainty quantification method [17].

I performed 50,000 (6,250 for each of the eight break size diameters: 0.00635m,
simulation runs in each of the Monte Carlo runs for Modes A, B, and C. I utilized 8,832 processors for each of the Monte Carlo runs. Each Monte Carlo run took 10-30 hours to complete. The average amount of time per simulation run is between 0.72 seconds and 2.16 seconds.

For each simulation run, the simulation ends when either the core is uncovered from the coolant (indicating the end of the mission time window) or when the simulation hits 24 hours after the initiating event has occurred. One day was selected as the threshold to cut off the simulation to allow for adequate time for the transient to progress. However, it should be noted that any mission times greater than 90 minutes for Case 1 and greater than 50 minutes for Case 2 would result in a probability of non reconfiguration value of 0.01 (the lowest possible value), as per Equations 3.3 and 3.4. In the latter runs where the simulation hits 24 hours, the core does not uncover from the coolant in that 24 hours. The Mission Time cannot be calculated for these runs because the time of core uncovery is unknown (it is greater than 24 hours). Therefore, these simulation runs are excluded from analysis.

Due to the exclusion of the runs in which the core was not uncovered from the coolant within 24 hours from the initiating event, the number of simulation runs analyzed are 44,592 runs for Mode A, 38,097 runs for Mode B, and 12,547 runs for Mode C. The Dakota and MAAP codes are listed in the Appendix B for Modes A, B, and C.
Chapter 4

Results

This chapter outlines the results from Case 1 and 2 for each power mode.

4.1 Case 1

In Case 1, a spurious fire alarm distracts the operator crew during a SBLOCA. The operator crew’s mission is to manually initiate safety coolant injection. The operator crew needs to recognize the error in entering fire procedures and needs to start the coolant injection. The probability of failing to recognize the error before core damage occurs is the probability of non-reconfiguration (\( P_{NR|ICA} \) or PNR). The PNR is a function of Mission Time (Equation 3.3).

For each operating mode, I first present a histogram of the Mission Time values, determined through propagating the uncertainty of the input parameters through the simulation, for all runs in which the core was uncovered by the coolant within one day (1440 minutes) from the initiating event (SBLOCA). Then, I present a normalized histogram of the Mission Times for break diameters of 0.00635m, 0.0127m, 0.0191m, 0.254m, 0.0318m, 0.0381m, 0.0445m, and 0.0508m, respectively. Then, I present the Mission Time mean values and range values as a function of the break diameter.

A Monte Carlo variance analysis was performed for the Mission Time distribution for each break size diameter in Case 1. This was performed in order to verify that adequate runs were performed and that undersampling did not occur. I performed
this analysis by calculating the variance of the distribution of Mission Times as a new Mission Time value from each simulation run was added. As my modeled distribution of Mission Time values approaches the real distribution, then the change in the variance value of the Mission Time distribution approaches zero. I show the plots of the percent change in the Mission Time distribution variance value as a function of the simulation run in Appendix C.

Finally, I present a comparison between three methods of determining the PNR value:

**PNR without Uncertainties:** The first, and least accurate, method of determining the PNR value is to estimate one value for each input of the uncertain parameters instead of using an uncertainty distribution. This means that the uncertainty is not considered and is not propagated through the simulation. This practice generates a single Mission Time value, and only that Mission Time value is utilized to calculate the PNR value. This method takes no uncertainties into account.

**PNR with Uncertainties (Averaged):** The second method of determining the PNR value is to use the distributions of uncertainty for the parameters outlined in Chapter 3 and Appendix A. Doing this generates many Mission Time values as the uncertainties are propagated through the simulation, utilizing a Monte Carlo method. I calculated the average (numerical mean values) of the Mission Time, and used this value in order to determine the PNR. While this method does take uncertainties into account, it only utilizes a single value (the numerical mean) to determine the PNR value.

**PNR with Uncertainties (Integrated):** The third, and most accurate, method of determining the PNR value is to use the distributions of uncertainty for the parameters outlined in Chapter 3 and Appendix A. However, instead of averaging the Mission Time values, I found the probability that a Mission Time value would be in each of the ranges specified in Equation 3.3. I calculated this result via Equation 4.1, where $n_i$ is the number of Mission Time values found
in a specific range and $N$ is the total number of Mission Time values generated in the simulation:

$$p_i = \frac{n_i}{N}. \quad (4.1)$$

The PNR value is then calculated as a weighted average of the PNR values in each range from Equation 3.3, as shown in Equation 4.2:

$$PNR = \sum_{i=1}^{\text{# of ranges}} (p_i \times P(NR|CICA)_i). \quad (4.2)$$

### 4.1.1 Mode A

Mode A is the operational state where the nuclear unit is at full power. Figure 4-1 depicts the Mission Time value histogram obtained from a total of 44,592 runs in which the core was uncovered by the coolant within one day (1440 minutes) from the initiating event (SBLOCA). The calculated Mission Time values range from about 30 minutes to one day.
The distribution in Figure 4-1 is not unimodal. There are five distinct local extrema. These different local extrema are results of the different break sizes used in the simulation. In order to examine the effect of the uncertainties for each of the break sizes, I examined the Mission Time histograms for each of the eight break sizes: 0.00635m, 0.0127m, 0.0191m, 0.254m, 0.0318m, 0.0381m, 0.0445m, and 0.0508m.

The histograms for the Mission Time values for each break size diameter are shown in Figures 4-2 to 4-9, respectively. The resulting range and mean values are shown in the caption of each figure.
Figure 4-2: Mission Time Normalized Histogram
Break Diameter: 0.00635 m
Mission Time Range = 28.2 min.
Mission Time Mean = 1323.4 min
Number of Runs = 842

Figure 4-3: Mission Time Normalized Histogram
Break Diameter: 0.0127 m
Mission Time Range = 752.6 min.
Mission Time Mean = 1031.7 min
Number of Runs = 6250

Figure 4-4: Mission Time Normalized Histogram
Break Diameter: 0.0191 m
Mission Time Range = 325.9 min.
Mission Time Mean = 460.9 min
Number of Runs = 6250

Figure 4-5: Mission Time Normalized Histogram
Break Diameter: 0.0254 m
Mission Time Range = 193.5 min.
Mission Time Mean = 261.1 min
Number of Runs = 6250
The probability distribution function of Mission Time values for a break diameter of 0.00635m (Figure 4-2) is roughly uniform until about 1330 minutes. After this, there is a sharp increase in the frequency of the Mission Time values. This is because the uniform values constitute the tail of the distribution. It is likely that the majority of the distribution values are above one day. However, more simulation runs for longer time periods than one day are necessary to make conclusions. A majority
of the simulation runs (5408 out of 6250) were discarded because the core was not uncovered from the coolant within one day from scram.

The distributions of Mission Time values for a break size diameters of 0.0127m to 0.0508m (Figures 4-3 to 4-9) are skewed to the left. The skew is a result of the propagation of the Low Pressurizer Pressure Trip Point (PPSL) uncertainty, which was modeled with a lognormal distribution. The higher the PPSL value is, the higher the pressure will be when the reactor scrams. This means that the coolant in the reactor will take longer to saturate because it is farther from the saturation pressure. Since Mission Time is calculated from the time at which the coolant is saturated to the time at which the core is uncovered from the coolant, if the saturation of the coolant is delayed then the total Mission Time will be shorter. Therefore, the skew to the right (higher pressures) of the lognormal PPSL uncertainty distribution is seen in the skew to the left (lower Mission Time values) in the distributions of the Mission Time values.

Figure 4-10 shows the dependence of Mission Time upon break diameter. The solid line shows the average of the runs and the shaded region represents the 90% interval of the runs. The 90% interval is reported so that the tails of the distribution, as well as any outliers, would be excluded from the interval. This allows the interval shown to more effectively describe the extend of the effect the TH uncertainties have upon the uncertainty in Mission Time values.
The Mission Time decreases as the break size increases. In addition, the range of Mission Times decreases as the break size increases. This is because as the break size increases, the speed of the transient increases. There is less time for the full effect of the uncertain parameters to be realized in the transient.

