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Risk Sensitivity to Human Error

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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

A sensitivity evaluation was conducted to assess the impact of human errors on the internal event risk parameters in the Oconee plant. The results provide the variation in the risk parameters, namely, core melt frequency and accident sequence frequencies, due to hypothetical changes in human error probabilities. Also provided are insights derived from the results, which highlight important areas for concentration of risk limitation efforts associated with human performance.

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EXECUTIVE SUMMARY

This report presents an evaluation of the sensitivity of nuclear power plant risk parameters to human errors that can occur during normal and accident conditions in the plant. The probabilistic risk assessment (PRA) of the Oconee-3 plant was the basis of the study and the human errors, whose impact are assessed, are those included in the PRA. This PRA was performed by the Electric Power Research Institute (EPRI) and published as NSAC-60 in 1984. The risk parameters chosen are the "internal event" accident sequence frequencies and the overall core-melt frequency. The sensitivity evaluations showed the changes in these risk parameters for systematic variation of all human error probabilities and for selected categories of human errors. Human error probabilities were varied in groups and over conservatively large ranges in order to obtain insights on the effect on risk, rather than to obtain realistic values for possible variations in CMF. Further, since 1984, plant modifications have been made at Oconee and a new PRA is in progress. Hence, the PRA used for this study does not represent the Oconee plant as currently configured and operated.

The importance of human error in determining risk from nuclear power plants is well known, and the purpose in performing this sensitivity evaluation was broader than merely verifying such importances. The sensitivity evaluation presented here provides a quantitative representation of changes in the human error probabilities, identifies the level of improvement to be obtained through reduction in these probabilities, identifies specific categories of human errors to seek such reductions, and seeks trends and patterns of risk significance in human behavior.

There are multifold justifications for performing this sensitivity evaluation. First, the estimation of human error probabilities is one of the most uncertain areas in quantification of PRAs and errors in the estimation can result in systematic under- or over-estimation of risk. Second, due to a lack of plant specific data, the human error probabilities are largely developed on a generic basis, in many cases, using expert judgment. The human error probabilities in a given plant may be significantly different from the generic or average estimates. Third, during the operating lifetime of a plant, there could be times when the performance of the plant crew is poor and other times when it is significantly better. Finally, the performance of the plant crew can be influenced by management's attitude resulting in human error probabilities that may be lower or higher.

Besides such strong justifications for conducting sensitivity evaluations with respect to human errors, the idea of a sensitivity evaluation is embedded in the desire to seek insights on the human role in plant risk variations. Of course, inherent in such sensitivity evaluations are a number of assumptions, which are directed to derive insights to seek improvements in plant safety levels and not to suggest the validity of such assumptions. For example, in this study, the probabilities of all the human errors and groups of human errors, signifying specific aspects of human interactions, are varied in separate evaluations to observe the risk parameter behaviors, implying an assumption that human error probabilities vary in such combinations in nuclear power plants. This implicit assumption of a sensitivity evaluation is not based on actual observation even though similar situations are not totally unlikely.

However, this approach to sensitivity evaluations has provided valuable insights. Based on a plant-specific application using the Oconee power plant PRA, the insights derived are significant both for plant-specific judgments and for generic implications. While conclusions regarding generic applicability cannot be overly broad at this time, the results presented in the Executive Summary appear to be generally applicable to most plants. These results will be re-evaluated as further studies are performed. The assumptions of sensitivity evaluations are taken into consideration in deriving these insights. The approach demonstrated in this study is also being used specifically for the LaSalle plant to obtain further foundations for the generic implications on the human role in plant risk.

The results of the human error sensitivity evaluations are presented in graphs showing the variation in the risk parameter due to changes in the human error probabilities. Figure (i) shows the sensitivity of Oconee core-melt frequency to the human error probabilities. An important aspect of the sensitivity evaluation is in seeking interpretation of these curves and in understanding the purpose of the study. One of the main purposes of the study was to extend our understanding of those aspects of human error which account for the most impact on risk. Observational data was not available to help establish realistic bounds on human error probabilities. For this reason, the displayed extreme values of core melt frequency should be regarded as hypothetical, resulting from extrapolation of PRA models beyond their originally intended purposes. Specific insights from this core-melt frequency curve are presented here. The details of interpretation of a number of such curves are presented in the report.

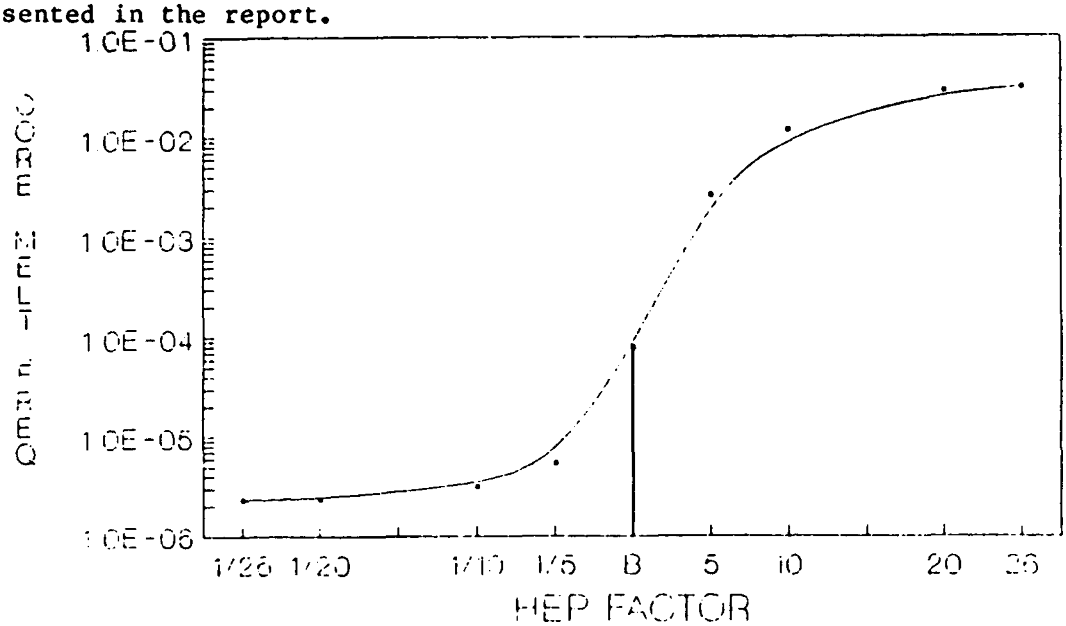


Figure i. Oconee CMF sensitivity to human error
(B = Base Case)

The Oconee core-melt frequency shows variation of over four orders of magnitude when all human error probabilities are simultaneously changed from their lower to their upper bounds. During plant operation, human error probabilities (HEPs) are not expected to vary to such extremes, and for practical considerations, small variations around the base error probabilities may be of more interest. Likewise, the core melt frequency is not realistically expected to vary to such high and low extremes, even as human performance in the plant varies. It is noteworthy that a significant increase and decrease in the risk parameters occurs when all human error probabilities are increased or decreased by small factors of 3 to 10 from base values. The curve also shows that significant improvement in core-melt frequency can be achieved through improvement in human performance. Relatively small improvement in HEPs (about a factor of two) can result in factors of five improvement in core-melt frequency. This finding suggests that risk reductions can be achieved by looking at specific errors modeled in the PRA and identifying those measures that may improve human performance.

The sensitivity of the dominant accident sequences in the Oconee plant showed a strong dependence on human errors. Also, the accident sequences with high initiating event frequency show strong sensitivity to human errors. These findings imply that events that are more likely to occur during the life of a plant can become significant safety concerns if the operators in the plant do not perform their roles adequately during these transients.

In this study, the human errors were categorized into various groups to understand the importance of various aspects of human behavior. The important insights derived from the sensitivity evaluation of human error categories can be summarized as follows:

a. Burden on the Operations Unit

In analyzing the errors that occur during accidents, including the recovery errors, it was apparent that there is a significant burden on the plant management and on the operating staff to control the risk from the plant. Following many accident initiating events, reactor operators have to conduct multiple activities, where more than one may involve coordination with non-licensed operators carrying out specific tasks outside the control room. In certain instances, such activities may have to be carried out without the benefit of detailed procedures.

b. Role of pre-accident human errors

The human errors modeled as occurring before an accident show sensitivity when increased from their base probabilities, but do not influence the core-melt frequency when decreased from their base values. This signifies that pre-accident errors need to be controlled at their base values to avoid adverse effects on plant safety. The base error probabilities for these errors are sufficiently low. Improvement in them from currently assumed values are not necessary unless the hardware in the plant is improved.

c. Significance of RO/NLO coordination

NLO activities in a plant can be divided into two groups; one, the activities that are carried out by the NLOs without any supervision and are typically required during normal operation of the plant, and two, the activities that are required to be carried our during an accident under specific request from ROs.

Among the various responsibilities of the reactor operators (ROs), we observed that their activities which involve coordination with non-licensed operators (NLOs) are as significant, if not more, as the actions carried out by ROs only. These activities are typically required during an accident, i.e., following the initiation of an event. This observation demonstrates the necessity of coordinating RO/NLO activities in assuring plant safety during accidents, and is also the reason for the importance of recovery errors, since RO/NLO actions are primarily required in carrying out the recovery actions.

d. Significance of NLO role

The NLO activities (carried out without the supervision of ROs) are not as important as RO activities, but they show a significant impact on CMF when increased from their base values. As discussed, these activities are pre-accident initiator activities. The other type of NLO activities supervised by ROs during an accident (recovery actions) also have a significant impact on CMF (discussion of item c. above). Overall, significant risk increases can be incurred in a plant due to NLO activities, and significant risk gain can be achieved by increasing the success probabilities of RO/NLO activities during accidents.

e. Dominance of "operator fails to" errors

The during-accident "Operator fails to" (or omission) type actions dominate the sensitivity curve. These include: a) operator fails to perform desired actions, and b) operator fails to recover. The during-accident commission errors (operator inhibits and inadvertent actions) have negligible influence on core-melt frequency. This is because the commission errors are highly unlikely events and even when their probabilities are increased, they are masked by errors of other types (e.g., omission errors, hardware faults) which are not being increased. Thus, significant improvements in risk levels can be achieved by assuring that operators perform the required actions during an accident sequence.

f. Significance of equipment restoration errors following test and maintenance

The human errors in test and maintenances (T/M) of components are due to (a) erroneous actions during T/M which result in failure of the component immediately or at some later time, and (b) failure of T/M personnel to restore the equipment to proper status following T/M. The PRA models, in general, and specifically the Oconee PRA model used in this study, implicitly model errors discussed in item (a), but explicitly model those under (b). Errors of type (a) are implicitly accounted for in the model in the hardware failure rates and in the initiator frequencies. Errors of type (b) are explicitly modeled as human errors. For this study, sensitivity evaluations were conducted only for the explicit errors of restoration following T/M. The effect of these errors on the core melt frequency was found to be minimal. However, when other types of human errors in maintenance are explicitly modeled for sensitivity evaluations, the result could be significantly different.

More detailed insights related to specific aspects of the study may be found in Chapters 5 and 6 of the report.

OVERVIEW SUMMARY

S.1 Introduction

The significance of human errors on nuclear power plant risk has been well recognized for several years. In analyzing the actual events observed in nuclear power plants, ranging from those of minor safety significance to those of major safety significance, it is seen that human errors played a role in almost all cases. Probabilistic risk assessments (PRAs) of nuclear power plants since the Reactor Safety study (WASH-1400) explicitly incorporate human intervention in assessing risks from nuclear power plants. These risk assessment models are the basis for analyzing the risk impact from human errors. Sensitivity of Risk Parameters to Human Errors in Reactor Safety Study for a PWR (Samanta et al., NUREG/CR-1879, 1981), conducted for the Surry plant, provided a methodology for assessment of the impacts of human error on plant risks through a sensitivity study recognizing the variability in human error probabilities.

The treatment of human errors in PRAs improved significantly along with the understanding of the error probabilities and the variabilities associated with them In this study, the basic approach of NUREG/CR-1879 is extended to assess the risk impact of human errors, using a current PRA with improved human error modeling. The PRA for the Oconee-3 nuclear power plant was chosen for evaluation. The choice of the Oconee-3 PRA was primarily based on the assessment that it contained an acceptable state-of-the-art treatment of human reliability analysis and not to imply that it is a representative plant whereby the results are applicable across all such plants. However, a number of insights are presented that have broad applications for addressing nuclear power plant risk from human errors and in modeling aspects of human reliability analysis.

This report presents the sensitivity evaluation for the Oconee plant and the various insights derived from the study. The approach used was based on plant-specific models and data, supplemented by generic data, and as such, provides a better understanding of the variations in the risk parameters, namely core melt frequency and accident sequence frequencies, of the Oconee plant due to changes in human performance. As the discussion in this report will reveal, the greatest benefit of sensitivity evaluation is in deriving insights which can be used to improve or reduce the risk parameter values. Some of the insights obtained have potential applicability to a wide variety of nuclear power plants. This potential applicability is due to the similarity in many aspects of the PRA modeling and plant design between Oconee and other plants. In the study, we compare the Oconee results to earlier results for Surry. Also, a follow-up study for the LaSalle nuclear power plant (a boiling water reactor) is currently underway.

In using the results of a sensitivity evaluation, the limitations of such evaluations should be taken into consideration. First, in any sensitivity evaluation, significant liberty is taken into varying the input parameters, in our case the human errors, to understand the behavior of the output parameters, which in this case, are the risk parameters. The primary objective here is to understand the behavior of the risk parameters and not to suggest that the variation in the input parameters are real or the manner in which they are varied is the way in which they will vary in real situations. To clarify further, in this sensitivity evaluation, all human error probabilities (HEPs) were varied together, but this is not to suggest that all such probabilities will change simultaneously, even though such possibilities exist. Human errors were also categorized into various groups and sensitivity evaluations were conducted varying such groups of human errors. This grouping provides insights on significance of various categories of human errors, but this is also not to suggest that the probabilities of human errors can only vary in such groups. Second, the results of the sensitivity evaluation in risk analyses depend on the risk models used. Any inadequacy of models will be reflected in the results that could also alter the insights to be derived. In this study, care was taken in deriving insights where doubts existed regarding the modeling adequacy and where the modeling consideration was considered of significant importance. Third, the base data in sensitivity evaluation have significant impact on the results of sensitivity evaluations. The use of a conservative data base will show stronger sensitivity compared to a non-conservative data base.

This section is an overview of all the important aspects of the study. It was written in addition to the Executive Summary because of the very detailed nature of this study. In a sense, it can be considered an enlarged, "summary and conclusions" section, that is usually presented at the end of a report, but here is presented at the front as a convenience to the reader. The main body of the report gives the technical details and an in-depth presentation of various aspects of the study.

S.2 Risk-Based Human Error Sensitivity Evaluation Process

Our methodology for sensitivity evaluation uses a plant specific probabilistic risk assessment (PRA). The Oconee-3 nuclear plant PRA conducted by EPRI and Duke Power Company (Sugnet et al., NSAC/60, 1984) was used to assess the sensitivity of its risk parameters to human errors. Only the portion of the PRA covering internal events was used for this analysis; sequences initiated by external events such as fires, earthquakes, and floods were not considered. Even though there are differences among nuclear power plant PRAs and in human reliability analysis (HRA) from one PRA to another, the process of sensitivity evaluation used to obtain the Oconee results is applicable to any PRA. Figure S.1 presents the broad elements in the human error sensitivity evaluation which consists of the following:

1. Identification of human errors and the associated probabilities in the PRA.

2. Categorization of the human errors.

3. Development of the range of human error probabilities for sensitivity evaluation.

4. Development of strategy for sensitivity evaluation.

RISK-BASED HE SENSITIVITY EVALUATION PROCESS

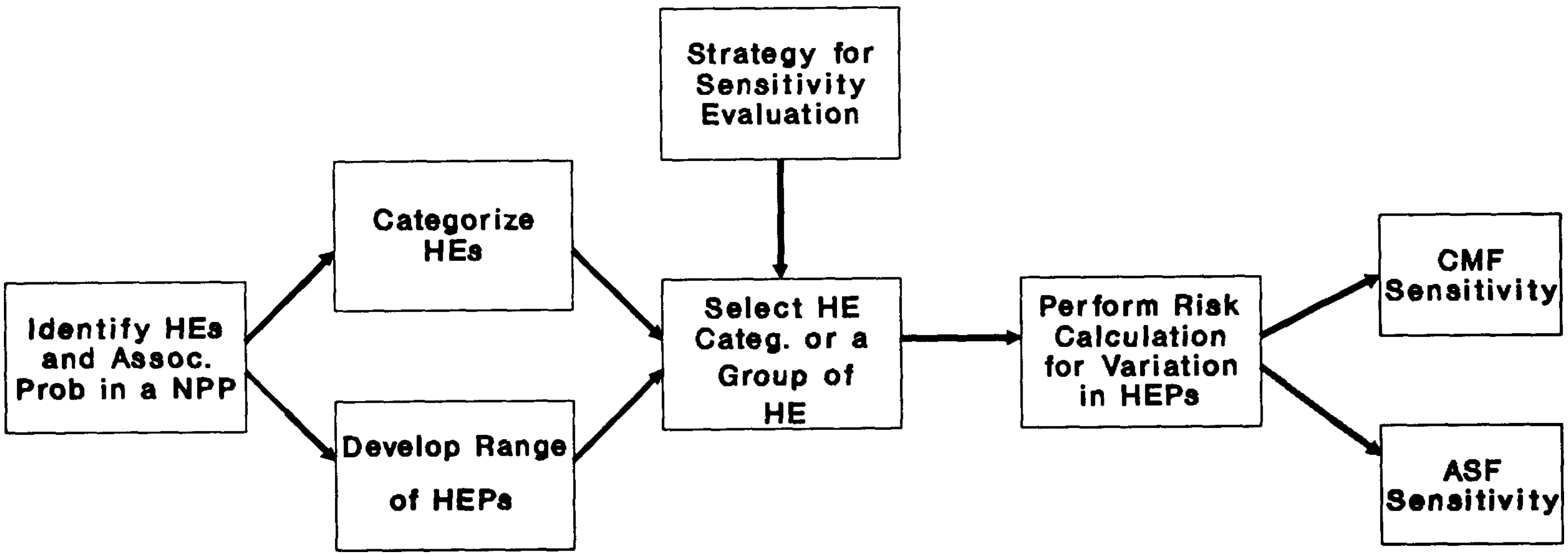


Figure S.1. Elements in human error sensitivity evaluation

5. Calculation of risk parameter values due to changes in HEP.

6. Assessment and interpretation of results.

In the following sections, these elements are discussed separately with examples from the Oconee application.

S.2.1. Identification of Human Errors (HEs)

The first step in carrying out the evaluation of human error sensitivity using a plant-specific PRA is to identify the human errors considered in the PRA. PRAs provide a systematic process of incorporating human errors that can lead to loss of safety functions in the plant affecting the plant risk level. Human errors appear in system fault trees and in the event trees for various initiating events. The errors of recovery (or failures to recover) are also considered as human errors, and they are obtained from the accident-sequence evaluations performed after initial quantification in the PRA. This step gives a complete list of human errors incorporated in the PRA and the associated mean probabilities used.

In the Oconee PRA, we identified 553 human errors, 64 of which were related to external events. Since the sensitivity evaluation focusses on the values of the risk parameters resulting from internal initiating events, this left an initial total of 489 human errors. Within this set of errors, many had little or no influence on the risk parameters, namely the core melt frequency and the accident sequence frequencies. In making the sensitivity evaluations, the errors that do not change the risk parameters even when their probabilities are significantly increased can be excluded. This was determined by examining the minimal cutsets (the minimal combination of basic events that cause the occurrence of an accident sequence) and excluding those human errors that appear in cutsets with frequencies of magnitude less than 10^{-10}. For the Oconee PRA, this process reduced the number of human errors to a set of 223, which was the final data base for the study.

S.2.2. Categorization of Human Errors

The primary purpose of a human error sensitivity evaluation is to seek patterns of human performance that alter the risk level in a plant. To show these patterns, various attributes of human errors need to be defined. We categorize human errors to define their characteristics so that each category provides a distinct perspective and defines the impact of the errors in that category on the risk parameters representing the risk significance of that aspect of the human error in the plant.

Table S.1 shows the categorization scheme used: Chapter 3 gives a more detailed discussion of the category and an example of each. The categorization scheme incorporates the categories used in other studies (Samanta et al., NUREG/CR-1879, 1981; Spettel et al., NUREG/CR-4103, 1986), and some new ones.

An examination of the categorization scheme reveals its utility for a sensitivity evaluation. For example, the "TIMING" category classifies the human errors in Oconee either as a pre-accident initiator error, or as a "during

Table S.1. Definitions of the Categories of Human Errors Relevant to Sensitivity Evaluation

CATEGORY	**DEFINITION**
TIMING	Classifies the timing of the human event relative to the accident initiating event or transient.
ACCINIT	Lists the accident initiating event(s) related to the human event.
SYSTEM	Defines the system where the human error occurs.
PERSONNEL	Identifies the individual(s) responsible.
OMCOM	Indicates whether the error is one of omission (human actions expected to be accomplished but not attempted) or of commission (human actions involving the completion of an improper action or an unsuccessful attempt to perform a desired action to achieve a specific goal).
EVENTTYPE	Relates the human event to the appropriate Oconee PRA established "Category of Human Error."
LOCATION	Identifies where the personnel most responsible for the human event is located.
ACTIVITY	Indicates the type of nuclear power plant activity that relates to the human event.
DEPEND	Identifies whether or not the outcome of a human event is dependent upon the outcome of another such event.
NRCPGM	Lists NRC Inspection areas which have the potential for detecting the occurrence of the human error.

accident" error, and shows the chronological relationship of the human error to the accident-initiating event. A sensitivity evaluation for this category provides the relative significance of pre-accident initiator error with respect to during accident initiator error.

Each of the Oconee human errors was coded according to the categorization scheme to identify the groups of errors belonging to each sub-element of the categories. In performing this task, each human error was analyzed and a distinct sub-element within each category that characterized the error was determined. For the categories defining the relationship to NRC Inspection Program (NRCPGM) and to accident initiators (ACCINIT), an error could be identified by more than one sub-element. Consider the error EFTDPP1H, Turbine-driven Emergency feedwater pump not restored following test or maintenance. This error, that results from the test and maintenance (T/M) activity before the initiation of an accident (Pre), is an omission type (OM) error, and the responsibility for the error lies with both reactor operator and maintenance personnel

(RO/MT). The NRC inspection categories that influence the error are Operations (Ops), Surveillance Testing (ST), System Walkdown (SW), and Maintenance (Maint.). Table S.2 shows the categorization of some of the human errors in the Oconee PRA.

It was quickly apparent that not all categories are independent. In many cases, there is a strong relationship among them which can be used to identify the specific characteristics of the human errors in the Oconee plant. For example, if a human error taken from the PRA was determined to be committed by a non-licensed operator (personnel category), by definition, the event occurred outside the control room (location category). Similarly, if an error was determined to be of the unavailability type (event type category), it occurred before an accident initiator (Pre in the timing category). Whenever this type of relationship was not evident, the judgment of the analyst defined the error. Figure S.2, called the linkage diagram, shows the breakdown of the Oconee human errors in terms of a number of categories whose interrelationships are also shown. In Section 2.3, there is an additional figure showing the relationship among other categories.

S.2.3. Development of Ranges of Human Error Probabilities

An important consideration in the evaluation is to define the entire range of variability of the input parameters whose significance is being evaluated in terms of their effect on the risk parameters. The range of variability of the HEPs in PRAs usually includes only the uncertainty in the data associated with these estimates. Ranges of the human error probabilities for sensitivity evaluation should consider the different causes of variability that can be assigned to the estimates.

In developing the ranges of the HEPs for sensitivity evaluation, different causes of variability defined in the literature (NUREG/CR-2300, NUREG/CR-1278) were considered and thus, the ranges defined are broader than those found in PRAs. We wanted to obtain a realistic, but also a conservative or broadest range, so that the sensitivity evaluation could cover the entire possible range, recognizing the different causes of variability.

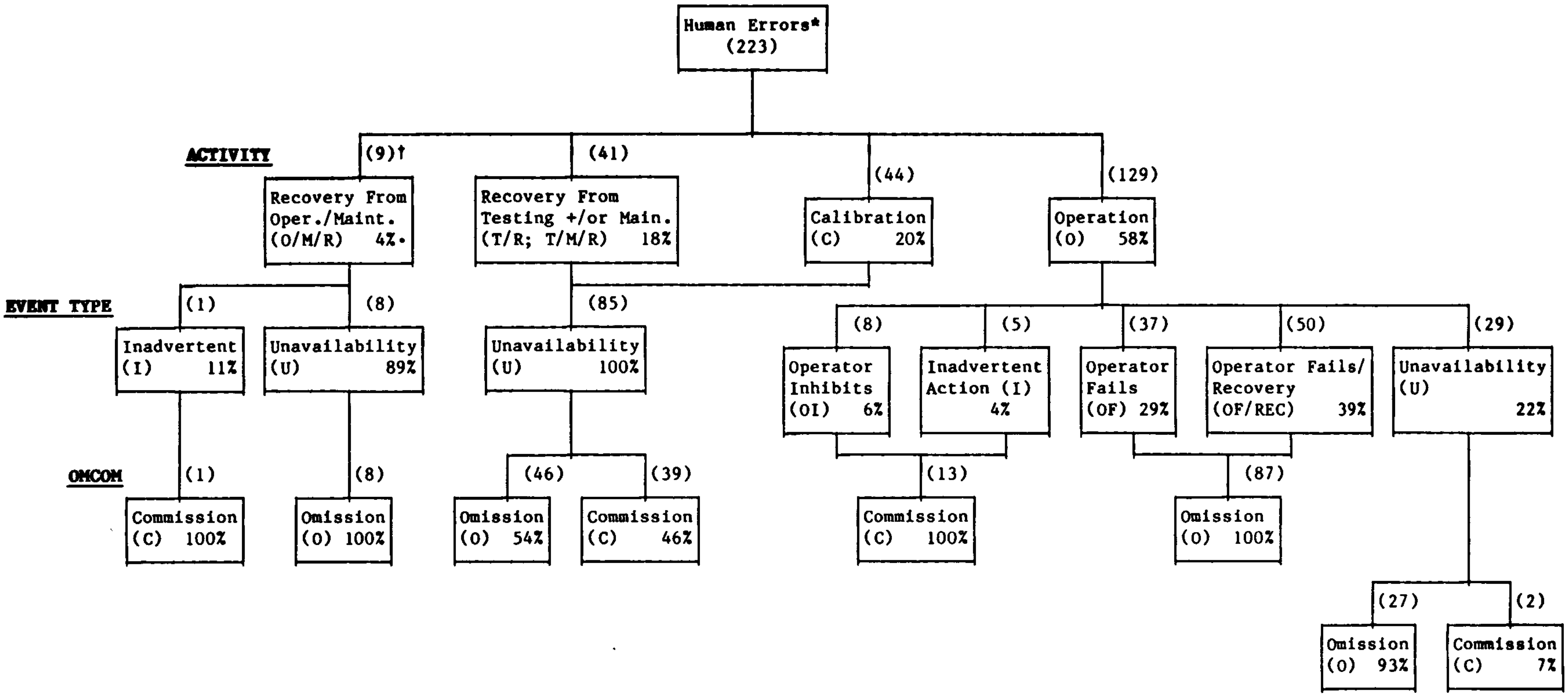
The methodology used to quantitatively determine the ranges of HEPs is drawn from the well-known statistical approach of analysis of variances (details are presented in Chapter 3). The influences of each of the causes of variability is defined in terms of error factors, and the variances in the HEP due to each cause are combined to obtain the overall variance in the HEP estimates. The overall variance is then used to obtain the range of the HEP. The error factors associated with each of the variability causes is defined subjectively. In this study, the error factors were defined using expert judgments which took into consideration the available data. This approach is considered adequate for sensitivity evaluation, since our objective is to develop realistic, but conservatively broad estimates of the ranges that account for the different causes of variability.

