Including Model Uncertainty in Risk-Informed
Decision-Making

by

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ABSTRACT

Model uncertainties can have a significant impact on decisions regarding licensing basis changes. We present a methodology to identify basic events in the risk assessment that have the potential to change the decision and are known to have significant model uncertainties. Because we work with basic event probabilities, this methodology is not appropriate for analyzing uncertainties that cause a structural change to the model, such as success criteria. We use the Risk Achievement Worth (RAW) importance measure with respect to both the core damage frequency (CDF) and the change in core damage frequency (ΔCDF) to identify potentially important basic events. We cross-check these with generically important model uncertainties. Then, sensitivity analysis is performed on the basic event probabilities, which are used as a proxy for the model parameters, to determine how much error in these probabilities would need to be present in order to impact the decision.

A previously submitted licensing basis change is used as a case study. Analysis using the SAPHIRE program identifies 20 basic events as important, four of which have model uncertainties that have been identified in the literature as generally important. The decision is fairly insensitive to uncertainties in these basic events. In three of these cases, one would need to show that model uncertainties would lead to basic event probabilities that would be between two and four orders of magnitude larger than modeled in the risk assessment before they would become important to the decision. More detailed analysis would be required to determine whether these higher probabilities are reasonable. Methods to perform this analysis from the literature are reviewed and an example is demonstrated using the case study. We then look at policy issues surrounding the effects of uncertainty in decision making related to nuclear power generation.

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Table of Contents

List of Acronyms

ATWS: Anticipated Transient without Scram initiating event

BWR: Boiling Water Reactor

CCFDB: The NRC's Common-Cause Failure Database

CDF: Core Damage Frequency

COL: Combined License

ECCS: Emergency Core Cooling System

EDG: Emergency Diesel Generator

FERC: Federal Energy Regulatory Commission

HRA: Human Reliability Analysis

IPE: Initial Plant Examination

ISI: In-Service Inspection

IST: In-Service Testing

KWh: Kilowatt-hour

LB: Licensing Basis

LERF: Large Early Release Frequency

LOCA: Loss of Coolant Accident initiating event

LOOP: Loss of Offsite Power initiating event

NRC: United States Nuclear Regulatory Commission

PACUA: Probabilistic Accident Consequence Uncertainty Analysis

PORV: Power-operated Relief Valve

PRA: Probabilistic Risk Assessment

PWR: Pressurized Water Reactor

RAI:	Request for Additional Information
RAW:	Risk Achievement Worth
RG:	NRC Regulatory Guide
RCP:	Reactor Coolant Pump
RHR:	Residual Heat Removal system
SAPHIRE:	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SCRAM:	Safety Control Rod Axe Man
SGTR:	Steam Generator Tube Rupture
SPAR:	Standardized Plant Analysis Risk
SSC:	System, Structure, or Component
SSHAC:	Senior Seismic Hazard Analysis Committee
TFI:	Technical Facilitator-Integrator
TRANS:	Transient initiating event
TS:	Technical Specification

List of Figures

List of Tables

CHAPTER I: INTRODUCTION

Very low-probability, high-consequence events are the focus of reactor safety studies. Because of the limited number of these events, there are large uncertainties regarding their probabilities of occurrence. Uncertainty in the Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) can be separated into three classifications; parameter uncertainty, model uncertainty, and completeness uncertainty. This thesis describes a methodology for the identification of basic events which have the potential to impact licensing basis decisions. We concentrate on applications of Level I, at-power, internal-events Probabilistic Risk Assessments (PRAs) and on the decision-making process related to licensing basis changes. Once these basic events are identified, sensitivity studies are performed to determine by how much the probability of each event must be increased to have an impact on the decision. Analysis must then be done to determine whether this increase is reasonable.

PRAs use models of a plant's structures, systems, and components (SSCs) to determine the probability of occurrence of various events. Sometimes, however, there is no consensus on the appropriate model to be used. It may be that, because the system or process is not sufficiently understood, there are differing opinions as to which model most accurately represents the system. This creates uncertainty in the model, which could be related to the structure of the model, or its numerical assessments. This uncertainty in the accuracy of the model introduces uncertainty in the output of the model and, therefore, uncertainty in the output of the PRA. It is this uncertainty that we refer to in this thesis as model uncertainty.[1]

Nuclear power plant licensees may use PRA information to apply for plant-specific licensing basis (LB) changes. Guidance for doing this is provided in Regulatory Guide (RG) 1.174[2] and includes a requirement to meet acceptance guidelines based on the plant's CDF and

LERF and the corresponding changes, ΔCDF and ΔLERF, resulting from the requested change. RG 1.174 and the regulatory guidance on the definition and treatment of model uncertainty will be discussed in Chapter II of this thesis. Model uncertainty is then discussed in detail in Chapter III. The intent is to establish a clear understanding of model uncertainty, its interpretation, theory, and how it is handled in practice, as well as a review and discussion of model uncertainties identified as generally important in the literature. The problem then becomes how to determine which uncertainties can affect the decision. The proposed methodology for this is presented in Chapter IV.

This methodology begins with using the PRA of the plant to determine the Risk Achievement Worth (RAW) importance measure[3] of each basic event. The use of RAW to determine the effect of an event on CDF is well understood. We use RAW in the same way, but we also evaluate RAW with respect to ΔCDF. From the importance measure information, we determine which basic events could possibly affect the decision either through CDF, ΔCDF or both.

Basic events that are identified as potentially important through the RAW analysis are cross-checked with those that have been identified in the literature as having generally important model uncertainty. This cross-check results in a list of basic events that have uncertainties that could possibly affect the decision. These basic events are then analyzed qualitatively and quantitatively to determine their impact on the specific decision.

The benefit of this methodology is that important basic events are identified with respect to the change-specific decision at hand rather than through the use of a general definition of importance. This methodology also reduces the number of uncertainties that must be analyzed extensively by allowing qualitative arguments based on change-specific conditions. In its present form, the proposed methodology deals with the uncertainties associated with the modeling of events that appear in the PRA; it does not deal with model uncertainty that may affect the logical structure of the PRA itself.

In order to illustrate the proposed method, a case study is provided in Chapter V. In Chapter VI, we evaluate the important basic events from the case study. This helps us to understand how important the uncertainty might be in the context of this case study and whether it warrants an in depth review of its probability and the assumptions underlying the calculation of its probability.

In Chapter VII, we look at the policy issues surrounding the effects of uncertainty in decision making related to nuclear power generation. Chapter VIII contains a summary and conclusions from this research.

In Appendix A, we summarize case studies of risk-informed licensing basis changes that were reviewed in preparation for this research. Appendix B contains a detailed procedure for calculating RAW with respect to ΔCDF, as proposed in this thesis. Appendices C and D contain the complete list of basic events from the case study used in this thesis, which come from the case study plant's PRA. Appendix C sorts the list in descending order according to the RAW with respect to CDF. Appendix D sorts the list according to RAW with respect to ΔCDF, as defined in this thesis.

CHAPTER II: RISK-INFORMED DECISION MAKING

The U.S. Nuclear Regulatory Commission (NRC) encourages the use of PRA methods where practical, consistent with the state-of-the-art, to support a risk-informed regulatory framework.[4] RG 1.174 is a key document in this framework. It presents five principles of risk-informed decision making to be used for making decisions regarding plant-specific changes to the licensing basis. These principles are:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption.
2. The proposed change is consistent with the defense-in-depth philosophy.
3. The proposed change maintains sufficient safety margins.
4. If the proposed change increases risk, the increase should be small.
5. The impact of the proposed change should be monitored using performance measurement strategies.

The first principle makes it clear that existing regulations not related to the requested change must still be met. The second and third principles account for some of the completeness uncertainty that exists when assessing nuclear power plant risk. This uncertainty is referred to as the "unknown unknowns," or the uncertainties that exist but have not been identified. Because the uncertainty has not been identified, it cannot be quantified. Traditional defense-in-depth measures and safety margins (the "structuralist" approach to safety[5]) are requirements designed to protect against these uncertainties. The fourth principle is the one we are concerned with in this thesis, requiring risk increases to be small. Risk and risk increases are quantified using PRA. The fifth principle ensures that the results of a licensing base change are as expected. Performance monitoring and measurement after the change provide feedback that can be used to identify and correct unexpected problems that result from the change and also to inform future changes. Also, the effects of uncertainty on all of these principles must be considered, whether the uncertainty is explicitly included in the model or not.

Risk increases must be small. Small changes are defined using the CDF and LERF acceptance guidelines of Figures 1 and 2, respectively. The values of CDF, ΔCDF, LERF, and ΔLERF to be used in these figures are supposed to be the mean values of the uncertainty distributions of these quantities. This thesis focuses on Level I PRAs, so we will focus on the CDF/ΔCDF guidelines. Referring to Figure 1, the horizontal axis represents the baseline CDF. This is the frequency at which core damage is expected to occur at the plant if no plant changes are made. The vertical axis represents the ΔCDF, the amount that the CDF is expected to increase, if the proposed LB change were made.

Each nuclear power plant has an associated plant-specific CDF. Each LB change that a plant desires to make has an associated plant-specific and change-specific ΔCDF. These two risk metrics place a proposed change in one of the three labeled regions in Figure 1. Uncertainty in the risk metric calculations prevents an exact placement of a change onto one of the three regions. Therefore, the values representing the dividing lines between the regions must be viewed as indicative, rather than definitive. Changes that have a ΔCDF placing them in Region III are classified as having a very small increase in risk and may be approved without the need for a quantification of CDF. Changes that are in Region II are classified as having a small increase in risk and may be approved, but may require a more stringent review. Region II also sets an upper bound of about 10^{-4} per reactor-year (ry^{-1}) on the baseline CDF. Changes in Region I do not meet the requirements of a small risk increase and will, in general, not be approved.

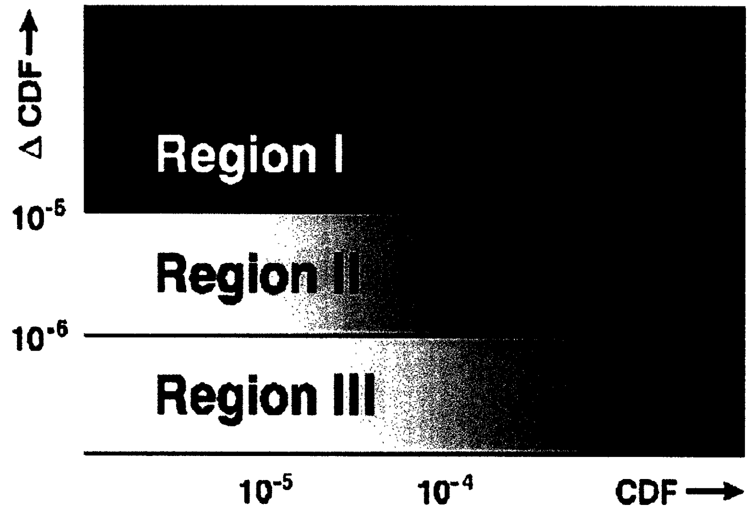

Figure 1. CDF Acceptance Guidelines

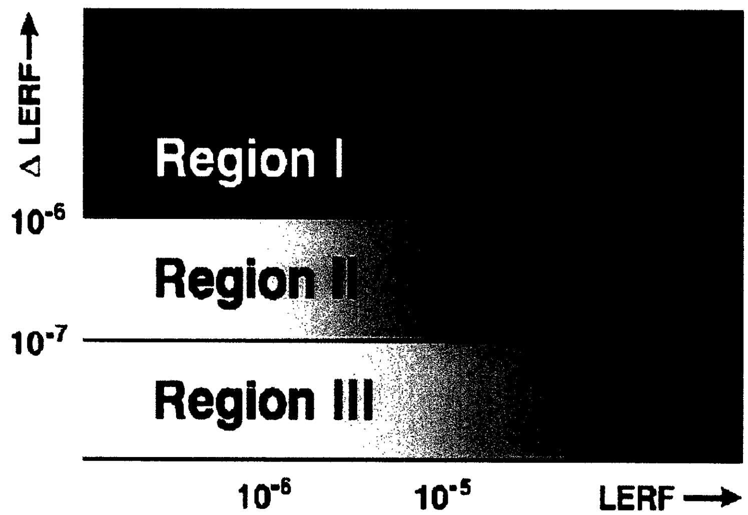

Figure 2. LERF Acceptance Guidelines

Figure 1 also shows gradual shading, darkening as one moves upward and to the right, representing CDF and ΔCDF combinations that are closer to the boundaries between regions. The darkness of the shading corresponds to the level of review that the application will be given, such that LB changes that have a representative point in areas of darker shading, i.e., near the region boundaries, warrant a review that is generally more intensive. "The closer the estimates of ΔCDF and ΔLERF are to their corresponding acceptance guidelines, the more detail will be required."[2]

RG 1.174 requires that all sources of uncertainty be identified and analyzed such that their impacts are understood at the technical element level, and on the CDF and LERF risk metrics. RG 1.200 states that "an essential aspect of the risk characterization is an understanding of the associated uncertainties.[6] Uncertainties in the PRA must be understood and accounted for, whether or not they are explicitly modeled.

RG 1.200 also provides two ways to ensure the technical adequacy of a PRA in support of risk-informed regulatory decisions. The first is to meet the criteria of the American Society of Mechanical Engineers (ASME) PRA standard,[7] as supplemented by the comments in RG 1.200. The second is to have the PRA peer reviewed using the Nuclear Energy Institute (NEI) peer review process,[8] as supplemented by the comments in RG 1.200.

II.A. Types of Uncertainty

Uncertainties can be categorized as either aleatory or epistemic uncertainties.[9] Aleatory uncertainty reflects our inability to predict random observable events. For example, the flip of a fair coin is generally accepted to yield heads with a probability of 0.50. However, the number of times that heads will occur as a result of 10 coin flips in unknown. Only the probability distribution of the number of heads is known. This is also referred to as 'randomness' or 'stochastic uncertainty.' Epistemic uncertainty represents our confidence in the model and the numerical values of its parameters, e.g., that the coin is fair so that the probability of heads can be taken to be 0.50. A process may not be sufficiently understood and, as such, a specific model may not be universally accepted as being the right model. This type of uncertainty is also called 'state-of-knowledge' uncertainty or just 'uncertainty.'[10] We note that, unlike aleatory uncertainties, epistemic uncertainties are associated with non-observable quantities, e.g., the parameters of models such as failure rates.

The distinction of uncertainty into aleatory and epistemic is largely due to "practical aspects of modeling and obtaining information."[11] At their core, they both refer to the problem of modeling real-world systems with mathematical formulas, whether deterministic or probabilistic.

Epistemic uncertainties can be roughly split into three categories and is done in RG 1.174. These are: parameter, model, and completeness uncertainty. Parameter uncertainty is that which relates to the parameters of the PRA, given a choice of model. Even with a known model, the parameter values may still be unknown. In situations where historical data is limited, this uncertainty may be quite large. Examples of parameter uncertainties include equipment failure rates, initiating-event frequencies, and human error probabilities.

In many cases, there is limited knowledge and some disagreement on the proper model to represent a system. The result is that for a particular process, there are multiple competing models, each of which necessarily produces a different approximation of the same real-world system. Because the correct model is unknown, there is additional uncertainty in the output of any model, representing the uncertainty in the model's itself. This uncertainty adds to the parameter uncertainty described above and is model uncertainty. The outputs of each reasonable model must be considered, according to the degree of belief in the appropriateness of each model, to prevent exclusion of valuable uncertainty data from consideration. Several methods have been proposed to accomplish this, including the linear or otherwise combination of models weighted by the analyst's belief that each model may be correct,[9] and the use of an adjustment factor on the single most likely model.[10] These and other methods will be discussed in Chapter III.

The PRA structure itself is model-dependent because model uncertainty can affect the choice of success criteria. For example, one model might say that two primary relief valves are required to prevent core damage during a loss of offsite power event, while another might say that only one primary relief valve is required. In this case, while there still remains (parameter) uncertainty in the failure rate of relief valves, there is also uncertainty in how many relief valves are required. This latter uncertainty is model uncertainty also. In cases where model uncertainty is treated by using a single, conservative model, the effects of alternate assumptions must be recognized. RG 1.174 recommends using sensitivity studies to determine whether or not there are any assumptions or models whose results would reduce confidence in the conservatism of the chosen model.

Completeness uncertainty is a type of model uncertainty, but is handled differently. It represents the uncertainty due to the portion of risk that is not explicitly included in the PRA. It may be that, for a particular risk contributor, the state-of-the-art has not evolved to the point where the risk can be modeled defensibly. This is the case with safety culture and organizational behavior in general. RG 1.174 states that "the influences of organizational performance cannot now be explicitly assessed." Completeness uncertainty also includes risks that have not been identified. This includes anything that has not been identified as a risk contributor, yet does contribute to risk. Due to the nature of this type of uncertainty, it is impossible to quantify. Conservatisms, such as defense-in-depth and safety margins largely exist to defend against this type of uncertainty, as stated earlier.

Referring back to the acceptance guidelines of Figure 1, the values of CDF and ΔCDF used are supposed to be epistemic mean values, i.e., the mean values of the distributions of CDF and ΔCDF. These distributions are largely the result of propagating through the PRA the epistemic distributions that represent the parameter uncertainties that were explicitly included in the PRA model. Because these mean values already include the effects of parameter uncertainty, as represented in the probability distributions of the input parameters, they are fairly insensitive to changes in these distributions. In contrast, model uncertainties generally have a greater potential to affect these metrics thus affecting the approval decision. Knudson and Smith support this argument by measuring the model uncertainty regarding the success criteria of Auxiliary Feedwater pumps and comparing it with the parameter uncertainty in the pump failure rates.[12] Model uncertainty is measured by varying the number of pumps required, such that the system may be a 1-out-of-3, 2-out-of-3, or 3-out-of-3 system, assigning a probability that each case is true. Looking at each uncertainty while ignoring the effects of the other, the parameter and model uncertainty provide the following 90th percent confidence intervals for the system failure rate.

Parameter uncertainty	$2.2*10^{-4}$ to $6.3*10^{-4}$ ry^{-1}
Model uncertainty	$1.9*10^{-5}$ to $1.5*10^{-3}$ ry^{-1}

While the parameter uncertainty range spans about a factor of three, the model uncertainty range spans about two orders of magnitude.

Bley, Kaplan, and Johnson[13] measure the impact of model uncertainty on CDF directly. In their work on a plant-specific PRA, they identified three model uncertainties that had the greatest potential to impact CDF: Reactor Coolant Pump (RCP) seal LOCA timing, low-end seismic fragility curves for piping and D/C electrical components, and seismically-induced relay chatter. They chose two alternate assumptions for each model and assigned probabilities that each was true. This resulted in eight different sets of assumptions when all three models were inserted into the PRA. Each set of assumptions resulted in a different mean value of CDF for the plant, which ranged from about $2*10^{-4}$ to $3*10^{-3}$ ry^{-1}, with the most likely value being about $2*10^{-4}$ ry^{-1}. The most likely value corresponds to the low end of the range because the set of assumptions with the highest probability of being true resulted in the lowest CDF. These results show that modeling assumptions can shift the mean value of CDF significantly.

CHAPTER III: MODEL UNCERTAINTY

As stated in Section II.A, there is no mathematical difference between different types of uncertainty. They all refer to unknowns, the limit of knowledge about a real-world phenomenon. "For the case of a finite number of alternative models, the model uncertainty is equivalent to parameter uncertainty,"[14] with reference to Savage's partition problem.[15] The theoretical overlap between model and parameter uncertainty can also be seen by creating a parameter whose value is dependent upon the model used.[10]

Methods to deal with model uncertainty include prediction expansion and model set expansion.[10] In prediction expansion, a single model is chosen as the best one to represent the system. However, it is recognized that this model has uncertainties and may model some characteristics of the system better than others. Sensitivity studies are performed on the various assumptions to analyze the effects of the choice of assumptions on the model output. This uncertainty is dealt with by applying an adjustment factor to the model results. The adjustment factor may be multiplicative or additive, or both may be necessary. The purely additive and multiplicative adjustment factor approaches can be seen in equations (1) and (2), respectively:

$$y = y^* + E_a^* \tag{1}$$

$$y = y^* * E_m^* \tag{2}$$

where y^* represents the model prediction, E^* represents the adjustment factor, and y represents the adjusted model output. E* may also have (aleatory or epistemic) uncertainty in its value, due to limited data, for example.