I calculated the probability of non-reconfiguration utilizing the three methods described in the beginning of the section. The dependences of the PNR values upon break diameter are shown in Figure 4-11.
Each method resulted in the same PNR value of 0.01 until a break diameter of 0.0318m. As the break size increases, the PNR values were not the same for each of the three calculation methods. The second method (PNR with Uncertainties (Averaged)) appears to overestimate the PNR value and the first method (PNR without Uncertainties) seems to underestimate the PNR. The first and second methods are restricted to a PNR value of 0.01 and 0.1 because only one Mission Time value is utilized in the calculation of the PNR value in these methods. The PNR is restricted to the values in Equation 3.3. However, since all of the calculated Mission Time values are used in method 3, the PNR increases from 0.01 (all Mission Time values are above 90 minutes) as soon as there is at least one Mission Time value below 90 minutes, which occurs at a break size diameter of 0.0381m.

### 4.1.2 Mode B

Mode B is the operational state of the reactor during the first 6.5 hours after shutting down the reactor for a planned outage. Figure 4-12 depicts the Mission Time histogram obtained from a total of 38,097 runs in which the core was uncovered by the coolant within one day (1440 minutes) from the initiating event (SBLOCA). The
calculated Mission Time values range from about 30 minutes to one day.

![Mission Time Histogram](image)

The distribution in Figure 4-12 is also not unimodal. It appears to have two local extrema. Taking a similar approach to Mode A, I examined the Mission Time histograms for each of the eight break sizes: 0.00635m, 0.0127m, 0.0191m, 0.254m, 0.0318m, 0.0381m, 0.0445m, and 0.0508m. These are shown in Figures 4-13 to 4-20, respectively. The Mission Time range and Mission Time mean values are in the caption below each figure.
Figure 4-13: Mission Time Normalized Histogram
Break Diameter: 0.00635 m
Range = 1215.1 min.
Mission Time Mean = 240.8 min
Number of Runs = 2723

Figure 4-14: Mission Time Normalized Histogram
Break Diameter: 0.0127 m
Range = 671.7 min.
Mission Time Mean = 1199.9 min
Number of Runs = 862

Figure 4-15: Mission Time Normalized Histogram
Break Diameter: 0.0191 m
Range = 1112.8 min.
Mission Time Mean = 860.6 min
Number of Runs = 3919

Figure 4-16: Mission Time Normalized Histogram
Break Diameter: 0.0254 m
Range = 1250.8 min.
Mission Time Mean = 661.6 min
Number of Runs = 5604
Break diameter 0.00635 (Figure 4-13) shows a trapezoidal distribution function. There is volatility from Mission time values of 1150 minutes to 1250 minutes. Break diameter 0.0127 (Figure 4-14) shows a distribution function with a positive slope. The positive slope is likely depicting the tail of actual Mission Time distribution. More simulation runs for longer time periods than one day are necessary to make
conclusions. A majority of the simulation runs (5388 out of 6250) were discarded because the core was not uncovered from the coolant within one day from scram.

The remaining break diameters (Figures 4-15 to 4-20) show a distribution function with a negative slope. This is a result of the propagation of the uncertainty in the time at which the transient is initiated. The uncertainty distribution of the time at which the transient is initiated is uniformly distributed from the time of reactor shutdown until 6.5 hours after the time of the reactor shutdown. Throughout this time range, there is an exponential decrease in core thermal power due to decay heat. This is estimated utilizing Equation 4.3 [21],

\[
\frac{Q}{Q_0} = 0.066t^{-0.2},
\]  

where \(Q\) is the core thermal power, \(Q_0\) is the core thermal power at operation, and \(t\) is the time since reactor shutdown. This is the exponential decrease that is seen in the Mission Time value distribution function. At the higher power, the transient moves more quickly and so the Mission Time value is smaller.

Like Mode A, the Mission Time values in Mode B are dependent on break size. Even considering the uncertainty in Mission Time, the slope has an overall negative value. Figure 4-21 shows the dependence on Mission Time on the break diameter. The solid line shows the average of the runs and the shaded region represents the 90% interval of the runs. The 90% interval is reported so that the tails of the distribution, as well as any outliers, would be excluded from the interval. This allows the interval shown to more effectively describe the extend of the effect the TH uncertainties have upon the uncertainty in Mission Time values.
As with Mode A, the Mission Time decreases with increasing break size. However, unlike Mode A, the range of the Mission Times increases until the 0.0254m diameter break size and then decreases. The initial increase in the Mission Time range is due to the simulation cut off time of one day. Most of the Mission Times will be larger than one day for the smaller break sizes. The Mission Time range reported here only extends to one day and therefore excludes the upper part of the range. If the entire range were to be reported, then the Mission Time range likely would decrease with increasing break diameter, which is the same behavior as Case A.

I calculated the probability of non-reconfiguration utilizing the three methods described in the beginning of the section. The PNR values a functions of break diameter are shown in Figure 4-22.
The PNR without Uncertainties method and the PNR with Uncertainties (Averaged) method both resulted in a uniform probability of non-reconfiguration of 0.01 across the break diameters. This indicates that both of these methods used a Mission Time of over 90 minutes (from Equation 3.3). Therefore, the conclusion from both of these methods is that the operator is unlikely to fail. However, the PNR with Uncertainties (Integrated) method shows a slight increase in the probability of non-reconfiguration for break diameters above 0.0381m. This is because the Mission Times below 90 minutes that are the result of the uncertainty propagation are considered in the calculation of the PNR values.

### 4.1.3 Mode C

Mode C is operational state of a reactor during the time from 6.5 to 13.5 hours after shutting down the reactor for a planned outage. Figure 4-12 depicts the Mission Time histogram obtained from a total of 12,547 runs in which the core was uncovered by the coolant within one day (1440 minutes) from the initiating event (SBLOCA). The calculated Mission Time values range from about 180 minutes to one day.
The distribution in Figure 4-23 is also unimodal. It appears to have two local extrema. Each local extremum is associated with a reactor temperature condition (see Figure 3-1 for the temperature profile during shutdown). The local extremum at the upper Mission Time results from the two hours in Mode C that the reactor primary loop is at 170°C. The local extremum at the lower Mission Time results from the one hour that the reactor primary loop is at 180°C. The higher temperature causes the transient speed to increase.

Taking a similar approach as with Mode A, I examined the Mission Time histograms for six of the eight break sizes: 0.0191m, 0.254m, 0.0318m, 0.0445m, and 0.0508m. These are shown in Figures 4-13 to 4-20, respectively. The Mission Time range and Mission Time mean values are in the caption below each figure. The results from the smallest two break diameters were not plotted because there were no simulation runs in which the core was uncovered from coolant within one day from the initiating event. Therefore, no Mission Time values could be calculated.
Figure 4-24: Mission Time
Normalized Histogram
Break Diameter: 0.0191 m
Mission Time Range = 405.4 min.
Mission Time Mean = 1258.5 min
Number of Runs = 121

Figure 4-25: Mission Time
Normalized Histogram
Break Diameter: 0.0254 m
Mission Time Range = 816.7 min.
Mission Time Mean = 1107.4 min
Number of Runs = 931

Figure 4-26: Mission Time
Normalized Histogram
Break Diameter: 0.0318 m
Mission Time Range = 1037.6 min.
Mission Time Mean = 957.3 min
Number of Runs = 1836

Figure 4-27: Mission Time
Normalized Histogram
Break Diameter: 0.0381 m
Mission Time Range = 1160.6 min.
Mission Time Mean = 817.1 min
Number of Runs = 2983
Break diameter 0.0191 (Figure 4-24) shows a sporadic distribution. This is due to the lower amount of simulation runs in which the core uncovered from coolant within one day (less than one percent of total simulation runs). More simulation runs are needed to refine this region of the distribution. The need for more simulation runs is verified with the variance analysis (see Appendix C, Figure C-17). The variance has not yet converged.