The reasons for variability in HEPs used in PRAs are discussed in PRA Procedures Guide (NUREG/CR-2300) and five major sources of uncertainties are defined:

Table S.2. Examples of the Categorization of Human Error

DESCRIPTION OF HUMAN ERROR	ERROR CODE	ERROR CATEGORIZATION				
		Timing	Personnel	Activity	Om/Com	Location
1. Operator fails to initiate ASW from SSF in 30 minutes from loss of feed-water	RESSFW30	During	RO/NL	Operations	Omission	CR/OCR
2. Operator fails to recover instrument air in one hour	REIA1	During	RO/NL	Operations	Omission	CR/OCR
3. Operator fails to attain or maintain HPI cooling after loss of all feed-water	UTHPIH	During	RO	Operations	Omission	CR
4. MOVs HP-24 and -25 (HPI suction valves) left un-available	HP2425MVH	During	RO	R	Omission	OCR
5. BWST suction valve LP-28 left closed after maintenance	LP28VVCH	During	RO/MT	R	Omission	OCR
6. Turbine driven emergency feed-water pump not restored after maintenance	EFTDPP1H	During	RO/MT	R	Omission	OCR

RO/NL: Reactor Operator and Non-Licensed Operator
RO: Reactor Operator
RO/MT: Reactor Operator and Maintenance Personnel
CR: Control Room
OCR: Outside Control Room
R: Restoration



* Human Errors considered in risk assessment of a nuclear power plant.
† Indicates the total number belonging to the category.
• Indicates the percentage of level above.

Figure S.2. Linkage diagram of human error categorization

1) Lack of Data
2) Inexactness of the Model
3) Difference in Task Description (application of generic HEPs)
4) Difference among Personnel
5) Skill and Knowledge of Human Reliability Analyst

In applying the methodology to determine ranges for HEPs in the Oconee PRA, the human errors were divided into groups depending upon the causes of variability and the associated error factors. Table S.3 lists the five groups along with their derived composite error factor.

Table S.3. Error Factor Associated with Types of Human Error

Type of Human Error	Error Factor
Dependent Human Errors	26
Test, Maintenance, & Calibration HEs with HEP > 1E-3	13
Test, Maintenance, & Calibration HEs with HEP $\leq$ 1E-3	22
Human Errors of Operations (act of Commission)	24
Human Errors of Operations (act of Omission)	21

Upper-bound and lower-bound HEPs for each human error event in a particular group are calculated by applying the error factor to the mean value of the HEP. For the last two error groups in which the base HEPs are $\geq$ 0.1, the use of the error factor resulted in an upper bound greater than 1.0, which was truncated to 1.0.

S.2.4. Strategy for Sensitivity Evaluation

In a risk-based sensitivity evaluation, the changes in the output parameters, such as the core melt frequency, accident-sequence frequency, are observed for changes in the input parameters, which, in our case, are the human error probabilities. For the objective of the evaluation, a specific strategy outlining the combinations of human errors (input parameters) and the output risk parameters needs to be defined. There are a large number of such combinations and the strategy specifically defines the combinations to be studied to effectively delineate the results being sought.

The specific objective of this study is to identify the quantitative impact of human errors on the plant's risk levels, to identify the specific aspects of human errors that have a higher risk impact, and to identify those categories of human errors whose improvement can provide significant risk benefits. The specific sensitivity evaluations performed and the significance of the evaluations are summarized in Table S.4.

S.2.5. Calculation of Risk Parameter Values

We calculated the major risk parameters in the plant, namely, the core melt frequency and the accident-sequence frequencies, due to change in the human error probabilities (HEPs) using the event tree and fault tree models of

Table S.4. Summary of Sensitivity Evaluations to Assess the Implications of Human Errors in Plant Risk

Sensitivity Evaluation	Significance of the Evaluation
1. Sensitivity with respect to all identified HEs in a plant	
a. CMF versus HEPs b. ASF versus HEPs c. Consequence Bin Frequency versus HEPs	i) identifies the role of HEs in plant risk ii) identifies the role of HEs in likelihood of accident sequences iii) identifies accident sequences that are most sensitive to HEs iv) identifies the role of HEs in consequences (bins) of accidents
2. Sensitivity of CMF to Errors of Recovery	Identifies the ability of operating staff to respond to an accident
3. Sensitivity of CMF to Categories of HEs	
a. TIMING Category	a. relative significance of during accident initiator, & pre-accident initiator HEs
b. LOCATION Category	b. role of HEs in and out of control rooms
c. PERSONNEL Category	c. risk significance of role of various types of personnel
d. ACTIVITY Category	d. risk significance of types of human activities
e. EVENTTYPE Category	e. risk significance of various types of actions
f. NRC INSPECTION Category	f. role of inspection categories
4. Relative likelihood of various accident sequences as HEPs vary	Identifies the dominance of accident sequences based on the performance of the plant crew.

the Oconee PRA. The process is similar to the evaluation of point estimates performed in PRAs. Individual accident sequence frequencies were computed for each set of changes in HEPs, and these frequencies were summed up to obtain the core melt frequency. To facilitate the large number of calculations needed we used the PAIRWISE computer program developed at Brookhaven National Laboratory. The PAIRWISE program is an interactive personal computer program where a select group of basic events (e.g., human errors) can be defined and their associated probabilities changed so that the corresponding accident sequence frequencies and core melt frequencies can be obtained. Further details of the calculation of risk parameters and of the PAIRWISE program are given in Section 4 and Appendix E, respectively.

In using PRA models for sensitivity evaluations where basic event probabilities (in our case, the HEPs) are significantly increased, certain precautions are necessary to calculate the risk parameter values appropriately.

The accident sequence models used in sensitivity evaluation are the minimal cutset expressions of the accident sequences. In PRAs, a large number of minimal cutsets are generated for each accident sequences, where a significant portion has a negligible contribution to the accident sequence frequency. For sensitivity evaluations, it is cumbersome to retain all the cutsets for repeated calculations and accordingly, only the cutsets that are the dominant contributors should be retained. Minimal cutsets that are the dominant contributors for estimating the expected accident sequence frequencies in PRAs are not the only cutsets required for sensitivity evaluations. Many cutsets that are not dominant when average HEPs are used can become dominant when calculated for increased HEPs. This is particularly so when a cutset contains multiple human errors where in a sensitivity evaluation, the probability estimates of these errors are increased simultaneously causing a significant jump in its frequency estimates, thereby making the cutset a dominant contributor.

To alleviate this problem, the dominant minimal cutset expressions for accident sequence frequencies were generated using HEPs equal to 1, and then using a truncation level of 10^{-10}. The cutsets that are eliminated in this process are negligible, even when the HEPs are increased to their maximum values.

S.3 Assessment and Interpretation of Results

S.3.1 Sensitivity of Core Melt Frequency to HEP Changes

One way to identify the role of human errors on plant risk is to assess the sensitivity of core melt frequency to changes in the human error probabilities in the plants. In this assessment, the probabilities of all the human errors that are judged to influence the core melt frequency are being changed together. The justifications for such an approach are multifold: (a) the assessment of HEPs in PRAs is subjective, and the HEPs may be systematically underestimated or overestimated, (b) the HEPs are average estimates and several causes may vary the HEPs, and (c) the operating staff of a nuclear power plant may give an improved performance or a degraded performance which are respectively signified by increased and decreased HEPs. However, such an approach in this study is to gain insights into the behavior of risk parameters, namely, the core melt frequency and accident sequence frequencies, and not to imply that probabilities of all the human errors change (increase or decrease) together during plant operation. As discussed previously, the range over which all the HEPs are varied is developed for individual HEPs and in actual situations all HEPs are not expected to reach upper or lower limits simultaneously.

Figure S.3 shows the sensitivity of the Oconee core melt frequency to multiplicative changes in the HEPs. The probability estimates of all the human errors included in the evaluation are increased or decreased by multiplicative factors until the respective upper or the lower bound of the HEPs is reached. The behavior of the core melt frequency is plotted on a logarithmic scale and it increases when the HEPs are increased and decreases when the HEPs are decreased, as expected. The shape of the curve, however, provides interesting insights on the human role in this power plant.

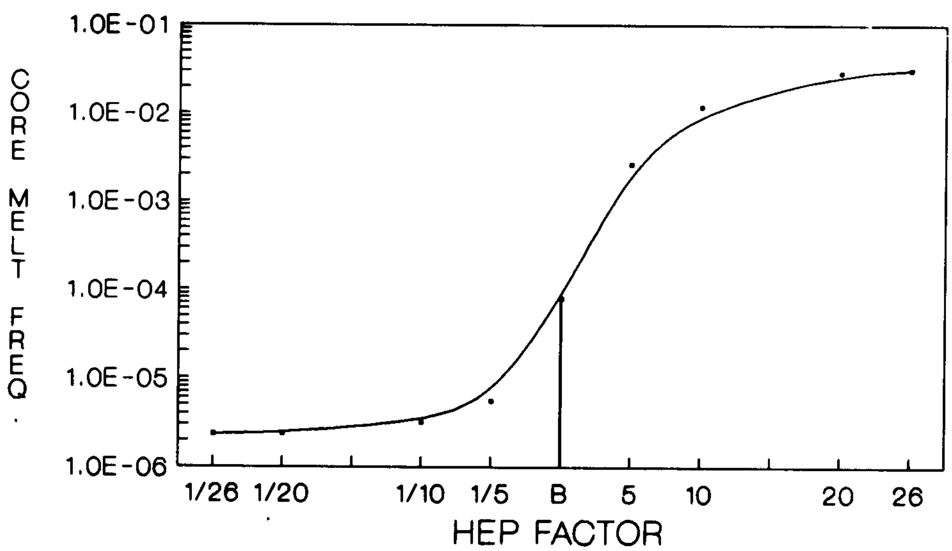


Figure S.3. Overall CMF sensitivity to human error

Range of Variation in Core Melt Frequency

The Oconee core melt frequency varies over four orders of magnitude (2.3E-6 and 3.1E-2) when all HEPs vary from the lower-bound to upper-bound values. Although a large variation in CMF due to changes in HEPs is not surprising, the significance of the Oconee CMF variation is partly attributable to plant-specific features. The dominance of the loss-of-instrument-air sequence in core melt frequency is specific to the Oconee plant. The sequence is particularly sensitive to Human Error and, at the upper bound of the HEPs, it contributes more than 60% of the CMF.

The behavior of the CMF curve shows the strong dominance of human errors on plant risk. The sharp increase and decrease of the CMF around the base HEPs signifies that the terms containing human errors dominate the CMF expression. Also, the large increase/decrease in CMF for a relatively small factor change in HEPs (factor of 33 increase in CMF for a factor of 5 increase in HEPs) signifies that the dominant terms (or cutsets) contain multiple human errors. The rate of increase of CMF due to increasing HEPs is partially dependent on the manner in which the HEPs were increased. An alternative method of varying HEPs from base values to upper- and lower-bound values is discussed and results are presented in Section 4.

Effect of Increased HEPs

The Oconee CMF shows a significant increase due to an increase in HEPs, but the increase in CMF is slower when HEPs are increased beyond a factor of 10. This happens because many HEPs with dominating influences reach their upper bounds when multiplied by a factor of 10. These are typically during accident errors with probabilities of 0.1 or greater, and such high probabilities are partly attributable to poor expectation of human performance and partly to lack of adequate information about them.

Limit of Reduction in CMF Due to Improvement in HEPs

For Oconee, the sensitivity curve of Figure S.3 reaches saturation when HEPs are decreased by factors of 10, i.e, any further decrease in HEPs does not result in any noticeable decrease in CMF. This is because the terms containing human errors are sufficiently small and no longer contribute significantly to the CM, and the hardware failures now dominate. It is also interesting to observe the contributions from hardware failures alone, i.e., the combination of hardware failures that will cause a core melt. For Oconee, with all HEPs are set to zero, signifying perfect human performance, the value is a core melt frequency of 2.3E-6 core melt events per reactor year. This value is about one-and-a-half orders of magnitude below the baseline CMF.

S.3.2 Sensitivity of Accident Sequence Frequencies (ASF) to HEP Changes

The sensitivity of individual accident sequences were analyzed for changes in the HEPs, and the sensitivity curves for several of the sequences are shown in Figure S.4. This curve shows the factor by which ASF varies as HEPs are varied by multiplicative factors to their upper and lower bounds. The shape is different than Figure S.3 since the y-axis is no longer logarithmic. Many of the accident sequence frequencies which contribute dominantly to the core melt frequency show significant variation with changes in HEPs. Typically, these sequences contain multiple human errors in their dominant cutsets. A variation in loss of instrument air sequence (T6BU) as high as seven orders of magnitude is observed when HEPs are varied from the lower bound to their upper bound. Detailed analyses of the sensitivity results of the accident sequences are presented in Chapter 5 and in Appendix F of this report. General observations on the influence of human errors in Oconee accident sequences are presented below:

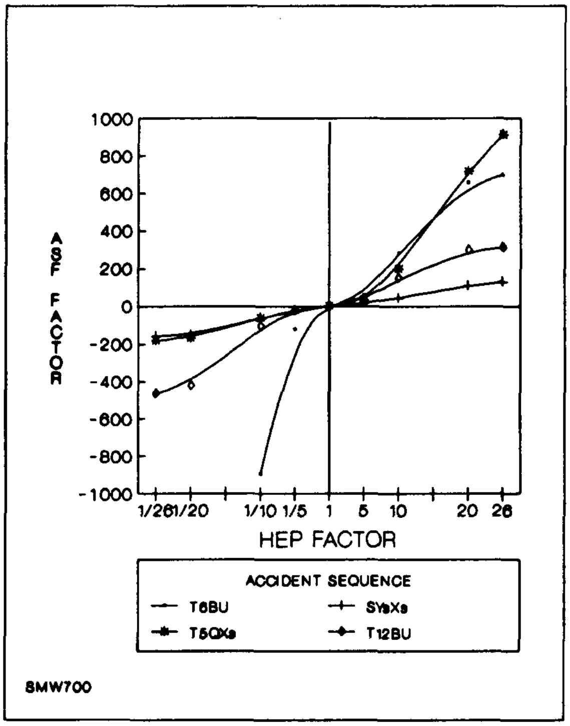


Figure S.4. Sensitivity of ASF to HEP variation

i) The dominant accident sequences in the Oconee PRA, for example, loss of instrument air (T_6BU), and loss of service water ($T_{12}BU$), are very sensitive to human errors. The strong sensitivity of these sequences to human errors contribute to the sensitivity of the Oconee CMF and relates to the presence of multiple human errors in the dominant cutsets. The loss of instrument air sequence (T_6BU) contains about 15 human errors and 90% of the sequence frequency is determined by cutsets containing triple human errors. Examples of these errors are: (a) failure to initiate the safe shutdown facility to provide feedwater within 30 minutes (RESSFW30), (b) failure to recover instrument air in one hour (REIA1), and (c) operator failure to attempt high pressure injection (HPI) cooling (UTHPIH). The base error probabilities of these human errors are in the range of 0.01 to 0.1. The factor increase in these errors results in the cubic growth of the accident sequence frequency.

ii) The dominant accident sequences show a significant decrease in their frequencies when HEPs are decreased. T_6BU is decreased by about 7,000 times, and $T_{12}BU$ is decreased by about 500 times when HEPs are decreased to their lower bounds from base probabilities. This is due to the dominant cutsets in these sequences containing multiple human errors with large assigned base case probabilities. As discussed above for the T_6BU sequence, the dominance of the cutsets containing triple human error terms also contributes to the cubic decline in the accident sequence frequency when the HEPs are decreased together. This results in the large potential for improvement in many dominant accident sequences through reduction in the human errors with large error probabilities. Another interesting feature of these accident sequences (involving multiple HEs) is that significant improvements in frequencies can be made for relatively small improvement in HEPs if such improvements are achievable.

For example, a factor of five improvement in HEPs decreases the T_6BU sequence frequency by a factor of 120, and the $T_{12}BU$ sequence frequency by a factor of 26. This is because multiple human errors appear in the dominant terms of the accident sequence frequency expression. For example, achieving the factor of 120 improvement in T_6BU frequency will require improvement in both RESSFW30 and REIA1 probabilities from 0.1 to 0.02 and UTHPIH from 0.01 to 0.002. Human factor studies can be undertaken to determine if such improvements in multiple errors are feasible. The subset of specific human errors that needs to be improved to lower these sequence frequencies is identified as a part of the sensitivity evaluations in Chapter 5.

iii) Transient-initiated accident sequences (T5, T6, and T12) show stronger sensitivity to human error compared with Loss-of-Coolant-Accident (LOCA) sequences (e.g., SY_SX_S). This is expected, and is considered to be of generic implications because of the following reasons. First, human actions are less effective in controlling LOCA sequences; second, transient-initiated accidents have greater chances of misdiagnosis by the operators; third, transients have much longer time-window for multiple operator actions following the initiating event. The interactive and/or relative effect of these reasons are not clearly known, but can be studied to develop a clearer understanding of the sensitivity of transient-initiated accident sequences to human errors. One of these sequences, T5QXS (transient-induced LOCA), is not dominant in the base case, yet increases by nearly three orders of magnitude and becomes important as HEPs reach their upper bound.

iv) The accident sequences with relatively higher initiating event frequencies show stronger sensitivity to human errors. In conducting this sensitivity evaluation, the initiating event frequency was assumed constant, even though it is generally agreed that initiating events are often caused by human errors. This linkage is implicitly in the initiating event data base, but not explicitly modeled in the PRAs to allow variation in a sensitivity evaluation. However, if such linkages were explicitly delineated in the PRA models, the sensitivity of the accident sequences would be further pronounced. The accident sequences, resulting from loss of main feedwater (0.5 events/yr), loss of instrument air (0.21 events/yr), loss of condenser vacuum (0.21 events/yr), loss of offsite power (0.12 events/yr) are among the accident sequences that are sensitive to human error. This implies that the events that are expected to occur during the lifetime of the plant have strong dependence on human errors and consequently, the frequencies of these accident sequences can be significantly lowered through improvement in the associated human error probabilities.

S.3.3 Insights on the Human Role in Plant Risk

S.3.3.1 Role of Operations-Related Errors

In evaluating the role of the operations-related errors in the Oconee plant, several sensitivity evaluations were conducted which provide valuable insights on the influence of the operations-related errors on the core melt frequency. Three sets of sensitivity evaluations were conducted based on the timing category, utility program activity category, and a category of recovery errors; the results are presented in Figures S.5 to S.7.

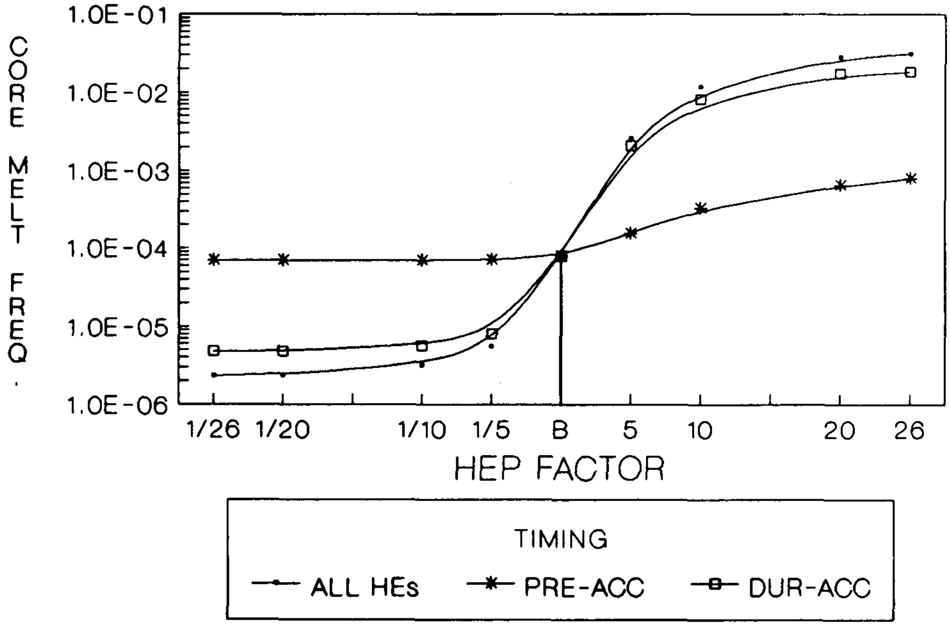


Figure S-5. Timing category

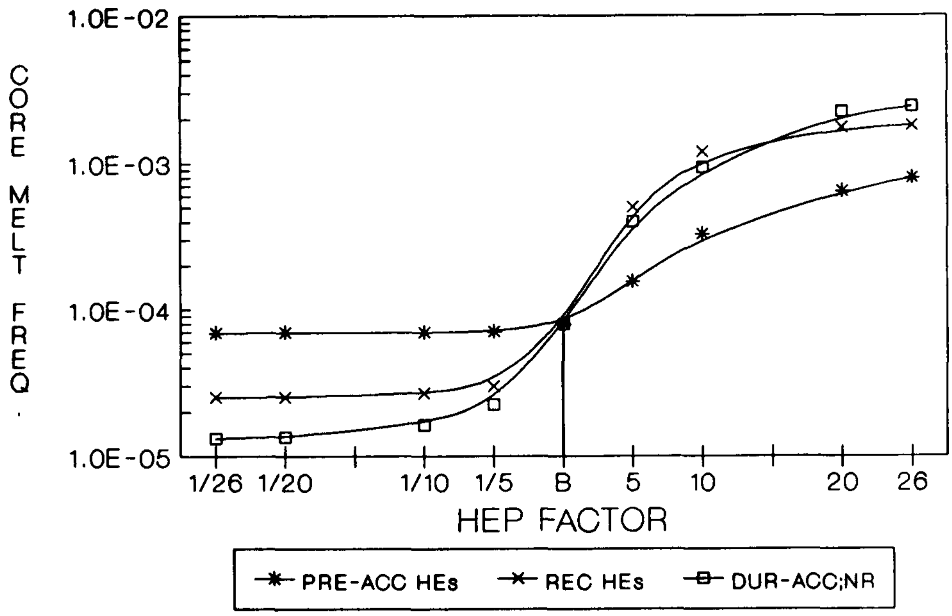


Figure S-6. Recovery error sensitivity

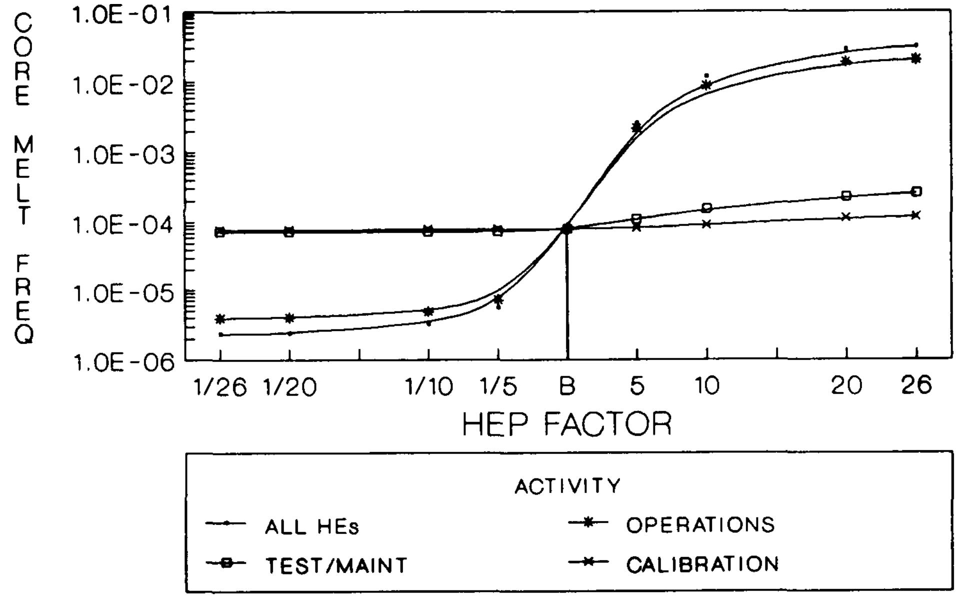


Figure S.7. Sensitivity to utility program activity

As discussed previously, the timing category categorized the human errors relative to the accident initiating event whereas the utility program activity category lists the types of nuclear power plant activity (restoration from test or maintenance, calibration, and operations) that relates to the human error. There are strong interrelations among the categories, for example, operations errors are predominantly during accident errors. These interconnections are evident in the sensitivity evaluations. One point to further emphasize is that the PRAs do not explicitly consider human errors during maintenance that could cause equipment unavailability (the human errors relating to maintenance explicitly included in PRAs are those errors in restoring the equipment in proper status following maintenance, called restoration from maintenance errors in this study). Explicit treatment of such implicit maintenance errors in the sensitivity evaluation is expected to change the sensitivity results.