In model set expansion, the characteristics of the system under consideration are analyzed and models are created in an attempt to emulate the system based on goodness-of-fit criteria. The models may use different assumptions, and require different inputs. Each model has its own advantages and disadvantages, including limitations on applicability. These models are then combined to produce a meta-model of the system.

Several methods have been proposed regarding the construction of this meta-model. They include mixture,[9] Bayesian updating,[16,17] the NUREG-1150 approach[18], the joint U.S./Commission of European Communities' (EC) Probabilistic Accident Consequence Uncertainty Analysis (PACUA) approach[19], and the Technical Facilitator-Integrator approach.[20] Of course, all of these methods rely on expert opinion.

In the mixture approach, the set of plausible models is agreed upon from expert opinion and these experts agree on probabilities that each model is correct. The models are then combined linearly, with their weights corresponding to the probability of correctness. The result is a weighted average of the probability distributions that result from each model. The multiple distributions should be presented before they are combined, thus allowing an analyst a more transparent look at the range of models that became the meta-model.

In the Bayesian approach, each model is integrated into the meta-model using Bayes' theorem, using the following formula:

$$f_p(x|f_1, \ldots f_k) = f(x) * g(f_1, \ldots f_k |x) \qquad (3)$$

$f_p(x|f_1, \ldots f_k)$ is the posterior distribution resulting from the combination of the individual models, $f(x)$ is the analysts' prior distribution, $f_i(x)$ is the distribution given by the i^{th} available model, and $g(f_1, \ldots, f_k|x)$ is the likelihood function. This method is theoretically very attractive due its mathematical rigor and ability to incorporate all types of information. However, the calculation rapidly becomes onerous and proves impractical.

In the NUREG-1150 approach, multiple experts are elicited to produce their own probability distribution of the system in question, based on their own opinion. The individual results are then combined linearly, with each expert given equal weight. The PACUA approach goes one step further by including information about the confidence in each expert. The experts are asked to produce distributions for seed variables, for which data is known, and their opinions are compared to the known data. Experts with superior performance when estimating the seed variable distribution are given higher weight when opinions regarding the system in question are elicited.

The final method under consideration here is the Technical Facilitator-Integrator (TFI) approach, which takes advantage of many of the lessons learned from previous expert elicitation exercises. In this approach, the experts are treated as a team, rather than individuals, each sharing their opinion separate from the consideration of the other experts. Individual elicitations are obtained. However, the team works together with the TFI to integrate the data, including the experts' knowledge of technical experts outside of their own group, into a meta-model that attempts to represent the current total body of knowledge. Part of the TFI's role is to mitigate problems identified in behavior science, such as the tendency for more dominant members of the group to be given undue weight on their opinion.

With this background on model uncertainty, the distinction between different classifications of uncertainty, the reason for these classifications, the theory behind the formalisms of model uncertainty, and practical applications, we now look at generic model uncertainties in PRAs that have been identified in the literature.

III.A: Generic Model Uncertainties

A literature review provided a fairly extensive, yet manageable, list of major model uncertainties pertaining to Level 1, at power, internal events PRAs. Insights from the literature review will be used to learn more about the uncertainties that were identified as important in the case study. Much of the data comes from NRC-sponsored studies. This review was not limited to NRC generated data, however, and a variety of sources was used. A discussion of the results is provided here, with an emphasis on basic events relating to Level 1, at-power, internal events PRAs.

NUREG/CR-4550[21] organized an expert panel to address several important uncertainties. Some were related to problems at individual plants and are excluded here. These are:

- Probability of the failure of two check valves in series in a PWR constituting a boundary between a high and a low pressure system.
- Emergency Core Cooling System (ECCS) failure rates due to venting or containment failure. This refers to the operability of components in hostile environments. PRAs normally assume 100% failure rate if equipment are operated above their qualification limit. This data shows expected failure rates with respect to different types of components, different operating condition, and different lengths of operation
- RCP seal LOCA probability. Results were given with respect to time after the initiating event and leak rate.
- Probabilities of innovative recovery actions for long-term sequences involving loss of containment heat removal. The panel concluded that success probabilities are highly dependent on plant specific features like climate, location, staff training, plant design, and layout. Results are given in terms of probabilities of repair versus time for various components.
- Failure probability of using high pressure service water spray in the dry well.
- Battery depletion time.
- Diesel Generator field flashing failure probability.

- Hydrogen ignition probability on restoration of A/C power.
- Human actions to shutdown the plant failure probability.
- Secondary safety valve demand and failure rates.
- Reactor Coolant System depressurization failure probability.
- Common-cause β-factor uncertainty ranges.
- Common-cause β-factor for Air-Operated Valves.

NUREG-1764[22] provided a list of uncertainties in the reliability of human actions that were either known to be risk-important or had the potential to be risk-important. They are broken into categories by plant type, Pressurized Water Reactor (PWR) or Boiling Water Reactor (BWR), as follows:

Pressurized Water Reactors

- Switch the ECCS from the injection mode to the recirculation mode in a LOCA scenario.
- Feed and bleed, particularly the use of pressurizer relief valves.
- Provide water supply for Auxiliary Feedwater by moving water from alternate sources into the Auxiliary Feedwater system when long-term cooling is needed.
- Trip the RCPs to prevent RCP seal LOCA on a loss of RCP cooling.
- Recover RCP seal cooling by aligning an alternative means of cooling.
- Recover emergency A/C or offsite power.
- Respond to an Anticipated Transient Without Scram (ATWS) - failure of the Reactor Protection System, particularly the initiation of boron injection and including manual scram of the reactor and ensuring turbine trip.
- Depressurize during a Steam Generator Tube Rupture (SGTR). This includes the depressurization of the primary and secondary systems and equalizing pressure between them.

- Isolate steam generator during a Main Steam Leak Break or a SGTR.
- Shut Power Operated Relief Valve (PORV) blocking valve during a stuck open PORV event.
- Isolate interfacing system LOCA during a LOCA in the Low Pressure Injection system.

Boiling Water Reactors

- Perform manual depressurization to allow injection with low pressure injection systems. This is typically done by operating the safety relief valves.
- Vent containment and align containment or suppression pool cooling during a LOCA.
- Control vessel level during an ATWS in order to control reactor power.
- Initiate standby liquid control during an ATWS.
- Inhibit the Automatic Depressurization System in order to prevent instabilities that occur at low pressures.
- Miscalibration of pressure switches that are important for initiating and controlling the ECCS.
- Initiate isolation condenser in BWR plants of early design.
- Control feedwater events. Control the feedwater system after a loss of feedwater event.
- Recover offsite power.
- Shed D/C loads after a Station Blackout in order to extend battery life.

Bley, Buttemer, and Stetkar argue that an adequate analysis of event sequence timing is important in PRA analysis for a couple of reasons.[23] Success criteria determination requires an understanding of sequence timing. How plant parameters change over time during an accident sequence determines what equipment is necessary to prevent core damage. Success criteria are often chosen based on deterministic thermohydraulic calculations using assumptions that are conservative. Calculation of the probability of recovery is also dependent on the results of a sequence timing analysis. Human performance is highly dependent on the time available for the operator to complete actions, and the time available is calculated using an analysis of sequence timing. The authors analyze a number of risk-important parameter and model uncertainties and reach the following conclusions: success criteria and recovery modeling are highly dependent on sequence timing; determinations of the factors that affect operator performance, including dependencies and competing demands, requires a detailed analysis; and simple analysis involving mass and energy balance to determine their effect on sequence timing is often sufficient for PRA applications.

RG 1.200[6] provides several examples of key uncertainties when determining the technical adequacy of a PRA. Uncertainties in success criteria, human reliability, and the choice of a RCP seal LOCA model are included. In these cases, the choice of the model and how it is used may have a significant impact on risk.

Sump performance was identified as important by the NRC's Advisory Committee on Reactor Safeguards.[24] Because of the nature of the sump, and the limited data that exist on how a sump might perform when needed, it is difficult to estimate the probability that it will perform successfully. Specifically, it is difficult to model how the strainer on the intake side of the pump will be affected by debris in the sump. There is some probability that the debris will clog the strainer and reduce the net positive suction head on the pump sufficiently to effectively disable the pump.

Interviews with NRC personnel also provided a number of important model uncertainties.[25] RCP seal LOCA probability, battery depletion time, common-cause failure modeling, and modeling of sump plugging and pool strainer plugging were identified.

Emergency diesel generator mission time and recovery modeling were identified also. This refers to how long the diesel generator is assumed to be needed in order to fulfill its mission, and also what mechanisms for recovery are credited in the PRA and the probability of these recovery mechanisms. Success criteria determination is important, specifically with regard to how many PORVs are required during a feed-and-bleed evolution. Support systems may be important in ways that are not obvious at first glance, especially when they have the ability to cause common-cause failures across many systems. Sometimes, the risk-importance of these systems is missed and they are either not modeled adequately, introducing model uncertainty, or left out of the PRA, introducing completeness uncertainty. Instrument air is an example of a support system that many components in multiple systems depend on, but that may not seem risk-important itself unless attention is brought to these dependencies. The modeling of SGTR event tree was also considered important.

CHAPTER IV: PROPOSED METHODOLOGY

The CDF is calculated as

$$CDF_{base} = \sum_{i} fr(MCS_{base,i}) \tag{4}$$

where CDF_{base} is the baseline CDF of the plant, as it is normally configured. However, this calculation can be performed for any plant configuration. $MCS_{base,i}$ is the i-th minimal cut set, and $fr(MCS_{base,i})$ is the frequency at which the i-th cutset occurs in the baseline PRA model.

Uncertainties surround the value of the baseline CDF, since there are uncertainties in the frequency of the initiating events and also in the conditional probabilities of occurrence of the basic events. These same uncertainties create uncertainties in the value of ΔCDF. The uncertainties in ΔCDF can have a significant impact on the decision whether or not to approve the change, as acceptance guidelines are provided as a combination of CDF and ΔCDF. The significance of this impact can be seen by looking at the definition of ΔCDF,

$$\Delta CDF = CDF_{after} - CDF_{base} \tag{5}$$

where CDF_{after} is the CDF of the plant after the proposed licensing change. Inserting Equation (4) into Equation (5), we find that

$$\Delta CDF = \sum_{i} fr(MCS_{after,i}) - \sum_{j} fr(MCS_{base,j}) \tag{6}$$

where $MCS_{after,j}$ is the frequency at which the j-th cutset occurs in the PRA model as it exists after the proposed licensing basis change. The proposed change will change the frequency at which some of the minimal cut sets occur. However, most of them will remain unchanged. Each minimal cut set that is unaffected will, therefore, appear in both terms on the right-hand side of equation (6) and drop out of the equation. It is clear that uncertainties affect the value of CDF and ΔCDF. It is also clear that these uncertainties can change the outcome of a decision based on the acceptance guidelines in Figure 1.

We propose a methodology for including these uncertainties in the decision making process used to make risk-informed licensing basis decisions in accordance with RG 1.174. This methodology begins with using the PRA of a plant to determine the RAW importance measure of each basic event.

IV.A: RAW with respect to CDF

Importance measures are used in the ranking and categorization of basic events modeled in a PRA.[3] The importance measure of most interest to us is the Risk Achievement Worth (RAW). RAW is defined as

$$RAW_j = \frac{R_j^+}{R} \tag{7}$$

where RAW_j is the value of RAW for basic event j, R is the value of the model's baseline risk metric, and R_j^+ is the value of the model's risk when basic event j is set to a logical TRUE. Each basic event, therefore, is assigned a value of RAW, by the PRA, that quantifies the factor by which a plant's risk would increase if the associated basic event were assumed to be completely unreliable. RAW is a bounding measure that provides the maximum level of risk that a basic event could cause.[3]

The meaning of RAW can also be viewed with respect to the logic structure of the PRA. A basic event that is completely unreliable serves no risk function in the PRA. It is as if the basic event were completely removed from the logic structure. Therefore, the RAW of a basic event represents the factor by which a plant's risk would increase if the basic event were removed from the plant. Since the risk metric in this case is the CDF,

$$RAW_{j,CDF-base} = \frac{CDF^{+}_{j,base}}{CDF_{base}} \tag{8}$$

The set of values for $RAW_{j,CDF\text{-}base}$ can easily be generated using the SAPHIRE[26] program.

IV.B: RAW with respect to ΔCDF

Importance measures can also be used to show areas in a PRA where uncertainty can have the greatest impact on the change in risk that is proposed by the licensing basis change. To represent this importance measure, we start with the definition of RAW above, Equation (7), and note that the risk metric R in this case is ΔCDF. Therefore,

$$RAW_{j,\Delta CDF} = \frac{\Delta CDF^{+}_{j}}{\Delta CDF} \tag{9}$$

Noting the definition of ΔCDF, Equation (5), we expand this equation to

$$RAW_{j,\Delta CDF} = \frac{CDF^{+}_{j,after} - CDF^{+}_{j,base}}{CDF_{after} - CDF_{base}} \tag{10}$$

Inserting equation (8) into equation (10), we see that

$$RAW_{j,\Delta CDF} = \frac{(RAW_{j,CDF-after})*(CDF_{after}) - (RAW_{j,CDF-base})*(CDF_{base})}{CDF_{after} - CDF_{base}} \tag{11}$$

The values on the right-hand side of equation (11) are fairly easy to generate. From the PRA, CDF_{base} is known directly. From the application of Equation (8), we calculate the set of values for $RAW_{j,CDF\text{-}base}$. To find the other values, we must modify the PRA to represent the plant as it would exist after the change. Using this model and repeating the steps used to calculate CDF_{base} and $RAW_{j,CDF\text{-}base}$, we calculate CDF_{after} and the set of values for $RAW_{j,CDF\text{-}after}$. Now, all of the variables on the right-hand side of this equation are known and the set of values for $RAW_{j,\Delta CDF}$ can be generated using, for example, sorting and arithmetic algorithms in a Microsoft Excel spreadsheet.

IV.C: Calculating RAW Thresholds

At this point, we have a complete set of basic events with their respective values for RAW with respect to CDF and RAW with respect to ΔCDF. Some threshold must be set to determine the value of RAW below which we deem the basic event to be not risk-important. It should be noted that traditionally in licensing basis change requests a threshold value of RAW is set at a value of two.[27] Using this method, basic events with a RAW of two or higher are deemed as potentially important and analyzed further, while those with a RAW less than two are classified as not risk-important. The methodology in this thesis proposes a simple method for determining a change-specific threshold value of RAW.

By referring to the acceptance guidelines in Figure 1 and the position of the proposed licensing change's risk on the figure, we see that there is some value of RAW that will move the plant's risk to the right on the figure until it enters a different region. The RAW with respect to CDF of each basic event indicates the maximum amount that uncertainty in this basic event can move the plant's CDF to the right. A small RAW might indicate that regardless of the reliability of a particular basic event, the CDF would not be in Region I, and therefore would not affect the decision. In other cases, the RAW may be large enough. It is only when the CDF moves into

Region I that the uncertainty becomes important to the decision. We, therefore, determine the threshold value of RAW with respect to CDF that will change the decision as follows,

$$RAW_{CDF,threshold} = \frac{CDF_{threshold}}{CDF_{base}} \tag{12}$$

where $RAW_{CDF,threshold}$ is the RAW value that will move the CDF to the right and into a different region, and $CDF_{threshold}$ is the value of CDF corresponding to the vertical lines between the regions of Figure 1. The threshold RAW value is dependent on the CDF of the plant and the ΔCDF of the proposed change and its value is change-specific. Remember that although the CDF has uncertainty and is represented by a distribution of values, RG 1.174 calls for the mean value to be used in Figures 1 and 2.

The threshold value for ΔCDF can be determined in a similar fashion. Referring to Figure 1, we see that there is a value of RAW with respect to the ΔCDF that will move ΔCDF upward in the figure until it enters a different region. It is this change between regions that changes the context of the decision and possibly the decision itself. We, therefore, determine the value of RAW with respect to ΔCDF that will change the decision as follows,

$$RAW_{\Delta CDF,threshold} = \frac{\Delta CDF_{threshold}}{\Delta CDF} \tag{13}$$

where $RAW_{\Delta CDF,threshold}$ is the RAW value that will move ΔCDF upward and into a different region, and $\Delta CDF_{threshold}$ is the value of ΔCDF corresponding to the horizontal line between the applicable regions of Figure 1. Just as the CDF_{base} used when calculating $RAW_{CDF,\ threshold}$ was a mean value, the ΔCDF value used here should be a mean value. This determination again differs from the traditional RAW threshold value of two used in licensing basis change decisions. The threshold value of RAW used here is change-specific.

IV.D: Cross-check of Important Basic Events

Of course, not all basic events have large uncertainties in their reliabilities. For example, motor-driven pumps have been used extensively in nuclear power plants for some time. Because of this, a sufficient amount of historical failure data has been accumulated such that their failure rates are known with a fair degree of certainty and the mechanisms by which they fail are fairly well understood. The methodology, therefore, cross-checks the basic events whose uncertainty may be important as identified by the method above to basic events that have been identified in the literature review as having generally important model uncertainty. The basic events that remain after the cross-check are those that are important to the plant-specific licensing basis change decision at hand. They have model uncertainties identified in the literature as generically important, and are close enough to a threshold value in Figure 1 that these uncertainties could possibly affect the decision. The generically important model uncertainties and their descriptions are included in Chapter III.

IV.E: Making the Decision

Having identified the important basic events with respect to both CDF and ΔCDF, we must now investigate their potential impact on the decision. Quantifying the model uncertainty would allow the decision maker to see how CDF and ΔCDF of the proposed change move and whether the change meets the acceptance guidelines. However, model uncertainty is quite difficult to quantify at this time. Instead, our proposed methodology employs sensitivity analysis to determine the degree to which a basic event's failure probability would need to change in order to violate the acceptance guidelines, in effect changing the approval decision. From this point, qualitative arguments remove some basic events from further consideration.

This does, of course, still rely on expert opinion as a tool for estimating plausible upper bounds for risk. However, expert opinion is used to a lesser degree because many uncertainties are eliminated from consideration because they are determined to be unimportant. Remaining basic events must be subjected to considerable scrutiny and the effects of uncertainty quantified before a decision can be made. Calculations needed for this thesis were performed using the Standardized Plant Analysis Risk (SPAR) model for this plant and the System Analysis Programs for Hands-On Integrated Reliability Evaluations (SAPHIRE) computer software.

Model uncertainties that require a detailed quantitative evaluation may be handled by using the methods of Chapter III. For example, one method was to use an adjustment factor. In this case, reasonable alternative assumptions to those used in the PRA are established, and their effects on the PRA output quantified. Expert judgment is then used to determine the probability distribution of the adjustment factor. The adjusted CDF from the PRA is then compared with the acceptance guidelines to determine if the licensing basis change decision is sensitive to these modeling assumptions.

The benefit of the proposed methodology is that important basic events are identified with respect to the change-specific decision at hand rather than using a general importance measure. This methodology also reduces the number of uncertainties that must be analyzed extensively by allowing qualitative arguments based on change-specific conditions. In its present form, however, this methodology is not sufficient when model uncertainty affects the logical structure of the PRA, as in the example provided earlier where uncertainty affected the success criteria.