The remaining break diameters (Figures 4-25 to 4-29) show bimodal distributions. Each local extremum is associated with a reactor temperature condition (see Figure 3-1 for the temperature profile during shutdown). The local extremum at the upper Mission Time is from the two hours in Mode C that the reactor primary loop is at 170°C. The local extremum at the lower Mission Time is from the one hour that the reactor primary loop is at 180°C. The higher temperature causes the transient speed to increase.

As with Modes A and B, the Mission Time is strongly dependent upon the input of break size. Figure 4-21 shows the dependence of Mission Time upon the break diameter. The shows the mean Mission Time value of the runs and the shaded region represents the 90% interval of the runs. The 90% interval is reported so that the tails of the distribution, as well as any outliers, would be excluded from the interval. This
allows the interval shown to more effectively describe the extend of the effect the TH uncertainties have upon the uncertainty in Mission Time values.

![Figure 4-30: Mission Time as a Function of Break Size (Mode C)](image)

As with Modes A and B, the Mission Time decreases with increasing break size. The range of the Mission Times increases slightly as break size increases. This is because the transients with a larger break size have an increased speed. This means that more of the simulation runs at a lower pressure will have the core uncover form the coolant within one day. Less simulation runs were excluded and so the true range is being captured.

I calculated the probability of non-reconfiguration utilizing the three methods described in the beginning of the section. The PNR values as functions of break diameter are shown in Figure 4-22.
Each of the three methods determined the probability of non-reconfiguration to be equal to 0.01 across the break diameters examined. This is because, as can be seen in Figure 4-23, there are no Mission Time values below 90 minutes (as per Equation 3.3). This means the operator is unlikely to fail to manually initiate safety coolant injection when the reactor is operating in Mode C.

### 4.2 Case 2

In Case 2, the operating crew tests for saturation of the primary system coolant before the saturated primary system state can occur during a SBLOCA. I assumed that this will occur if the Saturation Time (Equation 3.1) is above 3 minutes because I assume the operator will likely perform the saturation test within the first three minutes following scram [5] [22]. The mission is to realize the need to retest for the saturation of the primary system. The value of the probability of failing to realize this need is found as the probability of non-reconfiguration value ($P_{NR|CICA}$ or PNR). The PNR value is a function of Mission Time (Equations 3.2 and 3.4).

I present the results for Modes B and C. There are no simulation runs where Saturation Time is greater than 3 minutes in Mode A. Thus, this state is not plausible.
for Mode A. I first present a histogram of the Saturation Time values for all runs which has a Saturation Time less than one day (1440 minutes). Then, for those runs that had a Saturation Time of greater than 3 minutes, I present a histogram of the Mission Times. For both Modes B and C, there are no Mission Times less than 50 minutes. So, the value of the probability of non-reconfiguration is obtained as 0.01 for all break sizes as per Equation 3.4. Therefore, I concluded that there was no need to perform an independent analysis of each break size.

A Monte Carlo variance analysis was performed for the Saturation Time distribution for Modes B and C. This was performed in order to verify that adequate runs were performed and that undersampling did not occur. I performed this analysis by calculating the variance of the distribution of Saturation Time as a new Saturation Time value from each simulation run was added. As my modeled distribution of Saturation Time values approaches the real distribution, then the change in the variance value of the Saturation Time distributions approaches zero. I show the plots of the percent change in the Saturation Time distribution variance values as function of the simulation runs for Modes B and C in Appendix C.

4.2.1 Mode B

Figure 4-32 shows a histogram of the Saturation Time results for the N = 38,097 simulation runs. If the Saturation Time is greater than 3 minutes, I assume that the operator will not observe the saturation conditions.
The Saturation Time histogram has three visible local extrema. Each local extremum is a result of the different break sizes. The local extremum for the smallest Saturation Time is that of the largest break size. The amplitudes of the following local extrema are lower because the range is larger for the smaller break sizes and so the distributions have a larger spread. The histogram is normalized for each break size.

A total of 29,507 runs out of the 38,097 simulation runs had a Saturation Time above 3 minutes. A histogram of the Mission Times for these runs is in Figure 4-33.
There are no Mission Times below 50 minutes, and so the probability of non-reconfiguration is 0.01 as per Equation 3.4. The slope of the Mission Time distribution function is negative due to the uncertainty propagation of the time at which the transient is initiated. This uncertainty distribution is uniformly distributed from the time of reactor shutdown until 6.5 hours. For this time, there is an exponential decrease in core thermal power due to decay heat. This is the exponential decrease that is seen in the Mission Time value distribution function. At the higher power, the transient moves more quickly and so the Mission Time values are smaller.

4.2.2 Mode C

Figure 4-34 is a histogram of the $N = 12,547$ simulation runs for Saturation Time. If the Saturation Time is greater than 3 minutes, it is assumed that the operator will not observe the saturation in the initial saturation test. This is because the initial saturation test occurs in the first three minutes following the initiating event.
The slope of the distribution function of Saturation Time values is negative. This is due to the uncertainty propagation of the time at which the transient is initiated. This uncertainty distribution is uniformly distributed from 6.5 hours after reactor shutdown to 13.5 hours after shutdown. For this time, the temperature is held at 180°C for an hour. It is then decreased to 170°C for two hours. After this, there is a linear decrease in the temperature for the remaining 4 hours (see Figure 3-1 for the temperature profile during shutdown). The higher the temperature, the more quickly the coolant will reach saturation because the saturation pressure is higher for higher temperatures. The distribution function’s shape here reflects the time dependence of the temperature during shutdown. The distribution function is initially constant when temperature is constant. Then, the Saturation Time distribution function’s negative slope corresponds to the times in which there is a decreasing temperature.

A total of 11,326 runs out of the 12,547 simulation runs had a Saturation Time above 3 minutes. A histogram of the Mission Times for these runs is in Figure 4-35.
There are no Mission Times below 50 minutes, and so the probability of non-reconfiguration is 0.01 as per Equation 3.4. The distribution of Mission Time values has two local extrema. Each local extremum is associated with a reactor temperature condition (see Figure 3-1 for the temperature profile during shutdown). The local extremum at the upper Mission Time is from the two hours in Mode C that the reactor primary loop is at 170°C. The local extremum at the lower Mission Time is from the one hour that the reactor primary loop is at 180°C. The higher temperature causes the transient speed to increase.
Chapter 5

Discussion and Conclusion

In this Chapter, I discuss the results presented in the previous Chapter for both Case 1 and 2. I then compare the behavior of the failure probability values of these results to the failure probability values presented in the MERMOS Catalogue. I show that my calculated results and the results from the MERMOS Catalogue vary from each other by at least an order of magnitude. This demonstrates that the operator failure probability values can be sensitive to uncertainties in TH conditions. This is seen through the range of Mission Times. Operator failure is directly dependent upon the Mission Time. A consequence of this dependence is that a larger range in Mission Time values means that there will be a larger range in operator failure probabilities. The operator failure probability is more uncertain. This indicates a greater need to take TH uncertainties into consideration when performing an HRA.

The effects of the uncertainties upon the Mission Time values are most strongly determined by the break diameter. The second strongest effect from the TH uncertainties is the time from the shutdown in Modes B and C. This is because the primary system temperature and pressure as well as the core thermal power are functions of the time from shutdown. The other TH parameters, low pressurizer pressure trip point, break discharge coefficient and void fraction threshold, had a smaller effect on the Mission Time uncertainty.
5.1 Case 1

In Case 1, the operating crew is distracted during a SBLOCA by a spurious fire alarm. The mission of the crew is to recognize the error, and manually to initiate the safety coolant injection. The TH analysis of Case 1, presented in the Results Chapter, reveals range values for the Mission Times for eight break sizes. The range values are summarized in Table 5.1.