The sensitivity evaluation of the timing category, Figure S.5, shows the relative sensitivity of the pre-accident initiator and during-accident errors. Note that all recovery errors are categorized as during-accident errors. Figure S.6 shows the sensitivity of core melt frequency when the during-accident errors are split depending whether it is a recovery action or not. Figure S.7, the sensitivity curve for the activity category, shows the relative sensitivity of the errors associated with various types of plant activity - restoration following test or maintenance, test, calibration, and operations.

Based on the results of the sensitivity evaluations, there are a number of consistent observations from the three sets of curves:

1) Dominance of During-Accident Errors: During-accident errors have a strong influence on the core melt frequency. This is consistent with the sensitivity curve for activity category, where the role of the errors of the operations unit is most significant. The operators are primarily responsible for during-accident errors.

2) Control of pre-accident errors: The pre-accident initiator human errors show sensitivity when increased from their base probabilities, but they do not influence the core melt frequency when decreased from their base values. The reason being that the base probabilities associated with such errors are typically small (around 10^{-3} or less), which also contributed in making the cutsets to which they belonged less dominant. This signifies that pre-accident initiator errors need to be controlled at their base values to avoid adverse effects on plant safety, but improvement from currently assumed values is not necessary unless the hardware in the plant is also improved.

3) Importance of "Recovery" Actions: The "recovery errors" as defined in PRAs have strong influence on the core melt frequency. The term "recovery," as used in the Oconee PRA, refers to a manual action taken by operators to restore an interrupted function, usually by initiating alternative equipment, or sometimes, by repairing the equipment that has failed. These actions are taken primarily outside the control room, and are sometimes described in procedures. When during-accident errors are split into recovery errors and non-recovery errors, their sensitivities are significant and comparable. This result reveals an interesting insight on the role of the operations unit during an accident: the performance of procedure-based actions and the performance of those recovery actions, not generally covered in the procedures, are about equally important.

S.3.3.2 Risk Significance of Personnel Categories

During plant operation and accident response, reactor operators perform a number of activities that include their own actions, and coordinating other actions with non-licensed operator and maintenance personnel. Figure S.8, the sensitivity curve for the personnel category, shows the relative sensitivity of the errors according to the responsibility of the plant personel - reactor operators (ROs), non-licensed operators (NLOs), and instrumentation and control technicians (ICTs). Due to the risk significance of the during-accident errors and the reactor operators' role, a further sensitivity evaluation was conducted delineating the various responsibilities of the reactor operators. The sensitivity curves in Figure S.9 show the core melt frequency for changes in HEPs defined by reactor operator (RO) responsibility, reactor operator and non-licensed operator (RO/NLO), and dual reactor operator and maintenance personnel responsibility (RO/MT). Examples of various RO responsibilities in terms of the human errors considered in this study are described below. Operator failure to terminate reactor building spray operation during a small LOCA (YRBSH) is a RO responsibility since this action is to be carried by the reactor operator inside the control room and does not involve any other type of personnel. Failure of the operating staff to initiate Safe Shutdown Facility seal injection in approximately 30 minutes following a normal loss of seal injection (RESSFSI) is considered RO/NLO responsibility, where NLO assistance is

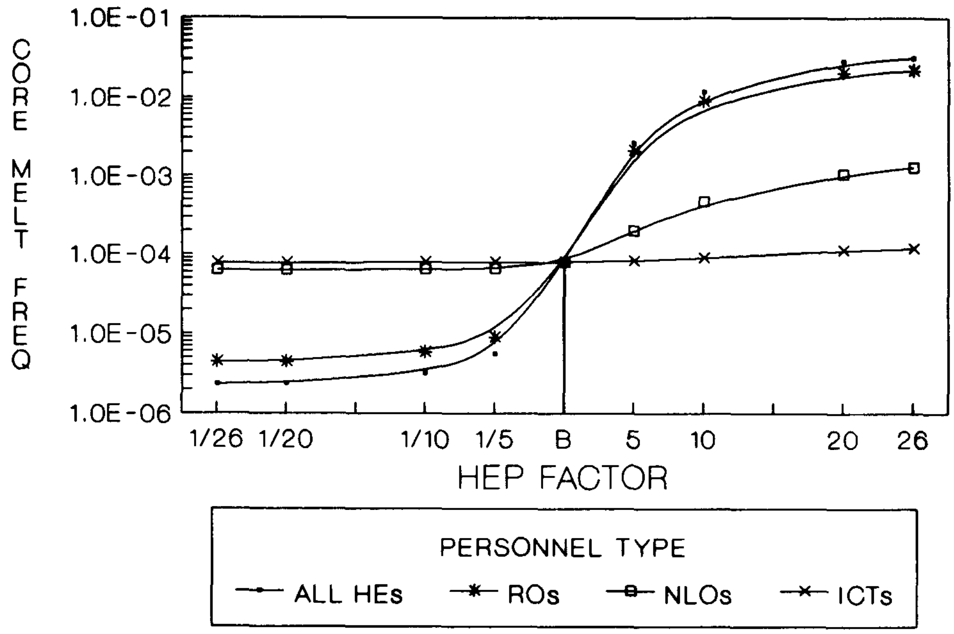


Figure S.8. Sensitivity to personnel category

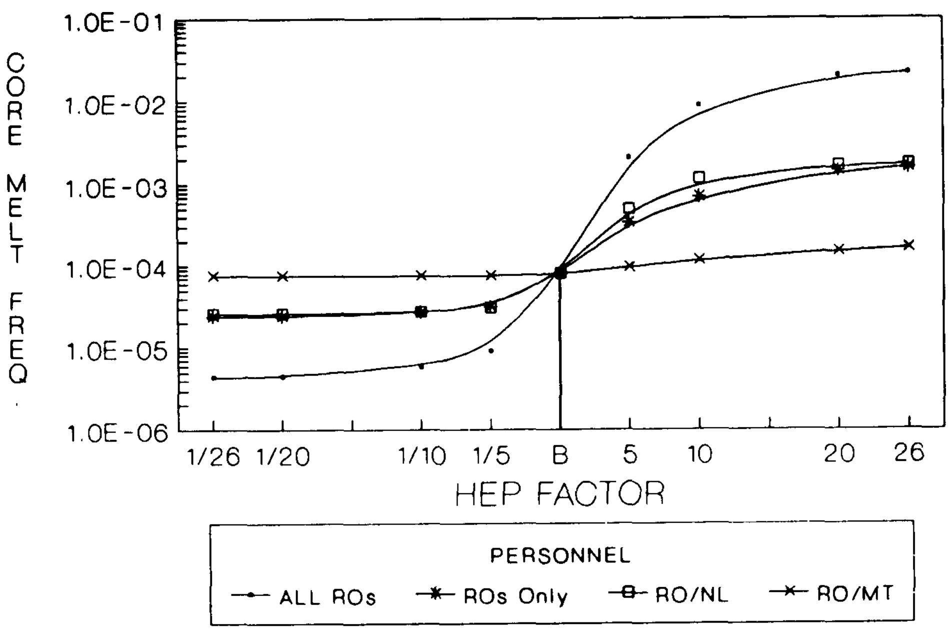


Figure S.9. Sensitivity of CMF to personnel: ROs and further breakdown within RO category required to carry out the action under the ROs direction (who remains the responsible person). Error LPSW-72 leaves the low pressure service water valve to low pressure injection coolers unavailable. This is an RO/MT error since the action is carried out by the operations or maintenance personnel, with operations having the final responsibility.

The insights obtained from these results are summarized as follows:

1) Significance of RO role: The sensitivity results of Figure S.8 show that the errors of reactor operators essentially dominate core melt frequency sensitivity evaluation. Those listed as RO are the errors where the RO has prime responsibility. This is also consistent with the previous observation that during-accident errors in the timing category and operations errors in the activity category are the significant contributors to core melt frequency. The reason for this observation is that almost all cutsets contain RO errors, signifying that ROs always have a role either through failure to perform the required action or committing erroneous action.

2) Significance of RO/NLO Coordination: Among the various responsibilities of ROs, it is observed that the activities of ROs in coordination with NLOs are as significant as those performed by ROs only. This signifies the necessity of coordinating RO/NLO activities in assuring plant safety before and during accidents. These results complement the ones in Figure S.6, showing the importance of recovery errors, since RO/NLO interactions are primarily required in carrying out the recovery actions.

3) Significance of NLO Role: The sensitivity results show the significant impact that NLOs have on plant risk. In Figure S.7, NLO activities (alone), even though not as important as RO activities, have a significant impact on CMF when increased from their base values. These activities are pre-accident initiator activities, and are not monitored by ROs. NLO activities supervised by ROs during an accident (discussed above) also show significant impact on CMF (Figure S-8). Overall, significant risk can be incurred in a plant due to NLO activities.

S.3.3.3 Risk Significance of the Operator Error Types

There are four types of operator errors specified in the Oconee PRA, which are included in this sensitivity evaluation: (i) operator fails to perform desired action, (ii) operator fails to perform recovery actions, (iii) operator inhibits the recovery action (intentionally defeating the function of a system after the initiating event because the situation has been misdiagnosed, and (iv) inadvertent actions (unintentionally defeating the function of a system during an event). The first two classes are omission errors whereas the last two are commission errors. Even though PRAs are criticized for not treating commission error adequately, an earnest effort in accounting for Operator Commission type errors was made in the Oconee PRA. Accordingly, a sensitivity evaluation on these categories show the significance of various types of operator actions. Figure S.10 presents the CMF sensitivity curves for various types of during-accident operator errors. In Figure S.10 all during-accident human errors (DUR-ACC HEs) are divided into: operator inhibits

(OpINHB), operator fails during recovery action (OF/RE), inadvertent action (INADV ACT), and operator fails only in a non-recovery action (OF only). Figure S.11 shows the sensitivity of omission/commission type of errors in the plant, where pre-accident initiator errors are also included.

Evaluation of these sensitivity curves result in the following observations:

1) Dominance of "Operator Fails to" Errors: "Operator fails to" type of actions, namely operator fails to perform desired actions and operator fails to perform recovery actions, dominate the sensitivity curve. The during-accident commission errors (operator inhibits and inadvertent actions) have negligible influence on CMF. For a conclusive judgment on this aspect, a better modeling and data base for operator commission errors will be needed in the PRAs. Current judgment depends on the assumption that PRAs have adequately incorporated at least the significant or important operator commission errors.

2) Significance of Omission versus Commission Errors: As shown in Figure S.11, errors of omission have a significantly stronger influence on the sensitivity curve than errors of commission. This is consistent with previous observations that during-accident errors and "operator fails to" errors have significant influence on core melt frequency. In this sensitivity evaluation, about 168 omission errors as opposed to 55 commission errors are included. Accordingly, it can be argued that the results are partly attributable to the relative treatment of these errors in the PRA.

S.4 Comparison with Previous Sensitivity Study (NUREG/CR-1879)

We compared the findings of this study with the previous human-error sensitivity study of the Surry plant (NUREG/CR-1879). The objectives behind the comparison are: (a) to identify any commonality of insights, (b) to identify any differences in observations and the reasoning behind such differences, (c) to identify any new insights that now may be derived, and (d) to seek any generic implications that may emerge from these two studies. To meet these objectives, three sets of comparisons were performed:

a) Comparison of Core Melt Frequency (CMF) Sensitivity to human errors.

b) Comparison of dominance of categories of human errors for the respective plants.

c) Comparison of sensitivity of accident-sequence frequencies to human errors.

Chapter 6 has a detailed discussion of these items.

The primary focus of this comparison is on the results of the respective evaluations. Although there are strong justifications for such comparisons, there are also significant differences that limit the insights that might be obtained. These differences relate to the following aspects which are discussed in detail in Chapter 6.

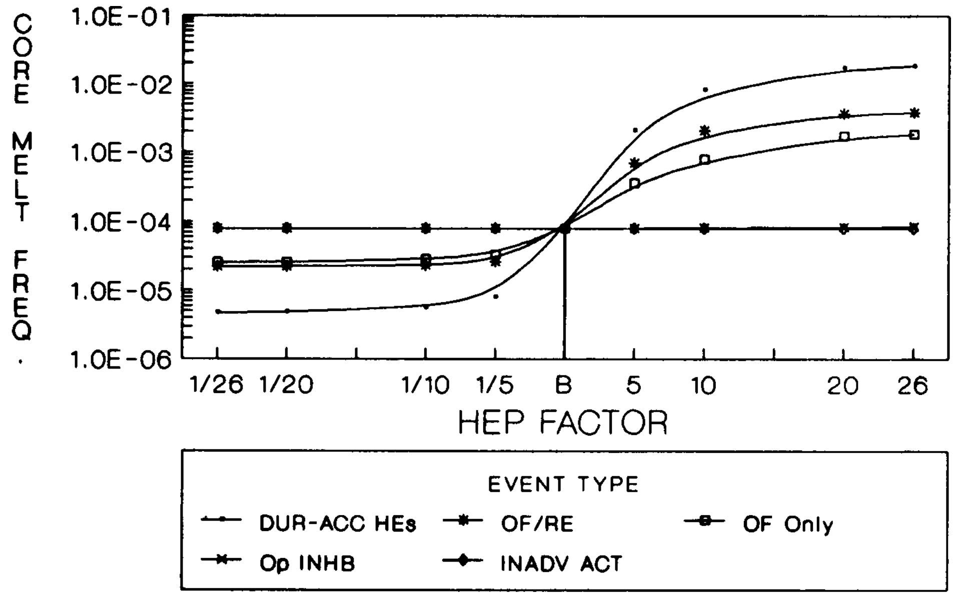


Figure S.10. Sensitivity of during-accident human errors sorted by event type

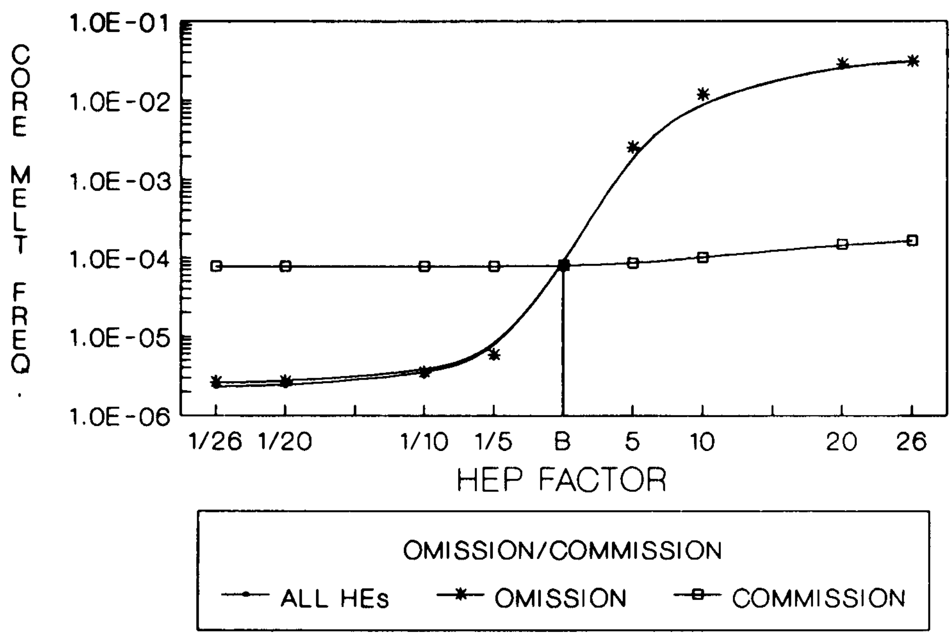


Figure S.11. Sensitivity of omission/commission errors

- Inclusion of larger number of human errors in Oconee PRA.
- Extended treatment of during-accident human errors in Oconee PRA.
- Incorporation of recovery errors in Oconee PRA.
- Reduced hardware failure contribution in Oconee PRA.

1) Stronger sensitivity of CMF to human errors in the Oconee PRA

For a similar variation in the HEPs, the Oconee CMF showed larger variation (about four orders of magnitude compared to about two orders of magnitude) variation in the Surry CMF. It can be argued that because of the design features of the plant, larger numbers of risk-significant human errors with associated high error probabilities are present in the Oconee PRA resulting in stronger sensitivity.However, the modeling of human errors in two PRAs are significantly different, which influenced the sensitivity curves. In particular, two aspects of the human reliability analysis in Oconee PRA are significantly different from the Surry WASH-1400 PRA, and they contribute to the stronger Oconee CMF sensitivity to human errors. First, improved modeling of operator's role during an accident increased the number of such errors and, because of the lack of actual evidence, these errors were assigned high estimates, resulting in stronger sensitivity. The modeling approach of the human errors in the Surry PRA was not as detailed as that in Oconee PRA which resulted in fewer HEs modeled (223 in Oconee compared to 110 in Surry). The other aspect relates to the incorporation of recovery errors, which reduced the base case core melt frequency by introducing other human errors to the combination of events that result in core melt. Thus, the cutsets for Oconee contained more HEs and a greater number of cutsets with multiple HEs. In the sensitivity calculations like ours where all HEs are simultaneously varied, this results in introduction of additional multiplicative factors. This resulted in a much faster increase in the Oconee CMF.

It could not be directly ascertained how much of the difference in core melt sensitivity curves between the Surry plant and the Oconee plant is due to differences in modeling human error. Nevertheless, even if all recovery errors are removed from the analysis, still Oconee CMF shows stronger sensitivity to human errors.

2) Greater Potential for Reducing CMF through Improved Human Performance in Oconee

The Oconee CMF sensitivity curve saturates at much lower HEPs than that of the Surry plant, signifying a stronger dominance of human errors in the Oconee plant. In the Surry plant, the CMF saturated much faster at relatively higher HEPs. This is due to the dominance of hardware related failures in the Surry plant. As HEPs are reduced, their influence relative to hardware failures reduces and at a certain point, hardware failures essentially determine the CMF (saturation of CMF). The results of Oconee sensitivity evaluation show that this saturation occurs at much lower CMF compared to the base CMF, thus providing a greater potential for reducing CMF through improved human performance.

3) Significant variation in CMF around the base HEPs in Oconee

The Oconee CMF showed much stronger sensitivity to human errors around the base value compared to the Surry results. This signifies that considerable lowering of core melt frequency can be achieved by improving (lowering) HEPs, and at the same time, increased HEPs showed a large factor increase in CMF. The Surry CMF showed only about a factor of two improvement for a decrease in HEPs by a factor of ten, whereas a similar decrease in all HEPs in Oconee resulted in approximately a factor of 33 improvement in Oconee CMF. Also, a small factor increase in all HEPs in Oconee resulted in a larger factor increase in Oconee compared to the Surry plant.

The reasoning for these differences in behavior of the curves are primarily: (a) the dominance of human errors in Oconee CMF, (b) multiple human errors in the combination of events that contribute to the CMF, and (c) large base probabilities assigned to human errors. These conditions in risk calculations are attributable both to the plant design features and to the modeling of human errors in the plant.

S.4.1 Comparison of Dominant Categories of Human Errors

Significant differences were observed between the Oconee and Surry plants when the sensitivity of core melt frequency was studied for categories of human errors. Chapter 6 has the specific comparison between timing of error, types of activity, omission/commission, and location of errors. The Oconee CMF showed stronger sensitivity to during-accident errors compared to pre-accident errors, whereas the Surry CMF showed exactly the opposite. The Oconee analysis also showed a stronger sensitivity to operational errors (in activity category) compared to restoration errors in tests and maintenance. In the Surry plant, the CMF was more sensitive to the restoration errors (called test and maintenance errors in NUREG/CR-1879) than operational errors. Since during-accident errors and operational errors are primarily control room errors, the sensitivity of risk parameters to control room errors at Oconee was more significant than non-control room errors. In the Surry evaluation, non-control room errors were more significant, which was expected because of the dominance there of pre-accident errors and restoration errors.

S.4.2 Comparison of Accident Sequence Frequencies

Sensitivity of the frequencies of the individual accident sequences to human errors were analyzed in this study and in NUREG/CR-1879. The way accident sequences were defined in these PRAs was somewhat different, hence, direct comparison is not possible. Nevertheless, a number of general observations can be made.

1. <u>Sensitivity of dominant accident sequences</u>

The dominant accident sequences in both plants are strongly sensitive to human errors. For the Surry plant, the very small LOCA sequence S_2C, and the transient sequence TMLB' were dominant and were among the human error sensitive sequences. In the Oconee plant, the dominant and sensitive sequences were Loss of Instrument Air, Loss of Service Water, and very small LOCA. In all these sequences, a significant increase in risk is observed when HEPs are increased, but for the Surry dominant sequences, the potential for decrease in the sequence frequencies was limited.

2. Dominance of transient-initiated sequences for increased HEPs

The transient-initiated accident sequences were strongly sensitive to human errors in both studies. Large LOCA-initiated sequences and Vessel Rupture sequence are among the least sensitive in both plants. When the HEPs are increased, the dominance of transient-initiated events increases in both plants. This is expected since the transient-initiated events have a significant human role compared to large LOCA or Vessel Rupture sequences.

S.5 Summary of Major Findings

A sensitivity evaluation was conducted to assess the impact of human errors on the risk parameters in the Oconee plant. The results show the variation in the risk parameters, namely core melt frequency and accident sequence frequencies, due to changes in human error probabilities. The major findings are summarized below. Additional key insights (particularly relating to the specific categories of Human Errors) are provided throughout this overview summary, and more detailed information is contained in the body of the report.

1) Significant variation of risk parameters on human errors

The sensitivity evaluations for core melt frequency and accident-sequence frequencies show variation over four orders of magnitude when human error probabilities vary from their lower to upper bound. During plant operation, human error probabilities are not expected to vary simultaneously to such extremes, and for practical considerations, variations within a short range surrounding the base error probabilities may be of more interest. Therefore, it is noteworthy that significant increase and decrease in risk (more than an order of magnitude) occurs when all human error probabilities are increased or decreased by factors of 3 to 10 from base values.

2) Sensitivity of dominant accident sequences

Many dominant accident sequences show strong dependence on human errors. Also, the accident sequences with a high frequency of initiating events show strong sensitivity to human errors. Thus, the events that are more likely to occur during the life of a plant can become significant safety concerns if employees do not perform their role adequately. Specific human actions that may be necessary in such events can be identified from a study such as this, and adequate procedures may be developed to help to train the operating personnel.

3) Level of improvement in plant risk due to improvement in human performance

The results also indicate that a significant improvement in the plant's risk parameters can be achieved through improvement in human performance. A relatively small improvement in HEPs (about a factor of two) can result in factors of 10 improvement in many accident sequences. Human factor studies can be conducted of those errors to identify specific measures to improve human performance.

4) Burden on the operations unit

In analyzing the during-accident errors, including recovery errors, it became apparent that the risk level in the Oconee plant strongly depends on the activities of the operations unit. Thus, there is a significant burden on the plant management and on the operating staff to control the risk from the operations of the plant. In many accident-initiating events, reactor operators have to conduct multiple activities, where more than one may involve coordination with non-licensed operators performing specific tasks outside the control room. In certain instances, such activities are to be carried out without the benefit of specific procedures.

1. INTRODUCTION

1.1 Background

The risk to the public from the operation of commercial nuclear power plants (NPPs) is a topic of much current interest. The majority of the risk to the public comes from low likelihood, high consequence events involving severe damaage to the reactor core of the nuclear power plant. In order to adequately understand these low likelihood events, a very detailed analysis must be performed. Analysis methods called probabilistic risk assessment (PRA) have evolved over the last 15 years in the nuclear power industry to address the risk from NPPs. The first commercial NPP PRAs were completed as part of the Reactor Safety Study (WASH-1400) in 1975. Since then, PRAs have been completed for 30 to 40 U.S. NPPs. These PRAs consist of a very detailed model of the plant hardware, components, and systems. The model also incorporates the interaction of the plant systems with each other and with the humans who operate and maintain the hardware.

As PRA modeling evolved over the last decade, several different and improved techniques for human performance modeling have been developed. Nonetheless, the human performance modeling, often called human reliability assessment (or HRA), remains difficult to accurately quantify and hence, has relatively large uncertainties associated with it. Increasingly, the analysis of actual events in nuclear power plants (NPPs) shows the significance of the human role in the risk from plant operations. Accordingly, increased attention is being paid to appropriately model human behavior and to understand the impact of any variation in human performance.

A number of different studies (Kelly, J.E., Parkinson, W.J., and NUREG/CR-1879) have indicated that the human contribution to overall NPP risk is substantial. NUREG/CR-1879 described a detailed sensitivity study to determine quantitatively how much plant risk would change as the human error probabilities (HEPs) were varied. The study evaluated the effect on risk using several risk measures, most notably the core melt probability per reactor year, and found significant increases as the HEPs were increased. Other more recent studies (e.g., Trager, E.A. Jr.) found that human errors contribute to over 50% of the significant events that have occurred at NPPs. Thus, human performance in nuclear power plants is a very important aspect of plant operation and can contribute significantly to overall plant risk.

1.2 Purpose

The specific purpose of this study was to identify the impact of human errors on plant risk levels, to identify specific aspects of human errors that have high impact on risk, and to identify categories of human errors whose improvement (in terms of lowered probabilities) can provide significant risk benefits. In addition, the management of nuclear power plants is considered to influence human errors in the plant, and this sensitivity evaluation is expected to provide some insights into the link in understanding the management's influence on plant risk. As an example, described in detail later, management affects training of operators, which in turn affects a significant number of risk important human errors.

Beyond the specific purposes mentioned above, this study, particularly when supplemented by follow-on studies, may be useful to NRC staff and NPP personnel in identifying the areas of human performance which are risk significant and hence, deserving of attention. The results also should be useful in providing insights to analysts in the PRA and the HRA areas.

The earlier sensitivity study (NUREG/CR-1879) used the probabilistic risk assessment (PRA) of the Surry PWR plant, which was one of the two reference plants in the original full-scale commercial nuclear power plant PRA (WASH-1400). Therefore, we chose a more recent, "state-of-the-art" PRA based on a different vendor design, namely a Babcock-Wilcox PWR plant. Since this PRA has the most complete HRA modeling available, it should come closest to providing realistic insights regarding the effect of human performance on risk. Also, since the accident sequences at Oconee are fairly typical of PWR PRAs, the certain results could also be generalized to other PWRs by using careful analysis. A follow-up sensitivity study will be performed on a recent BWR PRA (i.e., LaSalle).