CHAPTER V: THE CASE STUDY

To illustrate the methodology proposed in this thesis, we present a case study. This case study was a licensing basis change request submitted to the NRC. The request applied to a commercial PWR Westinghouse four-loop design and proposed to establish a risk-informed in-service testing (IST) program that would replace the existing IST requirements for a portion of the plant's valves. The IST program applied to 160 valves that made up various portions of 10 systems. These systems were:

Steam Generator Blowdown
Heating and Ventilation – Purge Air
Compressed Air – Control Air
Chemical and Volume Control
Safety Injection
Essential Raw Cooling Water
Component Cooling
Core Spray
Waste Disposal
Radiation Monitoring

The risk-informed IST program proposed that, for these valves, the IST frequency would be changed from once per quarter to once per refueling cycle, or approximately once per 18 months. The licensee states that it is conservative to assume that the failure probability of these valves increases linearly with time between inspections. Since the time between inspections increases by about a factor of six, the failure probability of each valve affected by the change is increased by a factor of six when modeling the effects of the change.

The point estimate baseline CDF and ΔCDF of the plant were:

$CDF = 6.8 \times 10^{-5}\ ry^{-1}$
$\Delta CDF = 6.9 \times 10^{-7}\ ry^{-1}$

These values represent a point in the CDF versus ΔCDF acceptance guidelines as shown in Figure 3. This pair of values places the point representing the proposed LB change in Region III of the acceptance guidelines. We note that the CDF and ΔCDF reported were "point" estimates, i.e., the licensee did not propagate the distributions of the input parameters to produce distributions for CDF and ΔCDF. As point estimates, the values of CDF and ΔCDF are sensitive to parameter uncertainties also. Since our objective is to investigate model uncertainties, we will treat these point estimates as if they were epistemic means.

Following our methodology, we must first generate the complete set of RAW values for the basic events at this plant. This includes the RAW with respect to CDF and the RAW with respect to ΔCDF.

V.A: Event RAW with respect to CDF

The point representing the proposed change is in Region III and the decision would be affected if uncertainties moved this point into Region I. Therefore, we are interested in the horizontal threshold between Region I and Region III, which is about 10^{-3} ry^{-1}. Although this is not a "bright line" boundary, 10^{-3} ry^{-1} is a reasonable value. We note that the NRC has a goal of keeping the CDF below 10^{-4} ry^{-1}, thus making 10^{-4} ry^{-1} another reasonable boundary value; this is also the value it would have to exceed to be in Region I if the uncertainties were to bring the ΔCDF up to Region II. Therefore, $CDF_{threshold}$ is given two values, equal to 10^{-3} and 10^{-4} ry^{-1}. The CDF_{mean} of the plant was 6.8×10^{-5} ry^{-1}. Therefore, the $RAW_{CDF,threshold}$ values were about 14.6 and 1.46, respectively, which we truncated to 14 and 1.4. Any basic event with a RAW greater than 14 has the potential to change the licensing basis decision because the actual CDF could be in Region I. Also, a RAW greater than 1.4 indicates that uncertainty in the basic event probability is important to the decision because the actual CDF could be greater than the NRC goal of keeping CDF less than 10^{-4}, and it has the potential of being important to the decision, depending on the ΔCDF value once uncertainties are included.

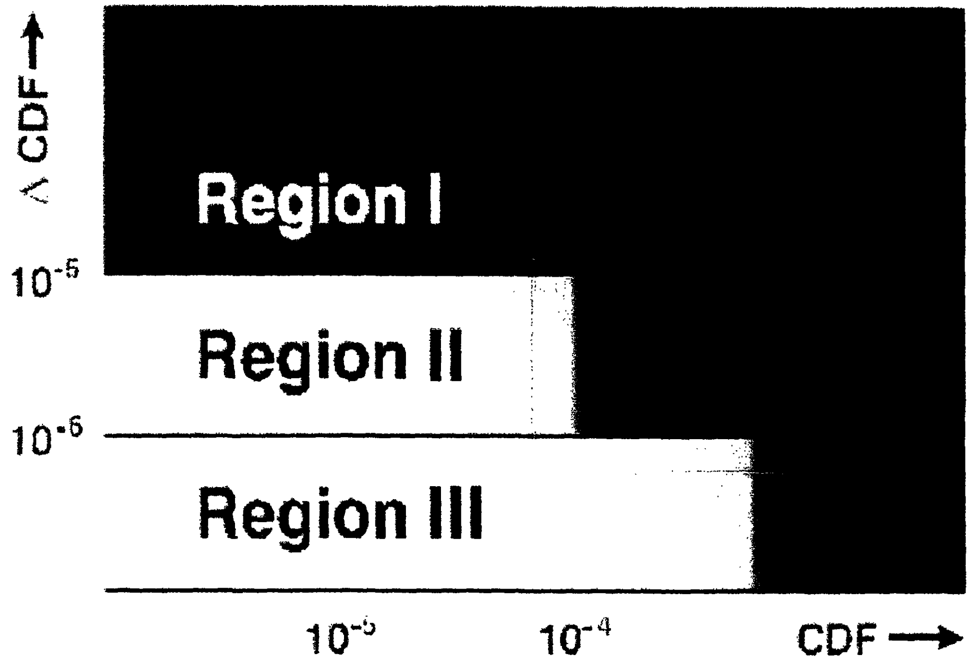

Figure 3. CDF Acceptance Guidelines with Representative Point

Using SAPHIRE, we determined that there were 12 basic events with RAW greater than 14. They are listed in Table I.

TABLE I. RAW with respect to CDF: RAW>14

BASIC EVENT	RAW_{CDF}
a) Control rods fail to insert	3,050
b) Common-cause diesel generator failure	271
c) Failure to depressurize due to hardware failure	218
d) Scram breakers fail to open	202
e) 4160V Bus 1B fails	197

f) Common-cause failure of Residual Heat Removal (RHR) pumps	134
g) Common-cause failure of RHR heat exchangers	134
h) Reserve Water Storage Tank not available	112
i) Common-cause Auxiliary Feedwater pump failure	26.6
j) Common-cause failure of Steam Generator discharge valves to open	26.5
k) Common-cause failure of the Steam Generator inlet check valves	26.0
l) 4160V Bus 1A fails	18.4

We also found that there were 32 additional basic events with RAW greater than 1.4. They are listed in Table II.

TABLE II. RAW with respect to CDF: $1.4 < RAW < 14$

BASIC EVENT	RAW_{CDF}
a) Diesel Generator B fails	12.1
b) Control rods remain energized	11.0
c) Diesel Generator A fails	11.0
d) Operator failure to depress below Steam Generator relief valve setpoints	8.89
e) Failure to recover offsite power before battery depletion	4.83
f) Failure to isolate faulty Steam Generator	4.26
g) Ruptured Steam Generator isolations fail	4.23
h) Turbine Boundary Valves and Condenser fail to cooldown the Reactor Coolant System	3.75
i) Common-cause failure of RHR suction valves	3.66
j) Operator fails to initiate RHR	3.66

k) Operator fails to isolate Reserve Water Storage Tank	3.65
l) RHR hotleg discharge valve A fails	3.65
m) RHR hotleg discharge valve B fails	3.65
n) PORV 1 fails to reclose	3.52
o) Operator failure to initiate cooldown below RHR tolerances	3.40
p) RCP seals fail	3.30
q) Common-cause failure of RHR to both High Pressure Injection isolation valves	2.93
r) Common-cause failure of sump recirculation valves	2.93
s) Common-cause failure of Reserve Water Storage Tank isolations	2.93
t) Sump failure	2.92
u) Operator failure to initiate High Pressure Recirculation	2.89
v) PORV 1 fails to open	1.96
w) Common-cause failure of Auxiliary Feedwater Motor-driven Pumps	1.87
x) Auxiliary Feedwater steam supply valves fail	1.86
y) Common-cause failure of Turbine Driven Pump steam supply valves to open	1.85
z) Turbine Driven Pump	1.84
aa) Operator fails to identify a SGTR	1.77
ab) Operator fails to initiate feed and bleed	1.77
ac) Operator fails to initiate Reactor Coolant System depressurization	1.76
ad) RHR Motor Driven Pump B fails	1.59
ae) RHR Motor Driven Pump A fails	1.53
af) Operator fails to manually scram reactor	1.43

V.B: Event RAW with respect to ΔCDF

In our case study, the decision would be affected if model uncertainties moved the representative point into Region I or Region II; so we are interested in the vertical boundary between Region III and Region II at about 10^{-6} ry^{-1} and between Region II and Region I at about 10^{-5} ry^{-1}. ΔCDF_{mean} was given in the licensee's application as 6.9×10^{-7} ry^{-1}. Our thresholds, $\Delta CDF_{threshold}$ of 10^{-6} ry^{-1} and $\Delta CDF_{threshold}$ of 10^{-5} ry^{-1}, yield $RAW_{\Delta CDF,threshold}$ values of 1.4 and 14, respectively. Therefore, any basic event with a RAW with respect to the ΔCDF greater than 1.4 has the potential to change the decision. If this RAW is 14, then the potential to change the decision is much higher because the representative point would be in Region I.

TABLE III. RAW with respect to ΔCDF: RAW>14

BASIC EVENT	$RAW_{\Delta CDF}$	RAW_{CDF}
a) Failure to isolate faulty Steam Generator	55.5	4.26
b) Mechanical failure of Steam Generator isolations	55.0	4.23
c) 4160V Bus 1B fails	**44.2**	**197**
d) Common-cause failure of RHR pumps	**35.6**	**134**
e) Failure to initiate High Pressure Recirculation	33.6	2.89
f) Common-cause failure of RHR supply to High Pressure injection isolation valves	33.6	2.94
g) Common-cause failure of sump recirculation valves	33.6	2.94
h) Common-cause failure of RHR Reserve Water Storage tank isolation valves	33.6	2.94
i) Sump failure	33.4	2.93
j) Common-cause failure of RHR Heat Exchangers	**32.8**	**134**

Using SAPHIRE, we determined that there were 10 basic events that had a RAW with respect to ΔCDF greater than 14. They are listed in Table III.

For each of these basic events, the RAW with respect to CDF, as calculated in Section IV.A, is provided for comparison. Out of these 10 events deemed important to the licensing basis decision because of their RAW with respect to ΔCDF, only three were identified as important because their RAW with respect to CDF exceeded 14. These are emphasized with bold font in Table III. This implies that although , individually, uncertainty in the remaining seven basic events cannot be sufficient to move the licensee's CDF horizontally into Region I of the acceptance guidelines, they each have uncertainty that may be sufficient to move the change vertically into Region I.

We also found that there were 28 additional basic events that had RAW with respect to ΔCDF greater than 1.4. They are listed in Table IV.

TABLE IV. RAW with respect to ΔCDF: 1.4<RAW<14

BASIC EVENT	$RAW_{\Delta CDF}$
a) Common-cause failures of High Pressure Injection flowpath	4.45
b) High Pressure Injection cold leg injection valve fails	4.38
c) Common cause failure of Reactor Coolant System cold leg discharge check valves	4.38
d) High Pressure Injection serial component failures	4.23
e) Common-cause failure of High Pressure Injection discharge check valves	4.16
f) 4160V Bus 1A fails	2.79
g) Reserve Water Storage Tank not available	2.35

h) Scram breakers fail to open	2.34
i) Operator fails to diagnose SGTR	1.94
j) Operator fails to initiate depressurization	1.93
k) Operator fails to throttle High Pressure Injection to reduce pressure	1.91
l) Common-cause failure of Chemical & Volume Control discharge valves A and B	1.70
m) Common-cause failure of Chemical & Volume Control discharge valves C and D	1.70
n) Charging system discharge check valves fail	1.70
o) Charging system suction check valves fail	1.70
p) Common-cause failure of charging pumps	1.70
q) Common-cause failure of Chemical & Volume Control suction valves	1.61
r) Common-cause failure of VCT isolation valves	1.61
s) Common-cause failure of Chemical & Volume Control pump check valves	1.60
t) RHR Motor Driven Pump B fails	1.45
u) RHR discharge valve B fails	1.43
v) Sump isolation valve B fails	1.43
w) Reserve Water Storage Tank isolation valve B fails	1.43
x) RHR discharge A fails	1.43
y) Sump isolation valve A fails	1.43
z) Reserve Water Storage Tank isolation valve A fails	1.43
aa) RHR Motor Driven Pump A fails	1.42
ab) Fail to depressurize due to hardware	1.40

V.C: Combined Importance with respect to CDF and ΔCDF

Basic events that have high RAW values with respect to both CDF and ΔCDF are especially important because their uncertainty can move the representative point both horizontally and vertically in Figure 1 simultaneously. In our case study, a factor of 14 increase in CDF or a factor of 14 increase in ΔCDF was sufficient to move the point into Region I. However, a factor of 1.4 increase in CDF in combination with a factor of 1.4 increase in ΔCDF would also move the representative point into Region I. This is because the ΔCDF required to enter Region I changes, depending on the value of CDF. If CDF is greater than about 10^{-4} ry^{-1}, then ΔCDF must be below about 10^{-6} ry^{-1} to remain out of Region I. Otherwise, ΔCDF may be as large as 10^{-5} ry^{-1}. This threshold RAW value of 1.4 is an order of magnitude lower than the previously required threshold RAW values of 14. Therefore, basic events with uncertainties that affect both the CDF and ΔCDF, but have a relatively weak effect on each, can still affect the licensing basis decision. The fact that the factor of 14 increase required is the same for both CDF and ΔCDF is purely coincidental.

For basic events that were important with respect to both CDF and ΔCDF, we divided them into three categories. There were three basic events that had both a RAW with respect to CDF and a RAW with respect to ΔCDF greater than 14. They are listed in Table V.

TABLE V. RAW with respect to CDF and RAW with respect to ΔCDF > 14

BASIC EVENTS	$RAW_{\Delta CDF}$	RAW_{CDF}
a.) 4160V Bus 1B fails	44.2	197
b.) Common-cause failure of RHR pumps	35.6	134
c.) Common-cause failure of RHR heat exchangers	32.8	134

In addition to those listed in Table V, there were 11 basic events that had both a RAW with respect to CDF and a RAW with respect to ΔCDF greater than 1.4. They are listed in Table VI.

TABLE VI. RAW with respect to CDF and RAW with respect to ΔCDF > 1.4

BASIC EVENTS	$RAW_{\Delta CDF}$	RAW_{CDF}
a) Operator failure to isolate a faulty Steam Generator	55.5	4.26
b) Ruptured Steam Generator Isolation Failures	55.0	4.23
c) Operator fails to initiate High Pressure Recirculation	33.6	2.89
d) Common-cause failure of RHR supply to High Pressure Injection valves	33.6	2.94
e) Common-cause failure of sump recirculation valves	33.6	2.94
f) Common-cause failure of Residual Heat Removal Reserve Waster Storage Tank Isolation valves	33.6	2.94
g) Sump failure	33.4	2.93
h) 4160V Bus 1A fails	2.79	18.4
i) Reserve Water Storage Tank not available	2.35	112
j) Scram breakers fail to open	2.34	202
k) Operator fails to diagnose SGTR	1.94	1.77
l) Operator fails to initiate depressurization	1.93	1.76
m) Operator fails to throttle High Pressure Injection to reduce pressure	1.91	1.77
n) RHR Motor Driven Pump 1B fails	1.45	1.60
o) RHR Motor Driven Pump 1A fails	1.42	1.53
p) Failure to depressurize due to hardware failure	1.40	218

CHAPTER VI: EVALUATION

Thus far, we have identified basic events whose probability has the potential to adversely affect the decision. Next, it must be determined whether there are model uncertainties associated with these basic events that can actually lead to a different decision. We start by matching the generically important model uncertainties from the literature with the basic events identified in the tables and then determine how far the probability of each basic event would have to shift in order to impact the decision. Expert opinion must be used to determine whether or not the required shift is reasonable. We provide some analysis here, but this is an area that requires further research, perhaps building upon the ideas presented in Chapter III.

The basic events in Table I had high RAW with respect to CDF, sufficient to move the representative point from the case study horizontally into Region I. Table I was only concerned with the effects of uncertainty on CDF, without regard to their effect on ΔCDF. Basic events b), f), g), i), j), and k) in Table I are all similar in that they refer to a common-cause failure mechanism. Basic event b) in Table I, "Common-cause diesel generator failure," is particularly interesting because it ranks second in importance with respect to CDF and is described in the literature as generally important. Specifically, the modeling of diesel generator mission time and recovery are important model uncertainties.[25] These uncertainties are related to how long the diesel is assumed to be needed in order to accomplish its mission, and how probabilities of recovering a failed diesel generator are calculated. In the model used for this analysis, a single bounding mission time of four hours was chosen and diesel generator failure probabilities were calculated using this value. Therefore, if an event required a mission time shorter than four hours, the calculated failure probability would be conservative. If an event required longer than four hours of operation, the failure probability would be optimistic. The mission time must therefore be chosen to bound reasonable mission time requirements in order to maintain conservatism. However, the reasonableness of this bound may change if, for example, confidence in the reliability of the municipal electric grid changes.

Modeling of diesel generator field flashing success probabilities have also been identified as generally important to diesel generator failure rates.[21] Because of diesel generator design, field flashing is a necessary component of a generator's ability to produce electricity and, therefore, has a large impact on diesel generator failure rates. During a station blackout, where offsite power and emergency A/C power have been lost, diesel generator field flashing power is drawn from station batteries. Therefore, the duration the battery is capable of supplying power before it is depleted is also a factor in determining diesel generator failure rates. Battery depletion time has also been identified as an important model uncertainty.[21]

Sensitivity to these model uncertainties can be found by varying the specific assumptions related to each. For example, the modeling of diesel generator field flashing is important. One could look at the model used to determine the probability that field flashing would fail, and question the assumptions that it makes. In lieu of this, we vary the failure rate of the basic event, effectively using the failure rate as a proxy for the modeling assumptions that went into its determination.

The basic event that we are concerned with here is "Common-cause diesel generator failure." Since there are two diesel generators, the failure rate of this event is calculated as follows:

$$\lambda_c = \beta\left(\lambda_{fts} + \lambda_{ftr} * t\right) \tag{14}$$

where λ_c is the total common-cause failure rate, λ_{fts} is the rate of independent diesel generator failure to start on demand, λ_{ftr} is the rate of independent diesel generator failure while running per hour, t is the mission time in hours, and β is the beta-factor, defined as:

$$\beta = \frac{\lambda_c}{\lambda_i + \lambda_c} \tag{15}$$

where λ_c is the common-cause failure rate and λ_i is the independent failure rate (represented by the sum of λ_{fts} and λ_{ftr}*t). Looking at equation (14), we see that the common-cause failure rate has two components, β and the independent failure rate. In the PRA, β is 0.038, λ_{fts} is $3.0*10^{-2}$/demand, λ_{ftr} is $2.0*10^{-3}$/hour and the mission time is four hours. The common-cause failure rate is, therefore,

$$\lambda_c = 0.038*[3.0*10^{-2}/\text{demand} + (2.0*10^{-3}/\text{hour})(4\text{ hour mission time})] = 1.44*10^{-3}\text{ per mission} \tag{16}$$

We tested the sensitivity of the CDF to variations in λ_c and found that a factor of 35 increase would change the common-cause failure rate to 0.051 and place the representative point in Region I of the acceptance guidelines. The question now is whether this increase is reasonable. Since λ_c is a product of two variables, we must question the values used for each variable in order to question to value used for λ_c.

We first look at the value of β. In the PRA, β is 0.038. The value of 0.10 is often used as a generic value for β. The NRC's Common-Cause Failure Database (CCFDB)[28] provides common-cause failure data from industry-wide operational experience. It provides failure-while-running and failure-to-start data for diesel generators. In this database, the value of β for the failure-while-running case has a mean of 0.0370 and a 95th percentile of 0.0499. β for the failure-to-start has a mean of 0.0263 and a 95th percentile of 0.0370. These are lower than the generic value of 0.10, indicating that diesel generators are somewhat robust with regard to common-cause failures. This is due to the fact that diesel generators are well known to be risk-important and focused efforts have been made to minimize the fraction of common-cause failures. Notably, the Station Blackout rule, 10 CFR 50.63, established the Emergency Diesel Generator reliability program. The value of β in the PRA is consistent with the CCFDB values.