<table>
<thead>
<tr>
<th>Break Diameter</th>
<th>Mission Time Range (max - min)</th>
<th>Mode A</th>
<th>Mode B</th>
<th>Mode C</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.00635m</td>
<td>28.2 min.</td>
<td></td>
<td>240.8 min.</td>
<td>n/a^a</td>
</tr>
<tr>
<td>0.0127m</td>
<td>752.6 min.</td>
<td>671.7 min.</td>
<td>n/a^a</td>
<td></td>
</tr>
<tr>
<td>0.0191m</td>
<td>325.9 min.</td>
<td>1112.8 min.</td>
<td>405.4 min.</td>
<td></td>
</tr>
<tr>
<td>0.0254m</td>
<td>193.5 min.</td>
<td>1250.8 min.</td>
<td>816.7 min.</td>
<td></td>
</tr>
<tr>
<td>0.0318m</td>
<td>121.2 min.</td>
<td>1322.5 min.</td>
<td>1037.6 min.</td>
<td></td>
</tr>
<tr>
<td>0.0381m</td>
<td>84.2 min.</td>
<td>1074.5 min.</td>
<td>1160.6 min.</td>
<td></td>
</tr>
<tr>
<td>0.0445m</td>
<td>60.7 min.</td>
<td>767.2 min.</td>
<td>1233.2 min.</td>
<td></td>
</tr>
<tr>
<td>0.0508m</td>
<td>46.9 min.</td>
<td>613.8 min.</td>
<td>1254.5 min.</td>
<td></td>
</tr>
</tbody>
</table>

^aThere were no Mission Times less than one day

Table 5.1: Mission Time Range for each Break Size

Mode A: In Operation
Mode B: Shutdown (0 to 6.5 hours after scram)
Mode C: Shutdown (6.5 to 13.5 hours after scram)

For Mode A, the Mission Time range decreases as the break size increases. The only exception to this is the smallest break diameter, 0.00635m. This exception is likely because the simulation ends at one day, and therefore there are no Mission Times beyond one day. The Mission Times for this break diameter are close to one day (the average is 1215.1 minutes). The range listed of 28.2 minutes does not include the actual range that extends beyond one day. The trend of decreasing Mission Time range with increasing break diameter indicates that the TH uncertainties affect the operator failure uncertainty more for smaller break sizes.
For Mode B, the range is greater than four hours (240 minutes) for all eight break diameters simulated. The Mission Time range increases as the break diameter increases to 0.0318m and then decreases as the break diameter increases to 0.0508m. This means that in an SBLOCA, a break size diameter of 0.0318m has the largest uncertainty in the value of the probability of the operator crew failing to retest for saturation.

For Mode C, the range increases as the break size increases. This is the opposite trend from Mode A. This means that the larger the break size, the more of an effect that the TH uncertainties will have on operator failure uncertainty.

The TH analysis of Case 1 reveals different Mission Times than those utilized by the MERMOS Catalogue. This is because I simulated the uncertain TH parameters to determine an uncertainty distribution of Mission Time values. The MERMOS Catalogue has a conservatively estimated Mission Time value of 24 minutes\(^1\) for Mode A, 21 minutes\(^2\) for Mode B, and 61 minutes for Mode C [5] [22]. Table 5.2 shows how much the failure probability values of the entire scenario failure are reduced in the first case study. The scenario failure probability is a function of the non-reconfiguration probability value (see Equation 2.2). As the probability of non-reconfiguration (the failure probability of the operator to realize his error) decreases, then the scenario failure probability also decreases.

In determining the scenario failure probability, I utilized the other scenario probabilities (CICA and PS values) probabilities listed in the MERMOS Catalogue [5]. If the effects of the TH uncertainties upon the other scenario probabilities are examined also (such as those of CICA or PS), then this result could change. If the CICA and PS probability values were to be reduced as a result of the TH uncertainties, then the scenario failure probability value would decrease. If the CICA and PS probability values were to be increased as a result of the TH uncertainties, then the scenario failure probability value would increase.

\(^1\)This value was determined from a Medium Break Loss of Coolant Accident analysis
\(^2\)This value was determined from a Medium Break Loss of Coolant Accident analysis
<table>
<thead>
<tr>
<th></th>
<th>Mode A</th>
<th>Mode B</th>
<th>Mode C</th>
</tr>
</thead>
<tbody>
<tr>
<td>MERMOS Catalog Scenario</td>
<td>0.22</td>
<td>0.22</td>
<td>0.012</td>
</tr>
<tr>
<td>Failure Probability</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Scenario Failure Probability</td>
<td>0.0052</td>
<td>0.0025</td>
<td>0.0022</td>
</tr>
<tr>
<td>Considering the Uncertainties</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Reduction</td>
<td>97.6%</td>
<td>98.9%</td>
<td>81.7%</td>
</tr>
</tbody>
</table>

Table 5.2: Summary of Failure Probabilities for Case 1

Mode A: In Operation
Mode B: Shutdown (0 to 6.5 hours after scram)
Mode C: Shutdown (6.5 to 13.5 hours after scram)

The highest reduction in the scenario failure probability value occurs in Mode B (Shutdown 0 to 6.5 hours after scram). The scenario failure probability value in Mode B is reduced by 98.9% (two orders of magnitude). The second highest reduction occurs in Mode A where the scenario failure probability value is reduced by 97.6% (nearly two orders of magnitude). These two reductions show the difference in having a PNR value (failure of the operator to realize that entering fire procedures is an error) equal to 1, which is very conservative, and a PNR value of 0.01. I am able to reduce the PNR value to almost 0.01 through demonstrating that the Mission Time is likely to be much larger than suggested by experts and initial models. The value of 0.01 comes from Equation 3.3. This is the lowest failure probability value. The scenario failure probability in Mode C is reduced by 81.7% (one order of magnitude).

5.2 Case 2

In Case 2, the initiating event in the nuclear unit is a SBLOCA, which causes an alarm indicating low pressure in the pressurizer. The safety coolant injection system fails to automatically actuate. The operating crew performs the test for saturation of the coolant in the primary system. However, if this test is performed before the saturation of the coolant occurs, then the crew will not realize that the initiating event is a SBLOCA. In this case, the crew’s mission is to recognize that the saturation test
was performed too early, to retest for saturation, and finally to manually initiate safety coolant injection before core damage occurs.

It is assumed that the saturation test will occur in the first three minutes following scram. Therefore, if the primary system is not saturated until after three minutes, the operators will test too early. If this occurs, they will continue through procedures to determine why the reactor scrammed. When they do not diagnose the problem correctly, they will return to test for saturation of the coolant again. This will likely occur about one hour after the first coolant saturation test.

The results for Case 2 are presented in the Results Chapter. For Mode A (reactor in operation), there are no simulation runs where the saturation of the primary system occurs after three minutes. Therefore, since the saturation of the primary system always occurs before the test for saturation, the Case 2 outcome is assumed to be impossible for Mode A.

For Modes B and C, there are simulation runs where the coolant saturation of the primary system occurs after three minutes. The Mission Times for these simulation runs are examined, and there are no Mission Times values under the 50 minute cut-off for the probability of non-reconfiguration (the operating crew failing to retest for saturation). Therefore, the failure probability is set at 0.01, which is the lowest value possible using the MERMOS method. There is no effect from the TH uncertainties because the range of Mission Times does not include any times below this 50 minute cut-off.

However, even though there is no effect on the PNR value from the TH uncertainties (there was an effect on the Mission Time Values), there is an important conclusion that can be drawn from the simulations of Case 2. The MERMOS Catalog reports that the scenario failure probability value for Mode C (shutdown 6.5 to 13.5 hours after scram) is 0.00045, which is the same as my calculated value (found utilizing Equation 2.2 and values for the other scenario parameters from the MERMOS Catalogue [5]). However, there was no value reported in the MERMOS catalog for scenario failure probability for Mode B (shutdown 0 to 6.5 hours after scram) because the analysts assumed the scenario failure probability to be negligible for Mode
B. I found the scenario failure probability for Mode B to be 0.0033. Not only is this not a negligible value, but it is larger than the scenario failure probability for Mode C. This demonstrates that utilizing advanced TH simulations to model uncertainties is notably important for examining the effects of the uncertainties. It also provides a stronger foundation for determination of the scenario failure probabilities.