The PRA for Oconee Unit 3 used for this study was performed jointly by the Nuclear Safety Analysis Center of the Electric Power Research Institute (EPRI) and by the Duke Power Company, the owner and operator of the Oconee NPP site. This PRA was published as NSAC-60 in June, 1984. The PRA was reviewed (and modified slightly) by Brookhaven National Laboratory in NUREG/CR-4374.

1.3 Scope and Limitations

To achieve the purpose of this study, the sensitivity evaluation technique was chosen. Based on this choice and the selection of the Oconee-3 PRA, the scope and limitations of the project can be defined. These are discussed together in the following paragraphs.

A sensitivity evaluation technique was selected because this allows one to see the effect on risk of large and small changes in human error probabilities (HEPs) without establishing conclusive reasons or methods for the changes used. One can vary all the HEPs or only certain categories, and hence, obtain a wide range of information about plant risk. Human error rates can vary considerably over time at a given plant or between plants, hence, this technique can give insights into the effect on risk of such variations. The selection of the Oconee-3 PRA in meeting the purpose of this study was discussed above in paragraph 1.2.

The human error sensitivity evaluation conducted in this study used a probabilistic risk assessment (PRA), and the limitations of PRAs are applicable to this sensitivity evaluation. The only human errors considered in this sensitivity evaluation are those modeled in the PRA. If there are additional human errors that were not modeled, then the sensitivity results would not be complete.

In determining how to vary the human error probabilities, primarily statistical techniques were used (see Chapter 4). The determination of an appropriate behavioral model for HEP variation and the verification of base case HEP values are both acknowledged to be important areas for research but were beyond the scope of this study.

This study focussed on risk parameter values resulting from "internal events" (as termed in PRA literature) such as plant transients, and loss-of-coolant accident. External event sequences (such as fires, floods, and earthquakes) also contribute significantly to the overall core melt frequency, but were not analyzed in this sensitivity evaluation. Also, the human actions which appear only in the external event sequences were not included in the study. A sensitivity evaluation on external event core melt frequency could be conducted in a manner similar to this study, but it was decided not to be based on the following reasons.

It was generally felt that while external events may present a noticeable risk, the likelihood of their occurrence (initiator frequency) is low. Also, equipment damage from the initiating external event (e.g., flood, fire, or earthquake) is much higher than in internal event sequences. As a result, the potential for human actions to recover failed equipment is much less. Hence, it was felt that there was less need to develop insights for these sequences and that the sensitivity to human actions would likely be lower than for internal event sequences.

The risk parameters chosen are the accident sequence frequencies and the core melt frequency. Other risk parameters, namely, the total risk (frequency x consequence), function unavailabilities and safety system unavailabilities can also be analyzed in a similar manner, but such analyses were outside the original scope of this study. Additionally, the type of insights sought from the study were found to be directly obtainable from those parameters studied. To address the impact of human actions on total risk, incorporating the consequences of an accident throughout a sensitivity evaluation would incur significant additional time and effort. Therefore, in this study, consequence bin frequencies were analyzed to obtain insights on the impact of change in human error probabilities on accident consequences.

This project was approached in an integrated team fashion, using people knowledgeable in human performance, plant operations, risk assessment, and NRC programs. This allowed the different disciplines to interact and produce overall results that are well-coordinated and considered the various aspects of NPP human performance.

1.4 Organization of the Report

Section 1 of the body of this report provides a basic introduction to the project and the report. Section 2 details the methodology developed and used for the study. Section 3 discusses the development of the Categorization scheme for the Oconee human errors (HEs), the coding of these HEs into the scheme, and provides the results of this categorization process. Section 4 develops appropriate ranges in both increase and decrease direction over which the various HE probabilities will be varied as part of the sensitivity evaluation. Section 5 describes the various sensitivity calculations performed, gives graphs of the results, and also interprets and summarizes the results of these calculations. Section 6 compares the extensive results of this study to the earlier study of NUREG/CR-1879. Section 7 discusses potential future research and Section 8 lists references. The Appendices provide additional details on specific aspects of the study.

2. METHODOLOGY

This section gives an overview of the full methodology employed in this project. Simply put, the sensitivity evaluation consists of varying the input parameters (human error probabilities) and determining the resultant variation in the output risk parameters, such as core melt frequency. The methodology is largely based on three tasks: (1) a determination of the full set of input parameters (human errors), (2) the consideration of the range over which the input parameters vary, and (3) an assessment of the sensitivity of the plant risk parameters to the input parameters. The sensitivity study was performed in four stages to identify the human role in various aspects of nuclear power plant operation and to indicate areas for potential improvement in human performance. The significant results might be used to derive useful insights for varied regulatory applications.

The three basic tasks of the sensitivity evaluation process are shown in Figure 2.1. They are then further subdivided into nine subtasks that constitute the detailed elements of the process.

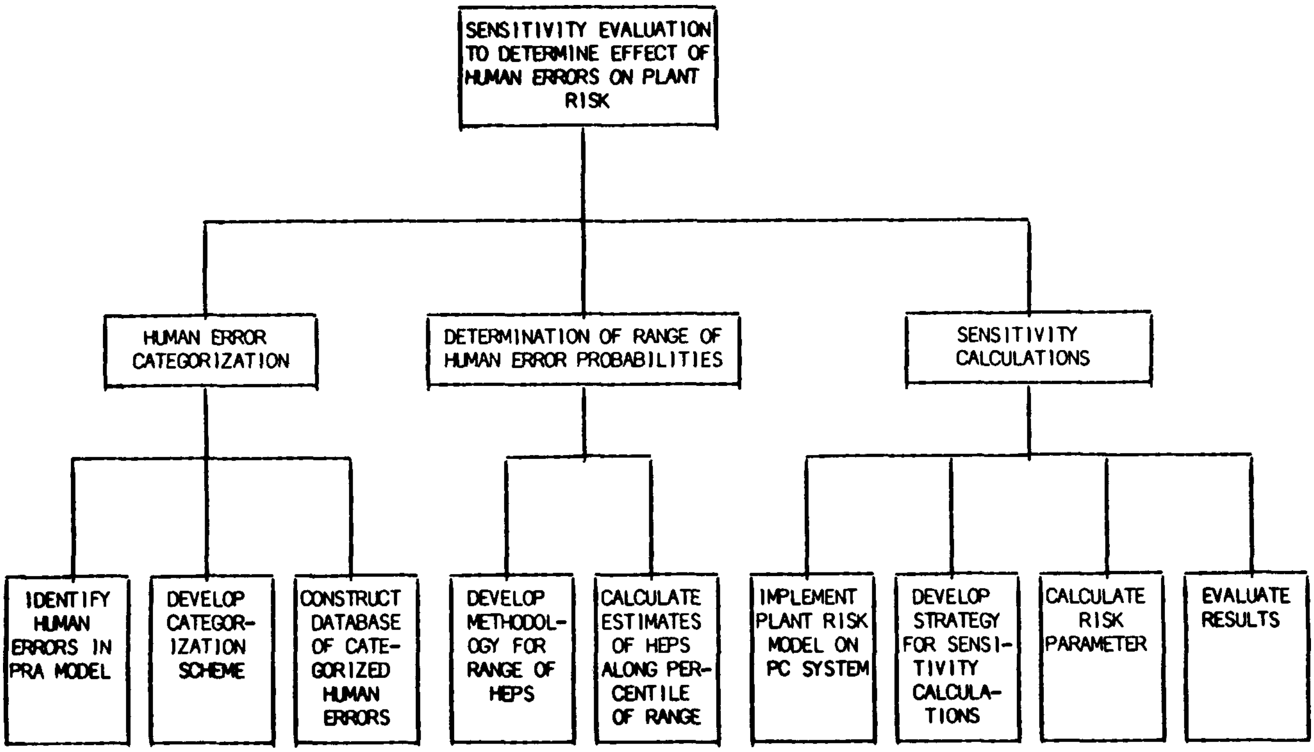


Figure 2.1. Sensitivity evaluation: basic tasks

The task that determined the set of input parameters entailed categorizing all human errors in the PRA (see Section 3 for details). In parallel with the extraction of human errors from the PRA, a categorization scheme was developed and the human errors were categorized in terms of types of activity, location, personnel involved, etc. Utilizing the categorization scheme, each human error was coded to identify specific characteristics that define the sub-element of a category. For example, a human error committed by auxiliary operators is coded as an error by a non-licensed operator (NLO) under the personnel category. A database was subsequently constructed on a PC-based data management utility, called "dBase III-plus," which allowed convenient analysis and quick sorting of the categorized human errors as input for the sensitivity study.

An important consideration in the risk-based sensitivity evaluation process is to define the entire range of variability of the input parameters, whose effects are being evaluated. To determine the range of the human error probabilities (HEPs), a methodology was developed, which incorporated the various known causes of uncertainty in HEP estimation. This methodology used the statistical approach of analysis of variances to first characterize the influences of each cause of variability in terms of error factors and variances, and then combined them all to obtain the overall range. The application of the derived error factors and ranges facilitates the determination of upper bound and lower bound estimates for each HEP.

Central to the assessment of the sensitivity of plant risk parameters is the performance of sensitivity calculations that show the change in plant risk level due to variations in human error probabilities up to their calculated error bounds. This task required the implementation of the plant risk model (or PRA) on a PC system to complete computations efficiently. The plant risk model, which contains the Boolean expressions of the minimal cutsets for various accident sequences, was created from the functional logic models using the event/fault tree methodology of the PRA.

Next, a detailed strategy for performing the sensitivity evaluations was needed to properly and logically guide the various calculations. This strategy, detailed in Section 5, allowed the program to concentrate on various aspects of the human error sensitivity issue, such as overall effect of human errors, effect of specific types of error, effect of groups of errors caused by different plant organizations, effect of variation in error rate on the types of accident sequences that dominate risk, and the effect of routine versus post-accident errors. Once the initial results were obtained, some minor additions and deletions were made to the strategy based on information learned. The risk parameters evaluated were the core melt frequency (CMF), the accident sequence frequencies, and the consequence bin frequencies. The CMF represents the overall plant risk which was obtained by the summation of frequencies of all event sequences leading to core melt. As such, the sensitivity of core melt frequency is an indicator of plant risk with generic implications, whereas the sensitivity of accident sequence frequency relates the dominant risk contributors in the likelihood of the particular accident sequence. The sensitivity of the consequence bin frequency relates the effects of in-plant consequences from the accident sequences defined in a core-melt bin. The calculation of these risk parameters for a set of HEP variations provided the output from which risk sensitivity curves were plotted.

3. CATEGORIZATION OF HUMAN ERRORS IN OCONEE PRA

3.1 Introduction

The purpose of this project was to identify, characterize, and determine the sensitivity of critical human performance actions/errors. Therefore, one necessary task was the detailed categorization of the human errors treated in the Oconee PRA. A categorization scheme is useful in providing a system of classification within which the categories can be specifically defined.

Several different categorization schemes can be developed to understand and evaluate the human errors modeled in a PRA. The modeling of human errors in one PRA as opposed to another, often results in different approaches to categorization. We developed a comprehensive scheme for this project that can be utilized across different PRAs, encompassing differential modeling of human errors. The scheme was developed with certain objectives:

a. Human errors were characterized with minimal ambiguity from the description given in the PRA. Implicit assumptions about the error were avoided wherever possible.

b. Categories were chosen for their applicability to different models of human reliability analysis.

c. Previously developed categories (e.g., NUREG/CR-1879) were included in the categorization scheme wherever appropriate.

3.2 Method/Approach

The first step in developing the categorization scheme for human errors in the Oconee PRA, was to closely examine the categories used in other studies. Specifically, the original work for this program, NUREG/CR-1879, used a set of categories to describe the human errors in that study. The categories included the system involved in the error, the component described in the error, whether the cause was an act of commission or omission, the timing of the error (pre- or post-accident), where the error occurred (location in the plant), the action involved in the error (operations, calibration and restoration from test or maintenance) and the probability assigned to that error.

NUREG/CR-4103 also used a classification scheme for human errors with similar-type categories, that included information about personnel, the system involved, whether the error was an act of omission or commission, the type of error, and the associated performance-shaping factors.

Only three of the categories were common to both schemes, the reported human error probability, the system identified with the error, and whether the error was an act of omission or commission. In developing the categorization scheme for this project, we used those common categories, several from each of the two schemes, and some new ones.

3.2.1 Identification of Categories

Table 3.1 lists the categories included in the scheme for this project, with a brief description of where the information was obtained and some of the codes used to describe each error. Section 3.3 has a detailed explanation of the categories and an example of each.

Table 3.1. Scheme for Categorizing Human Errors

Category	**Identification**
CODE	PRA Code
PAGENO	Location in PRA
TIMING	Pre-Accident Initiator (P), During Accident Initiator (D)
ACCINIT	Initiating Event (e.g., LBLOCA)
SYSTEM	Hardware System (e.g., DCPS)
COMPONENT	Unit of System (e.g., Valve)
PERSONNEL	Individual Involved (e.g., Reactor Operator (RO))
OMCOM	Omission (OM), Commission (COM)
EVENTTYPE	From PRA Model (e.g., unavailability, recovery)
LOCATION	Control Room (CR), Outside Control Room (OCR)
ACTIVITY	Utility Program Activity (e.g., Operations (O))
HEP	Numerical Human Error Probability in PRA
DEPEND	PRA Defined Dependencies Between Events
OCIMPT	PRA Defined Important Event
OTHERINF	Statistically Generated Category
NRCPGM	Relationship to NRC Inspection Program (e.g., Operations (OPS))
HIHEP	Upper Bound of Range
LOHEP	Lower Bound of Range

3.2.2 Relationships of Categories

In developing the categorization scheme, not all of the categories were independent of each other. Some categories were identified by the coding in others. For example, if a human error was determined to be committed by a non-licensed operator (personnel category), by definition, the event occurred outside the control room (location category). Similarly, if an error was determined to be of the unavailability type (event type category), it occurred before an accident initiator (P in the timing category). The interrelationships between some of the categories were very well-defined in the modeling of the PRA. When this type of relationship was not evident, it was implicitly defined in the categorization of the human error.

The lack of independence between some of the categories is important for interpreting the analyses (discussed later). Some analyses were performed with only one category and, in fact, but may represent at least two or three categories, depending upon their relationships. For example, the data describing errors of the unavailability type also describes errors occurring before an accident initiator. These relationships between categories are better defined in the specific discussions.

3.2.3 Process of Extraction of Human Error

3.2.3.1 Method of Extraction

The Oconee PRA (NSAC-60) and the Brookhaven National Laboratory review of the PRA (NUREG/CR-4374) were reviewed to identify all human errors. In the PRA, errors were extracted from system-fault trees and initiator-event trees. Human errors were generally (but not always) identified in the PRA by coding the errors with an "H." Five hundred and fifty three errors were identified, which subsequently were reviewed and edited. Sixty-four were deleted because they were not relevant to the program, e.g., errors concerned with external events. The remaining 489 errors constituted the initial database for further work.

3.2.3.2 Development of Database Using DBASE III Plus Software

A database was constructed for the 489 human errors with DBASE III Plus Software operating on an IBM PC for data entry and management. Each category of the 16-element categorization scheme was set up as a field, with a pre-determined size based on the category's coding. Each human error (HE) then was defined as a record with 16 fields and a size of characters. As the HEs were coded, as described below, they were entered into the database. This database provided excellent capability to manage, sort, count, and analyze the HEs.

3.2.3.3 Comparison with Oconee PRA Computer Data and Truncation

The Oconee PRA computer tapes used in the earlier BNL review of the Oconee PRA (NUREG/CR-4374) were re-run for the current project. Since the BNL review had modified the PRA (described in the NUREG/CR), in order to model risk more realistically, it was necessary to cross-check the human errors in the computer model with the PRA. Initially, 25 errors were identified from the computer tapes that had been added to the PRA model. These errors then were found in the NUREG/CR-4374 tables and trees to complete the human error list, and were coded to complete the data base. Also, many of the less important HEs from the PRA did not appear in the final computer model, either due to modeling changes or truncation of lower level cutsets. A final truncation process, described in Section S.2.5.1 was implemented, to obtain only those HEs which would have some measurable effect on risk. The reconciliation and truncation of the errors left 223 HEs in the final data base.

3.3 Application of Categorization

Table 3.1 in Section 3.2.1 above briefly identifies the categories developed at BNL for PRA-related human errors. This section provides documentation for the application (or coding) of the categories to the Oconee 3 PRA Human Errors. Each of the 16 categories are discussed, with at least one example taken directly from the Oconee PRA. Table 3.2 applies the Oconee PRA human errors to the categories by a series of codes, which are either fully contained in the table or detailed further in Appendix A. The following section discusses each category in Table 3.2 with examples from the Oconee PRA. Section 3.4 gives the summary and results of the categorization.

Table 3.2. Categorization Scheme and Codes for Human Errors from Oconee 3 PRA

Category	Codes For Each Category
1. CODE	Event Code (from Oconee PRA)
2. PAGENO	Page Number (from Oconee PRA)
3. TIMING	Pre-Initiator (P), During (or After) Initiator (D)
4. ACCINIT	LOSW, LBLOCA, SBLOCA, ATWS, FLB, LOIA, SGTR, LOOP, T/RT, TT, LOFW, OTH TRANS, RVR, INF LOCA - (Accident Initiating Event)
5. SYSTEM	ACPS, CFS, DCPS, ES, EFW, HPI/R, HVAC, IA, ICS, LPI/R, MFW, OFPWR, PCS, RBC, RBS, RCS, SSF, SW (Appendix, Table A-1 contains listing of Oconee 3 Plant Systems used in Oconee PRA)
6. COMPONENT	AOV, MOV, VV, PMP, SW/CONTROL, PS, BRKR, S, etc. (Appendix, Table A-2 contains listing of Oconee 3 Plant Components (Hardware Units) used in Oconee PRA)
7. PERSONNEL	Licensed Reactor Operator (RO), Nonlicensed Operator (NLO or NL), Maintenance Technician (MT), Instrumentation and Control Technician (ICT)
8. OMCOM	Omission (OM), Commission (COM)
9. EVENTTYPE	Unavailability (U), Operator Inhibits (OI), Inadvertent (I), Operator Fails (OF) and Operator Fails/Recovery (OF/RE) - (from Oconee PRA - defines TIMING and OMCOM codes)
10. LOCATION	Control Room (CR), Outside Control Room (OCR)
11. ACTIVITY	Operations (O), Restoration from Maintenance (M), Restoration from Testing (T), Calibration (C)
12. HEP	Numerical Human Error Probability (from Oconee PRA)
13. DEPEND	True (T)/False (F) (from Oconee PRA - dependencies between events)
14. OCIMPT	True (T)/False (F) (from Oconee PRA - important human event)
15. OTHERINF	Coded - 1,2,3
16. NRCPGM	OPS, P, TR, etc. (need more codes) (NRC Program relationships to Oconee PRA human events)
17. HIHEP	Upper Bound of Range
18. LOHEP	Lower Bound of Range

1. CODE - This alphanumeric category is the actual event-name identifier of the PRA human error developed in Appendix A (Section A1.3.5) of NSAC-60, the Oconee PRA. Several examples are SWC89VVH, RESW78, and EFPSIVH.

2. PAGENO - This category documents the number of the page in the Oconee PRA where the human error event-identifier is located. For example, the event SWC89VVH is found on page A14-49 of the Oconee PRA (Volume 3, Appendix A14, Table A14-2A). Note that the description of this event in the table is "Manual valve CCW-89 left closed by operator." If the human event has a PAGENO which starts with the letter "N," the human error probability (HEP) associated with the identifier will be found in Table A.1 of NUREG/CR-4374, Volume 1 ("A Review of the Oconee 3 PRA, Internal Events Core Damage") instead of the Oconee PRA. Moreover, if the PAGENO has A-32, A-33, A-34, or A-35 after the "N," then the human error event identifier itself was obtained from that page of NUREG/CR-4374, Volume 1, and not the Oconee PRA. For example, the PAGENO of NA-34 for identifier RESW78 indicates that the event is identified on page A-34 of NUREG/CR-4374, Volume 1. Finally, there are several identifiers developed in support of NUREG/CR-4374, which are not documented therein and not used in the original Oconee PRA. These identifiers have a PAGENO starting with an "N" followed by a page number for the Oconee PRA fault-tree where they would be added, and ending with an asterisk to signify their unique status. For example, the PAGENO for identifier EFPSIVH is NA10-32*.

3. TIMING - This category provides the timing of the human event in chronological relationship to that of the accident-initiating event or transient. A human event which is categorized as "Pre-Initiator" (P) is one that occurs before, while one which occurs during (or after) an accident-related initiating event or transient is categorized as "During (or After) Initiator" (D). The TIMING category code is determined by the EVENTTYPE category code assigned to each human event identified in the Oconee PRA. For example, SWC89VVH is an event which is defined by the EVENTTYPE category of "Unavailability Error" and as a result, is designated here as having a TIMING code of "P". Therefore, it occurs before an accident-related initiating event. The EVENTTYPE category gives more information about TIMING.

4. ACCINIT - This category lists the accident-initiating events. A human event is coded with those initiators corresponding to all sequences in which the event appears. The following accident initiators are listed as they appear in Table 5.9, the Oconee PRA (entitled "Oconee Updated Initiating-Event Frequencies"):

ACCINIT	DESCRIPTION
FLB	Feedwater-Line Break
LBL	Large Break Loss of Coolant Accident (LOCA)
LOC	Loss of Condenser Vaccuum
LOIA	Loss of Instrument Air
LOIC	Loss of ICS Power Bus KI
LOM	Loss of Main Feedwater
LOO	Loss of Offsite Power

ACCINIT	DESCRIPTION
LOS	Loss of Service Water
L4K	Loss of 4KV Switchgear 3TC
SBL	Small or Very Small Break LOCA
SEF	Spurious Engineered Safeguards Actuation Signal
SGT	Steam Generator Tube Rupture
SLP	Spurious Low Pressurizer Pressure Signal

For SWC89VVH, the associated ACCINT categories are LOC, LOIA, LOIC, LOM, LOO, LOS, L4K, SEF, and SLP.

5. SYSTEM - The SYSTEM category provides the Oconee 3 plant system associated with the PRA human event. Table A-1 in Appendix A gives a complete listing of all the Oconee 3 PRA systems identified by BNL as appropriate for one (or more) Oconee PRA-related human events. Using the example of SWC89VVH, the Oconee 3 system associated with this human error is identified as "SWCCW," the Service Water-Condenser Circulating Water System.

6. COMPONENT - This category provides the Oconee 3 plant system component (or "subcomponent-unit") associated with the human event. Table A-2 in the Appendix has a complete list of all appropriate BNL identified components for the Oconee PRA-related human events. For the SWC89VVH example, the appropriate component is "VV," a valve locally controlled by hand. Note that the component selected by BNL represents the principal unit that the person should be dealing with, not necessarily the associated control device that the person would have to physically manipulate. If a human event deals with multiple components of different types, then it is coded with an "S" for system.

7. PERSONNEL - The PERSONNEL category identifies the type of individual most responsible for the human event. The following is a complete listing of all PERSONNEL code entries developed by BNL for the Oconee PRA-related human event:

PERSONNEL	DESCRIPTION
RO	(Licensed) Reactor Operator
NLO (or NL)	Nonlicensed Operator (Equipment or Auxiliary Operator)
ICT	Instrumentation and Control Technician
RO/NL	Event involves both ROs and NLOs with the ROs assumed to be more responsible than the NLOs
NL/MT	Event involves both an NL and MTs with the NL assumed to be more responsible than the MT
RO/MT	Event involves both a Reactor Operator and a Maintenance Technician (MT) with the RO having primary responsibility.

Again, using SWC89VVH as an Oconee PRA-related human event, the associated PERSONNEL category code of "NLO" was used.

8. OMCOM - This category identifies human errors of omission (OM) or commission (COM). As used here, acts of omission involve actions which were expected to be accomplished, but were not even attempted (therefore, not completed). In other words, an act of omission is the failure to attempt to perform of a desired action. Conversely, an act of commission involves the completion of an improper action, or an unsuccessful attempt to perform a desired action (or series of associated actions) to achieve a specific goal. Like the TIMING category, the OMCOM category is determined for the EVENTTYPE category code assigned to each human event identified in the Oconee PRA. The example of SWC89VVH was assigned an EVENTTYPE code of "Unavailability Error", and hence was designated as an omission error (OM). In addition to OM or COM, the OMCOM category provides a very short narrative of explanation. For SWC89VVH, the "OM" is further described as "NOT OPEN."

9. EVENTTYPE - The EVENTTYPE category identifies "Categories of Human Error," established in the Oconee PRA. The Oconee PRA explains that four categories were chosen as representative of human behavior. They are Unavailability, Inadvertent Actions, Operator Inhibits, and Operator Fails To, and are defined below.

   - Unavailability Error (U) - Unavailability errors result in a system or a component being unavailable (or degraded) as the event or transient evolves, or may even be involved in initiating the event. In the Oconee PRA, unavailability errors occur before an initiating event or a transient (therefore, the associated TIMING category code is "P" for Pre-initiator). Also, the associated OMCOM category code is "OM" for an act of omission. An example of the "U" code is "a valve left in incorrect position after test or maintenance."

The three remaining "Categories of Human Error" are assumed to concern events that occur after the initiating event. Therefore, their associated TIMING code is "D" for During (or After) Initiator.

   - Inadvertent Actions (I) - human errors that unintentionally defeat the function of a system (associated OMCOM category code is "COM" for an act of commission) during an event. Typically, these errors are at the component level. An example of the "I" code is "a valve inadvertently closed."

   - Operator Inhibits (OI) - human errors where an "operator" intentionally defeats the function of a system (associated OMCOM category code is "COM") after the initiating event because the situation was misdiagnosed. Typically, these errors are at the system level. An example of the "OI" code is "operator inhibits the LPI system."

   - Operator Fails To (OF) - human errors where an "operator" fails to perform a necessary action (associated OMCOM category code is "OM" for an act of omission) during the event or transient. An example of the "OF" code is "operator fails to perform a required action." Note that "OF/RE" (Operator Fails To/Recovery) is a specific case of "OF," namely, to identify those events which have been designated as accident recovery actions.

For the SWC89VVH example, the Oconee PRA established EVENTTYPE category is "U," for an Unavailability Error.

10. LOCATION - This category identifies where the person considered most responsible for the human event (and its possible error) is located, that is, either in the Oconee 3 Control Room (CR) or Outside the Control Room (OCR). The CROCR LOCATION coding indicates that there is sufficient uncertainty as to where the personnel considered most responsible for the human event are located. The CROCR coding also included events that had multiple actions inside and outside the CR. The example SWC89VVH has a LOCATION category code of "OCR."