We next look at the value of λ_i, the independent diesel generator failure rate. In our case study, λ_i is quantified as,

$$\lambda_i = 3.0*10^{-2}/\text{demand} + (2.0*10^{-3}/\text{hour})(4 \text{ hour mission time}) = 0.038 \quad \text{per mission} \qquad (17)$$

The failure rate is a function of the probability that the diesel generator fails to start and the probability that it fails to run for the mission time. We compared the probabilities used in the case study with the probabilities used in representative PWR PRAs and found them to be consistent. However, these sources used the same NRC Accident Sequence Evaluation Program (ASEP) database, which uses industry-wide accumulated data. Plant-specific failure rates may vary considerably from the industry averages. Also, the electrical grid outage of August 14, 2003 has raised issues as to whether the current modeling assumptions are sufficient,[29] specifically, assumptions on the time to recover offsite power. Because of the extent of this outage, recovery times at some plant were quite long, raising concern that current recovery times that are modeled may not be long enough. Therefore, the basic event failure probability may be optimistic.

The important model uncertainties were diesel generator mission time and recovery modeling, diesel generator field flashing modeling, and battery depletion time modeling. Mission time modeling assumptions may be optimistic in our case study because of the recent question of whether mission times are long enough. Diesel generator field flashing and battery depletion time modeling may be optimistic in our case study for the same reasons. Longer outages require longer battery depletion times in order to prevent a station blackout. Also, the battery is required to supply diesel generator field flashing power.

We have concluded that a β value of 0.037 is reasonable. So, in order for the common-cause failure rate to be 0.051 (sufficient to affect the decision), the independent failure rate would have to increase to

$$\lambda_i = \frac{\lambda_c}{\beta} = \frac{0.051}{0.038} = 1.34 \text{ per mission} \tag{18}$$

This value is clearly unrealistic, therefore, model uncertainties regarding the diesel generators do not appear to be capable of affecting the decision.

Looking at the basic events that were important with respect to ΔCDF in Table III, we see that three of the uncertainties involve known important model uncertainties. They are listed in Table VII with their associated model uncertainties.

TABLE VII. Associating Basic Events with Model Uncertainties

BASIC EVENT	$RAW_{\Delta CDF}$	ASSOCIATED MODEL UNCERTAINTY
Failure to isolate faulty Steam Generator	55.5	Human reliability- failure to isolate faulty Steam Generator[22]
Failure to initiate High Pressure Recirculation	33.6	Human reliability- Switch ECCS from injection to recirculate[22]
Failure of sump	33.4	Sump plugging and pool strainer plugging modeling[25]

Basic event a) in Table III, "Failure to isolate faulty Steam Generator" is recognized as having the potential to be a risk-important human action after a Main Steam leak or a Steam Generator Tube Rupture initiating event. In the PRA, the conditional probability that this action will not be done when needed is 10^{-3}. For this analysis, we must increase this failure probability by a factor of 250 (thus making it 0.25) to achieve a RAW with respect to the ΔCDF of 14, thus placing the representative point in Region I. This same factor of 250 increases the CDF by only a factor of two. It is apparent, therefore, that the effect of an uncertainty on the CDF and the ΔCDF can be quite different.

Uncertainty in human reliability is well known to be important. Inputs to human reliability models, such as performance shaping factors, are difficult to quantify, the models are sensitive to these inputs, and different human reliability models with the same inputs may produce failure rates that span orders of magnitude. In the European Commission's Human Factors Reliability Benchmark Exercise,[30] 15 teams of analysts from different countries were asked to calculate human reliability for the crew's response to an operational transient at a nuclear power plant. One team produced results using different models ranged from about $1.5*10^{-2}$ to about $3.5*10^{-1}$. Across teams, results using the same model ranged from about $6*10^{-3}$ to about $3.5*10^{-1}$. In order for the "Failure to isolate faulty Steam Generator" basic event to be important to the decision, the probability of not performing this action would need to change from one in 1000 to one in four. To assess whether or not an error probability of 0.25 is reasonable for this event, one would need to look at operator training, time available, and other performance shaping factors. Such a high probability, however, does appear to be unreasonable.

Basic event e) of Table III, "Failure to initiate High Pressure Recirculation" is another human action and has the potential to be important to the decision. In the PRA, the conditional probability that this action will not be done when needed is 10^{-3}. For the analysis, we increased this failure probability by a factor of 400 to achieve a RAW with respect to the ΔCDF of 14, placing the representative point in Region I. This corresponds to a failure rate of 0.4 per demand. As was the case in basic event a) of Table III, an analysis is required to determine whether this error probability is reasonable, although we expect it to be unreasonable.

Basic event i) of Table III, "Sump failure" has model uncertainties that have the potential to be important to the decision. There has been significant debate over even whether sufficient data exists to measure sump performance. The PRA assigns a failure probability of 5×10^{-5}. This would need to increase by a factor of 8,500 to a failure probability of about 0.4 in order to impact the decision.

Looking at the basic events that have high values of RAW with respect to both CDF and ΔCDF from Table V, we see that basic event b) "Common-cause failure or RHR pumps" is risk-important. There are two RHR pumps. In the PRA, β is 0.15, λ_{fts} is $3.0*10^{-3}$/demand, λ_{ftr} is $3.0*10^{-5}$/hour and the mission time is 24 hours. The common-cause failure rate is, therefore,

$$\lambda_c = 0.15*[3.0*10^{-3}/\text{demand} + (3.0*10^{-5}/\text{hour})(24 \text{ hour mission time})] = 5.58*10^{-4} \text{ per mission} \quad (19)$$

We tested the sensitivity of the CDF to variations in the common-cause failure rate and found that a factor of 20 increase would change the common-cause failure rate to 0.011 and place the representative point in Region I of the acceptance guidelines. Since λ_c is the product of two variables, we must question the values used for each variable in order to question the value used for λ_c.

We first look at the value of β. The CCFDB lists a β for RHR pump failure-while-running having a mean of 0.0464 and a 95^{th} percentile of 0.0653. β for the failure-to-start has a mean of 0.0362 and a 95^{th} percentile of 0.0598. The value of β used in the case study application, 0.15, is considerably higher than the CCFDB database values. Therefore, the basic event probability may be very conservative. We next look at the value of independent RHR pump failure rate. We compared the probabilities used in the case study with the probabilities used in representative PWR PRAs and found them to be consistent. Unlike the diesel generator common-cause failure basic event, where there were several model uncertainties, common-cause failure modeling is the only model uncertainty that applies to the RHR pump common-cause failure basic event. We conclude that the factor of 20 increase necessary to affect this decision is not reasonable.

There were two other basic events listed in Table V. Basic event a) "4160V Bus 1B fails" had no associated model uncertainties that were identified in the literature review as generically important. An analysis of basic event c) from Table V, "Common-cause failure of RHR heat exchangers" produced similar results to that of the RHR pump common-failure analysis and is probably not important to this licensing basis change decision.

CHAPTER VII: POLICY ANALYSIS

Besides the technical aspects of including uncertainty in risk-informed decision-making, there are several policy aspects. A good policy analysis requires the inclusion of social values. Therefore, we must step back from the topic of how to include uncertainty in decision-making and ask the broader related policy questions, such as whether we should be expending resources in support of nuclear power, and if so, is the methodology presented in this thesis an appropriate way. One might argue that nuclear power is a generally poor method to produce electricity and that resources would be better spent on other alternatives. While a convincing framework in support of commercial nuclear power is not the topic of this thesis, a belief that nuclear power should continue as an industry, supported by regulation, is a necessary prerequisite. Therefore, we look at a few general societal issues related to the commercial use of nuclear power for electricity generation. We then look at how the methodology presented in this thesis can aid in addressing these issues.

VII.A: Opposition to Nuclear Power

Nuclear power was once hailed as a potential source of inexpensive electricity. On September 16th, 1954, Admiral Lewis L. Strauss, then Chairman of the US Atomic Energy Commission said that "'it is not too much to expect that our children will enjoy electrical energy in their homes too cheap to meter." Since then, nuclear energy has had its setbacks. So, the question of whether nuclear power should be a part of the U.S. energy mix is fundamental. Several oppositions have been raised, which are discussed here.

Opposition 1: There must be an acceptable solution to the disposal of high-level radioactive waste, or spent nuclear fuel. At present, this waste is stored onsite at commercial nuclear power plants. As spent fuel pools reach capacity, the waste has been transferred to dry casks, which are nominally licensed for 20 years. These are temporary measures intended only for use until a permanent repository is approved. Yucca Mountain in Nevada is the currently proposed solution. However, significant technical questions remain as to whether spent nuclear fuel can be safely stored for millennia, or even how many millennia should be considered sufficient. The effects of processes such as spent fuel container degradation mechanisms and groundwater transport parameters become highly uncertain over these time frames and when measuring transport of liquids over distances measured in tens of miles. There is also great concern for the risk of terrorist attacks at a national repository containing the majority of the nation's high-level nuclear waste. Quantification of the probability of a terrorist attack at a particular site is difficult, and very large uncertainty bounds are used. Therefore, an effective treatment of uncertainty is a necessary part of generating a defensible analysis.

Opposition 2: The public must be ensured that commercial nuclear power plants can be operated safety. While these plants have enjoyed an enviable safety record to date, there are still many concerns. The Union of Concerned Scientists[31] details several of their concerns. The effects of aging on nuclear plant components and the integrated effects on risk of the simultaneous aging of these components could present an increase in risk that has not been expected, since the earliest built plants are just now reaching the ends of their original 40-year licenses. Nuclear plants may be subjected to sabotage with associated contributions to CDF that are difficult to quantify. Operators may not be performing their duties in accordance with approved procedures. As the risks associated with the human actions assume that they are done according to procedure, the risk may be higher than modeled. Since it is unknown exactly how each human action is performed, it is impossible to model the risk contribution quantitatively.

There may also be accident sequences (cutsets) that exist and have not been identified. These cutsets contribute to risk just as any other, but are not included in the calculated CDF. Again, it is difficult to quantify the contribution of a cutset that has not been identified. Finally, latent design and construction errors may exist. As latent errors, they have not been identified, but still add to the plant's risk by making the plant more susceptible to natural disasters or sabotage. These risk contributions have not been identified and are difficult to quantify analytically. The above issues all have one thing in common; any sort of quantification is difficult and the estimates of these risks have large associated error bounds. Similar to Opposition 1, a defensible method of treating this uncertainty becomes critical to performing a defensible analysis.

Opposition 3: Commercial nuclear power production today requires the enrichment of Uranium in order to increase the fraction of the Uranium-235 isotope. In addition, the fission process produces a number of isotopes, including Plutonium-239. Both enriched Uranium-235 and Plutonium-239 can be used to produce nuclear weapons. Therefore, the use of nuclear power for peaceful purposes such as energy production has a negative impact on efforts to prevent the proliferation of nuclear weapons. Reprocessing complicates the issue. Spent nuclear fuel can be reused as fuel if it is reprocessed. This is done by separating out the fissionable components, including the Plutonium-239. The Plutonium is mixed into new fuel and reused, while the remainder of the spent fuel is stored for disposal. Since the Plutonium is separated out during reprocessing, and it is the Plutonium that is sought after to be used in nuclear weapons, reprocessing makes the nuclear fuel cycle more susceptible to proliferation.

Opposition 4: Electricity from nuclear power must be cost-competitive with electricity from other sources. The validity of this opposition depends highly on the choice of source data.

The most recent nuclear power plant in operation began construction over 20 years ago during a time when generators were predominantly a part of regulated utilities. These utilities received a rate-of-return, reducing some of the incentive to tightly control costs. Licensing of new nuclear plants also took place in two steps. A license was authorized for the construction of the plant. Once it was built, a separate license was issued for its operation. Given the complexity of licensing a nuclear power plant, this process was very lengthy, adding considerable expense to startup costs.

Since then, several things have changed. Much progress has been made in deregulating generators. They must now survive on the free market. This creates a strong market incentive to reduce costs. Figure 4 compares different types of generation, and shows the fraction of capital costs in nuclear power is far greater. This makes the cost-competitiveness of nuclear power highly dependent on the ability to control capital costs.

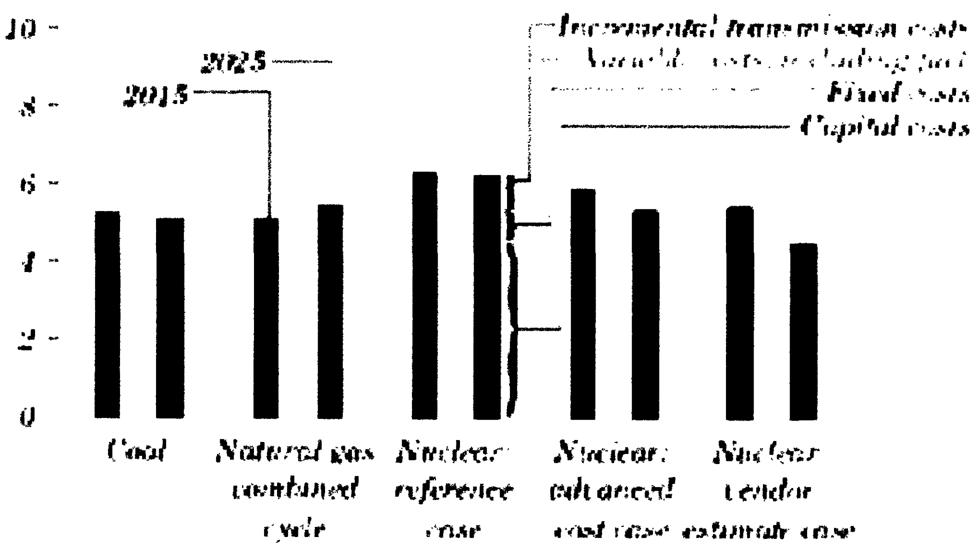

Figure 4. Levelized electricity costs for new plants by fuel type, 2015 and 2025[32]

The NRC has created a combined licensing (COL) process where many of the requirements for construction and operation licensing have been combined. The COL serves as a license for construction and a conditional license for operation. Before a generator secures a loan for the construction of a plant, a COL provides some assurance that operation will be allowed once the plant is constructed. This has the anticipated effect of reducing the approval time that must take place between the end of construction and the beginning of operation. This has the potential to decrease capital costs by reducing the number of years that interest on a project loan must be capitalized. However, the effects of these changes have not been proven, since construction on any new commercial nuclear plant has yet to begin in the U.S. Also, because of the length of time since any nuclear plants have been built in this country, learning effects have likely been reduced. This is likely to have a detrimental effect on construction costs.

VII.B: Support for Nuclear Power

Support 1: Energy is a critical infrastructure in the U.S., and as such, is vital to national security. Energy security is a major justification for promoting stability in the Middle East, including the war in Iraq in which over $100 billion has been spent. Its importance can be seen in the singular vision of the Federal Energy Regulatory Commission (FERC), "Dependable, affordable energy through sustained competitive markets." The importance of energy costs can also be seen in U.S. reluctance to commit to greenhouse reductions as part of the Kyoto Protocol. Economic costs are commonly stated as a reason for noncommittal.

Nuclear power supplies about 20% of the electricity in the U.S. and is the second-largest source of electricity after coal. This makes electricity from nuclear power an important part of the U.S. energy mix. In addition, attitudes towards nuclear power in other countries have been seen to have a striking effect on energy independence.[33] In Italy, the decision was made in 1987 to shutdown the nuclear power plants. As of 2002, Italy imports 51.5 billion kilowatt-hours (kWh) out of 294 kWh it consumes, or 17.5%. Germany has committed to early shutdown of its nuclear reactors. In 2003, Germany imported 45.8 billion kWh out of the 520 kWh it consumed, or 8.8%.

At the other extreme, France relies heavily on nuclear power and produces the largest fraction of its electricity from nuclear power compared with any other nation. They are also a net exporter of electricity. In 2002, France exported 79.9 billion kWh out of the 528.6 billion kWh that it produced, or 15.1%. From these examples, it seems that electricity from nuclear power may play a significant role in a nation's dependence on electricity imports.

The U.S. Energy Information Agency expects that the demand for electricity will increase at an average rate of 1.9% per year in the U.S. until the year 2025, as shown in Figure 5. Nuclear power may be a vital component of achieving this growth while maintaining a mix of fuel sources and preventing over-reliance on electricity imports.

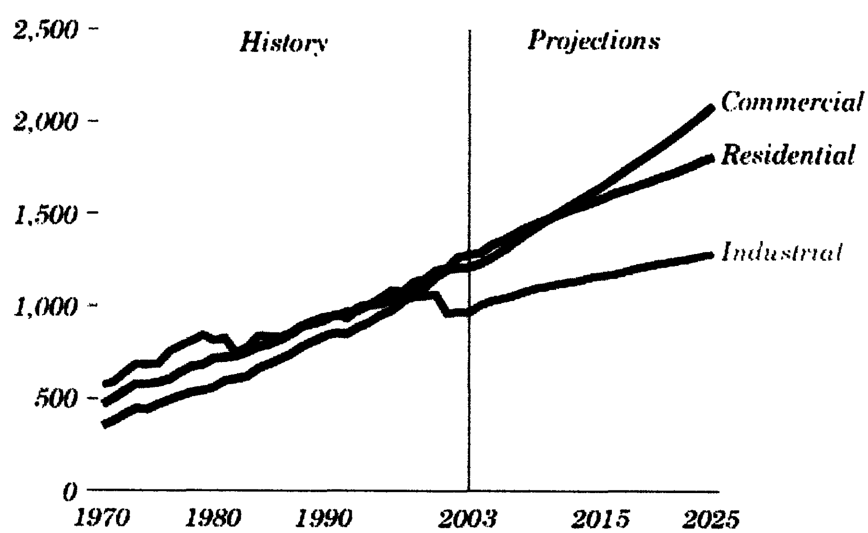

Figure 5. Annual electricity sales by sector, 1970-2025 (billion kilowatt-hours)[32]

Support 2: Electricity from nuclear power gains advantage over other sources as concern for global warming increases. There is growing international concern over the effects of global warming, such as rising ocean levels and the destructive power of increasingly erratic weather patterns. There is also a large consensus that much of global warming can be attributed to the anthropomorphic emissions of greenhouse gasses, of which carbon dioxide emissions from the burning of fossil fuels is a large contributor. However, electricity demand is tightly linked with economic growth.

In order to achieve acceptable economic growth while minimizing the effects of global warming, one must generate a sufficient amount of dependable, affordable electricity while minimizing the emissions of greenhouse gasses, such as carbon dioxide. Under these conditions, nuclear power has considerable advantage over other sources. A nuclear power plant can produce a massive amount of electricity and produces almost no greenhouse gasses. The electricity from a nuclear plant is also very dependable and while its cost-competitiveness is subject to debate, Figure 4 shows that the levelized cost of electricity from nuclear power is cost-competitive compared with coal and natural gas combined-cycle. These costs include that of spent nuclear fuel disposal which is paid by the nuclear power producers on a per kWh basis, and exclude the effects of any carbon tax that may be implemented in the future.

VII.C. Treatment of Uncertainty

We have established that the decision whether to continue the use of nuclear power generation as an industry is one of policy and presume that the industry will continue to exist. One must then decide whether the methodology to include model uncertainty in risk-informed decision-making is an appropriate way to support the nuclear power industry. Looking at the four oppositions to nuclear power presented in Section VII.A., we see that two of them are directly the result of uncertainty. Specifically, these were the uncertainty in operational nuclear plant safety analysis and the uncertainty in spent nuclear fuel repository safety analysis.