5.3 Usefulness of This Modeling Treatment

As stated in the beginning of this report, the purpose of my work is not to generate a new methodology to analyze HRA. Instead, it is to examine existing HRA methods and to suggest a treatment by which to improve estimates of failure probability through use of advanced thermal hydraulic analysis. It successfully demonstrated that this can be achieved.

In addition, the work demonstrates a relatively efficient method of using TH results to gain accuracy for the discussed cases. This is a method to increase the accuracy of HRA results (and possibly to reduce failure probability values) that were generated prior to computer clusters becoming available for executing simulation runs. However, there is a limitation due to required computer time. The simulations were run in parallel using a computing cluster, but the Monte Carlo analysis still took 10-30 hours to complete. If it were necessary to use more than eight uncertainty distributions, then more simulation runs would be needed and even more computational time would be needed. If more computational time was not available and there were more than eight uncertainties identified, then only eight of the uncertainties could be modeled with an uncertainty distribution and the others would need to be estimated with one parameter value.

5.4 Conclusions and Future Work

I have demonstrated that TH uncertainties in nuclear power units can have an effect upon operator failure probability values. The strengths of the effects of the uncertain-
ties depend upon the characteristics of the transient (such as break diameter), as well as the operating mode of the reactor (in operation, shutting down, etc.). It is important to determine whether TH uncertainties will affect scenario failure probabilities in an HRA analysis. One method of determining the magnitude of the effects is through coupling uncertainty analysis (such as using codes like Dakota) with advanced TH simulations (such as with MAAP).

If there are strong effects of the TH uncertainties upon the failure probabilities of the operator actions, then it will be important to determine the sensitivities of failure probabilities upon the TH uncertainties. While the method that I used in this work is able to calculate failure probabilities, it requires access to a computing cluster and adequate computing time. More work needs to be done to determine how to propagate uncertainty more quickly through an advanced TH simulation.

5.4.1 Effect of TH on Other Failure Probability Values

I studied the effect of TH on the probability of non-reconfiguration values. This is the probability that an operating crew will fail to realize and remedy an error. However, there are many other probability values that have an effect on the overall scenario failure probability (see Equation 2.2). In order to see the true effect of the TH uncertainties on the overall scenario failure probability, the effect of the TH uncertainties on the other scenario characteristic probability values must be examined. One example of this is the appearance of the shift supervisor in the control room. Whether the shift supervisor is present or not will effect other failure probabilities. MERMOS will allow a probability value of the shift supervisor being in the control room of 1, 0.5, and 0.1, depending on the Mission Time. There are many other scenario probability values that may be affected by TH uncertainties. More work can be done to refine these other scenario probability values based upon uncertain TH inputs.
5.4.2 Effect of Other Uncertain TH Parameters

I studied the effect of eight different TH uncertain parameters: Low Pressurizer Pressure Trip Point, Break Discharge Coefficient, Void Fraction Threshold, Initial Core Thermal Power, Break Size Diameter, Initial Primary Side Temperature, Initial Primary Side Pressure, and time since shutdown is initiated. However, there are many other TH parameters that will play a role in affecting an operating crew’s failure probability values. More work can be done in finding other TH uncertainties that will have an effect on the Mission Time values and the failure probabilities of the operating crew.

5.4.3 Uncertainty Propagation

I propagated uncertainties in TH parameters through a transient to determine the uncertainty in a Mission Time value as well as a probability of non-reconfiguration of the operating crew. I utilized a Monte Carlo method to propagate the uncertainties. However, since the simulation was performed on advanced thermal hydraulic software (MAAP), the total time to run the Monte Carlo scheme varied from 10 to 30 hours. Many methods of uncertainty propagation exist, such as performing sensitivity studies using correlation and variance-based indices as well as using surrogates as approximations to computer models. For more information on current uncertainty quantification methods, as well as an overview of research into uncertainty quantification methods, see "Simulation Credibility Advances in Verification, Validation, and Uncertainty Quantification" [23]. More work can be done in order to decrease the computational time needed to perform the propagation of TH uncertainties to operating crew failure probabilities.
### Appendix A

#### Uncertain Parameter Selection

<table>
<thead>
<tr>
<th>MAAP Variable</th>
<th>Reasoning</th>
</tr>
</thead>
<tbody>
<tr>
<td>PPSL: Low Pressurizer Pressure Trip Point</td>
<td>The mean was selected as the default MAAP4 value of 12.5 MPa. Standard deviation was selected as 0.1 MPa to give measurement error. [16]</td>
</tr>
<tr>
<td>FCDBRK: Break Discharge Coefficient</td>
<td>This captures the uncertainty in the shape of the break. The distribution was selected from the MAAP4 Application Guidance with the mode at the default MAAP value of 0.75. [16] [24]</td>
</tr>
<tr>
<td>VFSEP: Void Fraction Threshold</td>
<td>This is the void fraction above which the inventory in the primary system is no longer a homogenous two-phase mixture. Once the void fraction in the primary system exceeds this value, then the two-phase fluid separates into a relatively stagnant (gas over liquid) configuration. Distribution was selected from the MAAP4 Application Guidance with the mode at the default of 0.5. [16] [24]</td>
</tr>
<tr>
<td>QCR0: Initial Core Thermal Power</td>
<td>This was assumed to have a general load-following pattern. Rather than having a &quot;ramping up&quot; and &quot;ramping down&quot; period, I assumed that the reactor would be at full power (MAAP default of 3235 MW thermal) 70% of the time and at 70% full power at 30% of the time. [16]</td>
</tr>
</tbody>
</table>
In running the scenarios, I selected a parametric approach in which I modeled breaks from 0.00635m diameter to 0.0508m diameter at 0.00635m intervals. In integrating over the break diameters to get a resulting scenario failure probability, I used a probability distribution of break sizes from Fleming and Lydell (2013). [19]

I developed a linear function of time for the average for the temperature of the primary side as the reactor is shutting down (Modes B and C). To measure uncertainty in the shutdown path (as well as to capture the fact that the real path is not linear) we used an appropriate standard deviation depending on how much room there is for error in the shutdown profile from Coppolani et al. (2004). [20]

I developed a linear function of time for the average for the pressure of the primary side as the reactor is shutting down (Modes B and C). To measure uncertainty in the shutdown path (as well as to capture the fact that the real path is not linear) we used an appropriate standard deviation depending on how much room there is for error in the shutdown profile from Coppolani et al. (2004). [20]

I used a uniform distribution, assuming that a break is equally likely in any time after scram occurs.

Table A.1: Reasoning for Parameter Selection
Appendix B

Coding Scripts

B.1 Mode A

B.1.1 Dakota

```plaintext
# This is a basic DAKOTA input file
# It will be used in conjunction with MAAP
# It will be used to establish risk uncertainty

# Run dakota -i TSAT.in -o TSAT.out

environment
    tabular_graphics_data
        tabular_graphics_file = 'uncertainty.dat'

method,
    sampling
        samples = {MCruns}
        sample_type lhs

variables,
    lognormal_uncertain 1
        means 12500000
        std_deviations 100000
        descriptors 'PPSL'

    triangular_uncertain 2
```

75
modes          0.75     0.5 
lower_bounds   0.6      0.2 
upper_bounds   1.0      0.8 
descriptors    'FCDBRK' 'VFSEP'

histogram_point_uncertain
  integer        = 2
  pairs_per_variable = 2 8
  abscissas      = 2265 3236 1 2 3 4 5 6 7 8
  counts         = 75 165 1 1 1 1 1 1 1
  descriptors    = 'QCR0' 'ABB'

interface,      
  fork
  parameters_file 'param'
  results_file 'result'
  asynchronous evaluation_concurrency = {processors}
  analysis_drivers 'shell'
  file_tag file_save

responses,      
  response_functions = 2
  no_gradients
  no_hessians
B.1.2 MAAP4

DEBUG OFF
SENSITIVITY ON

TITLE
   maap_loca_test
END

PARAMETER FILE IS 4LOOP-PWR.par

ALIAS
   TIMER 1 AS CORE_UNCOVERED
   TIMER 2 AS FIRE_ALARM
   TIMER 3 AS SCRAM_TIME
   TIMER 4 AS CLAD_FAIL
   TIMER 5 AS LOW_PRESS
   TIMER 6 AS SATUR_TIME
END