11. ACTIVITY - The ACTIVITY category provides the type of activity being (or that should be) performed during the human event. The following is a complete listing of all code entries developed by BNL for the ACTIVITY category: "Operations" (O), "Restoration from Maintenance" (M/R), "Restoration from Testing" (T/R), and "Calibration" (C). These codes can occur in combination. The Activity code for the SWC89VVH example is "O". Like all PRAs to date, the only maintenance-related Human Errors explicitly modelled in the Oconee PRA are the errors of failure to properly restore components to their normal operational status after maintenance. The failure to properly restore valves as is common at NPPs is considered the primary responsibility of the operations department. This responsibility may fall on the RO or NLO, depending on the location of the valve. Also, maintenance personnel often have a secondary responsibility. Errors committed during maintenance, which would cause equipment to fail later, when required to operate, are only included implicitly in the data on hardware failure rates.

12. HEP - This category gives the numerical value of the Human Error Probability assigned by the Oconee PRA or NUREG/CR-4374 for each human event error. For the SWC89VVH example, the HEP is 0.0008.

13. DEPEND - The DEPEND category provides a declaration of dependency between human events. This code records dependency by a "T," if True and no dependency by a "F," if False. This declaration was defined by the Oconee PRA. The DEPEND category code for the SWC89VVH example is "F."

14. OCIMPT - This category identifies the Oconee PRA defined important events by a "T," if true; if not, a "F" for False is used. The Oconee Important Human Errors are described in Appendix C of NSAC-60. For the example SWC89VVH, the OCIMPT category code is "T."

15. OTHERINF - The OTHERINF category, generated by BNL, provides Other Influences. This category was statistically generated, using a multivariate analysis, specifically a principal components factor analysis (Harris, 1975) with the categorical data already developed in the human error database. Two distinct groups of human errors were identified. One group is composed of errors of operations. Licensed reactor operators (RO) are the personnel most often committing these errors, and the errors are largely the result of operators failing to recover. Increased NRC Inspection in the areas of training and operations could lead to a higher rate of detection of these types of errors. Errors falling into this group were coded with a 1.

The second group, coded with a 2, is composed of errors of omission committed by the licensed reactor operators and maintenance technicians (RO/MT). These errors result from the recovery of test or maintenance actions, and the components most often involved are valves. Increased NRC inspection in the areas of operations, system walkdown, maintenance, and surveillance testing might detect the occurrence of these errors.

The human errors which did not fall distinctly into either of these categories were coded with a 3.

16. NRCPGM - This category provides information about which NRC Inspection Program area was judged to effect the human event failure probability or HEP. An attempt was made to list all those NRC inspection programs which could have an effect, and then code the error with those that apply. The codes are listed below. The secondary code was assigned with the primary, where appropriate.

NRC PGM CODE	**DESCRIPTION**
	Primary
ST	Surveillance Testing
C	Calibration
M	Maintenance
TR	Training
Q	Quality Assurance
OPS	Operations
OPP	Operations Policy
SW	System Walkdown
	Secondary
P	Procedures
O	Observation

For the example of SWC89VVH, the NRC PGM codes are OPS(P) and SW. This means that NRC inspections in the operations procedures area and in the system walkdown area could help to lower the HEP. The implicit assumption is made that increased NRC inspection would result in increased attention by the utility and hence improvements. This code was used to determine those areas which could effect risk; the results should not be used quantitatively, since the magnitude of improvement in HEP from NRC inspection is extremely variable.

3.4 Results of Categorization

The primary purpose of this project was to identify, characterize, and determine the sensitivity of critical human performance errors of major risk significance. By examining the overall impact on risk of different categories of human errors using sensitivity analysis, this purpose was met. The categorization scheme was utilized in the analysis by varying the magnitude of errors in different categories, and evaluating their importance to overall risk as measured by core melt frequency (CMF), accident sequence frequency (ASF) or bin frequency.

Categories were analyzed either singly; for example, examining all pre-accident initiator errors, or in combination with each other; for example, all pre-accident initiator errors performed by licensed reactor operators. A strategy was developed based upon the type of information that could be extracted and what it would mean in terms of the objectives of this project. The strategy, outlined in the Overview Summary, is presented in detail in Section 5.

3.4.1 Sorts of Categories

A total of 489 human errors were coded and sorted, giving the initial database before truncation (the Overview Summary gives a detailed account of the truncation process). After truncation, 223 errors remained from the original database, and these also were sorted.

One purpose in sorting both the larger and the smaller database was to determine whether the two files were truly representative of each other, and to see what type of errors had been truncated as being not significant to risk. The two databases were surprisingly similar the differences that exist-are discussed below.

3.4.1.1 Results of Sorting on Human Errors

The results of the sorting for each category (refer to Table 3.2 for a full description of the coding scheme used) are described below. The discussion includes pie charts, displaying the categorical data of the 223 human errors.

<u>Timing</u>

Forty-four percent of the 223 dominant human errors of the Oconee PRA were modeled to occur during an accident, while 56% were modeled as pre-accident errors.

<u>System</u>

Figure 3.1 gives both the percent and number of errors in the system and shows that the errors are fairly well distributed across the system categories, with the highest percentage (22%) occurring in the Low Pressure Injection (LPI) system. A high percentage of errors also fall in the category of other (10%). This category is made up of the small number of errors that occur in several different systems.

The system category is where the most change occurs between the original 489 human errors and the sorted 223 errors. There were a total of 18 systems originally used in the categorization scheme for system. The LPI and AC Power

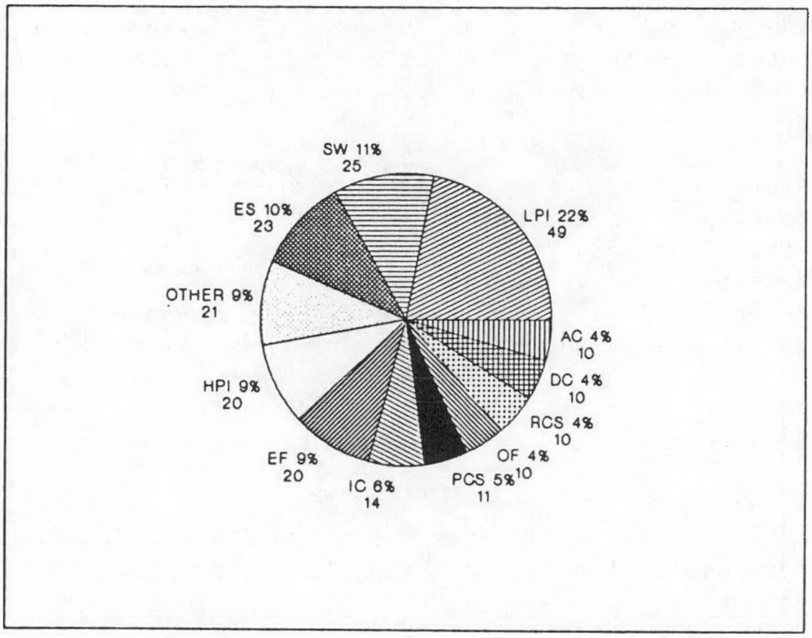


Figure 3.1. Human errors in the Oconee PRA - SYSTEM category
NOTE: See Appendix A for abbreviations.

systems contained the most human errors, each accounting for 13%. The Emergency Feedwater (EF) system contained 10%, and the IC, PCS, DC, and SW each contained 8% of the errors. The ES and HPI systems accounted for 7% and 5%, respectively. The remaining human errors, 22%, were distributed across nine systems, each having less than 5% of the errors.

The discrepancy between the two data sets is most likely due to the fact that many of the systems in which human errors were truncated are highly mechanized, and therefore the opportunity for a human error to occur in them and have an impact on that system's failure rate is significantly lessened.

Components

Valves are the component with the largest percentage of human errors (41%) (Figure 3.2). Other significant categories are Instrumentation + Control, with 26% of the human errors, and system, which contains 17% of the errors. The high percentage of errors in valves and Instrumentation + Control is partially due to the large number of these types of components in the Oconee power plant. The number of errors in the systems category is also important because this represents top_level errors, many of which are the result of multiple erroneous actions dealing with multiple component types. Comparison with the 489 error database was not significantly different.

Personnel

The reactor operator is the personnel mainly responsible for many of the errors modeled in the Oconee PRA (Figure 3.3). The sum of all errors coded

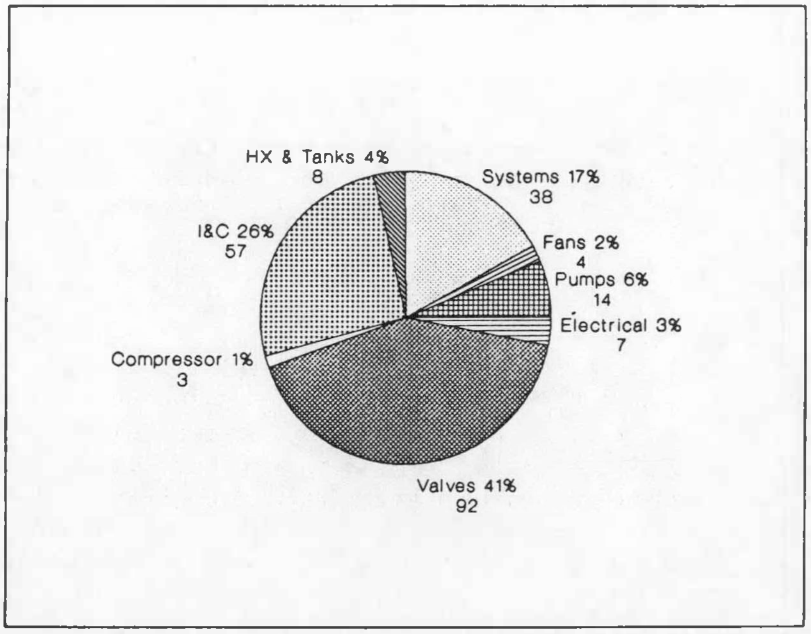


Figure 3.2. Human errors in the Oconee PRA - COMPONENT category

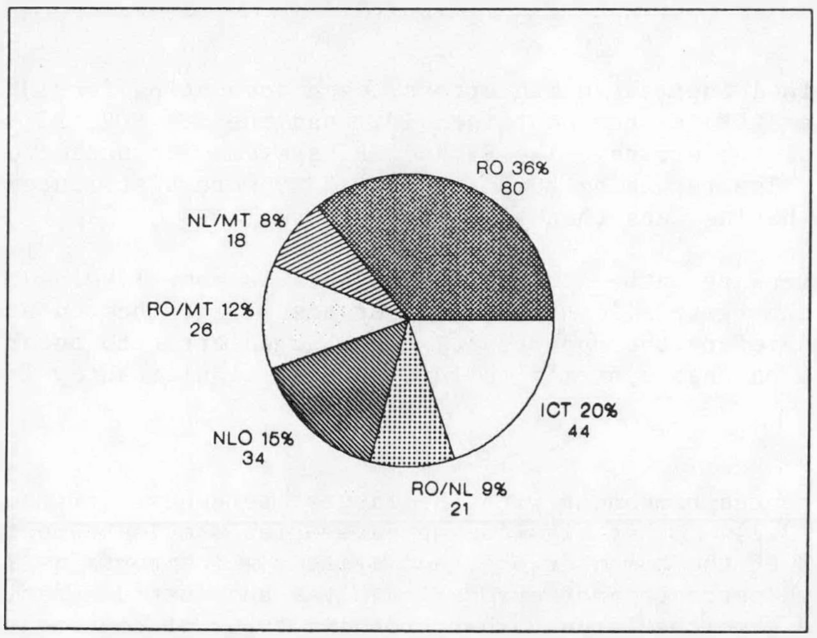


Figure 3.3. Human errors in the Oconee PRA - PERSONNEL category

with RO, RO/NL, or RO/MT exceeds 50%. The non-licensed operator is accountable for 23% of the errors, and the Instrument Control Technician is responsible for 20%. These were predictable results, although not altogether realistic. The Oconee PRA did not model human errors resulting from inadequate and/or preventive maintenance, maintenance personnel do not hold the main responsibility for the occurrence of an error. These results are very similar to those obtained when doing sorts on the total 489 human errors for the Personnel category.

Omission/Commission

Seventy-five percent of the 223 human errors were coded as Omission, and 25% were coded as Commission. This value is similar to other PRAs, in that the modeling of commission errors is much more difficult and is, therefore, often avoided. The percentage of Omission and Commission errors was exactly the same for the 223 errors as for the 489 errors.

Event Type

EVENTTYPE is an Oconee PRA-defined category. Fifty-two percent of the 223 human errors fell under the event type of "U," unavailable, and 30% of the errors resulted from the "OF," operator failing or the operator failing to recover (Figure 3.4). Only 3% of the errors fell under "OI," operator inhibits and 2% under "I," inadvertent action. A large number (12%) of the errors fell under the implicit category. These errors were not given an event type by Oconee. Therefore, they were coded by the BNL experts, and an "I" was added on the end to indicate implicit. These percentages were similar in the total database of 489 human errors.

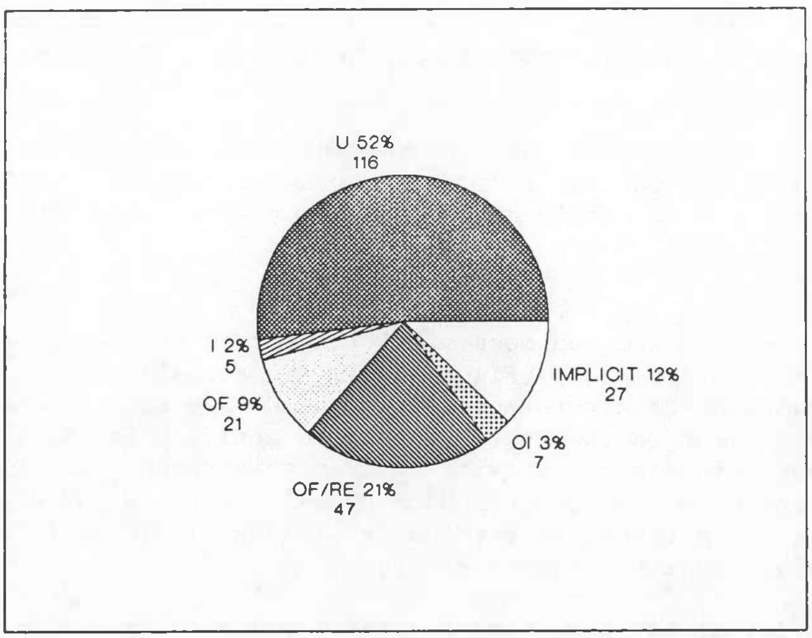


Figure 3.4. Human errors in the Oconee PRA - EVENTTYPE category

Location

Most human errors occur (48%) in the control room (Figure 3.5), while 31% of the errors are coded as outside the control room. The category with the smallest percentage, 21%, Control Room/Outside Control Room, represents another level of uncertainty. These errors either consisted of multiple actions, some of which occurred in the control room and others which occurred outside the control room, or the error could have been committed in either place and therefore was coded to represent this fact.

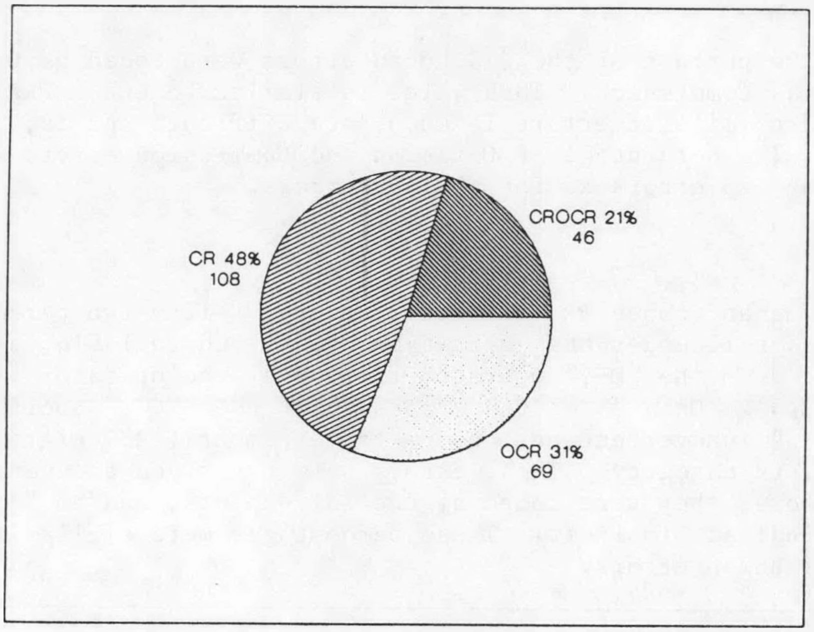


Figure 3.5. Human errors in the Oconee PRA - LOCATION category

Of the 489 human errors, 53% originated in the control room, 28% were committed outside the control room, but for a large number (19%), a specific location could not be determined and thus, they were coded CROCR.

Activity

Operation errors are the most significant type of error, accounting for over half of all errors (58%) (Figure 3.6). In actuality there are no true Test and/or Maintenance errors modeled in the Oconee PRA as human errors. The closest type of error to the Test/Maintenance activity is the error of restoration from Test/Maintenance activities. For this reason, an "R" has been added to the end of all T/M, O/M, and T codes to signify restoration from Test and/or Maintenance actions, as previously discussed. These errors were analyzed separately from the operaton errors.

In the original 489 human errors, 64% involve an operations activity. T/M and C account for 16% of the errors each, and the remaining categories of O/M, T, and M account for 2%, 1%, and 1%, respectively.

Dependency

The category was coded with the Oconee PRA information. The percentage of errors they believed to be dependent were only 6% of the 223 errors.

Of the original data set of 489 human errors, only 3% were considered dependent. The increase in percentages of dependent errors is due to the small number of dependent errors that were lost after truncation.

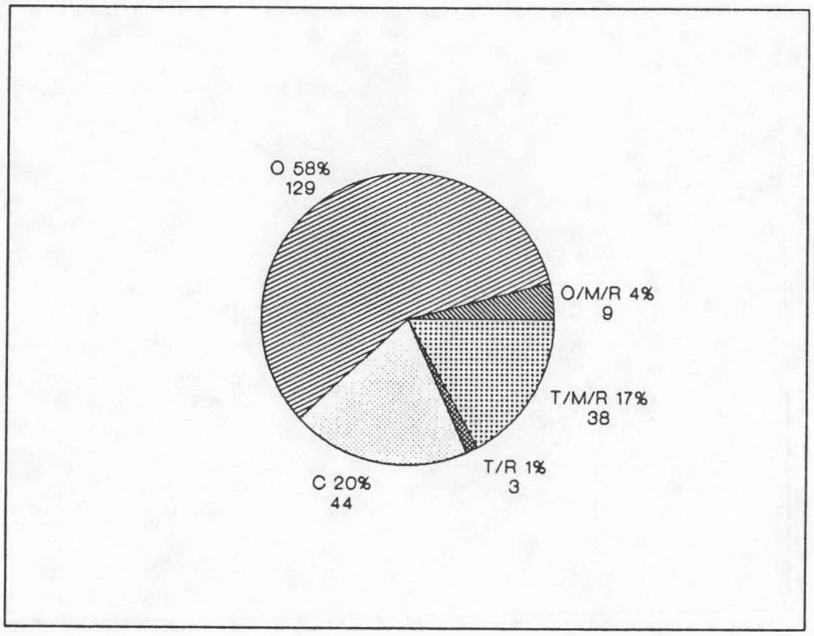


Figure 3.6. Human errors in the Oconee PRA - ACTIVITY category

OCIMPT

Forty-one percent of the 223 human errors, or 18%, were considered by the Oconee PRA as being important.

In the total database of 489 errors, only 47, or 3%, were considered to be Oconee-important. The large increase in the percentage of important errors after the truncation (although the number decreased slightly) was because these errors were specifically set apart and labeled as important in the Oconee PRA. However, some of these important errors are for external events or occur in sequences we did not consider (e.g., ATWS sequence). Therefore, the total number of errors decreased slightly, while the percentage rose.

NRC Inspection

Because this category involved a multiple coding scheme, the percentage of errors in each category is not given in Figure 3.7. However, of the 223 human errors coded, the majority (84%) were coded with Operations (OPS). Training (TR) was used in the coding schemes of 43% of the errors, while System Walkdown (SW) occurred in the coding schemes of 33%. The remaining codes, Surveillance Testing (ST), Maintenance (M), Operations Policy (OPP), Calibration (C), and Quality Assurance (Q), occurred in 14%, 18%, 5%, 16%, and 1% respectively.

In the total database of 489 human errors, the percentage occurring in each category was very similar to the percentages obtained after truncation. Eighty-seven percent of the errors fell under operations, 46% under training, 32% under system walkdown, 24% under maintenance, 16% under surveillance testing, and 12% under calibration. Only 5% fell under operations policy, and less than 1% under Quality Assurance.

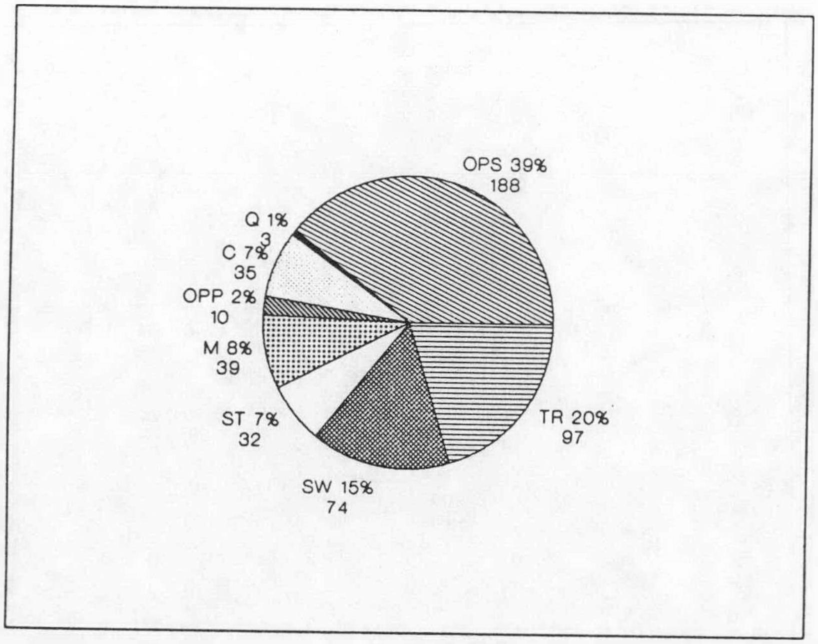


Figure 3.7. Human errors in the Oconee PRA - NRC inspection category
NOTE: See Section 3.3 for abbreviations.

3.4.1.1.1 Summary comments on sorting of categories

Based on our sorting, we concluded that the final data set of 223 human errors is an appropriate sample of the total of 489 in the Oconee PRA. The differences observed can largely be explained by the nature of those errors that occur in the top cutsets. These are the errors which remained in the data set after the truncation process, while other less significant cutsets containing human errors dropped out.

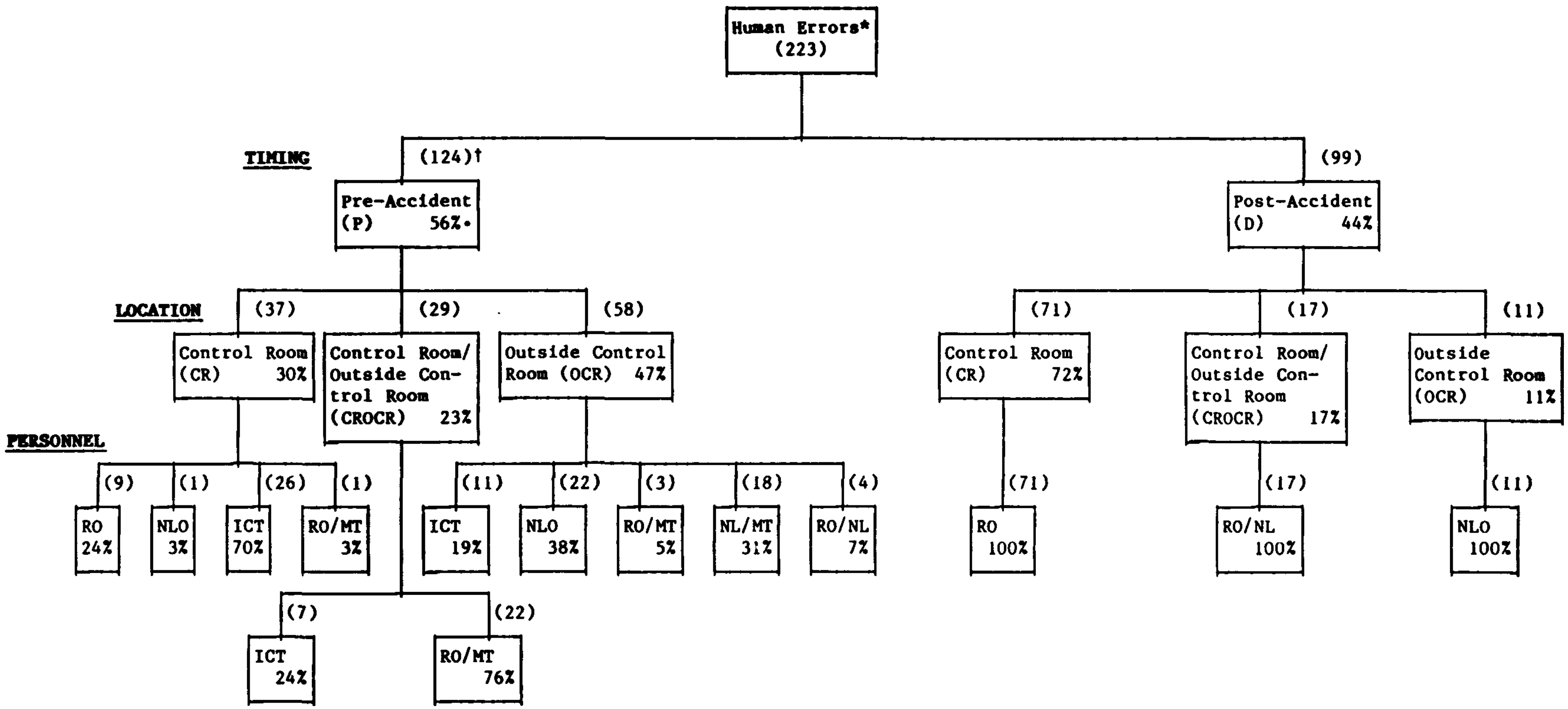
A final important point to note in comparing the two databases, is that the percentage of errors coded with a level of uncertainty increased fairly significantly. This change was due to the fact that many human errors occurring in the highest sequence cutsets involve a number of different actions and personnel, therefore, the coding of these errors was often based on BNL expert opinion.