Uncertainty is also a part of the remaining oppositions. The risk of proliferation is not well quantified. There are large uncertainties in how the presence of nuclear power for peaceful purposes impacts the threat of proliferation, to what extent nations commit resources in the pursuit of nuclear weapons, and in the level of assurance that key technologies can be controlled to prevent enriched Uranium and Plutonium from being used to create nuclear weapons. With regards to the cost-effectiveness of nuclear power generation, uncertainty in the ability of a nuclear power plant to remain operational and generate revenue affects the interest rate that developers must pay to finance nuclear power projects. Uncertainty in the future costs of fossil fuels and uncertainty in whether a carbon tax or emissions trading scheme might be imposed affect the future cost of electricity from fossil fuels, thereby affecting the relative cost of electricity from nuclear power.

An effective treatment of uncertainty will promote the NRC's policy[4] on using PRA in decision-making. This policy states that "the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach." The methodology proposed in this thesis supports this policy by helping the regulation to become more "rationalist,[5]" relying more heavily on quantitative processes and moving away from the predominantly "structuralist" system that is present today.

Therefore, an effective treatment of uncertainty is beneficial to nuclear regulation. The proposed methodology attempts to aid in the inclusion of model uncertainty in decisions on whether to approve risk-informed licensing basis changes at nuclear power plants. Fundamentally, it involves the identification of potentially significant uncertainties, sensitivity studies to determine the probability of an event at which the risk becomes too great, and an analysis of whether this required probability is reasonable. When done with transparency, this process allows the public to see that identified uncertainties are being treated in a systematic, formal way. This may improve the public's confidence that nuclear power plants are being regulated in an effective manner and that an acceptable level of safety is assured. Therefore, this methodology helps to address the societal issues related to nuclear power generation and aids the NRC's policy on the use of PRA in regulatory activities.

VIII: CONCLUSIONS

We have sought to identify basic events where the value of their probability can change the decision, and are known to have significant model uncertainty. We focused on Level I, at power, internal events PRAs, and the decision making process related to licensing basis changes. The acceptance guidelines with respect to a plant's CDF and ΔCDF of a proposed change have been clearly defined by RG 1.174 and the need to address all uncertainties in the decision-making process has been established. Once the basic events of interest are identified, they are analyzed to determine what their probability would need to be to affect the decision. Then, an analysis using expert opinion must be used to determine if this change is reasonable. We referred to several methods to accomplish this and provided an example.

In our case study, a total of 12 basic events had RAW with respect to CDF showing that their uncertainty could place the licensing basis change's representative point in Region I of the acceptance guidelines, in which case the change would generally not be approved. The model uncertainties in one of these basic events have been found to be important in a review of the literature. 10 basic events have RAW with respect to ΔCDF showing that their uncertainties could place the change's representative point in Region I. Of these, three have been found in the literature review. Two basic events were common to both lists, showing high importance with respect to both the CDF and ΔCDF. Therefore, a total of 20 basic events were identified as important.

The decision seems to be fairly insensitive to uncertainties in all but one of these basic events. In order to move the representative point into Region I, the probabilities of "failure to isolate faulty Steam Generator," "failure to initiate High Pressure Recirculation," and "failure of sump" would need to increase considerably. An evaluation of the reasonableness of the increases would be required. The decision is much more sensitive to uncertainty in the basic event "Common-cause diesel generator failure." A factor of 35 increase in the failure probability of this basic event has the effect of moving the representative point such that the change would probably not be approved.

We also performed a sensitivity of success criteria related to Auxiliary Feedwater pumps for illustration, where we changed the assumption that one feedwater was sufficient to ensure success to an assumption that either the turbine-driven pump or both motor-driven pumps were required. The alternative assumption produced a CDF of $6.85*10^{-5}$, or 0.36% higher than the baseline case.

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APPENDIX A: CASE STUDIES

Three license base change requests, issued by nuclear power plants in the United States during the last five years, provide insights on important sources of uncertainty. These case studies include:

A PWR risk-informed In-Service Testing (IST) program
A BWR risk-informed In-Service Inspection (ISI) program
A PWR risk-informed Technical Specification (TS) Change for the Emergency Diesel Generators (EDG) Allowed Outage Time (AOT)

It should be noted that these requests were issued prior to the release of the NRC's RG 1.200, which established guidelines on determining the technical adequacy of PRAs. Therefore, the primary guidance for approval of risk-informed applications came from earlier NRC documents, namely Regulatory Guides 1.174 through 1.178.

A.1: A PWR Risk-Informed IST Program

A PWR submitted a licensing basis change application to extend the test interval of 160 valves in 10 systems from the frequency of once per quarter, as specified in the ASME Code, to a new frequency of once per refueling cycle, or once per eighteen months. The failure rate was assumed to be linearly proportional to the test interval. The list of valves was compiled to include good test performers that were either modeled in the plant's PSA model or similar in plant configuration to valves that were modeled. Good performance was based on a low rate of corrective maintenance and a low failure rate for the previous three to five years.

Valves that were only used during refueling and cold shutdown operations were removed from consideration because this was an at-power application. Motor-operated valves were removed from consideration because there was a separate initiate to risk-inform their maintenance requirements. The application also stipulated that the list of valves that would be covered under this risk-informed IST program would be reevaluated every second refuel.

The baseline risk at the plant and the change in risk attributed to the proposed change were as follows:

Table VIII. CDF and LERF Expected Values for Case Study 1.

	Baseline	Change
CDF	$3.80*10^{-5}$	$2.0*10^{-7}$
LERF	$4.45*10^{-6}$	$1.1*10^{-8}$

This license base change request was not approved for a number of reasons. The safety significance of valves was not considered when valves were considered for the IST program. The component of risk while the plant was in a shutdown configuration was not addressed. While risks associated with plant configurations other that at-power were previously considered negligible, these risks were later found to be significant and risks associated with all plant configurations may need to be addressed.

Whether this plant's PRA model was peer-reviewed was not addressed in the application, which raised concerns as to the accuracy of the model. It was also not described how the PRA model was updated as plant conditions changed due to periodic modification of equipment configurations and operating procedures. Current PRAs are required to be "living" analyses, with procedures in place describing how it is updated. Also, significant differences between the plant's original Initial Plant Examination (IPE) and the PRA used in this application existed and were not explained.

This power plant consists of two units with some systems that could be cross-tied between them. Cross-unit dependencies were not addressed in this application. These dependencies were applicable since Unit 1 could be at-power while Unit 2 could be in a refueling outage with vital, cross-unit systems out of service. Since this configuration was addressed, it created a component of risk that was not quantified.

Not all potentially important failure modes for the valves in this application were considered. Also, although the impact on CDF for each system was determined, the plant did not address the possibility that the aggregate impact of the IST program on multiple systems simultaneously could be greater than the sum of the individual impacts. In other words, there may be dependencies between the systems that could amplify and have a significant effect on the overall risk. The applicant also did not calculate the expected cumulative change in LERF. The change in LERF was calculated for systems that had the most impact on the containment, but not for the other systems that were affected by the IST program. During the NRC's safety evaluation report, the staff estimated the cumulative change in LERF from all of the systems. This estimate was higher than the applicant's estimate and placed the change's risk in Region II of the acceptance criteria. The NRC also noted that uncertainties and sensitivities to key PSA modeling parameters and assumptions were not addressed.

A.2: A BWR Risk-Informed ISI Program

A BWR submitted a licensing basis change request to reduce the number of required piping weld inspections. This request was applicable to Class 1 and Class 2 piping welds, while maintaining the ASME requirements of examination technology, examination frequency, and acceptance criteria. This request was approved after one Request for Additional Information

(RAI) by the NRC, and resulted in a 60% reduction in the number of required piping weld inspections that were covered by this ISI program.

Piping weld inspection locations were determined by determining the susceptibility of each weld to failure and by determining the severity of the consequence if the weld did fail. These characteristics were used to place in welds into a risk matrix where they were risk ranked. There were some exceptions to this process. Welds that were susceptible to Flow Accelerated Corrosion were assigned high-risk status. Welds that were susceptible to Intergranular Stress Corrosion Cracking were assigned a high- or medium-risk status. If there was no equipment to mitigate the effects of a weld failure, then the weld was assigned a high- or medium-risk. In the IST program, 25% of the high-risk welds and 10% of the medium-risk welds would be inspected.

This application followed the EPRI template for establishing risk-informed ISI programs for Class 1 and Class 2 piping weld inspections. This document is the "Revised Risk Informed ISI Evaluation Procedure," Topical Report 112657. The template is pre-approved by the NRC as a methodology. Therefore, the application is shortened for expedited NRC review.

The plant deviated from EPRI's approved methodology regarding the use of the Monte Carlo method for calculating risk impact. Because of this, the plant was required to submit detailed information regarding all formulas that were used, and what made these formulas appropriate.

The plant was credited with recognizing that multiple failure mechanisms could exist, even in a single pipe segment, and that the effect of the risk-informed ISI program would have a different effect on failure frequency dependent on the failure mechanism. Because of this, some pipe segments were removed from consideration for the program, while others were subsumed into the new ISI program.

The plant also recognized that there could be synergistic effects between multiple failure mechanisms. For some failure mechanisms, the effects were treated independently and their effects added, indicating no dependency between mechanisms. For other mechanisms, the effects were added, and then the sum was multiplied by a factor of three to account for synergies.

The plant performed sensitivity studies. They showed that even with no nondestructive examination of the welds, ΔCDF would be less than 10^{-7} and ΔLERF would be less than 10^{-8}. They also accounted for uncertainties that may come to light in the future by showing how the ISI program would be updated periodically based on new information.

A.3: A PWR Risk-Informed TS Change for the EDG AOT

A PWR submitted a licensing basis change to extend the AOT of an EDG from three days to ten days during at-power operations. This change made use of the Combustion Engineering Owners Group Joint Applications Report for EDG AOT extensions (CE NPSD-996).

The PRA was shown to be a living analysis, updated formally and routinely. The baseline risk at the plant and the change in risk attributed to the proposed change were as follows:

Table IX. CDF and LERF Expected Values for Case Study 3.

	Baseline	Change
CDF	$2.08*10^{-5}$	$4.0*10^{-7}$
LERF	$6.30*10^{-7}$	$1.2*10^{-8}$

Many compensatory measures qualitatively reduced the risk impact of this change and ensured that defense-in-depth and safety margins were maintained. An alternate A/C diesel generator had previously been added at the plant and credit had not been taken for it in the PRA. If an EDG was taken out of service during at-power operations, the alternate diesel generator would be verified available every eight hours. The alternate diesel generator and the emergency feedwater pump would be reclassified as protected train components. The alternate diesel generator would be limited to perform only safety functions. This reduced use had the intention of increasing the generators reliability. No discretionary switchyard maintenance would be allowed while an EDG was out of service during at-power operations. The system dispatcher would be informed of the alternate diesel generator's status daily. The alternate diesel generator would be manned during severe weather in order to speed its connection to the vital busses when needed. Also, no EDG would be taken out of service at-power during severe weather.

APPENDIX B: RAW WITH RESPECT TO ΔCDF CALCULATION

Part of the methodology proposed in this thesis was calculating the RAW importance with respect to the change in risk. We proposed a definition for this importance measure in Chapter IV, and a brief description of how it may be calculated using existing software. In this Appendix, we provide a step-by-step procedure for performing this calculation.

Looking back at the discussion on Chapter IV, we defined RAW with the respect to ΔCDF as defined in Equation (11),

$$RAW_{j,\Delta CDF} = \frac{(RAW_{j,CDF-after})*(CDF_{after})-(RAW_{j,CDF-base})*(CDF_{base})}{CDF_{after}-CDF_{base}}$$

where: $RAW_{j,CDF\text{-}after}$ is the RAW with respect to CDF for basic event j after the proposed change has been made.

$RAW_{j,CDF\text{-}base}$ is the RAW with respect to CDF for basic event j before the proposed change has been made.

CDF_{after} is the CDF of the plant after the proposed change has been made.

CDF_{base} is the CDF of the plant before the proposed change has been made.

Step-by-step procedure to calculate RAW with respect to ΔCDF:

This procedure uses the SAPHIRE program and Microsoft Excel and assumes the applicable plant's SPAR model has been opened in SAPHIRE.

1. In SAPHIRE, select Generate from the toolbar
2. Select Generate in the Generate window. This allows SAPHIRE to perform the calculations necessary for this procedure

3. Select End State from the toolbar
4. Select CD from the End State List
5. Right-click on CD from the End State List
6. Select Gather from the drop-down menu
7. Uncheck Cut Set Probability Truncation in the Cut Set Generation window. This allows the entire list of basic events to be listed regardless of their probability of occurrence.
8. Select OK in the Cut Set Generation window
9. Record the Total value in the Cut Set Generation Results window. This is the value for CDF_{base} to be used in our calculation.
10. Select OK in the Cut Set Generation Results window.
11. Repeat steps 4 and 5.
12. Select Display from the drop-down menu, then Importance from the drop-down submenu, then Ratio from the next drop-down submenu.
13. Select NAME in the Sort drop-down menu. This sorts all of the basic events alphabetically by their name.
14. Select the entire list of basic events and importance measures in the Importance Measures window by left-clicking on the first basic event listed, then while holding down the Shift key, left-clicking on the last basic event listed.
15. Copy the list of basic events by holding down the Ctrl key and typing the "c" key.
16. Paste the list of basic events into an Excel spreadsheet. This will create a list of all basic events, along with the number of times that they occur in cutsets, their probability of occurrence, and their Fussell-Vesely, Risk Reduction Worth, and RAW importance measures. The RAW importance measure values are the $RAW_{j,CDF\text{-}base}$ values needed for our calculation.
17. Select Generate from the SAPHIRE toolbar.
18. Create a change set which reflects the change that is proposed by the licensing basis change. This process varies according to the change that is proposed.
19. Repeat steps 1 through 16. In this case, Step 9 provides the value for CDF_{after} to be used in our calculation. Also, Step 16 provides the RAW importance measure values to be used as the values of $RAW_{j,CDF\text{-}after}$ in our calculation.

20. Organize the spreadsheet such that all of the data for a particular basic event are located on the same row. This amounts to aligning the two lists that were pasted into excel, such that the information for a basic event before the proposed change is on the same row as the information for the same basic event after the proposed change.
21. At this point, all of the necessary data is available in the spreadsheet. Using the formula tool in Excel, write a formula for each basic event that performs the calculation of Equation (11). The result of this formula is the RAW with respect to ΔCDF for the basic event ($RAW_{j\text{-}\Delta CDF}$).

APPENDIX C: CASE STUDY BASIC EVENT IMPORTANCE MEASURES SORTED BY RAW WITH RESPECT TO CDF

Basic Event	Probability	Fussell-Vesely	Risk Reduction Worth	RAW with respect to CDF	RAW with respect to deltaCDF
	Basic Event Description				
IE-LOOP	1.60E-05	8.31E-01	5.92E+00	5.19E+04	0.00E+00
	Loss of offsite power initiating event				
IE-SGTR	1.63E-06	6.71E-02	1.07E+00	4.12E+04	5.56E+05
	Steam generator tube rupture initiating event				
IE-SLOCA	2.33E-06	9.39E-02	1.10E+00	4.03E+04	3.97E+04
	Small loss of collant accident initiating even				
RPS-VCF-FO-MECH	8.90E-08	2.72E-04	1.00E+00	3.05E+03	0.00E+00
	Control rod assemblies fail to insert				
EPS-DGN-CF-ALL	1.44E-03	3.90E-01	1.64E+00	2.71E+02	0.00E+00
	Common cause failure of diesel generators				
PCS-PSF-HW	1.00E-05	2.17E-03	1.00E+00	2.18E+02	1.40E+00
	Hardware failures causing failure to depressurize				
RPS-BKR-FC-FTO	5.70E-06	1.14E-03	1.00E+00	2.02E+02	2.34E+00
	RPS breakers fail to open				
ACP-BAC-LP-1B	9.00E-05	1.77E-02	1.02E+00	1.97E+02	4.42E+01
	Division 1B AC power 4160V bus fails				
RHR-HTX-CF-AB	1.40E-05	1.86E-03	1.00E+00	1.34E+02	3.28E+01
	Failure of heat exchanges due to common cause failure				
RHR-MDP-CF-ALL	5.60E-04	7.45E-02	1.08E+00	1.34E+02	3.56E+01
	RHR pump common cause failures				
HPI-TNK-VF-RWST	2.70E-06	3.00E-04	1.00E+00	1.12E+02	2.35E+00
	RWST not available				

IE-TRANS	2.65E-04	7.90E-03	1.01E+00	3.08E+01	5.44E+00
	Transient initiating event				
AFW-PMP-CF-ALL	5.60E-05	1.43E-03	1.00E+00	2.66E+01	1.23E+00
	Common cause failure of AFW pumps (all types)				
AFW-AOV-CF-DIS	1.85E-05	4.72E-04	1.00E+00	2.65E+01	1.19E+00
	Common cause failure of SGs discharge AOVs to open				
AFW-CKV-CF-DIS	1.84E-06	4.59E-05	1.00E+00	2.60E+01	8.33E-01
	Common cause failure of AFW inlet check valves into SGs				
ACP-BAC-LP-1A	9.00E-05	1.56E-03	1.00E+00	1.84E+01	2.79E+00
	Division 1A AC power 4160V bus fails				
EPS-DGN-FC-1B	3.80E-02	4.37E-01	1.78E+00	1.21E+01	1.08E+00
	Diesel generator B fails				
RPS-VCF-FO-ELEC	4.30E-04	4.32E-03	1.00E+00	1.10E+01	9.23E-01
	Control rod drives remain energized				
EPS-DGN-FC-1A	3.80E-02	3.93E-01	1.65E+00	1.10E+01	1.12E+00
	Diesel generator A fails				
RCS-XHE-RECOVER	3.50E-03	2.77E-02	1.03E+00	8.89E+00	9.92E-01
	Operator fails to depressurize RCs below SG SRV given A				
OEP-XHE-NOREC-BD	7.40E-02	3.06E-01	1.44E+00	4.83E+00	9.83E-01
	Operator fails to recover offsite power before battery				
MSS-XHE-XM-ERROR	1.00E-03	3.27E-03	1.00E+00	4.26E+00	5.55E+01
	Operator fails to isolate faulted steam generator				
MSS-VCF-HW-ISOL	1.00E-02	3.27E-02	1.03E+00	4.23E+00	5.50E+01
	Ruptured steam generator isolation failures				
PCS-VCF-HW	3.00E-03	8.26E-03	1.01E+00	3.75E+00	3.07E-01
	TBVs/COND/Cir failures				
RHR-MOV-CF-SUCT	2.64E-04	7.02E-04	1.00E+00	3.66E+00	1.00E+00
	RHR suction valve common cause failures				
RHR-XHE-XM	1.00E-03	2.66E-03	1.00E+00	3.66E+00	1.03E+00