FUNCTION
   HUM = MBRKPS/1520 // Humidity in the Containment
   STARTTIME = {time}
END

PARAMETER CHANGE
   ABB = (.000506707 * {breaksize} ^ 2) M**2
   FBB = 7
   ZBB = 8.0 M**2
   FCDBRK = {brkcoef}
   VFSEP = {vdfrac}
   PPSL = {ppsl}
   QCR0 = ({power} * 1000000) W
END

START TIME IS 0 S
END TIME IS 86400 S
PRINT INTERVAL IS 600 S

FUNCTION
   TCLADMAX=max(TCLN(1),TCLN(2),TCLN(3),TCLN(4),TCLN(5),TCLN(6),TCLN(7))
   TCLADMAX=max(TCLADMAX,TCLN(8),TCLN(9),TCLN(10),TCLN(11),TCLN(12))
   TCLADMAX=max(TCLADMAX,TCLN(13),TCLN(14),TCLN(15),TCLN(16),TCLN(17))
   TCLADMAX=max(TCLADMAX,TCLN(13),TCLN(14),TCLN(15),TCLN(16),TCLN(17))

77
TCLADMAX = max(TCLADMAX, TCLN(18), TCLN(19), TCLN(20), TCLN(21), TCLN(22))
TCLADMAX = max(TCLADMAX, TCLN(23), TCLN(24), TCLN(25), TCLN(26), TCLN(27))
TCLADMAX = max(TCLADMAX, TCLN(28), TCLN(29), TCLN(30), TCLN(31), TCLN(32))
TCLADMAX = max(TCLADMAX, TCLN(33), TCLN(34), TCLN(35), TCLN(36), TCLN(37))
TCLADMAX = max(TCLADMAX, TCLN(38), TCLN(39), TCLN(40), TCLN(41), TCLN(42))
TCLADMAX = max(TCLADMAX, TCLN(43), TCLN(44), TCLN(45), TCLN(46), TCLN(47))
TCLADMAX = max(TCLADMAX, TCLN(48), TCLN(49), TCLN(50), TCLN(51), TCLN(52))
TCLADMAX = max(TCLADMAX, TCLN(53), TCLN(54), TCLN(55), TCLN(56), TCLN(57))
TCLADMAX = max(TCLADMAX, TCLN(58), TCLN(59), TCLN(60), TCLN(61), TCLN(62))
TCLADMAX = max(TCLADMAX, TCLN(63), TCLN(64), TCLN(65), TCLN(66), TCLN(67))
TCLADMAX = max(TCLADMAX, TCLN(68), TCLN(69), TCLN(70), TCLN(71), TCLN(72))
TCLADMAX = max(TCLADMAX, TCLN(73), TCLN(74), TCLN(75), TCLN(76), TCLN(77))
TCLADMAX = max(TCLADMAX, TCLN(78))
END

INITIATORS
IEVNT(209) = 1 // Loss of coolant accident
IEVNT(222) = 1 // Upper and lower compartment sprays are locked off
IEVNT(217) = 1 // LPI train 1 is locked off
IEVNT(254) = 1 // LPI train 2 is locked off
IEVNT(216) = 1 // HPI locked off
IEVNT(232) = 1 // Charging pumps locked off
IEVNT(257) = 1 // No scram when charging pumps off
END

C Timers:

WHEN HUM > 1
  SET FIRE_ALARM
END

WHEN IEVNT(13) IS TRUE
  SET SCRAM_TIME
END

WHEN ZWPZ < 2.38
  SET LOW_PRESS
END

WHEN IEVNT(20) IS TRUE
  SET SATUR_TIME
END

WHEN IEVNT(49) IS TRUE
  SET CORE_UNCOVERED
WHEN TCLADMAX > 1477.15
    SET CLAD_FAIL
END

WHEN CLAD_FAIL > 300
    TILAST = TIM
END
B.2 Mode B

B.2.1 Dakota

# This is a basic DAKOTA input file
# It will be used in conjunction with MAAP
# It will be used to establish risk uncertainty

# Run dakota -i TSAT.in -o TSAT.out

environment
    tabular_graphics_data
        tabular_graphics_file = 'stateb.dat'

method,
    sampling
        samples = {MCruns}
        sample_type lhs

variables,
    normal_uncertain 2
        means 0 0
        std_deviations 1 1
        lower_bounds -2 -2
        upper_bounds 2 2
        descriptors 'Z_P' 'Z_T'

    uniform_uncertain 1
        lower_bounds 0
        upper_bounds 330
        descriptors 'time_B'

    triangular_uncertain 2
        modes 0.75 0.5
        lower_bounds 0.6 0.2
        upper_bounds 1.0 0.8
        descriptors 'FCDBRK' 'VFSEP'

    histogram_point_uncertain
        integer = 1
        pairs_per_variable 8
        abscissas 1 2 3 4 5 6 7 8
        counts 1 1 1 1 1 1 1 1
        descriptors 'ABB'
interface,
  fork
    parameters_file 'param'
    results_file 'result'
    asynchronous evaluation_concurrency = {processors}
    analysis_drivers 'shell'
    file_tag file_save
responses,
  response_functions = 2
  no_gradients
  no_hessians
B.2.2 MAAP4

DEBUG OFF
SENSITIVITY ON

TITLE
STATE_B_SHUTDOWN
END

PARAMETER FILE IS 4LOOP-PWR.par

ALIAS
    TIMER 1 AS CORE_UNCOVERED
    TIMER 2 AS FIRE_ALARM
    TIMER 3 AS SCRAM_TIME
    TIMER 4 AS CLAD_FAIL
    TIMER 5 AS LOW_PRESS
    TIMER 6 AS SATUR_TIME
END

FUNCTION
HUM = MBRKPS/1520 // Humidity in the Containment
STARTTIME = {time}
END

PARAMETER CHANGE

C Decay Curve
FQINHF = 1 // Manual Decay Heat Entry
NFQHF = 8 // Eight Points
TIFQHF(1) = 0 // Time Steps
TIFQHF(2) = 1800.0
TIFQHF(3) = 5400.0
TIFQHF(4) = 9000.0
TIFQHF(5) = 12600.0
TIFQHF(6) = 19600.0
TIFQHF(7) = 41400.0
TIFQHF(8) = 138600.0
FQHF(1) = 1.0 // Power Fractions
FQHF(2) = {F2}
FQHF(3) = {F3}
FQHF(4) = {F4}
FQHF(5) = {F5}
FQHF(6) = \{F6\}
FQHF(7) = \{F7\}
FQHF(8) = \{F8\}

C Primary System
QCR0 = \{power\}  // Initial Core Thermal Power: State b1
ZWPZ0 = \{pzrlevel\}  // Initial Pressurizer Level: State b1
TWPSNM = \{temperature\} K  // Initial Temperature: State b1
TWPS0 = TWPSNM  // Initial Temperature
PPSNOM = \{pressure\} PA  // Initial Pressure: State b1
PPS0 = PPSNOM  // Initial Pressure
PPZHT0 = PPSNOM  // Pressurizer Set Point
FCDBRK = \{brkcoef\}
VFSEP = \{vdfrac\}

C Secondary System
MWS0 = 120.2 \times \{density\} \times 0.4285  // Initial Mass in Steam Generator
PSG0 = PSAT(TWPS0 - 2)  // Initial Pressure in Steam Generator
ZWSG0 = 13.2  // Initial Level in Steam Generator
PSGRV = PSG0

C Break Information
ABB = (.000506707 \times \{breaksize\}^2) M**2
FBB = 7
ZBB = 8.0