3.4.1.2 Linkage Diagrams

Based on the sorts, linkage diagrams were constructed using the data set of 223 human errors to illustrate some of the relationships between various categories (see Figures 3.8 and 3.9).

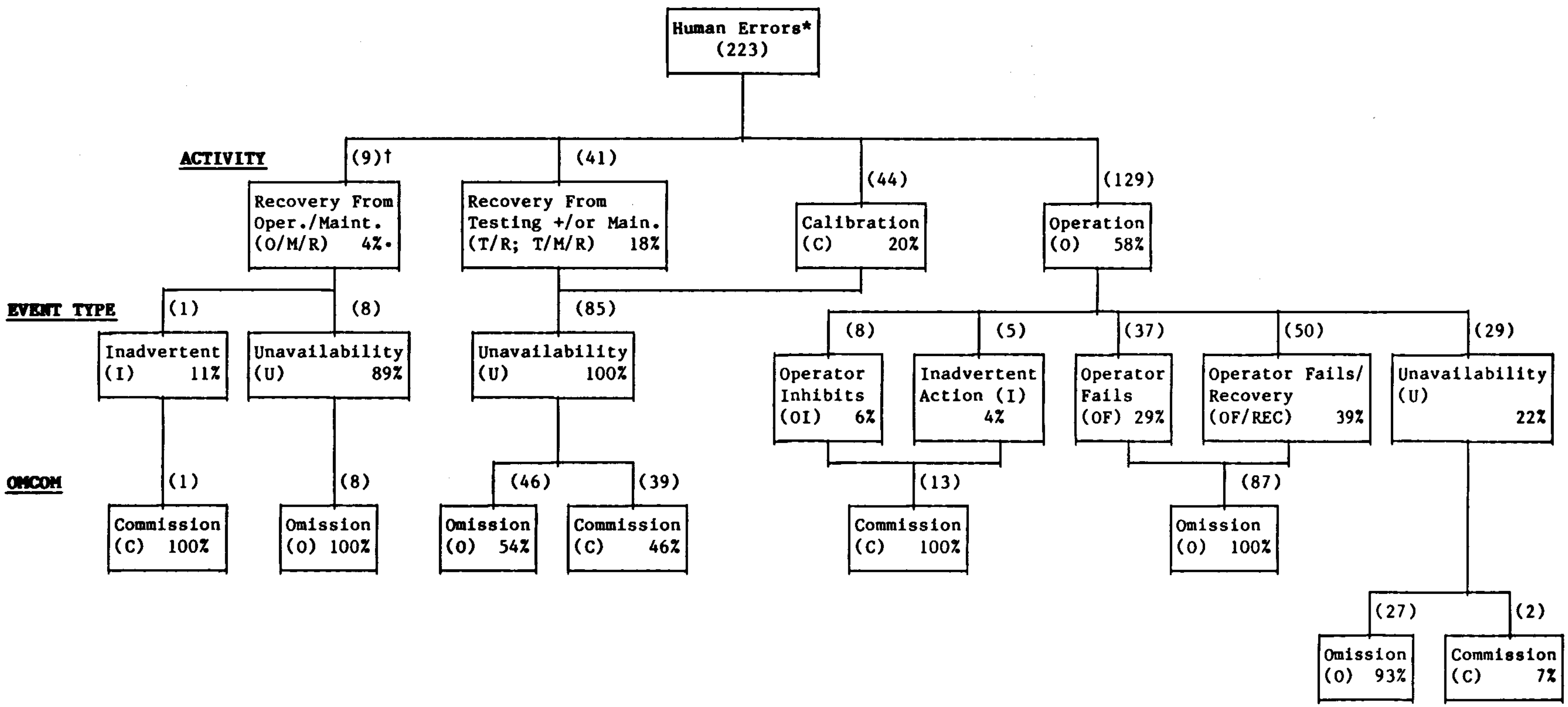
3.4.2 Comparison with Other PRAs

The categorization scheme developed and applied for the 223 human errors in the Oconee PRA is usable for the human errors modeled in published PRAs depending on the detail of documentation given for each human error. The more explicit and detailed the documentation, the more accurate and complete will



* Human Errors considered in risk assessment of a nuclear power plant.
† Indicates total number of errors belonging to the category.
• Indicates the percentage of level above.

Figure 3.8. Linkage diagram of human error categorization based on timing, location, and personnel



* Human Errors considered in risk assessment of a nuclear power plant.
† Indicates the total number belonging to the category.
• Indicates the percentage of level above.

Figure 3.9. Linkage diagram of human error categorization based on activity, event type, and OMCOM

be the categorization. Also, the categories EVENTTYPE and OCIMPT generated from the Oconee PRA will have to be handled differently for human errors in other PRAs, or not applied at all.

In comparison to 489 human errors from the Oconee PRA (which already excludes those 64 concerned with external events), the number of human errors modeled in previous PRAs is significantly less. Based on the NUREG/CR-4103, all human errors modeled in 19 PRAs were identified. In all 19, only 1976 records of errors were found, which averages out to slightly more than 104 human errors per PRA.

3.4.3 Oconee LER Review

3.4.3.1 Purpose and Scope

To obtain information on the level of realistic modeling of the Oconee PRA human errors, we undertook a review of recent Oconee Station Licensee Event Reports (LERs). All LERs for three-and-one-half years, from 1984 through the first half of 1987, were reviewed to extract the human errors. The errors identified were compared with those human errors modeled in the Oconee PRA to determine (1) if such a comparative approach was reasonable and achievable, (2) if similar errors appeared in both, and (3) to obtain insights on the types of errors that actually occurred and on the modeling of human errors in the PRA.

3.4.3.2 Review Process

Information on all Oconee LERs from January 1984 through mid-1987 was obtained from the Sequence Coding Search System (SCSS), which is maintained by the Nuclear Safety Information Center at Oak Ridge National Laboratory. For each LER, the SCSS provided an abstract and a decoded step matrix. The matrix represented a very detailed analysis of each step of the LER event, identifying each failed component and system, and each human error that occurred. The SCSS LER data allowed a relatively rapid analysis of most LER errors, and for those that were not adequately defined by SCSS, full LER texts were obtained and reviewed. Each human error occurring in an LER was evaluated to determine if it or a similar error was included in the Oconee PRA model. If the error was not included, then an evaluation was made as to whether or not it was appropriate for the error to be included in the model. For example, some LERs contained administrative type human errors or errors that were insignificant to risk and thus, would not appropriately be modeled in a PRA.

3.4.3.3 Results

Figure 3.10 illustrates, by year, the number of LERs at the Oconee site, the number of those LERs that contained a human error, and the total number of human errors (HEs) identified in the LERs. Some LERs contained multiple HEs, the most being four. The totals over the 3-1/2 years show that 57 of the 82 LERs, or 70%, contained human errors. Overall, 81 HEs were identified. Figures 3.11 a, b, and c illustrate the distribution of LERs and HEs for each of the three Oconee units. Unit 1 has more LERs than Units 2 or 3. This difference appeared to be partially due to the reporting practice, whereby generic

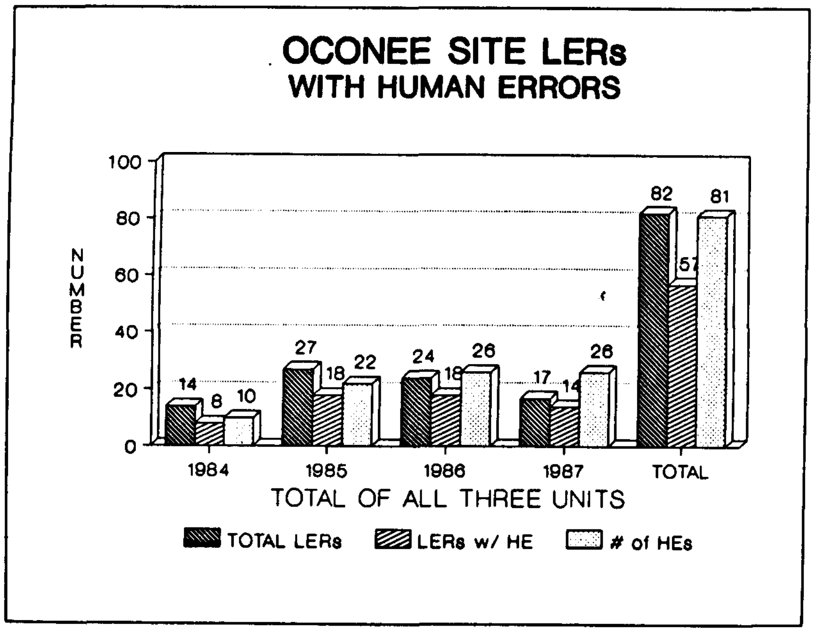


Figure 3.10. Oconee site LERs with human errors

LERs applicable to all units are preferentially reported under Unit 1. These LERs were not then entered in the SCSS as applicable to all units. There were no other significant trends in the numbers of occurrences over time in HEs or in LERs.

Figure 3.12 compares the human errors from the Oconee LERs with the human errors in the Oconee PRA. Sixty of the 81 total HEs were not included in the PRA: these consisted of 45 inappropriate for inclusion, 10 design type HEs, and five appropriate for inclusion, as reviewed by BNL. Twenty-one of the 81 HEs were included in the PRA in some form. Fourteen of these were included as initiators, six had similar errors modeled in the PRA, and one appeared exactly.

As noted above, the majority (45 of 81) of the human errors in the LERs were not of an appropriate type to be included in the PRA. Examples of such errors were: failure to sample radwaste, moving heavy equipment over spent fuel, exceeding the cooldown rate on plant shutdown, exceeding technical specification surveillance internal on fire protection equipment, and problems with tendon surveillance program in the reactor building.

Ten of the 81 HEs were design errors (see Table 3.3). Design errors are generally not included in today's PRAs, due to their low probability and the difficulty of modeling them. However, some analysts maintain that this is a deficiency in current PRAs. The LER review showed that even for a mature plant such as Oconee, design errors continue to be identified.

A relatively small number (5 of 81) of the HEs in the LERs were found to be appropriate for inclusion in the PRA but were not modeled (Table 3.4). No attempt was made to estimate the effect on risk of adding these errors to the PRA.

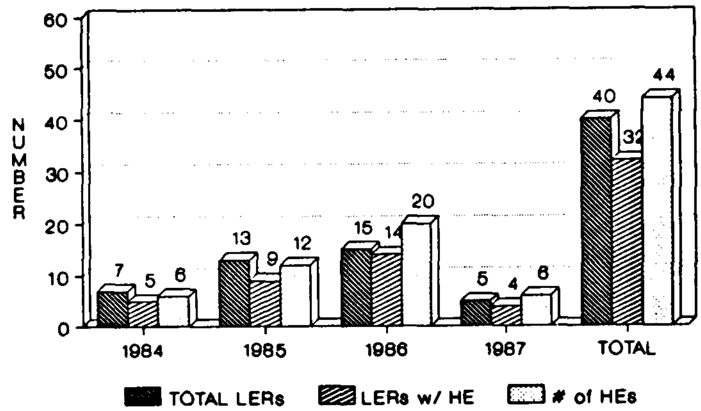


Figure 3.11a. Oconee-1 LERs with human errors

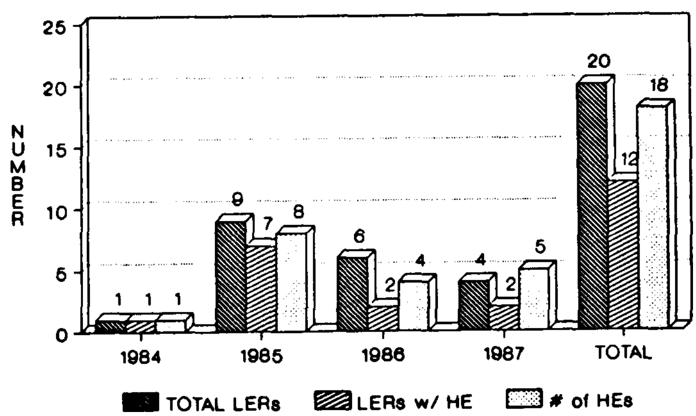


Figure 3.11b. Oconee-2 LERs with human errors

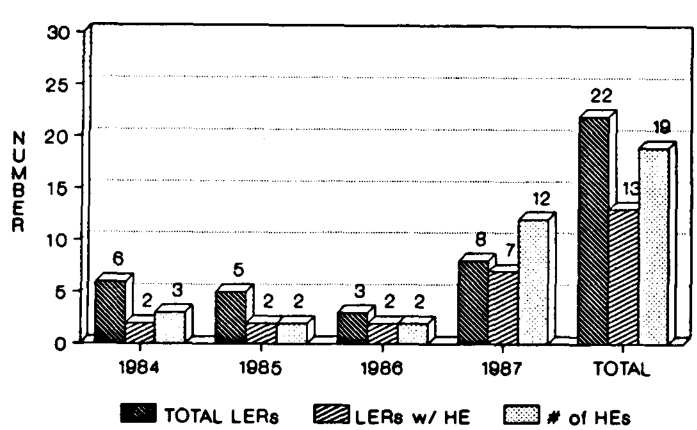


Figure 3.11c. Oconee-3 LERs with human errors

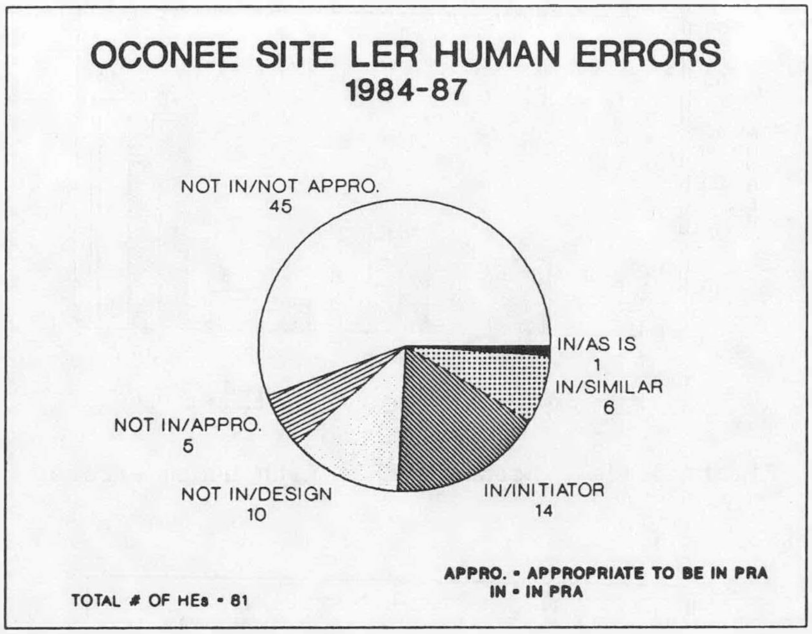


Figure 3.12. Human errors in the Oconee site

Table 3.3. Design Related Human Errors in Oconee LERs

1) Engineered Safety Feature Actuation System (ESFAS) setpoint for HPI initiation was specified to be too low. Revised setpoint from 1550 psig to 1600 psig.

2) Insufficient terminal voltage for MOVs to ensure operability in a degraded voltage situation.

3) Inadequate Seismic Design.

4) Auxiliary Feedwater (AFW) pump had inadequate protection against pump runout.

5) Suction design of the Low Pressure Service Water System was inadequate.

6) Inadequate design of Standby Shutdown Facility (SSF).

7) Cabling error to SSF.

8) Inadequate safety valve sizing on AFW.

9) Seismic design inadequacy of Keowee battery racks.

10) Improper voltage monitoring by Inverter and Static Transfer Switch for Essential AC bus resulting in unnecessary power loss.

Table 3.4. Human Errors in Oconee LERs Appropriate for PRA but not included in Oconee PRA

1) Boron Concentration in Core Flood Tank found to be below minimum Technical Specification due to water leaking in.

2) Moderator Temperature Coefficient was allowed to exceed FSAR values.

3) Two Reactor Protection System (RPS) Channels inadvertently removed from service due to poor procedures.

4) Inadequate testing of Emergency Condenser Circulating Water and Low Pressure Service Water over the years failed to detect the inability of the system to provide decay heat removal in Station Blackout scenario.

5) Load shedding (Source B) fuse block not installed, making part of load shed circuit inoperable.

A notable number, 14, of the human errors judged to be included in the PRA were initiators. These types of HEs are included implicitly in the initiator frequency data used for the PRA. However, the effect of the human error cannot be distinguished without additional work to determine which portion of each initiator is due to human error and then splitting that portion out for further sensitivity analysis. The 14 human errors in the LERs led to the following initiators:

3 - Turbine Trip/Reactor Trip
3 - Loss of Feedwater
3 - Reactor Trip
2 - Loss of Offsite Power
2 - Small Break LOCA
1 - Engineered Safety Feature Actuation

Six of the human errors from the LERs had similar errors modeled in the PRA (listed in Table 3.5). Only one error from the LERs was included, as is, in the PRA. At Oconee 3 the breakers for the High Pressure Injection (HPI) suction valves (valves 3HP-24 and 3HP-25) from the Borated Water Storage Tank (BWST) were inadvertently left tagged open with the valves left shut. This was modeled in the PRA as error HP2425MVH at a probability of 5×10^{-5}. No attempt was made to modify the base case human error probabilities as a result of errors noted in the LERs.

When the review of all of the HEs identified in the Oconee LERs was completed, the HEs from the Oconee PRA were considered. Clearly, a very small percentage of the 500 human errors in the PRA appeared in LERs, for two main reasons. One is that most of the HEs from the PRA do not constitute a reportable event in an LER. For example, common PRA HEs such as single valve mispositioning and miscalibrations are not reportable. Second, many of the PRA HEs are "During" accident errors, such as recovery actions. These would only appear as a real event in an LER, if the plant experienced the accident initiating event and then also had the HE. This combination of events is relatively rare. Thus, we would not expect to see most of the PRA HEs in LERs.

Table 3.5. Human Errors from LERs that had Similar Errors in PRA

1) Operator error on Integrated Control System when transferring from manual to automatic mode.

2) During test, both trains of Reactor Building Spray were inoperable for two minutes, due to the opening of circuit breaker for MOV.

3) Failure to take manual control of Emergency Feedwater valves before starting Main Feedwater System.

4) Incorrect Torque Switch Settings for Decay Heat Removal valve.

5) Incorrect Limit Switch Setting for MOV.

6) All three Reactor Building Cooling Units left isolated on startup after completion of shutdown period.

3.4.3.4 Summary of LER Review

This section summarizes the insights drawn from studies of the LERs. The review of Oconee LERs found a high percentage of LERs containing human errors (HEs) and a large number of HEs occurred within those LERs. Hence, human error appears important, and a sensitivity study to determine th impact of human error on risk appears beneficial. Of those LER HEs found to be appropriate for inclusion in the PRA, the majority were included in some form in the Oconee PRA. Hence, the use of the Oconee PRA for a sensitivity study on human errors also appears reasonable. Regarding the large numbers of HEs in the PRA, not found in the LERs, there are justifiable reasons discussed previously, so that should not affect the validity of the sensitivity study. However, it should be realized that not all possible HEs are modeled in the sensitivity study (e.g., the 15 HEs found in LERs), and that the effect of humans on risk may be higher than shown. Finally, it must be kept in mind that comparisons between human errors in LERs and a PRA have limitations because of the LER reporting requirements and the low probability of many events modeled.

3.5 Conclusions

This section discusses some overall conclusions, uses, and limitations of the Oconee human error categorization.

The categorization scheme was necessary for several reasons. First, the human errors modeled in the PRA had to be broken down into analyzable components based upon recognizable behavior. By developing a set of categories, each human error from the PRA could be described by the same scheme. Second, with the categorization scheme in place, different categories of errors could be compared with respect to their sensitivity to risk. Third, the categorization scheme defined relationships between categories in their sensitivity to risk.

The primary use for the categorization scheme of human errors was to vary different groups of errors, and observe their differential sensitivity to risk, for which the scheme is very suitable. An inherent limitation, however, in this analysis was the categorization scheme itself. The sensitivity analyses performed were only as descriptive as were the categories. Since sometimes the error was treated superficially in the PRA, the categories were somewhat superficial although ambiguity and assumption were avoided whenever possible. The categorization scheme will be useful for other studies of human error and for data bank purposes.

From the sorting of the human errors that was done based on the categorization scheme, several conclusions can be made. The conclusions are based on the numbers of errors and not their risk significance, which is addressed in Section 5.

1) There are approximately equal number of pre-accident and during-accident errors in the Oconee PRA.

2) Operations errors are dominant in the human error modeling in the Oconee PRA.

3) The subset of human errors remaining after initial truncation (223 out of 489) are representative of the larger database as demonstrated by similar statistics on the sortings of categories that were conducted.

4) Human errors that were lost in truncation were usually associated with highly mechanized systems.

5) Comparisons of LERs with PRA/HRA modeling are feasible and informative.

4. DEVELOPMENT OF RANGES FOR HUMAN ERROR PROBABILITIES

4.1 Introduction

In performing sensitivity evaluation for risk assessments to identify the variations in the risk parameters, due to the variation in human error rates the range over which the input parameters can vary should be carefully defined. The method of defining the range of variability of the input parameters significantly influences the shape and the limit of variability of the output parameters.

This section defines the ranges for the human error probabilities (HEPs) used for the sensitivity evaluation in the Oconee nuclear power plant, and discusses the methodology used in defining those ranges. The methodology is generally applicable in developing the ranges of HEPs for sensitivity evaluation in PRAs.

It is important for a risk-based sensitivity evaluation to define the entire range of variability of the input parameters whose effect on the risk parameters is being evaluated. The range of the input parameters on human error probabilities (HEPs) was developed considering the various causes of variability that can be assigned to the estimates. The range of variability, expressed in terms of error factors of the median estimate of the HEPs, in PRAs usually includes only the data uncertainty associated with these estimates.

In developing the ranges of the HEPs for this sensitivity evaluation, various causes of variability defined in the literature (NUREG/CR-2300, NUREG/CR-1278) were taken into consideration and thus, the ranges defined are broader than those typically found in PRAs. The attempt also was to obtain a conservative or broadest range, so that the sensitivity evaluation can cover the entire range possible, recognizing the various causes of variability.

The methodology presented for quantitative determination of the ranges of HEPs is drawn from a well-known statistical approach of analysis of variances. The influences of each of the causes of variability is defined in terms of error factors and the variances in the HEP due to each of the causes are combined to obtain the overall variance in the HEP estimates. The resulting overall variance is then used to obtain the range of the HEP. Subjective judgments are involved in defining the error factors associated with each of the variability causes and in this study, the error factors were defined using available data sources and the expert judgments of two human factors research specialists, a PRA specialist, and a nuclear engineer. This approach is considered adequate for sensitivity evaluation since, as discussed, the objective is to develop conservative estimates of the ranges that account for various causes of variability. Conservatism is introduced by providing conservative estimates of the error factors for the causes of variability.

In the following section, the causes of variability included in the calculation of the ranges are discussed and the methodology for quantitatively combining the variability due to different causes is presented in detail. The assignment of the error factors for each cause is also discussed. Finally, the application of the methodology for determining ranges of HEPs in the Oconee PRA is presented.

4.2 Method/Approach

4.2.1 Sources of Variability in Human Error Probability

The reasons for variability in HEPs used in PRAs are discussed in PRA Procedures Guide (NUREG/CR-2300), and five major sources of uncertainties are defined. In this study, these same sources are defined as the causes of variability. These causes of variability are considered to adequately determine a very large percentage of the overall variability that can be accounted for in a sensitivity analysis due to the fact that many of the causes implicitly include a number of other variables, such as Performance Shaping Factors (PSFs). The range of HEPs are developed considering these causes of variability; however, care was taken to define the applicability of the for each group of human errors. For example, the variability due to differences in task description was not considered applicable for human errors of operation. A brief description of each of the variability causes is given below. Further details can be obtained in PRA Procedures Guide (NUREG/CR-2300).

1) Lack of actual data

   This cause of variability reflects the sparsity of data relevant to human performance in NPPs. Even where the information is available, (for example, Licensee Event Reports), and the incidents involving real human errors can be obtained, we do not know the number of opportunities for making such an error, thus causing uncertainty in the estimate of the HEP. A further complication arises due to a lack of adequate description of the human errors in such incident reports.

2) Inexactness of the Model

   This variability cause represents the inherent weaknesses in modeling human performances. Although various models are used in quantifying HEPs in nuclear power plants, their validity or accuracy is known only to the extent that they are an approximate representation of the real-world situation. However, this applies to all models, and human reliability models are no exceptions.

3) Difference in Task Description (application of generic HEPs)

   This cause of variability arises because often the same error probability is assigned for similar components, although there are differences in the actual task and work conditions. The data is inadequate to distinguish among such situations. Another factor in this cause of variability is that, in some cases, the error probability was obtained from similar tasks in non-nuclear industry, and the performance shaping factors applicable in non-nuclear industry can be vastly different from those in nuclear power plants.

4) Difference among Personnel

   This cause accounts for the variability in human performance due to individual differences. An average person is assumed in developing estimates for PRA evaluations, but differences exist from one person to another.

5) Skill and Knowledge of Human Reliability Analyst

The human reliability analyst is a cause of variability in the HEP estimate. The experience of the analyst and the level of detail used in analyzing the errors can both influence the HEP estimates. Furthermore, the analyst usually does not have complete knowledge of the work situation in the plant, nor necessarily know the makeup of the team conducting human activities in the plant.

4.2.2 HEP Range Development Using Variability Causes

In this section, the methodology is presented for defining the ranges for HEPs using the variability causes associated with a human error. The information available for developing the ranges is limited. Essentially, the methodology requires three inputs:

a. the central estimate of the HEP for which a range is to be established,
b. the assumed distribution for the HEP, and
c. the error factors (EFs) associated with the HEP for each cause of variability.