	Operator fails to initiate RHR system				
RHR-MOV-CC-HOTA	3.00E-03	7.98E-03	1.01E+00	3.65E+00	9.95E-01
	Discharge MOV 74-1 from hot legs fails				
RHR-MOV-CC-HOTB	3.00E-03	7.98E-03	1.01E+00	3.65E+00	9.95E-01
	Discharge MOV 74-2 from hot legs fails				
RHR-MOV-OO-RWST	3.00E-03	7.98E-03	1.01E+00	3.65E+00	9.95E-01
	Failure to isolate RWST				
PPR-SRV-OO-PRV1	3.00E-02	7.79E-02	1.08E+00	3.52E+00	1.03E+00
	PORV 1 fails to reclose after opening				
PCS-XHE-XM-RCOOL	1.00E-03	2.40E-03	1.00E+00	3.40E+00	9.98E-01
	Operator fails to initiate RCS cooldown below RHR				
RCS-MDP-LK-SEALS	1.30E-01	3.44E-01	1.52E+00	3.30E+00	1.02E+00
	RCP seals fail without cooling and injection				
HPR-MOV-CF-RHR	2.64E-04	5.10E-04	1.00E+00	2.93E+00	3.36E+00
	Common cause failure of RHR supply to HPI isolation valves CV8804A & SI8804				
HPR-MOV-CF-SMP	2.64E-04	5.10E-04	1.00E+00	2.93E+00	3.36E+00
	Common cause failure of sump recirculation MOVs				
RHR-MOV-CF-RWST	2.64E-04	5.10E-04	1.00E+00	2.93E+00	3.36E+01
	Common cause failure of RWST isolation MOVs				
HPR-SMP-FC-SMP	5.00E-05	9.58E-05	1.00E+00	2.92E+00	3.34E+00
	Failure of sump				
HPR-XHE-XM	1.00E-03	1.89E-03	1.00E+00	2.89E+00	3.36E+00
	Operator fails to initiate HPR				
PPR-SRV-OO-PRV2	3.00E-02	2.98E-02	1.03E+00	1.96E+00	1.01E+00
	PORV 2 fails to reclose of opening				
AFW-MDP-CF-AB	2.10E-04	1.83E-04	1.00E+00	1.87E+00	1.02E+00
	Common cause failures of AFW motor driven pumps				
AFW-MOV-CC-STM	3.10E-03	2.69E-03	1.00E+00	1.86E+00	9.97E-01
	Steam supply valves fail				
AFW-AOV-CF-STM	1.00E-04	8.54E-05	1.00E+00	1.85E+00	9.98E-01

	Common cause failure of TDP STM supply line AOVs to open				
AFW-TDP-FC-1A	3.26E-02	2.83E-02	1.03E+00	1.84E+00	9.72E-01
	AFW Turbine driven pump fails				
RCS-XHE-DIAG	6.80E-03	5.26E-03	1.01E+00	1.77E+00	1.94E+00
	Operator fails to diagnose SGTR to start procedures				
HPI-XHE-XM-THRTL	1.00E-02	7.73E-03	1.01E+00	1.77E+00	1.91E+00
	Operator fails to throttle HPI to reduce pressure				
RCS-XHE-XM-SG	2.10E-02	1.62E-02	1.02E+00	1.76E+00	1.93E+00
	Operator fails to initiate RCS depressurization				
RHR-MDP-FC-1B	3.90E-03	2.33E-03	1.00E+00	1.59E+00	1.45E+00
	RHR MDO 1B fails				
RHR-MDP-FC-1A	3.90E-03	2.09E-03	1.00E+00	1.53E+00	1.42E+00
	RHR motor driven pump 1A fails				
RPS-XHE-XM-SCRAM	1.00E-02	4.32E-03	1.00E+00	1.43E+00	9.94E-01
	Operator fails to manually trip the reactor				
PCS-XHE-XM-CDOWN	1.00E-03	3.52E-04	1.00E+00	1.35E+00	1.01E+00
	Operator fails to initiate cooldown				
OEP-XHE-NOREC-2H	1.20E-01	4.63E-02	1.05E+00	1.34E+00	1.02E+00
	Operator fails to recover offsite power within 2 hours				
PPR-SRV-CO-L	1.60E-01	4.78E-02	1.05E+00	1.25E+00	9.91E-01
	PORVs/SRVs open during loop (1.0 probability)				
HPI-MDP-CF-ALL	7.80E-04	1.88E-04	1.00E+00	1.24E+00	4.45E+00
	Common cause failure of HPI flowpath				
OEP-XHE-NOREC-SL	6.30E-01	3.88E-01	1.64E+00	1.23E+00	9.97E-01
	Operator fails to recover offsite power (Seal loca)				
CVC-XHE-XM-BOR	1.00E-03	2.26E-04	1.00E+00	1.23E+00	9.95E-01
	Operator fails to initiate emergency boration				
HPI-MOV-OC-DISCH	4.00E-05	8.47E-06	1.00E+00	1.21E+00	4.38E+00
	HPI cold leg injection valves MOV 63-22 fails				
HPI-CKV-CF-CL	2.96E-05	6.15E-06	1.00E+00	1.21E+00	4.38E+00

	Common cause failure of RCS cold leg discharge check valves				
PPR-SRV-CC-RCS	4.40E-04	8.28E-05	1.00E+00	1.19E+00	1.01E+00
	Relief valves fail to open to limit RCS pressure				
RCS-PHN-MODPOOR	1.40E-02	2.37E-03	1.00E+00	1.17E+00	9.94E-01
	Moderator temp coefficient not enough negative				
HPI-MOV-OC-RWST	1.40E-04	2.32E-05	1.00E+00	1.17E+00	4.23E+00
	HPI serial component failures				
HPI-CKV-CF-PMPS	1.00E-05	1.30E-06	1.00E+00	1.13E+00	4.16E+00
	Common cause failure of HPI pump discharge check valves				
SLOCA-XHE-NOREC	4.30E-01	9.22E-02	1.10E+00	1.12E+00	1.12E+00
	Operator fails to recover from A SLOCA in short term				
CVC-MDP-CF-ALL	1.51E-04	1.78E-05	1.00E+00	1.12E+00	1.70E+00
	Common cause failure of charging pumps (HPI)				
CVC-MOV-CF-DIS1	2.64E-04	3.11E-05	1.00E+00	1.12E+00	1.70E+00
	Common cause failure of CVC discharge MOVS 63-39 63-40 to open				
CVC-MOV-CF-DIS2	2.64E-04	3.11E-05	1.00E+00	1.12E+00	1.70E+00
	Common cause failure of CVC discharge MOVS 63-25 63-26 to open				
PPR-XHE-XM-BLK	1.00E-03	1.14E-04	1.00E+00	1.11E+00	1.06E+00
	Operator fails to close block valves				
CVC-MOV-CF-SUC	2.64E-04	2.99E-05	1.00E+00	1.11E+00	1.61E+00
	Common cause failure of CVC suction MOVS to open				
CVC-MOV-CF-VCT	2.64E-04	2.99E-05	1.00E+00	1.11E+00	1.61E+00
	Common cause failure of VCT isolation MOVS to close				
CVC-CKV-CC-DIS	1.00E-04	1.10E-05	1.00E+00	1.11E+00	1.70E+00
	Charging system discharge check valve fails				
CVC-CKV-CC-SUC	1.00E-04	1.10E-05	1.00E+00	1.11E+00	1.70E+00
	Charging system suction check valve fails				
CVC-CKV-CF-PMPS	1.00E-05	9.98E-07	1.00E+00	1.10E+00	1.60E+00
	Common cause failure of CVC pump discharge check valves				
PPR-SRV-CO-TRAN	4.00E-02	4.00E-03	1.00E+00	1.10E+00	1.04E+00

	PORVs/SRVs open during transient				
PPR-SRV-CO-SBO	3.70E-01	5.59E-02	1.06E+00	1.10E+00	1.01E+00
	PORVs/SRVs open during station blackout				
PPR-SRV-CC-PRV1	6.30E-03	5.86E-04	1.00E+00	1.09E+00	9.77E-01
	PORV1 fails to open on demand				
PPR-SRV-CC-PRV2	6.30E-03	5.86E-04	1.00E+00	1.09E+00	9.77E-01
	PORV2 fails to open on demand				
HPR-MOV-CC-RHRB	3.00E-03	2.56E-04	1.00E+00	1.09E+00	1.43E+00
	RHR discharge MOV 63-11 fails				
HPR-MOV-CC-SMPB	3.00E-03	2.56E-04	1.00E+00	1.09E+00	1.43E+00
	Sump isolation MOV 63-73 fails to open				
RHR-MOV-OO-RWSTB	3.00E-03	2.56E-04	1.00E+00	1.09E+00	1.43E+00
	RWST isolation MOV 74-21 fails to close				
CVC-CKV-CF-CL	1.85E-06	1.57E-07	1.00E+00	1.08E+00	1.01E+00
	Common cause failure of CVC cold leg discarge check valves				
CVC-TNK-VF-BIT	2.40E-06	2.00E-07	1.00E+00	1.08E+00	1.01E+00
	Boron injection tank unavailable				
LOOP-18-22-NREC	2.72E-01	3.07E-02	1.03E+00	1.08E+00	9.95E-01
	LOOP sequence 18-22 nonrecovery probability				
PPR-MOV-OO-BLK2	3.00E-03	2.47E-04	1.00E+00	1.08E+00	1.03E+00
	PORV 2 block valve fails to close				
OEP-XHE-NOREC-ST	5.30E-01	8.66E-02	1.10E+00	1.08E+00	9.90E-01
	Operator fails to recover offsite power in short term				
HPI-MOV-OO-RWST	3.00E-03	2.22E-04	1.00E+00	1.07E+00	1.31E+00
	HPI isolation valve from RWST fails				
HPI-MDP-FC-1B	3.80E-03	2.57E-04	1.00E+00	1.07E+00	1.01E+00
	HPI MDP 1B fails				
HPR-MOV-OO-MFLB	3.00E-03	1.85E-04	1.00E+00	1.06E+00	1.00E+00
	Failure to minflow MOV 63-175 to close				
LOOP-18-09-NREC	8.00E-01	2.45E-01	1.33E+00	1.06E+00	1.00E+00

	LOOP sequence 18-09 nonrecovery probability				
HPR-XHE-XM-L	1.00E-03	5.48E-05	1.00E+00	1.06E+00	9.97E-01
	Operator fails to initiate HPR during loop				
HPI-XHE-XM-FB	1.00E-02	5.05E-04	1.00E+00	1.05E+00	9.92E-01
	Operator fails to initiate feed and bleed cooling				
LOOP-18-02-NREC	8.00E-01	1.93E-01	1.24E+00	1.05E+00	1.02E+00
	LOOP sequence 18-02 nonrecovery probability				
HPI-XHE-XM-FBL	1.00E-02	4.26E-04	1.00E+00	1.04E+00	1.01E+00
	Operator fails to initiate feed and bleed cooling				
LOOP-18-18-NREC	8.00E-01	1.44E-01	1.17E+00	1.04E+00	1.01E+00
	LOOP sequence 18-18 nonrecovery probability				
PPR-MOV-OO-BLK1	3.00E-03	1.01E-04	1.00E+00	1.03E+00	1.03E+00
	PORV 1 block valve fails to close				
AFW-MDP-FC-1A	3.94E-03	1.26E-04	1.00E+00	1.03E+00	1.00E+00
	AFW motor-driven pump 1A fails				
LOOP-18-11-NREC	8.00E-01	1.13E-01	1.13E+00	1.03E+00	9.99E-01
	LOOP sequence 18-11 nonrecovery probability				
HPR-MOV-CC-RHRA	3.00E-03	7.11E-05	1.00E+00	1.02E+00	1.43E+00
	RHR discharge MOV 63-8 fails				
HPR-MOV-CC-SMPA	3.00E-03	7.11E-05	1.00E+00	1.02E+00	1.43E+00
	Sump isolation MOV 185A fails to open				
RHR-MOV-OO-RWSTA	3.00E-03	7.11E-05	1.00E+00	1.02E+00	1.43E+00
	RWST isolation MOV 74-3 fails to close				
AFW-MDP-FC-1B	3.94E-03	7.31E-05	1.00E+00	1.02E+00	1.00E+00
	AFW motor-driven pump 1B fails				
LOOP-18-20-NREC	8.00E-01	5.59E-02	1.06E+00	1.01E+00	9.85E-01
	LOOP sequence 18-20 nonrecovery probability				
MSS-MOV-OO-ADV	3.00E-03	3.50E-05	1.00E+00	1.01E+00	1.19E+00
	ADV block valve fails to close				
MSS-XHE-XM-BLK	1.00E-02	1.19E-04	1.00E+00	1.01E+00	1.19E+00

	Operator fails to close ADV block valve				
MFW-XHE-ERROR	5.00E-02	5.93E-04	1.00E+00	1.01E+00	1.01E+00
	Operator fails to restore MFW flow				
RCS-CKV-CF-TRNA	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	Common cause failure of RCS CL inlet TRN A check valves				
RCS-CKV-CF-TRNB	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	Common cause failure of RCS CL inlet TRN B check valves				
RHR-AOV-OC-DISA	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	RHR discharge AOV A fails to provide flow				
RHR-AOV-OC-DISB	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	RHR discharge AOV A fails to provide flow				
RHR-MOV-OC-DISA	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	RHR dischagre MOV A fails to provide flow				
RHR-MOV-OC-DISB	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	RHR discharge MOV A fails to provide flow				
RHR-MOV-OC-SUCA	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	RHR train A suction MOV 74-3 fails				
RHR-MOV-OC-SUCB	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	RHR train A suction MOV 74-3 fails				
CVC-MOV-OO-RWSTA	3.00E-03	2.07E-05	1.00E+00	1.01E+00	1.15E+00
	Failure to isolate RWST (MOV 135)				
CVC-MOV-OO-RWSTB	3.00E-03	2.07E-05	1.00E+00	1.01E+00	1.15E+00
	Failure to isolate RWST (MOV 136)				
CVC-MOV-CF-RWST	2.64E-04	1.20E-06	1.00E+00	1.01E+00	1.12E+00
	Common cause failure to isolate RWST from charging system				
HPI-MDP-FC-1A	3.80E-03	1.90E-05	1.00E+00	1.01E+00	1.03E+00
	HPI MDP 1A fails				
RPS-XHE-ERROR	2.00E-01	1.14E-03	1.00E+00	1.01E+00	9.76E-01
	Operator fails to de-energize MG sets				
TRANS-20-NREC	2.18E-01	1.13E-03	1.00E+00	1.00E+00	1.00E+00

	Transient sequence 20 nonrecovery probability				
LOOP-17-NREC	2.18E-01	9.63E-04	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 17 nonrecovery probability				
MFW-SYS-UNAVAIL	2.00E-01	6.39E-04	1.00E+00	1.00E+00	9.74E-01
	Main feedwater system unavailable given an ATWS				
CVC-MDP-FC-1A	8.20E-04	1.44E-06	1.00E+00	1.00E+00	1.00E+00
	Charging pump 1A fails				
CVC-MDP-FC-1B	3.80E-03	6.05E-06	1.00E+00	1.00E+00	1.00E+00
	Charging pump 1B fails				
CVC-MOV-CC-DISA	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	Charging system MOV 63-39 fails to remain open				
CVC-MOV-CC-DISB	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	Charging system MOV 63-40 fails to remain open				
CVC-MOV-CC-DISC	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	Charging system MOV 63-25 fails to remain open				
CVC-MOV-CC-DISD	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	Charging sytem MOV 63-26 fails to remain open				
CVC-MOV-CC-SUCA	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	Charging system MOV 62-135 fails to open				
CVC-MOV-CC-SUCB	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	Charging system MOV 62-136 fails to open				
CVC-MOV-OO-VCTA	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	VCT isolation valve 62-132 fails to close				
CVC-MOV-OO-VCTB	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	VCT isolation valve 62-133 fails to close				
MFW-XHE-NOREC	2.00E-01	5.81E-04	1.00E+00	1.00E+00	1.00E+00
	Operator fails to recover MFW flow				
OEP-XHE-NOREC-6H	3.60E-02	7.16E-05	1.00E+00	1.00E+00	1.00E+00
	Operator fails to recover offsite power within 6 hours				
AFW-TDP-CC-STMA	1.10E-03	1.03E-06	1.00E+00	1.00E+00	1.00E+00

	Steam supply trains from SG A fail				
AFW-TDP-CC-STMC	1.10E-03	1.03E-06	1.00E+00	1.00E+00	1.00E+00
	Steam supply trains from SG C fail				
MSS-AOV-OO-ADV	1.00E-01	1.54E-04	1.00E+00	1.00E+00	1.03E+00
	ADV fails to reclose after opening				
AFW-CKV-CC-SGB	2.00E-04	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Discharge check valves 922 or 862 fail				
AFW-CKV-CC-SGD	2.00E-04	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Discharge check valves 921 or 861 fail				
AFW-MDP-FLOW	1.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of MDP flow into steam generators				
AFW-PSF-CC-MSGA	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG A (from MDP) fails				
AFW-PSF-CC-MSGB	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG B (from MDP) fails				
AFW-PSF-CC-MSGC	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG C (from MDP) fails				
AFW-PSF-CC-MSGD	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG D (from MDP) fails				
AFW-PSF-CC-TSGA	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG A (from TDP) fails				
AFW-PSF-CC-TSGB	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG B (from TDP) fails				
AFW-PSF-CC-TSGC	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG C (from TDP) fails				
AFW-PSF-CC-TSGD	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG D (from TDP) fails				
AFW-TNK-FC-CST1	1.30E-06	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	AFW condensate storage tank A fails				
AFW-TNK-FC-CST2	1.30E-06	0.00E+00	1.00E+00	1.00E+00	1.00E+00

	AFW condensate storage tank B fails				
AFW-XVM-OC-CST	0.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	CST common discharge valve 3-800 fails				
HPI-MOV-OC-SUCA	4.00E-05	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	HPI suction MOV 63-47 fails				
HPI-MOV-OC-SUCB	4.00E-05	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	HPI suction MOV 63-48 fails				
HPR-MOV-CC-PDA	3.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	MOV 63-6 fails to open				
HPR-MOV-CC-PDB	3.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	MOV 63-7 fails to open				
HPR-MOV-CF-PDIS	2.64E-04	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Common cause failure of RHR discharge MOVs				
HPR-MOV-OO-MFLA	3.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of minflow MOV 63-4 to close				
HPR-MOV-OO-MFLAB	3.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of minflow MOV 63-3 to close				
HPR-XVM-OC-RHR	0.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Manual valve 63-531 plugs				
LOOP-05-NREC	1.00E+00	1.25E-03	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 05 nonrecovery probability				
LOOP-07-NREC	1.00E+00	8.77E-06	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 07 nonrecovery probability				
LOOP-09-NREC	1.00E+00	4.65E-02	1.05E+00	1.00E+00	1.00E+00
	LOOP sequence 08 nonrecovery probability				
LOOP-10-NREC	8.40E-01	4.71E-05	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 10 nonrecovery probability				
LOOP-13-NREC	2.60E-01	2.53E-05	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 13 nonrecovery probability				
LOOP-16-NREC	2.60E-01	7.27E-05	1.00E+00	1.00E+00	1.00E+00

	LOOP sequence 16 nonrecovery probability				
LOOP-18-05-NREC	8.00E-01	2.67E-04	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 18-05 nonrecovery probability				
LOOP-18-07-NREC	8.00E-01	1.11E-06	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 18-07 nonrecovery probability				
LOOP-18-08-NREC	6.72E-01	6.56E-07	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 18-08 nonrecovery probability				
LOOP-18-14-NREC	8.00E-01	1.56E-04	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 18-14 nonrecovery probability				
LOOP-18-16-NREC	8.00E-01	2.99E-07	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 18-16 nonrecovery probability				
LOOP-18-17-NREC	6.72E-01	2.00E-07	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 18-17 nonrecovery probability				
LOOP-19-NREC	1.00E+00	1.83E-04	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 19 nonrecovery probability				
LPR-XHE-XM	1.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Operator fails to initiate LPR system				
MFW-SYS-TRIP	8.00E-01	5.81E-04	1.00E+00	1.00E+00	1.00E+00
	Main feedwater system unavailable given Rx trip				
MSS-ADV-FC-BLK	0.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	ADV block valve is closed during full power				
MSS-SRV-OO-SGS	1.00E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of steam generator SRV to reclose				
PCS-XHE-XO-SEC	2.00E-01	5.37E-05	1.00E+00	1.00E+00	1.00E+00
	Operator fails to establish secondary cooling				
PCS-XHE-XO-SECL	3.40E-01	5.63E-05	1.00E+00	1.00E+00	1.00E+00
	Operator fails to establish secondary cooling during LO				
PPR-MOV-CC-BLK1	3.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	PORV 1 block valve fails to open				
PPR-MOV-CC-BLK2	3.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00