END

START TIME IS 0.0
END TIME IS 86400.0
PRINT INTERVAL IS 600

FUNCTION
TCLADMAX = \max(TCLN(1), TCLN(2), TCLN(3), TCLN(4), TCLN(5), TCLN(6), TCLN(7))
TCLADMAX = \max(TCLADMAX, TCLN(8), TCLN(9), TCLN(10), TCLN(11), TCLN(12))
TCLADMAX = \max(TCLADMAX, TCLN(13), TCLN(14), TCLN(15), TCLN(16), TCLN(17))
TCLADMAX = \max(TCLADMAX, TCLN(13), TCLN(14), TCLN(15), TCLN(16), TCLN(17))
TCLADMAX = \max(TCLADMAX, TCLN(18), TCLN(19), TCLN(20), TCLN(21), TCLN(22))
TCLADMAX = \max(TCLADMAX, TCLN(23), TCLN(24), TCLN(25), TCLN(26), TCLN(27))
TCLADMAX = \max(TCLADMAX, TCLN(28), TCLN(29), TCLN(30), TCLN(31), TCLN(32))
TCLADMAX = \max(TCLADMAX, TCLN(33), TCLN(34), TCLN(35), TCLN(36), TCLN(37))
TCLADMAX = \max(TCLADMAX, TCLN(38), TCLN(39), TCLN(40), TCLN(41), TCLN(42))

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TCLADMAX=\text{max}(TCLADMAX, TCLN(43), TCLN(44), TCLN(45), TCLN(46), TCLN(47))
TCLADMAX=\text{max}(TCLADMAX, TCLN(48), TCLN(49), TCLN(50), TCLN(51), TCLN(52))
TCLADMAX=\text{max}(TCLADMAX, TCLN(53), TCLN(54), TCLN(55), TCLN(56), TCLN(57))
TCLADMAX=\text{max}(TCLADMAX, TCLN(58), TCLN(59), TCLN(60), TCLN(61), TCLN(62))
TCLADMAX=\text{max}(TCLADMAX, TCLN(63), TCLN(64), TCLN(65), TCLN(66), TCLN(67))
TCLADMAX=\text{max}(TCLADMAX, TCLN(68), TCLN(69), TCLN(70), TCLN(71), TCLN(72))
TCLADMAX=\text{max}(TCLADMAX, TCLN(73), TCLN(74), TCLN(75), TCLN(76), TCLN(77))
TCLADMAX=\text{max}(TCLADMAX, TCLN(78))

END

INITIATORS

IEVNT(214)=1 \quad // \text{Accumulators Shut Off}
C IEVNT(215)=1 \quad // \text{Main Coolant Pumps Shut Off}
IEVNT(216)=1 \quad // \text{HPI Shut Off}
IEVNT(217)=1 \quad // \text{LPI Shut Off}
C IEVNT(226)=1 \quad // \text{Pressurizer Heaters Shut Off}
C IEVNT(223)=1 \quad // \text{Pressurizer Sprays Shut Off}
IEVNT(251)=1 \quad // \text{Auxiliary Feedwater Pumps Forced On}
IEVNT(252)=1 \quad // \text{Auxiliary Feedwater Pumps Forced On}
IEVNT(228)=1 \quad // \text{Loss of Main Feedwater}
IEVNT(227)=1 \quad // \text{Reactor Scram}
IEVNT(230)=1 \quad // \text{Accumulators Isolated}
IEVNT(232)=1 \quad // \text{Charging Pumps Shut Off}
IEVNT(235)=1 \quad // \text{Main Steam Isolation Valves Forced Closed}
IEVNT(209)=1 \quad // \text{LOCA}

END

C Turn on Auxiliary Feedwater
WHEN TIM > 0.
\quad \text{WVAFW}(1) = 0.0175
END

WHEN HUM > 1
\quad \text{SET FIRE_ALARM}
END

WHEN IEVNT(13) IS TRUE
\quad \text{SET SCRAM_TIME}
END

WHEN IEVNT(20) IS TRUE
\quad \text{SET SATUR_TIME}
WHEN IEVNT(49) IS TRUE
  SET CORE_UNCOVERED
END

WHEN TCLADMAX > 1477.15
  SET CLAD_FAIL
END

WHEN ZWPZ < 2.38
  SET LOW_PRESS
END

WHEN CLAD_FAIL > 300
  TILAST = TIM
END
B.3 Mode C

B.3.1 Dakota

# This is a basic DAKOTA input file
# It will be used in conjunction with MAAP
# It will be used to establish risk uncertainty

# Run dakota -i TSAT.in -o TSAT.out

environment
  tabular_graphics_data
    tabular_graphics_file = 'stateb.dat'

method,
  sampling
    samples = {MCruns}
    sample_type lhs

variables,
  normal_uncertain 2
    means 0 0
    std_deviations 1 1
    lower_bounds -2 -2
    upper_bounds 2 2
    descriptors 'Z_P' 'Z_T'

  uniform_uncertain 1
    lower_bounds 6.5
    upper_bounds 13.5
    descriptors 'time_B'

  triangular_uncertain 2
    modes 0.75 0.5
    lower_bounds 0.6 0.2
    upper_bounds 1.0 0.8
    descriptors 'FCDBRK' 'VFSEP'

  histogram_point_uncertain
    integer = 1
    pairs_per_variable 8
    abscissas 1 2 3 4 5 6 7 8
    counts 1 1 1 1 1 1 1 1
    descriptors 'ABB'
interface,
    fork
    parameters_file 'param'
    results_file 'result'
    asynchronous evaluation_concurrency = {processors}
    analysis_drivers 'shell'
    file_tag file_save
responses,
    response_functions = 2
    no_gradients
    no_hessians
B.3.2 MAAP4

DEBUG OFF
SENSITIVITY ON

TITLE
  STATE_B_SHUTDOWN
END

PARAMETER FILE IS 4LOOP-PWR.par

ALIAS
  TIMER 1 AS CORE_UNCOVERED
  TIMER 2 AS FIRE_ALARM
  TIMER 3 AS SCRAM_TIME
  TIMER 4 AS CLAD_FAIL
  TIMER 5 AS LOW_PRESS
  TIMER 6 AS SATUR_TIME
END

FUNCTION
  HUM = MBRKPS/1520 // Humidity in the Containment
  STARTTIME = {time}
  TUP = TWPSNM + 1
  TLO = TWPSNM - 1
  PUP = PPSNOM - 100000
  PLO = PPSNOM - 300000
  RHRFLOW = 1.5e-5 * QDECAY ** 0.5 - 5.e-8 * (TWPS - 500) ** 3
  RHRFLOWUP = 1.5 * RHRFLOW
  RHRFLOWLO = 0.8 * RHRFLOW
END

PARAMETER CHANGE

C Decay Curve
  FQINHF = 1 // Manual Decay Heat Entry
  NFQHF = 8 // Eight Points
  TIFQHF(1) = 0 // Time Steps
  TIFQHF(2) = 1800.0
  TIFQHF(3) = 5400.0
  TIFQHF(4) = 9000.0
  TIFQHF(5) = 12600.0
  TIFQHF(6) = 19600.0
TIFQHF(7) = 41400.0
TIFQHF(8) = 138600.0
FQHF(1) = 1.0 // Power Fractions
FQHF(2) = {F2}
FQHF(3) = {F3}
FQHF(4) = {F4}
FQHF(5) = {F5}
FQHF(6) = {F6}
FQHF(7) = {F7}
FQHF(8) = {F8}

C Primary System
QCR0 = {power} // Initial Core Thermal Power: State b1
ZWPZ0 = {pzrlevel} // Initial Pressurizer Level: State b1
TWPSNM = {temperature} K // Initial Temperature: State b1
TWPS0 = TWPSNM // Initial Temperature
PPSNOM = {pressure} PA // Initial Pressure: State b1
PPS0 = PPSNOM // Initial Pressure
PPZHT0 = 3000000 // Pressurizer Heater Set Point
ZPZHT = 0
FCDBRK = {brkcoef}
VFSEP = {vdfrac}

C Secondary System
MWSG0 = 166.2 * {density} * 0.4285 // Initial Mass in Steam Generator
PSG0 = PSAT(TWPS0 − 2) // Initial Pressure in Steam Generator
ZWSG0 = 13.2 // Initial Level in Steam Generator