The methodology uses the available information on the central estimate of the HEP used in probabilistic risk assessments, and expert judgments, as appropriate, for the error factors associated with each cause of variability and for the nature of the HEP distribution. To develop the ranges for the HEP, let us consider the mean value used in PRA as the grand mean, μ, i.e., the mean value is obtained considering the mean values resulting from the various causes of variability.

The various causes of variability (as discussed above) can be assumed to effect the grand mean. Following the approach of analysis of variance, the effect of any cause, j, is defined as the deviation of μ_j, the mean due to the variability cause j, and is given by:

$$\mu_j = \mu + \alpha_j \qquad (1)$$

where α_j is the effect on the mean value due to the variability cause j.

For the five causes of variability defined for the HEPs, if one is able to define the effect on the mean due each of the causes, one obtains:

$$\mu \text{ is a function of } \left\{ \begin{array}{c} \mu_1 \\ \mu_2 \\ \cdot \\ \cdot \\ \mu_5 \end{array} \right.$$

where μ_j's are given by Eqn. (1).

The values of μ_j or α_j cannot be obtained directly. Based on expert judgment and on limited studies, the ranges over which the HEP will lie due to a particular cause of variability are assigned or defined. For example, assuming an error factor (EF_j) due to the variability cause j, using the grand mean, μ, the bounds are obtained over which the mean value lies. However, this requires an assumption for the HEP distribution. In the following, the use of both lognormal and uniform distributions (used in developing Oconee HEP ranges) is presented.

4.2.2.1 Lognormal Distribution

Let μ be the mean HEP defined for a particular human action and let EFj be the error factor associated with a variability cause j. Assuming a lognormal distribution for the HEP:

$$f(\chi) = \frac{1}{\sqrt{2\pi}\sigma_n\chi} \exp - \left(\frac{(\ln\chi-\mu_n)^2}{\sigma_n^2} \right) \quad ; \chi > 0 \qquad \text{\_\_\_\_\_\_(2)}$$

This gives,

$$\text{median} = e^{\mu_n} = \text{HEP}$$

$$\text{mean} = \mu_j = e^{(\mu_n + \sigma_n^2)/2} \qquad \text{\_\_\_\_\_\_(3)}$$

$$\sigma_n = \ln(\text{EFj})/1.645$$

The variance V_j associated with the variability cause j is given by:

$$V_j = e^{2\mu_n + \sigma_n^2} \left[e^{\sigma_n^2} - 1 \right]$$

$$\text{or } V_j = \mu_j^2 \left[\exp\left(\ln(\text{EFj})/1.645\right)^2 - 1 \right] \qquad \text{\_\_\_\_\_\_(4)}$$

4.2.2.2 Uniform Distribution

The density function of an uniform distribution is given by:

$$f(x) = \frac{1}{b - a} \quad ; a \leq x \leq b \qquad \text{\_\_\_\_\_\_(5)}$$

where a and b are, respectively, the lower and upper bound of the variate. The parameters of the distribution are:

$$\mu = \frac{b + a}{2}, \text{ and } \sigma^2 = \frac{(b - a)^2}{12} \qquad \text{\_\_\_\_\_\_(6)}$$

In using the uniform distribution to obtain the overall range combining the variability causes based on the error factors associated with each cause, the variance associated with each of the causes needs to be determined. As defined in Eqn. (6), the variance is obtained from upper and lower bounds. The upper and lower bounds are obtained by respectively multiplying and dividing the mean by the logarithm of the error factor, since the error factors are based on an assumption of logarithmic distribution.

Let a_j and b_j, respectively, be the lower and upper bounds of the HEP associated with the variability cause j. Then the variance associated with the cause is obtained as:

$$V_j = (b_j - a_j)^2/12 \qquad \text{\_\_\_\_\_\_(7)}$$

4.2.2.3 Combining the Variability Due to Various Causes

The variance, Vj, obtained from each of the variability cause is to be combined to obtain the overall variance of the HEP. The overall variance, V, can be obtained under two assumptions: (a) no interaction among the variability causes, and (b) complete interaction among the causes. The overall variance, V, assuming no interaction can be obtained using the expression:

$$V = \sum_j V_j = \sum_j s_j^{\,2} \qquad \text{\_\_\_\_\_\_(8)}$$

where sj is the standard deviation. For complete interaction, V is given by:

$$V = \sum_j s_j^{\,2} + \sum_{i \neq j} s_i \; s_j \qquad \text{\_\_\_\_\_\_(9)}$$

4.2.2.4 Development of the Overall Range

The overall variances obtained (Eqns. (8) and (9)) can now be used to obtain the range of HEPs. For lognormal distributions, one can obtain the upper (UHEP) and lower (LHEP) bounds using the following expressions:

$$\sigma^2 = \ln\,(\,1 + V/\mu^2)$$

$$EF = \exp\,(1.645\sigma)$$

$$LHEP = (\text{median})/(EF), \text{ and}$$

$$UHEP = (\text{median}) \cdot (EF) \qquad \text{\_\_\_\_\_\_(10)}$$

Similarly, for a uniform distribution, LHEP and UHEP are obtained from the solution of the following equations:

$$V = (UHEP - LHEP)^2/12, \text{ and} \qquad \text{\_\_\_\_\_\_(11)}$$
$$\mu = (LHEP + UHEP)/2$$

4.3 Application of Methodology for Development of Oconee HEP Ranges

4.3.1 Categorization of Oconee HEPs for Development of the Range

In applying the methodology for developing ranges for HEPs in the Oconee PRA, the human errors were divided into groups depending upon the variability causes and the associated error factors. All HEs were placed into one of the following five groups:

1. HEs for calibrations and restoration from test and maintenance (T, M, & C), with HEP > 1E-3,
2. HEs for calibration, and restoration from test and maintenance (T, M, & C), with HEPs ≤ 1E-3,
3. HEs of Omission for Operations,
4. HEs of Commission for Operations,
5. Dependent HEs.

The errors were grouped in this way due to the similarity of the modeling of these types of errors within each group in the Oconee PRA.

The discussion in the next section provides the attributable variability causes for each of these categories and the assigned error factors. As presented in Table 4.1, each group of human errors has a distinct set of error factors. Nevertheless, a more refined range for human errors within a group of errors might be defined or additional groups of errors created if more distinct sets of error factors could be defined. Considering the available information on human errors, the grouping of the errors for the range calculation is considered adequate for sensitivity evaluations.

Table 4.1. Error Factors Associated with Groups of HEs for Each of the Variability Causes

Variability Cause	HEP for T, M, & C		HEP for Operation		Dependent HEP
	HEP >1E-3	HEP<1E-3	Omission	Commission	
1. Lack of Actual Data	5	10	10	10	10
2. Inexactness of the Model	3	3	5	5	10
3. Differences in Task Description	3	3	-	-	3
4. Differences among Personnel	2	2	5	10	5
5. Skill and Knowledge of HRA Analyst	2	2	3	3	3

4.3.2 Selection of Error Factors (EFs) for Human Errors

In this section, there is a brief discussion on the EFs defined for the various groups of human errors broken into each of the variability causes. Table 4.1 provides the EFs used in deriving the ranges for Oconee HEPs.

Lack of Actual Data

The EFs associated with HEPs due to lack of actual data is usually considered in PRAs. Typically, an EF of 3 or 10 is assigned. In this application, the choice of the EF was similar; a factor of 10 was assigned for all groups of HEs except for calibration and restoration from test and maintenance errors with probabilities greater than 10^{-3}, where a factor of 5 was used. The reason was that for such errors, there are adequate data, and the factor of 5 signifies that there is more information for these errors. Limited data are available for other groups where a factor of 10 is assigned which represents a larger variability in the estimate due to this cause.

Inexactness of the Model

The variability in the HEP estimate due to the modeling of the error is difficult to quantify, and there is very limited information on this in the literature. One approach to infer such variability due to modeling has been to apply different models and observe the range over which the calculated estimate lies. Samanta and Mitra (NUREG/CR-2211) studied the dependent human failure probability assuming various underlying distributions that may describe the phenomenon and observed a factor of 10 variation. The same factor was used in the Oconee study for this variability cause due to the large amount of uncertainty associated with the quantification of dependent human errors. The choice of factors of 3 and 5 for other groups are based on the relative difficulty of modeling those errors. Same factors, however, are assigned where similar types of models are involved; for example, Errors of Operations are assigned a factor of 5 and calibration/restoration from test and maintenance, errors are assigned a factor of 3, irrespective of groupings within these classes of errors.

Differences in Task Description

This variability cause is not applicable to HEPs for operation since these estimates are based on nuclear power plant procedures and the available data from nuclear industry; none of these estimates resulted from experiences in non-nuclear industry. For other groups, a factor of 3 was used. This assessment is subjective; however, it is considered realistic since such a variability is typically observed when comparisons are made among different but similar errors.

Difference Among Personnel

Both PRA Procedures Guide (NUREG/CR-2300) and Oconee PRA (NSAC-60) refer to Wechsler to describe the variability due to differences among personnel. Wechsler data indicate that for routine and well-defined tasks the ratio of performance scores for personnel at the top to those at the bottom is about

3:1. Accordingly, an EF of 2 ($\sqrt{3} = 1.7$) is used for HEs of T, M, and C. The EFs for other groups of HEs were proportionately assigned higher values based on expert judgements. Although many operation tasks are also routine and well-defined, many that are included in the PRA are not routinely performed and therefore, higher values were assigned to these groups of errors to show much larger variation, depending on the skill of the personnel, compared to variation among Personnel in Test, Maintenance, and Calibration activities.

Skill and Knowledge of HRA Analyst

The assignment of EFs for this variability cause was based on subjective judgement. An EF of 3 for Operational errors and dependent errors represents the relative difficulty of analyzing these errors compared to HEs of T, M, and C, where an EF of 2 was used.

4.3.3. Generic Considerations in Developing Oconee HEP Ranges

In applying the methodology presented to develop the ranges of the Oconee HEPs for sensitivity evaluation, additional considerations and assumptions are necessary. These primarily result from the human error modeling approach taken in Oconee PRA. The following presents the generic considerations for applying the methodology:

a) Treatment of modeling detail in developing HEP ranges

The Oconee human reliability analysis developed models to obtain the HEPs. The range methodology can be applied directly to the estimate of the HEP in the Oconee PRA with an associated error factor for inexactness of the model. For example, calibration errors are modeled as an unavailability, incorporating restoration and verification errors, using an equation:

$$U = \frac{1}{T} \sum_{i=1}^{N} P_i t_i$$

where, P_i = average probability that the component is not restored

t_i = length of the period for which P_i applies

N : number of periods

T : time between manipulation

For this model, the range was obtained directly for U with the assignment of EFs for each of the causes of variability, as defined previously.

In other situations where the estimate of the HEP is obtained as a sum of different types of errors, the associated EF for each of the causes could be different for each term. Following the methodology, the variability in each term can be developed separately and then combined to obtain the overall range. For example, the generic model for "operator fails to" is a fault tree of the human error with four possible basic contributors to the error. The probability of the error is given by:

$$P = P_1 \times P_2 + P_3 + P_4$$

P_1 = probability of failure to decide to take action based on event diagnosis
P_2 = probability of failure to decide to take action based on rules
P_3 = probability of failure to take action at correct time based on surveillance
P_4 = probability of uncorrected failure to manipulate controls

Typically, in the Oconee PRA such an HEP was dominated by one of the three terms (P_1P_2, P_3 or P_4). The range was obtained directly for P; however, where EFs can be separately assigned for each of the terms, the choice remains to obtain the overall variability by combining the variability of each of the terms.

b) Treatment of performance shaping factors

Performance shaping factors (PSFs) were considered in deriving the HEPs in the Oconee PRA. Also, various levels of stress factors were incorporated to modify the HEPs based on optimal level of stress. In developing the ranges, PSFs are not considered separately. The effect of PSFs are considered to be incorporated into the various variability causes defined. For example, three of the five causes of variability: lack of actual data, inexactness of the model, and differences among personnel incorporate various elements of PSFs to varying degrees.

c) Treatment of dependent human failure probabilities in the PRA model

The development of ranges for dependent HEPs was conducted using the estimate of the dependent HEP, and not by delineating the specific dependency factor used. It is, however, recognized that larger modeling uncertainty is associated with dependent HEPs, and accordingly, larger error factors were used for both lack of actual data and inexactness of the model. A more refined development of range can be performed by assigning EFs for the dependency factor and the individual HEP.

d) Choice of distributions for developing HEP ranges

As identified in the methodology, the distributions for HEPs need to be defined in obtaining the ranges. In this study, two types of distributions were used - lognormal and uniform. Following the standard practice in PRA studies, lognormal distribution was primarily used for HEs with base case probabilities less than 0.1 (as defined in Oconee PRA). This included a large portion of the HEs consisting of all human errors of calibration/restoration from T&M, and some of the human errors of operation. In all cases where the HEP >.1, a combination of lognormal and uniform distribution was used to obtain the lower and upper bounds of the range. For these cases, the distribution was assumed to have lognormal behavior as it approached the lower bound, but have a uniform behavior when approaching the upper bound.

e) Application of Overall Variance

The methodology used determines two solutions for overall variance, V: (a) no interaction among the variability causes, and (b) complete interaction among the variability causes. In this study, complete interaction among the variability causes was applied in order to derive the most conservative range of variance around the median HEP.

4.4 Example Application

The methodology described above was applied to the Oconee HEPs included in the risk-based sensitivity evaluation. Appendix B has a listing of the upper and lower bounds of the HEPs calculated using the methodology and using the sensitivity evaluation. In this section, the details of the calculations are given for example cases. Tables 4.2 through 4.7 present the steps in calculation for HEPs representing each of the groups of errors described in Section 4.3.1.

General observations on the application of the methodology are as follows:

a) The range of HEP obtained is strongly dependent on the error factor assigned for each of the variability causes. This signifies the importance of the assignment of these factors, which is largely based on expert judgements. Care was taken to be conservative, resulting in a broader range of the HEPs than would realistically be expected.

b) The range of HEP obtained was insensitive to the base HEP estimate as long as the associated EFs for the variability cause was the same. A number of applications were carried out for different base HEPs within a group defined by the same set of EFs to indicate minimal change in the overall error factor. Accordingly, a single error factor was used for every HE within that group. This error factor was then used to obtain the lower and upper bounds based on the base HEP.

c) For HEPs with base probabilities greater than 0.1, the use of the methodology resulted in upper bounds greater than 1, which were always truncated at 1. Also, for many HEPs where a base probability of 1 was used, a corresponding lower bound was calculated.

4.5 Summary of Range Development for Oconee HEPs

In this chapter, the methodology and quantification of the ranges of Oconee HEPs used in the sensitivity evaluation was presented. The defined ranges for the HEP play a significant role in a sensitivity evaluation as they define the limits of such an evaluation. The ranges of the HEPs were developed incorporating various causes of variability in the estimation of these probabilities.

The methodology presented can be applied to human errors used in probabilistic risk assessments; in this study, it was applied to Oconee HEPs. Table 4.8 summarizes the overall error factor for various types of human errors. The results also show that the error factor, which defines the range,

Table 4.2. Calculation of Range of HEP in Calibration/Restoration from T&M Activity

Human Error in T, M, and C
Example Case: Valve CW20AVMMH.
median estimate = 3×10^{-4}

Variability Cause	**EF**	σ_n	μ_j	V_j	s_j
1. Lack of Actual Data	10	1.4	8×10^{-4}	3.9E-6	1.98E-3
2. Inexactness of the Model	3	0.67	3.75×10^{-4}	7.97E-8	2.82E-4
3. Differences in Task Description	3	0.67	3.75×10^{-4}	7.97E-8	2.82E-4
4. Differences Among Personnel	2	0.42	3.28×10^{-4}	2.08E-8	1.44E-4
5. Skill & Knowledge of HRA analyst	2	0.42	3.28×10^{-4}	2.08E-8	1.44E-4

$s^2 = \sum_j s_j^2 = 4.1E-6$ EF = 18.1

$s^2 = \sum_j s_j^2 + \sum_{i \neq j} s_i s_j = 6.05E-6$ EF = 21.4

HIHEP = (HEP)(EF)
= (3E-4)(21) = 6.3E-3

LOHEP = (HEP)/EF
= 3E-4/21 = 1.4286-5

Table 4.3. Calculation of Range of HEP in Calibration/Restoration from T&M Activity

Human Error in T, M, and C
Example Case: Valve CW205VH.
median estimate = 3×10^{-3}

Variability Cause	**EF**	σ_n	μ_j	V_j	s_j
1. Lack of Actual Data	5	0.98	4.85E-3	3.8E-5	6.2E-3
2. Inexactness of the Model	3	0.67	3.75E-3	7.97E-6	2.8E-3
3. Differences in Task Description	3	0.67	3.75E-3	7.97E-6	2.8E-3
4. Differences Among Personnel	2	0.42	3.28E-3	2.08E-6	1.44E-3
5. Skill & Knowledge of HRA analyst	2	0.42	3.28E-3	2.08E-6	1.44E-3

$$s^2 = \sum_j s_j^2 = 5.8\text{E-}5 \qquad \text{EF} = 8.2$$

$$s^2 = \sum_j s_j^2 + \sum_{i \neq j} s_i s_j = 1.37\text{E-}4 \qquad \text{EF} = 12.5$$

$$\begin{aligned}\text{HIHEP} &= (3\text{E-}3)(13)\\ &= 3.9\text{E-}2\end{aligned}$$

$$\begin{aligned}\text{LOHEP} &= 3\text{E-}3/13\\ &= 2.3077\text{E-}4\end{aligned}$$

Table 4.4. Calculation of Range of HEP for Dependent Human Errors

Dependent Human Error in Restoration from Maintenance
Example Case: LWD99103H
Valves LWD-99 and LWD-103 Left Open.
median estimate = 3×10^{-4}

Variability Cause	EF	σ_n	μ_j	V_j	s_j
1. Lack of Actual Data	10	1.4	8×10^{-4}	3.9E-6	1.98E-3
2. Inexactness of the Model	10	1.4	8×10^{-4}	3.9E-6	1.98E-3
3. Differences in Task Description	3	0.67	3.75×10^{-4}	7.97E-8	2.82E-4
4. Differences Among Personnel	5	0.98	4.84×10^{-4}	1.45E-7	3.81E-4
5. Skill & Knowledge of HRA analyst	3	0.67	3.75×10^{-4}	7.97E-8	2.82E-4

$$s^2 = \sum_j s_j^2 = 8.1E-6 \qquad EF = 19.5$$

$$s^2 = \sum_j s_j^2 + \sum_{i \neq j} s_i s_j = 1.61E-5 \qquad EF = 26.1$$

HIHEP = (3E-4)(26) = 7.8E-3

LOHEP = (3E-4)/(26) = 1.1538E-5

Table 4.5. Calculation of Range of HEP Using the Method of Uniform Distribution for Operational Errors of Omission

Human Error in T, M, and C
Example Case: YRBSH - Operator fails to terminate RB spray.
mean estimate = 0.5

Variability Cause	EF	ℓn(EF)	upper bound, b	lower bound, a	$\sigma^2 = \frac{(b-a)^2}{12}$
1. Lack of Actual Data	10	2.3	1.0	0.22	.05
2. Inexactness of the Model	5	1.6	0.8	0.31	.02
3. Differences Among Personnel	5	1.6	0.8	0.31	.02
4. Skill & Knowledge of HRA analyst	3	1.1	0.55	0.45	.001

$$s^2 = \sum s_i^2 = .09 \qquad 1 \leq \chi \leq 0.02;\ b = 1,\ a = 0.02$$

$$s^2 = \sum s_i^2 + \sum_{i \neq j} s_i s_j = 0.186 \qquad 1 \leq \chi \leq 0;\ b = 1,\ a = 0$$

Considering a uniform distribution:

$$\mu = \frac{a+b}{2}$$

$$\sigma^2 = \frac{(b-a)^2}{12}$$

HIHEP = 1.0 (assumes uniform distribution)

LOHEP = 0.02 (assumes lognormal distribution)

Table 4.6. Calculation of Range of HEP for Operational Errors - Acts of Commission

Human Error of Commission for Operations
Example Case: LP12MVCH - Operator inadvertently throttles valve closed
median estimate = 3E-3

Variability Cause	**EF**	σ_n	μ_j	V_j	s_j
1. Lack of Actual Data	10	1.4	8×10^{-3}	3.89E-4	1.97E-2
2. Inexactness of the Model	5	0.98	4.84×10^{-3}	3.76E-5	6.13E-3
3. Differences in Task Description	--	--	--	--	--
4. Differences Among Personnel	10	1.4	8.0×10^{-3}	3.89E-4	1.97E-2
5. Skill & Knowledge of HRA analyst	3	0.67	3.75×10^{-3}	7.90E-6	2.81E-3

$$s^2 = \sum_j s_j^2 = 8.24E-4 \qquad EF = 18.3$$

$$s^2 = \sum_j s_j^2 + \sum_{i \neq j} s_i s_j = 1.58E-3 \qquad EF = 24.3$$

HIHEP = (3E-3)(24) = 7.2E-2

LOHEP = (3E-3)/(24) = 1.25E-4

Table 4.7. Calculation of Range of HEP for Operational Errors - Acts of Omission

Human Error of Omission for Operations
Example Case: XHPR2H - Operator fails to initiate HPR
median estimate = 3E-3

Variability Cause	**EF**	σ_n	μ_j	V_j	s_j
1. Lack of Actual Data	10	1.4	8×10^{-3}	3.89E-4	1.97E-2
2. Inexactness of the Model	5	0.98	4.84×10^{-3}	3.76E-5	6.13E-3
3. Differences in Task Description	--	--	--	--	--
4. Differences Among Personnel	5	0.98	4.84×10^{-3}	3.76E-5	6.13E-3
5. Skill & Knowledge of HRA analyst	3	0.67	3.75×10^{-3}	7.90E-6	2.81E-3

$$s^2 = \sum_j s_j^2 = 4.72E-4 \qquad EF = 16.2$$

$$s^2 = \sum_j s_j^2 + \sum_{i \neq j} s_i s_j = 8.42E-4 \qquad EF = 20.9$$

HIHEP = (3E-3)(21) = 6.30E-2

LOHEP = (3E-3)/(21) = 1.43E-4

Table 4.8. Error Factor Associated with Types of Human Error

Type of HE	Error Factor	
	Uncorrelated	Correlated
Dependent HEs	20	26
T, M, & C HEs with HEP > 1E-3	8	13
T, M, & C HEs with HEP $\leq$ 1E-3	18	22
Operation HE/act of Commission	18	24*
Operation HE/act of Omission	16	21*

* These factors are used to obtain the lower bound when the base probability is $\geq$ 0.1, and upper bound is directly obtained using the assumption of uniform distribution.

can be obtained for a group of human error defined by a set of individual EFs for the attributable variability causes. The improvement needed in this process is in defining the applicable error factors for each of the variability causes for a type of human error.

5. SENSITIVITY CALCULATIONS

5.1 Introduction

This section gives the detailed results of the sensitivity calculations, and an analysis of those results. A synopsis of the more important results is provided in the Overview Summary at the beginning of this report.

The overall methodology employed for performing these sensitivity calculations is described in Section 2. As described in detail in Section 3, the Human Errors (HEs) postulated in the Oconee-3 PRA were extracted, categorized and entered into the HE database program. Section 4 of this report then describes the development of ranges for the human error probabilities (HEPs), including lower bounds and upper bounds, which were also entered into the database. These bounds defined the interval limits of the HEPs to be varied in the sensitivity calculations. A strategy for what type of HE sorts and the sensitivity calculations that would be performed was then developed (Samanta, January 1988). Table 5.1 summarizes the sensitivity evaluations to be performed in accordance with this strategy. The Oconee PRA (NSAC-60) model, which was to be used for the sensitivity calculations, was first reconstructed on the BNL mainframe computer using the SETS computer code. A brief description of this model is provided in Appendix C, "Oconee-3 PRA Computer Model." For calculational ease and speed, and for compatibility with the Dbase program containing the HEs, the Oconee PRA then was converted to a personal computer or PC-based model. This model used the PAIRWISE computer algorithm, developed at BNL, and is fully described in Appendix D. Once these preliminary steps were completed, the actual sensitivity calculations were performed.

5.2 Method/Approach

5.2.1. Sensitivity Evaluation

The sensitivity evaluations performed here are intended to explore, among other things, the influence of human errors on the various plant risk parameters. Human error rates are believed to vary considerably between plants and among personnel. As an example, certain "good" plants are believed to have low error rates while certain "problem" plants are conceived to have higher human error rates. Various factors, such as education, training, management, and motivation, can affect the human error rates or probabilities. One technique for analyzing these issues, which affect the performance of a large number of personnel in a plant, is to conduct sensitivity studies using a PRA model whereby the human error probabilities (HEPs) are varied to determine the potential human error dependencies. This allows one to vary all HEPs as may occur in a plant when performance is affected by some top-level factor such as management or training.

A sensitivity study of this type, which is based on observing the variation in risk due to HEP changes without regard to the actual cause of the change in HEPs, allows the analyst to address those assumptions suspected of having a potentially significant impact on plant risk. Therefore, one can study the impact of certain selected human errors or groups of human errors. One can also examine the effect on risk due to different plant organizational

Table 5.1. Summary of Sensitivity Evaluations to Assess Implications of Human Errors on Plant Risk

Sensitivity Evaluation	**Significance of the Evaluation**
1. Sensitivity of Risk Parameters a. CMF versus HEPs b. RCF versus HEPs c. ACF versus HEPs	i. identifies the role of HEs in plant risk ii. identifies the role of HEs in consequences of accidents iii. identifies the role of HEs in likelihood of accident sequences
2. Sensitivity of Risk Parameters to Errors of Recovery	Identifies the ability of operating staff to respond to an accident situation
3. Sensitivity of Risk Parameters to "Routine" Human Activity	i. identifies the perturbation in the risk level due to variation in the performance level of operating staff ii. identifies the human errors deserving special attention during plant operation
4. Sensitivity of Risk Parameters to Groups of Human Errors Signifying Various Types of Activity	i. identifies the operators' role in maintaining risk level ii. identifies the significance of negligent plant practices iii. identifies the need for adequate training (also refer to Table 1 of the report)
5. Sensitivity of Risk Parameters to Errors of Diagnosis	Identifies the effect of misdiagnosis on plant risk
6. Sensitivity Analysis With Respect to Recovery from T,M & C Errors	Identifies the role of a disciplined crew in detecting and restoring errors from human mistakes
7. Sensitivity Analysis to Obtain Relative Ranking of Human Error Categories	i. identifies the role of various types of personnel ii. identifies the role of inspection activities iii. identifies the role of human error in and out of control room
8. Sensitivity Analysis to Obtain Relative Likelihood of Various Accidents	Identifies the dominance of accident sequences based on the performance of the plant crew

structures and different types of human errors. Notwithstanding that a sensitivity evaluation is subjective to recognizable assumptions, meaningful insights can be gleaned from its results. In addition, a review of actual human error occurrence data for Oconee 3, as described in Section 3, was performed to help establish the validity of the sensitivity study using the Oconee-3 PRA.