	PORV 1 block valve fails to open				
PPR-MOV-FC-BLK1	0.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	PORV 1 block valve is closed during full power				
PPR-MOV-FC-BLK2	0.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	PORV 1 block valve is closed during full power				
PPR-SRV-CO-1	1.00E+00	5.29E-06	1.00E+00	1.00E+00	1.00E+00
	PORVs/SRVs open during transient (AFW failed)				
PPR-SRV-OO-SR1	1.60E-02	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of SRV 1 to reclose				
PPR-SRV-OO-SR2	1.60E-02	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of SRV 2 to reclose				
PPR-SRV-OO-SR3	1.60E-02	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of SRV 3 to reclose				
PPR-SRV-OO-SRV1	1.00E-01	1.49E-05	1.00E+00	1.00E+00	1.00E+00
	SRV-1 fails to reclose after passing water				
PPR-SRV-OO-SRV2	1.00E-01	1.49E-05	1.00E+00	1.00E+00	1.00E+00
	SRV-2 fails to reclose after passing wayer				
PPR-SRV-OO-SRV3	1.00E-01	1.49E-05	1.00E+00	1.00E+00	1.00E+00
	SRV-3 fails to reclose after passing water				
PPR-SRV-OO-WTR1	1.00E-01	2.03E-05	1.00E+00	1.00E+00	1.00E+00
	PORV 1 fails to reclose after passing water				
PPR-SRV-OO-WTR2	1.00E-01	1.14E-07	1.00E+00	1.00E+00	1.00E+00
	PORV 2 fails to reclose after passing water				
RCS-CKV-CF-ALL	1.85E-06	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Common cause failure of RCS cold leg discharge check valves				
RCS-PHN-PL	9.00E-01	2.37E-03	1.00E+00	1.00E+00	1.00E+00
	Power at high level				
RHR-CKV-CF-PMPS	1.00E-05	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Common cause failure of RHR check valves				
SGTR-03-NREC	1.00E+00	2.54E-02	1.03E+00	1.00E+00	1.00E+00

	SGTR sequence 03 nonrecovery probability				
SGTR-04-NREC	1.00E+00	9.21E-03	1.01E+00	1.00E+00	1.00E+00
	SGTR sequence 04 nonrecovery probability				
SGTR-05-NREC	1.00E+00	2.09E-03	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 05 nonrecovery probability				
SGTR-08-NREC	1.00E+00	1.07E-03	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 08 nonrecovery probability				
SGTR-09-NREC	1.00E+00	3.89E-04	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 09 nonrecovery probability				
SGTR-10-NREC	1.00E+00	7.91E-05	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 10 nonrecovery probability				
SGTR-11-NREC	1.00E+00	2.77E-02	1.03E+00	1.00E+00	1.00E+00
	SGTR sequence 11 nonrecovery probability				
SGTR-13-NREC	8.40E-01	7.14E-06	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 13 nonrecovery probability				
SGTR-14-NREC	8.40E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 14 nonrecovery probability				
SGTR-16-NREC	8.40E-01	1.28E-07	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 16 nonrecovery probability				
SGTR-17-NREC	8.40E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 17 nonrecovery probability				
SGTR-18-NREC	8.40E-01	4.28E-08	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 18 nonrecovery probability				
SGTR-21-NREC	2.60E-01	4.99E-07	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 21 nonrecovery probability				
SGTR-22-NREC	2.60E-01	1.85E-07	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 22 nonrecovery probability				
SGTR-23-NREC	2.60E-01	4.28E-08	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 23 nonrecovery probability				
SGTR-26-NREC	2.60E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00

	SGTR sequence 26 nonrecovery probability				
SGTR-27-NREC	2.60E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 27 nonrecovery probability				
SGTR-28-NREC	2.60E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 28 nonrecovery probability				
SGTR-29-NREC	2.60E-01	2.28E-07	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 29 nonrecovery probability				
SGTR-31-NREC	2.18E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 31 nonrecovery probability				
SGTR-32-NREC	2.18E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 32 nonrecovery probability				
SGTR-34-NREC	2.18E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 34 nonrecovery probability				
SGTR-35-NREC	2.18E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 35 nonrecovery probability				
SGTR-36-NREC	2.18E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 36 nonrecovery probability				
SGTR-39-NREC	2.60E-01	1.28E-07	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 39 nonrecovery probability				
SGTR-41-NREC	2.60E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 41 nonrecovery probability				
SGTR-42-NREC	2.60E-01	2.37E-06	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 42 nonrecovery probability				
SGTR-43-NREC	2.18E-01	4.01E-06	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 43 nonrecovery probability				
SGTR-44-NREC	1.00E+00	1.16E-03	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 44 nonrecovery probability				
SLOCA-04-NREC	1.00E+00	9.04E-02	1.10E+00	1.00E+00	1.00E+00
	SLOCA sequence 04 nonrecovery probability				
SLOCA-06-NREC	1.00E+00	1.39E-03	1.00E+00	1.00E+00	1.00E+00

	SLOCA sequence 06 nonrecovery probability				
SLOCA-07-NREC	8.40E-01	4.52E-04	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 07 nonrecovery probability				
SLOCA-11-NREC	2.60E-01	1.95E-06	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 11 nonrecovery probability				
SLOCA-13-NREC	2.60E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 13 nonrecovery probability				
SLOCA-14-NREC	2.18E-01	1.43E-08	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 14 nonrecovery probability				
SLOCA-17-NREC	2.60E-01	2.42E-07	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 17 nonrecovery probability				
SLOCA-19-NREC	2.60E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 19 nonrecovery probability				
SLOCA-21-NREC	2.60E-01	1.57E-07	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 21 nonrecovery probability				
SLOCA-22-NREC	2.18E-01	6.02E-06	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 22 nonrecovery probability				
SLOCA-23-NREC	1.00E+00	1.65E-03	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 23 nonrecovery probability				
TRANS-05-NREC	1.00E+00	3.98E-03	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 05 nonrecovery probability				
TRANS-07-NREC	1.00E+00	1.58E-05	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 07 nonrecovery probability				
TRANS-08-NREC	8.40E-01	1.08E-06	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 08 nonrecovery probability				
TRANS-13-NREC	2.60E-01	5.29E-06	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 13 nonrecovery probability				
TRANS-15-NREC	2.60E-01	1.43E-08	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 15 nonrecovery probability				
TRANS-16-NREC	2.18E-01	1.14E-07	1.00E+00	1.00E+00	1.00E+00

	Transient sequence 16 nonrecovery probability				
TRANS-19-NREC	2.60E-01	3.05E-05	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 19 nonrecovery probability				
TRANS-21-04-NREC	1.00E+00	3.95E-05	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-04 nonrecovery probability				
TRANS-21-06-NREC	1.00E+00	2.57E-07	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-06 nonrecovery probability				
TRANS-21-07-NREC	1.00E+00	1.88E-04	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-07 nonrecovery probability				
TRANS-21-11-NREC	1.00E+00	7.23E-06	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-11 nonrecovery probability				
TRANS-21-13-NREC	1.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-13 nonrecovery probability				
TRANS-21-14-NREC	1.00E+00	3.76E-05	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-14 nonrecovery probability				
TRANS-21-15-NREC	1.00E+00	1.30E-05	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-15 nonrecovery probability				
TRANS-21-16-NREC	1.00E+00	2.45E-03	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-16 nonrecovery probability				

APPENDIX D: CASE STUDY BASIC EVENT IMPORTANCE MEASURES SORTED BY RAW WITH RESPECT TO deltaCDF

Basic Event	Probability	Fussell-Vesely	Risk Reduction Worth	RAW with respect to CDF	RAW with respect to deltaCDF
	Basic Event Description				
IE-SGTR	1.63E-06	6.71E-02	1.07E+00	4.12E+04	5.56E+05
	Steam generator tube rupture initiating event				
IE-SLOCA	2.33E-06	9.39E-02	1.10E+00	4.03E+04	3.97E+04
	Small loss of coolant accident initiating even				
MSS-XHE-XM-ERROR	1.00E-03	3.27E-03	1.00E+00	4.26E+00	5.55E+01
	Operator fails to isolate faulted steam generator				
MSS-VCF-HW-ISOL	1.00E-02	3.27E-02	1.03E+00	4.23E+00	5.50E+01
	Ruptured steam generator isolation failures				
ACP-BAC-LP-1B	9.00E-05	1.77E-02	1.02E+00	1.97E+02	4.42E+01
	Division 1B AC power 4160V bus fails				
RHR-MDP-CF-ALL	5.60E-04	7.45E-02	1.08E+00	1.34E+02	3.56E+01
	RHR pump common cause failures				
RHR-MOV-CF-RWST	2.64E-04	5.10E-04	1.00E+00	2.93E+00	3.36E+01
	Common cause failure of RWST isolation MOVs				
RHR-HTX-CF-AB	1.40E-05	1.86E-03	1.00E+00	1.34E+02	3.28E+01
	Failure of heat exchanges due to common cause failure				
IE-TRANS	2.65E-04	7.90E-03	1.01E+00	3.08E+01	5.44E+00
	Transient initiating event				
HPI-MDP-CF-ALL	7.80E-04	1.88E-04	1.00E+00	1.24E+00	4.45E+00
	Common cause failure of HPI flowpath				
HPI-CKV-CF-CL	2.96E-05	6.15E-06	1.00E+00	1.21E+00	4.38E+00
	Common cause failure of RCS cold leg discharge check valves				

HPI-MOV-OC-DISCH	4.00E-05	8.47E-06	1.00E+00	1.21E+00	4.38E+00
	HPI cold leg injection valves MOV 63-22 fails				
HPI-MOV-OC-RWST	1.40E-04	2.32E-05	1.00E+00	1.17E+00	4.23E+00
	HPI serial component failures				
HPI-CKV-CF-PMPS	1.00E-05	1.30E-06	1.00E+00	1.13E+00	4.16E+00
	Common cause failure of HPI pump discharge check valves				
HPR-MOV-CF-RHR	2.64E-04	5.10E-04	1.00E+00	2.93E+00	3.36E+00
	Common cause failure of RHR supply to HPI isolation valves CV8804A & SI8804				
HPR-MOV-CF-SMP	2.64E-04	5.10E-04	1.00E+00	2.93E+00	3.36E+00
	Common cause failure of sump recirculation MOVs				
HPR-XHE-XM	1.00E-03	1.89E-03	1.00E+00	2.89E+00	3.36E+00
	Operator fails to initiate HPR				
HPR-SMP-FC-SMP	5.00E-05	9.58E-05	1.00E+00	2.92E+00	3.34E+00
	Failure of sump				
ACP-BAC-LP-1A	9.00E-05	1.56E-03	1.00E+00	1.84E+01	2.79E+00
	Division 1A AC power 4160V bus fails				
HPI-TNK-VF-RWST	2.70E-06	3.00E-04	1.00E+00	1.12E+02	2.35E+00
	RWST not available				
RPS-BKR-FC-FTO	5.70E-06	1.14E-03	1.00E+00	2.02E+02	2.34E+00
	RPS breakers fail to open				
RCS-XHE-DIAG	6.80E-03	5.26E-03	1.01E+00	1.77E+00	1.94E+00
	Operator fails to diagnose SGTR to start procedures				
RCS-XHE-XM-SG	2.10E-02	1.62E-02	1.02E+00	1.76E+00	1.93E+00
	Operator fails to initiate RCS depressurization				
HPI-XHE-XM-THRTL	1.00E-02	7.73E-03	1.01E+00	1.77E+00	1.91E+00
	Operator fails to throttle HPI to reduce pressure				
CVC-CKV-CC-DIS	1.00E-04	1.10E-05	1.00E+00	1.11E+00	1.70E+00
	Charging system discharge check valve fails				
CVC-CKV-CC-SUC	1.00E-04	1.10E-05	1.00E+00	1.11E+00	1.70E+00
	Charging system suction check valve fails				

CVC-MDP-CF-ALL	1.51E-04	1.78E-05	1.00E+00	1.12E+00	1.70E+00
	Common cause failure of charging pumps (HPI)				
CVC-MOV-CF-DIS1	2.64E-04	3.11E-05	1.00E+00	1.12E+00	1.70E+00
	Common cause failure of CVC discharge MOVS 63-39 63-40 to open				
CVC-MOV-CF-DIS2	2.64E-04	3.11E-05	1.00E+00	1.12E+00	1.70E+00
	Common cause failure of CVC discharge MOVS 63-25 63-26 to open				
CVC-MOV-CF-SUC	2.64E-04	2.99E-05	1.00E+00	1.11E+00	1.61E+00
	Common cause failure of CVC suction MOVS to open				
CVC-MOV-CF-VCT	2.64E-04	2.99E-05	1.00E+00	1.11E+00	1.61E+00
	Common cause failure of VCT isolation MOVS to close				
CVC-CKV-CF-PMPS	1.00E-05	9.98E-07	1.00E+00	1.10E+00	1.60E+00
	Common cause failure of CVC pump discharge check valves				
RHR-MDP-FC-1B	3.90E-03	2.33E-03	1.00E+00	1.59E+00	1.45E+00
	RHR MDO 1B fails				
HPR-MOV-CC-RHRA	3.00E-03	7.11E-05	1.00E+00	1.02E+00	1.43E+00
	RHR discharge MOV 63-8 fails				
HPR-MOV-CC-RHRB	3.00E-03	2.56E-04	1.00E+00	1.09E+00	1.43E+00
	RHR discharge MOV 63-11 fails				
HPR-MOV-CC-SMPA	3.00E-03	7.11E-05	1.00E+00	1.02E+00	1.43E+00
	Sump isolation MOV 185A fails to open				
HPR-MOV-CC-SMPB	3.00E-03	2.56E-04	1.00E+00	1.09E+00	1.43E+00
	Sump isolation MOV 63-73 fails to open				
RHR-MOV-OO-RWSTA	3.00E-03	7.11E-05	1.00E+00	1.02E+00	1.43E+00
	RWST isolation MOV 74-3 fails to close				
RHR-MOV-OO-RWSTB	3.00E-03	2.56E-04	1.00E+00	1.09E+00	1.43E+00
	RWST isolation MOV 74-21 fails to close				
RHR-MDP-FC-1A	3.90E-03	2.09E-03	1.00E+00	1.53E+00	1.42E+00
	RHR motor driven pump 1A fails				
PCS-PSF-HW	1.00E-05	2.17E-03	1.00E+00	2.18E+02	1.40E+00
	Hardware failures causing failure to depressurize				

HPI-MOV-OO-RWST	3.00E-03	2.22E-04	1.00E+00	1.07E+00	1.31E+00
	HPI isolation valve from RWST fails				
AFW-PMP-CF-ALL	5.60E-05	1.43E-03	1.00E+00	2.66E+01	1.23E+00
	Common cause failure of AFW pumps (all types)				
AFW-AOV-CF-DIS	1.85E-05	4.72E-04	1.00E+00	2.65E+01	1.19E+00
	Common cause failure of SGs discharge AOVs to open				
MSS-MOV-OO-ADV	3.00E-03	3.50E-05	1.00E+00	1.01E+00	1.19E+00
	ADV block valve fails to close				
MSS-XHE-XM-BLK	1.00E-02	1.19E-04	1.00E+00	1.01E+00	1.19E+00
	Operator fails to close ADV block valve				
CVC-MOV-OO-RWSTA	3.00E-03	2.07E-05	1.00E+00	1.01E+00	1.15E+00
	Failure to isolate RWST (MOV 135)				
CVC-MOV-OO-RWSTB	3.00E-03	2.07E-05	1.00E+00	1.01E+00	1.15E+00
	Failure to isolate RWST (MOV 136)				
CVC-MOV-CF-RWST	2.64E-04	1.20E-06	1.00E+00	1.01E+00	1.12E+00
	Common cause failure to isolate RWST from charging system				
EPS-DGN-FC-1A	3.80E-02	3.93E-01	1.65E+00	1.10E+01	1.12E+00
	Diesel generator A fails				
SLOCA-XHE-NOREC	4.30E-01	9.22E-02	1.10E+00	1.12E+00	1.12E+00
	Operator fails to recover from A SLOCA in short term				
EPS-DGN-FC-1B	3.80E-02	4.37E-01	1.78E+00	1.21E+01	1.08E+00
	Diesel generator B fails				
PPR-XHE-XM-BLK	1.00E-03	1.14E-04	1.00E+00	1.11E+00	1.06E+00
	Operator fails to close block valves				
PPR-SRV-CO-TRAN	4.00E-02	4.00E-03	1.00E+00	1.10E+00	1.04E+00
	PORVs/SRVs open during transient				
HPI-MDP-FC-1A	3.80E-03	1.90E-05	1.00E+00	1.01E+00	1.03E+00
	HPI MDP 1A fails				
MSS-AOV-OO-ADV	1.00E-01	1.54E-04	1.00E+00	1.00E+00	1.03E+00
	ADV fails to reclose after opening				

PPR-MOV-OO-BLK1	3.00E-03	1.01E-04	1.00E+00	1.03E+00	1.03E+00
	PORV 1 block valve fails to close				
PPR-MOV-OO-BLK2	3.00E-03	2.47E-04	1.00E+00	1.08E+00	1.03E+00
	PORV 2 block valve fails to close				
PPR-SRV-OO-PRV1	3.00E-02	7.79E-02	1.08E+00	3.52E+00	1.03E+00
	PORV 1 fails to reclose after opening				
RHR-XHE-XM	1.00E-03	2.66E-03	1.00E+00	3.66E+00	1.03E+00
	Operator fails to initiate RHR system				
AFW-MDP-CF-AB	2.10E-04	1.83E-04	1.00E+00	1.87E+00	1.02E+00
	Common cause failures of AFW motor driven pumps				
LOOP-18-02-NREC	8.00E-01	1.93E-01	1.24E+00	1.05E+00	1.02E+00
	LOOP sequence 18-02 nonrecovery probability				
OEP-XHE-NOREC-2H	1.20E-01	4.63E-02	1.05E+00	1.34E+00	1.02E+00
	Operator fails to recover offsite power within 2 hours				
RCS-MDP-LK-SEALS	1.30E-01	3.44E-01	1.52E+00	3.30E+00	1.02E+00
	RCP seals fail without cooling and injection				
CVC-CKV-CF-CL	1.85E-06	1.57E-07	1.00E+00	1.08E+00	1.01E+00
	Common cause failure of CVC cold leg discarge check valves				
CVC-TNK-VF-BIT	2.40E-06	2.00E-07	1.00E+00	1.08E+00	1.01E+00
	Boron injection tank unavailable				
HPI-MDP-FC-1B	3.80E-03	2.57E-04	1.00E+00	1.07E+00	1.01E+00
	HPI MDP 1B fails				
HPI-XHE-XM-FBL	1.00E-02	4.26E-04	1.00E+00	1.04E+00	1.01E+00
	Operator fails to initiate feed and bleed cooling				
LOOP-18-18-NREC	8.00E-01	1.44E-01	1.17E+00	1.04E+00	1.01E+00
	LOOP sequence 18-18 nonrecovery probability				
MFW-XHE-ERROR	5.00E-02	5.93E-04	1.00E+00	1.01E+00	1.01E+00
	Operator fails to restore MFW flow				
PCS-XHE-XM-CDOWN	1.00E-03	3.52E-04	1.00E+00	1.35E+00	1.01E+00
	Operator fails to initiate cooldown				