C Break Information
ABB = (.000506707 * {breaksize} ^ 2) M**2
FBB = 7
ZBB = 8.0

C RHR System
NESF = 1
NSLP1 = 1
NDLP1 = 2
NPOI2 = 1
ZHDPI2(1) = 120.0
ZHDR2(1) = 3.05
FHXP1 = 1 // straight tube heat exchanger type
NTU1 = 0.989 // heat exchanger NTU 0.989
TDLPI = 0
WVPME(1) = 1.5e−5 * QCR0 ** 0.5 − 5.e−8 * (TWPS0 − 500) ** 3
START TIME IS 0.0
END TIME IS 86400.0
PRINT INTERVAL IS 600

FUNCTION
TCLADMAX = max(TCLN(1), TCLN(2), TCLN(3), TCLN(4), TCLN(5), TCLN(6), TCLN(7))
TCLADMAX = max(TCLADMAX, TCLN(8), TCLN(9), TCLN(10), TCLN(11), TCLN(12))
TCLADMAX = max(TCLADMAX, TCLN(13), TCLN(14), TCLN(15), TCLN(16), TCLN(17))
TCLADMAX = max(TCLADMAX, TCLN(18), TCLN(19), TCLN(20), TCLN(21), TCLN(22))
TCLADMAX = max(TCLADMAX, TCLN(23), TCLN(24), TCLN(25), TCLN(26), TCLN(27))
TCLADMAX = max(TCLADMAX, TCLN(28), TCLN(29), TCLN(30), TCLN(31), TCLN(32))
TCLADMAX = max(TCLADMAX, TCLN(33), TCLN(34), TCLN(35), TCLN(36), TCLN(37))
TCLADMAX = max(TCLADMAX, TCLN(38), TCLN(39), TCLN(40), TCLN(41), TCLN(42))
TCLADMAX = max(TCLADMAX, TCLN(43), TCLN(44), TCLN(45), TCLN(46), TCLN(47))
TCLADMAX = max(TCLADMAX, TCLN(48), TCLN(49), TCLN(50), TCLN(51), TCLN(52))
TCLADMAX = max(TCLADMAX, TCLN(53), TCLN(54), TCLN(55), TCLN(56), TCLN(57))
TCLADMAX = max(TCLADMAX, TCLN(58), TCLN(59), TCLN(60), TCLN(61), TCLN(62))
TCLADMAX = max(TCLADMAX, TCLN(63), TCLN(64), TCLN(65), TCLN(66), TCLN(67))
TCLADMAX = max(TCLADMAX, TCLN(68), TCLN(69), TCLN(70), TCLN(71), TCLN(72))
TCLADMAX = max(TCLADMAX, TCLN(73), TCLN(74), TCLN(75), TCLN(76), TCLN(77))
TCLADMAX = max(TCLADMAX, TCLN(78))

END

INITIATORS
IEVT(214) = 1 // Accumulators Shut Off
IEVT(215) = 1 // Main Coolant Pumps Shut Off
IEVT(216) = 1 // HPI Shut Off
IEVT(221) = 1 // Fan Coolers Shut Off
IEVT(222) = 1 // Containment Spray Shut Off
IEVT(223) = 1 // Pressurizer Sprays Shut Off
IEVT(227) = 1 // Reactor Scram
IEVT(228) = 1 // Main Feedwater Off
IEVT(230) = 1 // Accumulators Isolated
IEVT(232) = 1 // Charging Pumps Shut Off
IEVT(235) = 1 // Main Steam Isolation Valves Forced Closed
IEVT(209) = 1 // LOCA
C IEVT(217) = 1 // LPI Shut Off
IEVT(213) = 1 // LPI (RHR) On
WHEN TWPS > TUP
    WVPM2(1) = RHRFLOWUP
    REPEAT
END

WHEN TWPS < TLO
    WVPM2(1) = RHRFLOWLO
    REPEAT
END

WHEN HUM > 1
    SET FIRE_ALARM
END

WHEN IEVNT(13) IS TRUE
    SET SCRAM_TIME
END

WHEN IEVNT(20) IS TRUE
    SET SATUR_TIME
END

WHEN IEVNT(49) IS TRUE
    SET CORE_UNCOVERED
END

WHEN TCLADMAX > 1477.15
    SET CLAD_FAIL
END

WHEN ZWPZ < 2.38
    SET LOW_PRESS
END

WHEN CLAD_FAIL > 300
    TILAST = TIM
END
Appendix C

Variance Analysis

A Monte Carlo variance analysis was performed for the Mission Time distribution for each Monte Carlo runs for each break size diameter. A Monte Carlo variance analysis was performed for the Saturation Time for Modes B and C. This was performed in order to verify that adequate runs were performed and that undersampling did not occur. I performed this analysis by calculating the variances of the distributions of the Mission Time and Saturation Time as the new values from each new simulation run were added. As my modeled distribution approaches the real distribution, then the change in the variance value of the distribution approaches zero.

I show first in this Appendix the plots of the percent change in the Mission Time distribution variance value as a function of the simulation run. In almost all of the cases, the percent change approaches zero. The percent changes in variance are around 0.1% at the end of the Monte Carlo simulations. The exception is in Mode C with a break size diameter of 0.0191m. The percent change is about 1% at the end of the Monte Carlo simulation. In this case, undersampling has occurred and therefore no valid conclusions can be made from the distribution.

I then show the plots of the percent change in the Saturation Time distribution variance values as a function of the simulation run for Modes B and C. For both of these operating modes, the percent change approaches zero. The percent changes in variance are around 0.01% at the end of the Monte Carlo simulations.
C.1 Mission Time Distribution: Mode A

Figure C-1: Percent Change in Mission Time Variance
Break Diameter = 0.00635 m
Mode A

Figure C-2: Percent Change in Mission Time Variance
Break Diameter = 0.0127 m
Mode A

Figure C-3: Percent Change in Mission Time Variance
Break Diameter = 0.0191 m
Mode A

Figure C-4: Percent Change in Mission Time Variance
Break Diameter = 0.0254 m
Mode A
Figure C-5: Percent Change in Mission Time Variance
Break Diameter = 0.0318 m
Mode A

Figure C-6: Percent Change in Mission Time Variance
Break Diameter = 0.0381 m
Mode A

Figure C-7: Percent Change in Mission Time Variance
Break Diameter = 0.0445 m
Mode A

Figure C-8: Percent Change in Mission Time Variance
Break Diameter = 0.0508 m
Mode A
C.2 Mission Time Distribution: Mode B

Figure C-9: Percent Change in Mission Time Variance
Break Diameter = 0.00635 m
Mode B

Figure C-10: Percent Change in Mission Time Variance
Break Diameter = 0.0127 m
Mode B

Figure C-11: Percent Change in Mission Time Variance
Break Diameter = 0.0191 m
Mode B

Figure C-12: Percent Change in Mission Time Variance
Break Diameter = 0.0254 m
Mode B
Figure C-13: Percent Change in Mission Time Variance
Break Diameter = 0.0318 m
Mode B

Figure C-14: Percent Change in Mission Time Variance
Break Diameter = 0.0381 m
Mode B

Figure C-15: Percent Change in Mission Time Variance
Break Diameter = 0.0445 m
Mode B

Figure C-16: Percent Change in Mission Time Variance
Break Diameter = 0.0508 m
Mode B
C.3 Mission Time Distribution: Mode C

Figure C-17: Percent Change in Mission Time Variance
Break Diameter = 0.0191 m Mode C

Figure C-18: Percent Change in Mission Time Variance
Break Diameter = 0.0254 m Mode C

Figure C-19: Percent Change in Mission Time Variance
Break Diameter = 0.0318 m Mode C

Figure C-20: Percent Change in Mission Time Variance
Break Diameter = 0.0381 m Mode C
C.4 Saturation Time Distribution

Figure C-23: Percent Change in Saturation Time Variance Mode B

Figure C-24: Percent Change in Saturation Time Variance Mode C
Bibliography