PPR-SRV-CC-RCS	4.40E-04	8.28E-05	1.00E+00	1.19E+00	1.01E+00
	Relief valves fail to open to limit RCS pressure				
PPR-SRV-CO-SBO	3.70E-01	5.59E-02	1.06E+00	1.10E+00	1.01E+00
	PORVs/SRVs open during station blackout				
PPR-SRV-OO-PRV2	3.00E-02	2.98E-02	1.03E+00	1.96E+00	1.01E+00
	PORV 2 fails to reclose of opening				
RCS-CKV-CF-TRNA	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	Common cause failure of RCS CL inlet TRN A check valves				
RCS-CKV-CF-TRNB	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	Common cause failure of RCS CL inlet TRN B check valves				
RHR-AOV-OC-DISA	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	RHR discharge AOV A fails to provide flow				
RHR-AOV-OC-DISB	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	RHR discharge AOV A fails to provide flow				
RHR-MOV-OC-DISA	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	RHR dischagre MOV A fails to provide flow				
RHR-MOV-OC-DISB	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	RHR discharge MOV A fails to provide flow				
RHR-MOV-OC-SUCA	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	RHR train A suction MOV 74-3 fails				
RHR-MOV-OC-SUCB	4.00E-05	3.28E-07	1.00E+00	1.01E+00	1.01E+00
	RHR train A suction MOV 74-3 fails				
AFW-CKV-CC-SGB	2.00E-04	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Discharge check valves 922 or 862 fail				
AFW-CKV-CC-SGD	2.00E-04	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Discharge check valves 921 or 861 fail				
AFW-MDP-FC-1A	3.94E-03	1.26E-04	1.00E+00	1.03E+00	1.00E+00
	AFW motor-driven pump 1A fails				
AFW-MDP-FC-1B	3.94E-03	7.31E-05	1.00E+00	1.02E+00	1.00E+00
	AFW motor-driven pump 1B fails				

AFW-MDP-FLOW	1.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of MDP flow into steam generators				
AFW-PSF-CC-MSGA	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG A (from MDP) fails				
AFW-PSF-CC-MSGB	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG B (from MDP) fails				
AFW-PSF-CC-MSGC	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG C (from MDP) fails				
AFW-PSF-CC-MSGD	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG D (from MDP) fails				
AFW-PSF-CC-TSGA	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG A (from TDP) fails				
AFW-PSF-CC-TSGB	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG B (from TDP) fails				
AFW-PSF-CC-TSGC	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG C (from TDP) fails				
AFW-PSF-CC-TSGD	1.10E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Inlet valve train to SG D (from TDP) fails				
AFW-TDP-CC-STMA	1.10E-03	1.03E-06	1.00E+00	1.00E+00	1.00E+00
	Steam supply trains from SG A fail				
AFW-TDP-CC-STMC	1.10E-03	1.03E-06	1.00E+00	1.00E+00	1.00E+00
	Steam supply trains from SG C fail				
AFW-TNK-FC-CST1	1.30E-06	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	AFW condensate storage tank A fails				
AFW-TNK-FC-CST2	1.30E-06	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	AFW condensate storage tank B fails				
AFW-XVM-OC-CST	0.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	CST common discharge valve 3-800 fails				
CVC-MDP-FC-1A	8.20E-04	1.44E-06	1.00E+00	1.00E+00	1.00E+00
	Charging pump 1A fails				

CVC-MDP-FC-1B	3.80E-03	6.05E-06	1.00E+00	1.00E+00	1.00E+00
	Charging pump 1B fails				
CVC-MOV-CC-DISA	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	Charging system MOV 63-39 fails to remain open				
CVC-MOV-CC-DISB	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	Charging system MOV 63-40 fails to remain open				
CVC-MOV-CC-DISC	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	Charging system MOV 63-25 fails to remain open				
CVC-MOV-CC-DISD	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	Charging sytem MOV 63-26 fails to remain open				
CVC-MOV-CC-SUCA	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	Charging system MOV 62-135 fails to open				
CVC-MOV-CC-SUCB	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	Charging system MOV 62-136 fails to open				
CVC-MOV-OO-VCTA	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	VCT isolation valve 62-132 fails to close				
CVC-MOV-OO-VCTB	3.00E-03	5.36E-06	1.00E+00	1.00E+00	1.00E+00
	VCT isolation valve 62-133 fails to close				
HPI-MOV-OC-SUCA	4.00E-05	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	HPI suction MOV 63-47 fails				
HPI-MOV-OC-SUCB	4.00E-05	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	HPI suction MOV 63-48 fails				
HPR-MOV-CC-PDA	3.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	MOV 63-6 fails to open				
HPR-MOV-CC-PDB	3.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	MOV 63-7 fails to open				
HPR-MOV-CF-PDIS	2.64E-04	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Common cause failure of RHR discharge MOVs				
HPR-MOV-OO-MFLA	3.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of minflow MOV 63-4 to close				

HPR-MOV-OO-MFLAB	3.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of minflow MOV 63-3 to close				
HPR-MOV-OO-MFLB	3.00E-03	1.85E-04	1.00E+00	1.06E+00	1.00E+00
	Failure to minflow MOV 63-175 to close				
HPR-XVM-OC-RHR	0.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Manual valve 63-531 plugs				
LOOP-05-NREC	1.00E+00	1.25E-03	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 05 nonrecovery probability				
LOOP-07-NREC	1.00E+00	8.77E-06	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 07 nonrecovery probability				
LOOP-09-NREC	1.00E+00	4.65E-02	1.05E+00	1.00E+00	1.00E+00
	LOOP sequence 08 nonrecovery probability				
LOOP-10-NREC	8.40E-01	4.71E-05	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 10 nonrecovery probability				
LOOP-13-NREC	2.60E-01	2.53E-05	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 13 nonrecovery probability				
LOOP-16-NREC	2.60E-01	7.27E-05	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 16 nonrecovery probability				
LOOP-17-NREC	2.18E-01	9.63E-04	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 17 nonrecovery probability				
LOOP-18-05-NREC	8.00E-01	2.67E-04	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 18-05 nonrecovery probability				
LOOP-18-07-NREC	8.00E-01	1.11E-06	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 18-07 nonrecovery probability				
LOOP-18-08-NREC	6.72E-01	6.56E-07	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 18-08 nonrecovery probability				
LOOP-18-09-NREC	8.00E-01	2.45E-01	1.33E+00	1.06E+00	1.00E+00
	LOOP sequence 18-09 nonrecovery probability				
LOOP-18-14-NREC	8.00E-01	1.56E-04	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 18-14 nonrecovery probability				

LOOP-18-16-NREC	8.00E-01	2.99E-07	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 18-16 nonrecovery probability				
LOOP-18-17-NREC	6.72E-01	2.00E-07	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 18-17 nonrecovery probability				
LOOP-19-NREC	1.00E+00	1.83E-04	1.00E+00	1.00E+00	1.00E+00
	LOOP sequence 19 nonrecovery probability				
LPR-XHE-XM	1.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Operator fails to initiate LPR system				
MFW-SYS-TRIP	8.00E-01	5.81E-04	1.00E+00	1.00E+00	1.00E+00
	Main feedwater system unavailable given Rx trip				
MFW-XHE-NOREC	2.00E-01	5.81E-04	1.00E+00	1.00E+00	1.00E+00
	Operator fails to recover MFW flow				
MSS-ADV-FC-BLK	0.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	ADV block valve is closed during full power				
MSS-SRV-OO-SGS	1.00E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of steam generator SRV to reclose				
OEP-XHE-NOREC-6H	3.60E-02	7.16E-05	1.00E+00	1.00E+00	1.00E+00
	Operator fails to recover offsite power within 6 hours				
PCS-XHE-XO-SEC	2.00E-01	5.37E-05	1.00E+00	1.00E+00	1.00E+00
	Operator fails to establish secondary cooling				
PCS-XHE-XO-SECL	3.40E-01	5.63E-05	1.00E+00	1.00E+00	1.00E+00
	Operator fails to establish secondary cooling during LO				
PPR-MOV-CC-BLK1	3.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	PORV 1 block valve fails to open				
PPR-MOV-CC-BLK2	3.00E-03	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	PORV 1 block valve fails to open				
PPR-MOV-FC-BLK1	0.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	PORV 1 block valve is closed during full power				
PPR-MOV-FC-BLK2	0.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	PORV 1 block valve is closed during full power				

PPR-SRV-CO-1	1.00E+00	5.29E-06	1.00E+00	1.00E+00	1.00E+00
	PORVs/SRVs open during transient (AFW failed)				
PPR-SRV-OO-SR1	1.60E-02	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of SRV 1 to reclose				
PPR-SRV-OO-SR2	1.60E-02	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of SRV 2 to reclose				
PPR-SRV-OO-SR3	1.60E-02	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Failure of SRV 3 to reclose				
PPR-SRV-OO-SRV1	1.00E-01	1.49E-05	1.00E+00	1.00E+00	1.00E+00
	SRV-1 fails to reclose after passing water				
PPR-SRV-OO-SRV2	1.00E-01	1.49E-05	1.00E+00	1.00E+00	1.00E+00
	SRV-2 fails to reclose after passing wayer				
PPR-SRV-OO-SRV3	1.00E-01	1.49E-05	1.00E+00	1.00E+00	1.00E+00
	SRV-3 fails to reclose after passing water				
PPR-SRV-OO-WTR1	1.00E-01	2.03E-05	1.00E+00	1.00E+00	1.00E+00
	PORV 1 fails to reclose after passing water				
PPR-SRV-OO-WTR2	1.00E-01	1.14E-07	1.00E+00	1.00E+00	1.00E+00
	PORV 2 fails to reclose after passing water				
RCS-CKV-CF-ALL	1.85E-06	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Common cause failure of RCS cold leg discharge check valves				
RCS-PHN-PL	9.00E-01	2.37E-03	1.00E+00	1.00E+00	1.00E+00
	Power at high level				
RHR-CKV-CF-PMPS	1.00E-05	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Common cause failure of RHR check valves				
RHR-MOV-CF-SUCT	2.64E-04	7.02E-04	1.00E+00	3.66E+00	1.00E+00
	RHR suction valve common cause failures				
SGTR-03-NREC	1.00E+00	2.54E-02	1.03E+00	1.00E+00	1.00E+00
	SGTR sequence 03 nonrecovery probability				
SGTR-04-NREC	1.00E+00	9.21E-03	1.01E+00	1.00E+00	1.00E+00
	SGTR sequence 04 nonrecovery probability				

SGTR-05-NREC	1.00E+00	2.09E-03	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 05 nonrecovery probability				
SGTR-08-NREC	1.00E+00	1.07E-03	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 08 nonrecovery probability				
SGTR-09-NREC	1.00E+00	3.89E-04	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 09 nonrecovery probability				
SGTR-10-NREC	1.00E+00	7.91E-05	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 10 nonrecovery probability				
SGTR-11-NREC	1.00E+00	2.77E-02	1.03E+00	1.00E+00	1.00E+00
	SGTR sequence 11 nonrecovery probability				
SGTR-13-NREC	8.40E-01	7.14E-06	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 13 nonrecovery probability				
SGTR-14-NREC	8.40E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 14 nonrecovery probability				
SGTR-16-NREC	8.40E-01	1.28E-07	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 16 nonrecovery probability				
SGTR-17-NREC	8.40E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 17 nonrecovery probability				
SGTR-18-NREC	8.40E-01	4.28E-08	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 18 nonrecovery probability				
SGTR-21-NREC	2.60E-01	4.99E-07	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 21 nonrecovery probability				
SGTR-22-NREC	2.60E-01	1.85E-07	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 22 nonrecovery probability				
SGTR-23-NREC	2.60E-01	4.28E-08	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 23 nonrecovery probability				
SGTR-26-NREC	2.60E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 26 nonrecovery probability				
SGTR-27-NREC	2.60E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 27 nonrecovery probability				

SGTR-28-NREC	2.60E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 28 nonrecovery probability				
SGTR-29-NREC	2.60E-01	2.28E-07	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 29 nonrecovery probability				
SGTR-31-NREC	2.18E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 31 nonrecovery probability				
SGTR-32-NREC	2.18E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 32 nonrecovery probability				
SGTR-34-NREC	2.18E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 34 nonrecovery probability				
SGTR-35-NREC	2.18E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 35 nonrecovery probability				
SGTR-36-NREC	2.18E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 36 nonrecovery probability				
SGTR-39-NREC	2.60E-01	1.28E-07	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 39 nonrecovery probability				
SGTR-41-NREC	2.60E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 41 nonrecovery probability				
SGTR-42-NREC	2.60E-01	2.37E-06	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 42 nonrecovery probability				
SGTR-43-NREC	2.18E-01	4.01E-06	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 43 nonrecovery probability				
SGTR-44-NREC	1.00E+00	1.16E-03	1.00E+00	1.00E+00	1.00E+00
	SGTR sequence 44 nonrecovery probability				
SLOCA-04-NREC	1.00E+00	9.04E-02	1.10E+00	1.00E+00	1.00E+00
	SLOCA sequence 04 nonrecovery probability				
SLOCA-06-NREC	1.00E+00	1.39E-03	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 06 nonrecovery probability				
SLOCA-07-NREC	8.40E-01	4.52E-04	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 07 nonrecovery probability				

SLOCA-11-NREC	2.60E-01	1.95E-06	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 11 nonrecovery probability				
SLOCA-13-NREC	2.60E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 13 nonrecovery probability				
SLOCA-14-NREC	2.18E-01	1.43E-08	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 14 nonrecovery probability				
SLOCA-17-NREC	2.60E-01	2.42E-07	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 17 nonrecovery probability				
SLOCA-19-NREC	2.60E-01	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 19 nonrecovery probability				
SLOCA-21-NREC	2.60E-01	1.57E-07	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 21 nonrecovery probability				
SLOCA-22-NREC	2.18E-01	6.02E-06	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 22 nonrecovery probability				
SLOCA-23-NREC	1.00E+00	1.65E-03	1.00E+00	1.00E+00	1.00E+00
	SLOCA sequence 23 nonrecovery probability				
TRANS-05-NREC	1.00E+00	3.98E-03	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 05 nonrecovery probability				
TRANS-07-NREC	1.00E+00	1.58E-05	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 07 nonrecovery probability				
TRANS-08-NREC	8.40E-01	1.08E-06	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 08 nonrecovery probability				
TRANS-13-NREC	2.60E-01	5.29E-06	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 13 nonrecovery probability				
TRANS-15-NREC	2.60E-01	1.43E-08	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 15 nonrecovery probability				
TRANS-16-NREC	2.18E-01	1.14E-07	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 16 nonrecovery probability				
TRANS-19-NREC	2.60E-01	3.05E-05	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 19 nonrecovery probability				

TRANS-20-NREC	2.18E-01	1.13E-03	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 20 nonrecovery probability				
TRANS-21-04-NREC	1.00E+00	3.95E-05	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-04 nonrecovery probability				
TRANS-21-06-NREC	1.00E+00	2.57E-07	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-06 nonrecovery probability				
TRANS-21-07-NREC	1.00E+00	1.88E-04	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-07 nonrecovery probability				
TRANS-21-11-NREC	1.00E+00	7.23E-06	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-11 nonrecovery probability				
TRANS-21-13-NREC	1.00E+00	0.00E+00	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-13 nonrecovery probability				
TRANS-21-14-NREC	1.00E+00	3.76E-05	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-14 nonrecovery probability				
TRANS-21-15-NREC	1.00E+00	1.30E-05	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-15 nonrecovery probability				
TRANS-21-16-NREC	1.00E+00	2.45E-03	1.00E+00	1.00E+00	1.00E+00
	Transient sequence 21-16 nonrecovery probability				
LOOP-18-11-NREC	8.00E-01	1.13E-01	1.13E+00	1.03E+00	9.99E-01
	LOOP sequence 18-11 nonrecovery probability				
AFW-AOV-CF-STM	1.00E-04	8.54E-05	1.00E+00	1.85E+00	9.98E-01
	Common cause failure of TDP STM supply line AOVs to open				
PCS-XHE-XM-RCOOL	1.00E-03	2.40E-03	1.00E+00	3.40E+00	9.98E-01
	Operator fails to initiate RCS cooldown below RHR				
AFW-MOV-CC-STM	3.10E-03	2.69E-03	1.00E+00	1.86E+00	9.97E-01
	Steam supply valves fail				
HPR-XHE-XM-L	1.00E-03	5.48E-05	1.00E+00	1.06E+00	9.97E-01
	Operator fails to initiate HPR during loop				
OEP-XHE-NOREC-SL	6.30E-01	3.88E-01	1.64E+00	1.23E+00	9.97E-01
	Operator fails to recover offsite power (Seal loca)				

CVC-XHE-XM-BOR	1.00E-03	2.26E-04	1.00E+00	1.23E+00	9.95E-01
	Operator fails to initiate emergency boration				
LOOP-18-22-NREC	2.72E-01	3.07E-02	1.03E+00	1.08E+00	9.95E-01
	LOOP sequence 18-22 nonrecovery probability				
RHR-MOV-CC-HOTA	3.00E-03	7.98E-03	1.01E+00	3.65E+00	9.95E-01
	Discharge MOV 74-1 from hot legs fails				
RHR-MOV-CC-HOTB	3.00E-03	7.98E-03	1.01E+00	3.65E+00	9.95E-01
	Discharge MOV 74-2 from hot legs fails				
RHR-MOV-OO-RWST	3.00E-03	7.98E-03	1.01E+00	3.65E+00	9.95E-01
	Failure to isolate RWST				
RCS-PHN-MODPOOR	1.40E-02	2.37E-03	1.00E+00	1.17E+00	9.94E-01
	Moderator temp coefficient not enough negative				
RPS-XHE-XM-SCRAM	1.00E-02	4.32E-03	1.00E+00	1.43E+00	9.94E-01
	Operator fails to manually trip the reactor				
HPI-XHE-XM-FB	1.00E-02	5.05E-04	1.00E+00	1.05E+00	9.92E-01
	Operator fails to initiate feed and bleed cooling				
RCS-XHE-RECOVER	3.50E-03	2.77E-02	1.03E+00	8.89E+00	9.92E-01
	Operator fails to depressurize RCs below SG SRV given A				
PPR-SRV-CO-L	1.60E-01	4.78E-02	1.05E+00	1.25E+00	9.91E-01
	PORVs/SRVs open during loop (1.0 probability)				
OEP-XHE-NOREC-ST	5.30E-01	8.66E-02	1.10E+00	1.08E+00	9.90E-01
	Operator fails to recover offsite power in short term				
LOOP-18-20-NREC	8.00E-01	5.59E-02	1.06E+00	1.01E+00	9.85E-01
	LOOP sequence 18-20 nonrecovery probability				
OEP-XHE-NOREC-BD	7.40E-02	3.06E-01	1.44E+00	4.83E+00	9.83E-01
	Operator fails to recover offsite power before battery				
PPR-SRV-CC-PRV1	6.30E-03	5.86E-04	1.00E+00	1.09E+00	9.77E-01
	PORV1 fails to open on demand				
PPR-SRV-CC-PRV2	6.30E-03	5.86E-04	1.00E+00	1.09E+00	9.77E-01
	PORV2 fails to open on demand				

RPS-XHE-ERROR	2.00E-01	1.14E-03	1.00E+00	1.01E+00	9.76E-01
	Operator fails to de-energize MG sets				
MFW-SYS-UNAVAIL	2.00E-01	6.39E-04	1.00E+00	1.00E+00	9.74E-01
	Main feedwater system unavailable given an ATWS				
AFW-TDP-FC-1A	3.26E-02	2.83E-02	1.03E+00	1.84E+00	9.72E-01
	AFW Turbine driven pump fails				
RPS-VCF-FO-ELEC	4.30E-04	4.32E-03	1.00E+00	1.10E+01	9.23E-01
	Control rod drives remain energized				
AFW-CKV-CF-DIS	1.84E-06	4.59E-05	1.00E+00	2.60E+01	8.33E-01
	Common cause failure of AFW inlet check valves into SGs				
PCS-VCF-HW	3.00E-03	8.26E-03	1.01E+00	3.75E+00	3.07E-01
	TBVs/COND/Cir failures				
EPS-DGN-CF-ALL	1.44E-03	3.90E-01	1.64E+00	2.71E+02	0.00E+00
	Common cause failure of diesel generators				
RPS-VCF-FO-MECH	8.90E-08	2.72E-04	1.00E+00	3.05E+03	0.00E+00
	Control rod assemblies fail to insert				
IE-LOOP	1.60E-05	8.31E-01	5.92E+00	5.19E+04	0.00E+00
	Loss of offsite power initiating event				