Concomitant with the sensitivity calculations to produce risk variation curves, a more in-depth analysis was performed to identify the dominant human errors in the minimal cutsets of the dominant accident sequences. The cutset analyses identified specific human errors that contribute significantly to risk in the various accident sequences, as well as those minimal cutsets containing multiple human errors. This process allows one to explore human error coupling with hardware failures in each accident sequence in more depth than would be the case of observing risk variation in sensitivity curves alone. In general, the cutset analysis provided good agreement with the results of the sensitivity evaluations in describing the important types and groups of human errors. Details of the cutset analyses to identify the significant human errors in the three most dominant accident sequences of the Oconee-3 PRA model are provided in Appendix F. Some specific results are mentioned throughout the Section.

5.2.2 Methods of Varying HEPs

To vary the human error probabilities (HEPs) from their base case values to their upper and lower bounds, an appropriate method of variation had to be determined. A number of methods were explored, and two methods were selected. Since it is not known precisely how the HEPs will vary as overall human performance improves or degrades, two possibilities were selected to show the effect produced on the risk parameters from different methods of HEP variation over the same range. The two methods, the Factor Method and the Range Method, are described below, and selected sensitivity results for both are shown. However, most results are derived and presented using the factor method so that comparisons can easily be made to glean insights.

5.2.2.1 Factor Method

In this method, the HEPs are varied in a multiplicative fashion. To obtain a selected set of HEPs for sensitivity calculations using the "factor method," the base case HEP for each error is multiplied by a fixed constant factor, and then a new CMF is calculated using these new HEPs. When the multiplicative process causes the HEP to exceed its upper bound or the value of 1.0, the HEP is set at the value of its upper bound or 1.0. For values between the lower bound and base case value, the same method applies, wherein the base HEP is divided by the constant factor rather than multiplied. This method results in HEPs increasing to their upper bound relatively quickly and in CMF also increasing quickly. This method assumes that a set of HEPs change together at the same rate regardless of their base case values or how close to unity (1.0) they are.

5.2.2.2 Range Method

The "range" method of varying the HEPs is devised such that all HEPs will vary and reach their upper bound or lower bound at the same time. This is particularly important for errors with HEPs that lie in the interval between 0.1 and 1.0. Consider an error with a base case HEP of 0.5. This error would reach 1.0 by the factor method when all errors are at twice their base values, but would not reach 1.0 in the range method until all other errors had reached their upper bounds (for some of which this would mean 26 times their base values). To achieve this, a method was devised to move the HEPs in percentiles along the lognormal distribution equally until they reached either 95% (upper bound) or 5% (lower bound). To normalize the distribution for various percentiles, a statistically derived z-score transformation was used to obtain the scaling of the error factors that was determined from a lognormal distribution of HEPs. The area under the curve for each percentile plotted, e.g., .1 through .9, was obtained from a z-score distribution. The z-score values were obtained from Tables for the standard normal distribution function. These values were then divided by the upper (or lower) bound z-scale value (e.g., $Z_{.95}$). For percentiles greater than .5, the ratio of these two values was positive, for those less than .5, the ratios were negative. The value of the error factor to be used at each percentile was computed as follows:

$$ef_i = \exp - \left\{\left[\frac{z_i}{z_{lb}}\right] \ln \left[\frac{\bar{x}}{lb}\right]\right\} \text{ for } 0.05 < i < 0.5$$

$$ef_i = \exp + \left\{\left[\frac{z_i}{z_{ub}}\right] \ln \left[\frac{ub}{\bar{x}}\right]\right\} \text{ for } 0.5 < i < 0.95$$

where i = ith percentile, ub = upper bound value ($z_{ub} = z_{.95}$), lb = lower bound value ($z_{lb} = z_{0.05}$), and x = base case HEP. The resulting error factor for each percentile was multiplied by x (base case value) at each point to provide the HEP estimates for calculation of either the core melt or accident sequence frequency.

This method results in all HEPs reaching their upper bound (or lower bound) at the same time, even if the upper bound is 1.0. Errors that are close to 1.0 or close to their upper bound will increase at a slower rate than other errors, and hence, results in CMF increasing at a slower rate than with the factor method.

5.3 Results of Sensitivity Calculations

5.3.1 Organization of Results

Sensitivity evaluations, summarized in Table 5.1, were performed to determine the effect of human errors on plant risk parameters. Each sensitivity evaluation addresses some aspect of human performance in nuclear power plant operation. Sensitivity curves of the various risk parameters are plotted from the calculated data for each analysis. Appendix E gives the actual data, on which the curves are based.

The results, based on the two methods of HEP variation, and the interpretation of risk variation curves produced for each specific evaluation are presented in the following subsections. Additionally, individual accident sequence-level cutsets were reviewed to identify the specific human errors which affect the magnitude of risk sensitivity. Details of the cutset analyses are presented in Appendix F.

Subsection 5.3.2 discusses the overall sensitivity of various risk parameters (e.g., core melt frequency, accident sequence frequency, and "core-melt bin" frequency) to HEP variations. The sensitivity of plant risk to various categories of human errors is discussed in subsection 5.3.3. To compare the sensitivity of accident types, the sensitivity of selected accident sequences to human error impact is examined in subsection 5.3.4. Sensitivity evaluations to address special situations such as the impact of recovery events and routine human actions on plant risk are described in subsection 5.3.5. The assessment and interpretation of important results with meaningful insights are highlighted in Section S.3 of the Overview Summary.

5.3.2 Overall Sensitivity of Risk Parameters

A method to identify the role of human errors on plant risk is to assess the sensitivity of risk parameters to changes in HEPs in a nuclear plant. In this assessment, the probabilities of all human errors that are considered to influence a risk parameter are being changed together. The justification for this approach is multifold: (a) the determination of HEPs in PRA studies are subjective, wherein there may be systematic underestimation or overestimation in them, (b) the HEPs are average estimates and there are a number of causes that may vary the HEPs, and (c) a nuclear power plant may experience an improved performance or a degraded performance by its operating staff which are respectively signified by decreasing and increasing HEPs.

The following subsections described the sensitivity of various risk parameters to HEP variations.

5.3.2.1 CMF Sensitivity to Human Errors

Figure 5.1 illustrates the dependence of CMF on variations in human error probability when all HEPs are changed simultaneously over their ranges. Within these ranges HEPs are varied by multiplicative factors. In addition, the effects of varying HEPs for recovery errors and non-recovery errors upon CMF are displayed. The set of recovery errors considered here consists of all human actions to restore the operation of a failed system, or to find alternative systems. Non-recovery errors refer to all other operator errors.

Even though CMF varies over four orders of magnitude from changes in HEPs, the largest change in the CMF is observed within a factor of 10 increase in base case HEPs. This effect is due to HEPs with large initial values, e.g., recovery actions with HEPs of 0.1 to 0.5, reaching their upper bounds within this interval. As detailed in Appendix F, the review of minimal cutsets shows that the HEPs of sequence-dependent recovery errors, e.g., operator failure to recover instrument air (REIA1), tend to effect the increase in CMF sensitivity. On the other hand, reduction in HEPs by constant factors results in a significant decrease in CMF until hardware failure contributions supersede the human error impact.

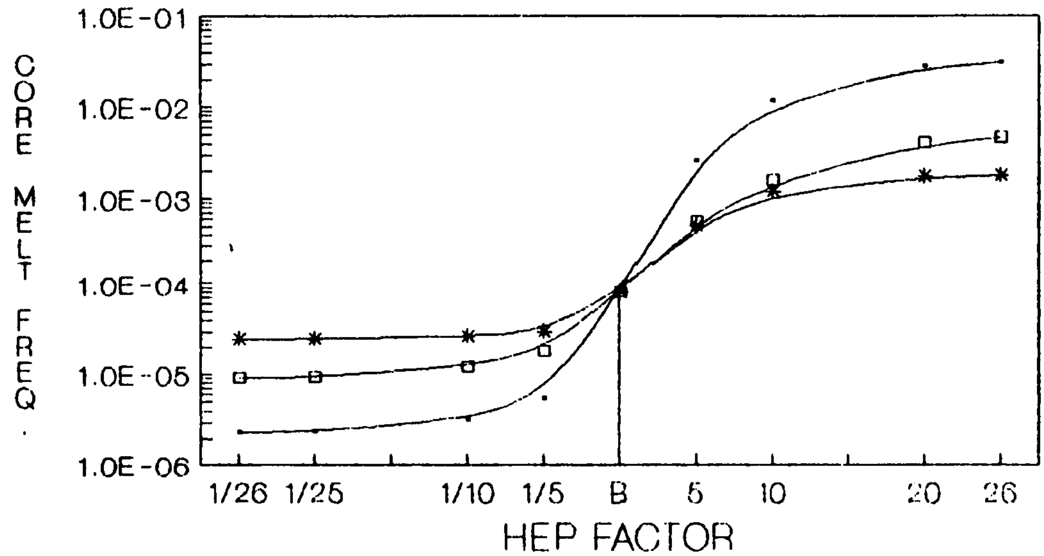


Figure 5.1. Sensitivity of core melt frequency to HEP variations (HEP factor)

In Figure 5.2, the overall CMF results are shown using the range method of HEP variation. The increase in CMF is much more gradual than with the factor method in Figure 5.1. Nonetheless, the final endpoints are the same. As mentioned earlier, it is not known precisely how the HEPs will vary. However, it is likely that the actual increase or decrease in CMF will be bounded by the two methods shown here. The rest of this section will present the results using the factor method, with a few further examples of the range method.

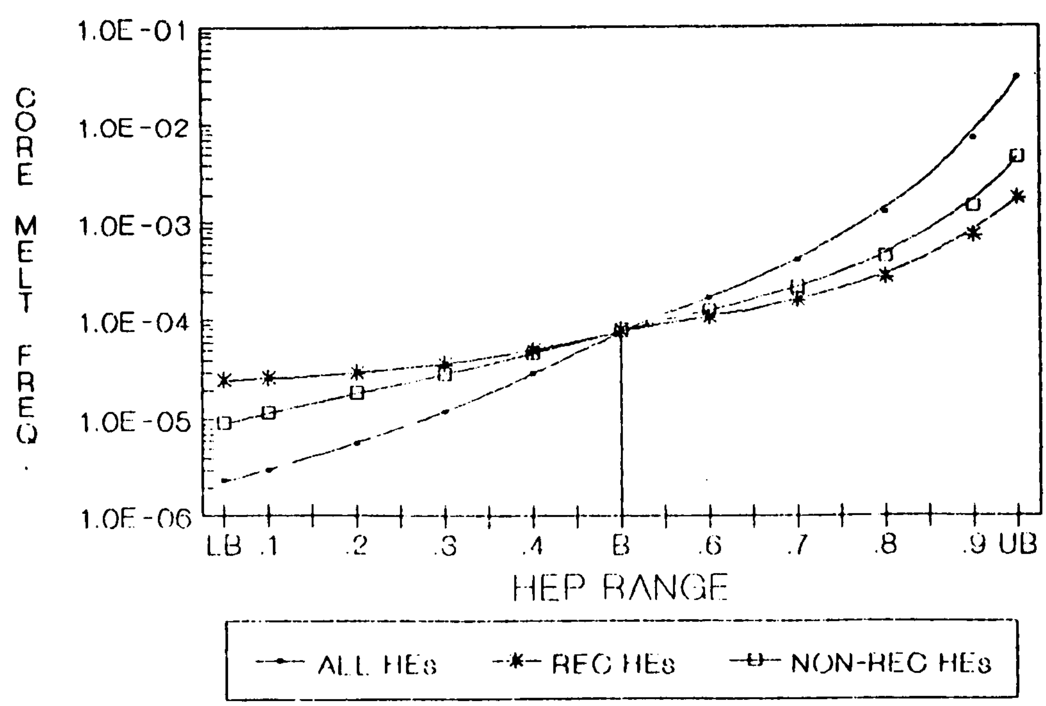


Figure 5.2. Sensitivity of core melt frequency to HEP variations (HEP range)

5.3.2.2 Accident Sequence Sensitivity to Human Error

As HEPs vary over their full range, the relative likelihood of occurrences of different accident events can be obtained to show the distribution of accident risk levels due to changes in HEPs. The relative percentage contribution to overall core melt frequency from different types of accident sequences can be derived to show the dominance of accident events as HEPs are varied. Accident sequence types are characterized by a specific initiating event, e.g., a loss of instrument air system, that causes the accident to progress. The relative contribution of each type of accident sequence at Oconee versus HEP is shown on the next three Figures (5.3 to 5.5) and is discussed below. Following these discussions plots are given, which show the absolute variation of accident sequence frequency with HEP variation. In reviewing the relative distribution curves, one should realize that as HEPs increase and the sequence frequency increases, the relative contribution of that sequence may still drop if other sequences' frequency increases faster.

Figure 5.3 shows that the relative contribution to core melt frequency of three of the dominant types of accident sequences at Oconee. The loss of instrument air (LOIA) sequences increase from 36% to about 64% as HEPs increase from base case to upper bound values. Also, the relative contribution to core melt frequency from LOIA sequences decreases to 0.6% when HEPs are set at lower bound values. For loss of service water (LOSW) sequences, the relative contribution to overall plant risk increases from 0.8 to 19 percent as HEPs are increased from the lower bound to the 70th percentile of the range. However, the relative contribution to core melt frequency from LOSW sequences is about 15% when HEPs are at upper bound values. The reduced sensitivity for LOSW is due to the number of significant human error contributors being smaller and the upper bound values of these contributors being generally lower than those for the dominant sequence types, e.g., LOIA.

The relative contribution of steam generator tube rupture (SGTR) sequences to plant risk decreases from 21% to 6% as HEPs are varied from lower to upper bound values. This is because SGTR sequence cutsets do not contain a significant number of human errors. Similar trends in the relative distribution of accident risk are observed for small and large-break loss-of-coolant accident (LOCA) sequences as shown in Figure 5.4. Again, the reason is that these sequences are not affected by human errors to the same extent as the LOIA sequences.

Where there is no impact of human error, for a reactor vessel rupture sequence, the relative distribution of accident risk varies from 30% to zero over the HEP range from the lower bound to upper bound values. These observations imply that accident sequences characterized by hardware failures and malfunction of automatic safety systems are not driven by human errors to the same extent as transient event sequences. Thus, the contribution from these sequences is largely a function of hardware reliability rather than human errors.

For less dominant accident sequences, Figure 5.4 shows that the contribution from transient-initiated sequences (e.g., loss of main feedwater or loss of offsite power events) is again higher than that for sequences involving hardware failures (e.g., large feedwater line break) when HEPs are at upper

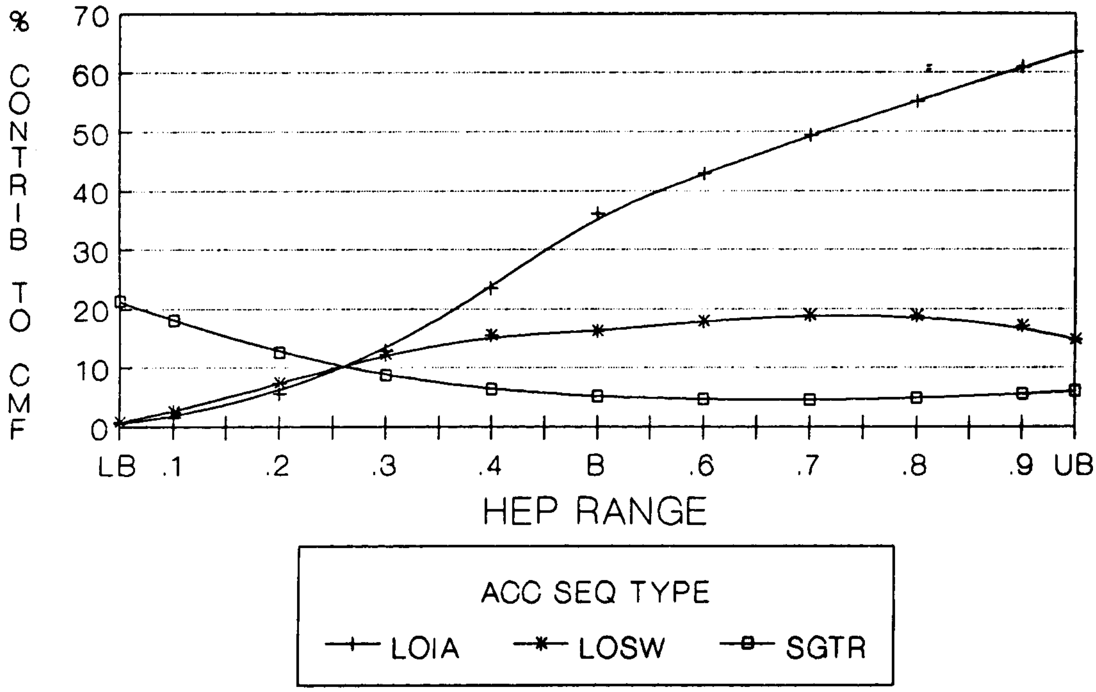


Figure 5.3 Relative distribution of accident risk over HEP range

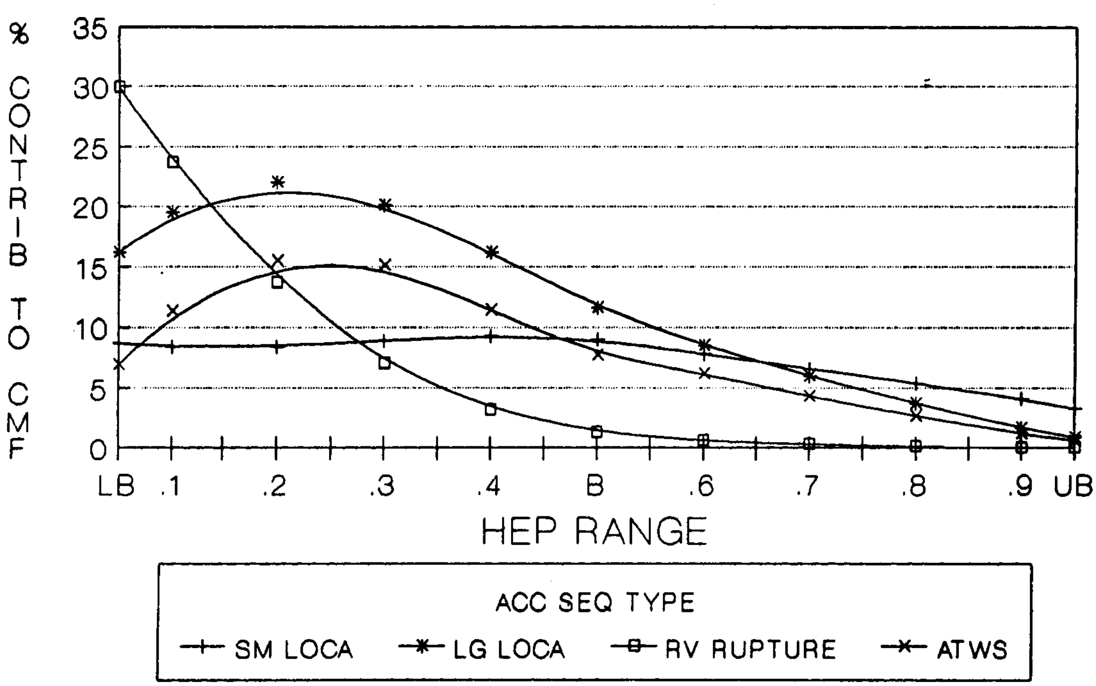


Figure 5.4. Relative distribution of accident risk over HEP range

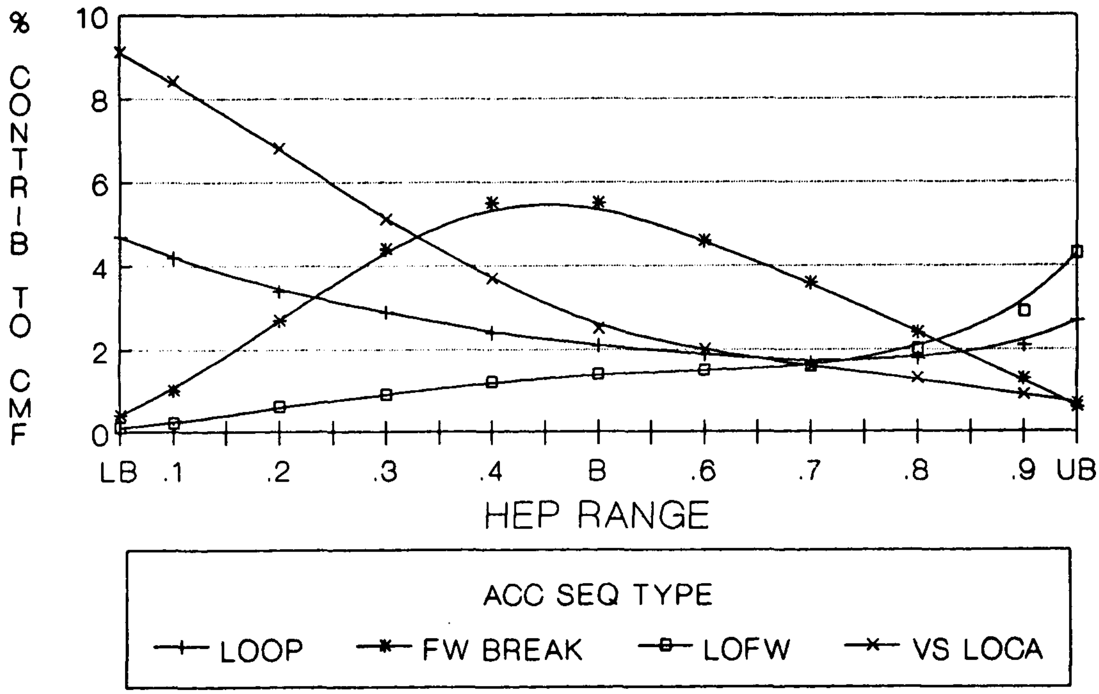


Figure 5.5. Relative distribution of accident risk over HEP range

bound values. This increasing percentage contribution from these sequences is attributed to more human errors being modeled. Therefore, it can be seen that degraded human performance has greater influence on sequences initiated by transient events.

Figures 5.6 and 5.7 show the sensitivity of the dominant accident sequences to changes in HEPs. In general, all dominant accident sequences are sensitive to human error and vary over seven orders of magnitude as all HEPs increase from lower bound to upper bound values. The sensitivity curves show that transient-initiated accident sequences such as the loss of instrument air (T_6BU) and loss of service water ($T_{12}BU$) sequences have significant human error dependence. Therefore, probabilities of such sequences have the potential for being reduced by reducing human error rates, especially those HEPs for sequence-dependent recovery errors. The decrease in failure probabilities of recovery errors can be influenced by training, well-developed procedures, and operating practices. For sequences dominated by hardware failures such as large break LOCA (AX_a) and small break LOCA (SY_sX_s) sequences, the sensitivity curves show that there are no great reductions in their probabilities when the contributions from human errors are decreased significantly. This effect indicates that accident sequence likelihood due to design-basis accidents dependents heavily on hardware reliability as well as contributions from human error.

Additionally, it is important to observe that increasing human error probabilities from the base values greatly increases the sequence probabilities to varying extents depending on the involvement of human actions in each sequence. The increasing accident sequence likelihoods due to human errors identify the role of degraded human performance in accident risks.

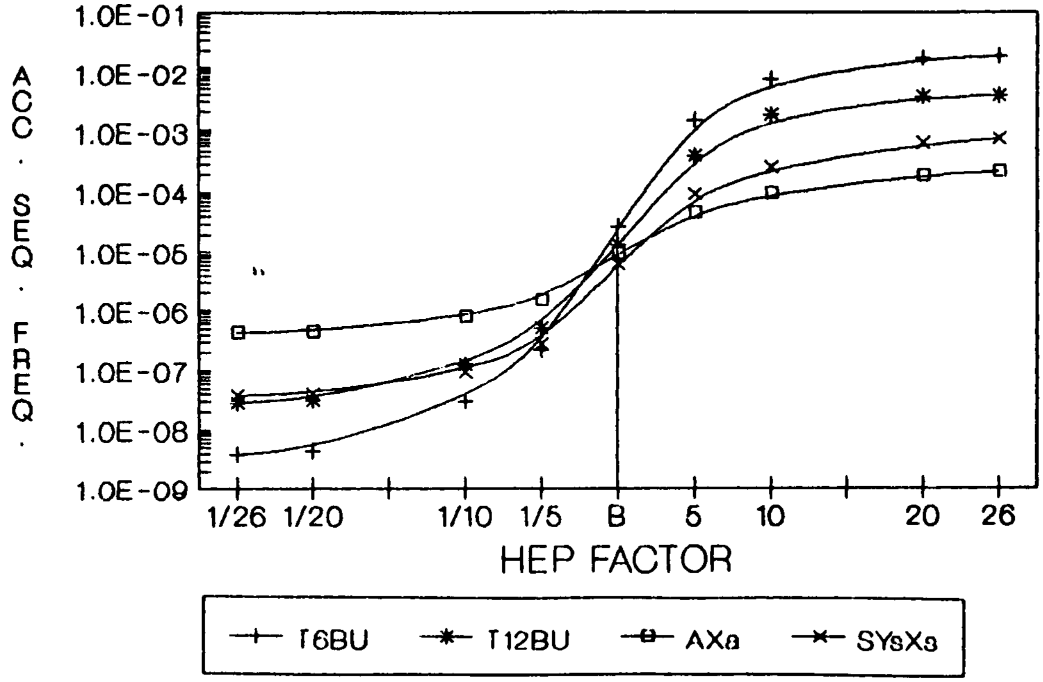


Figure 5.6. Sensitivity of accident sequence frequency to HEP variations (HEP factor)

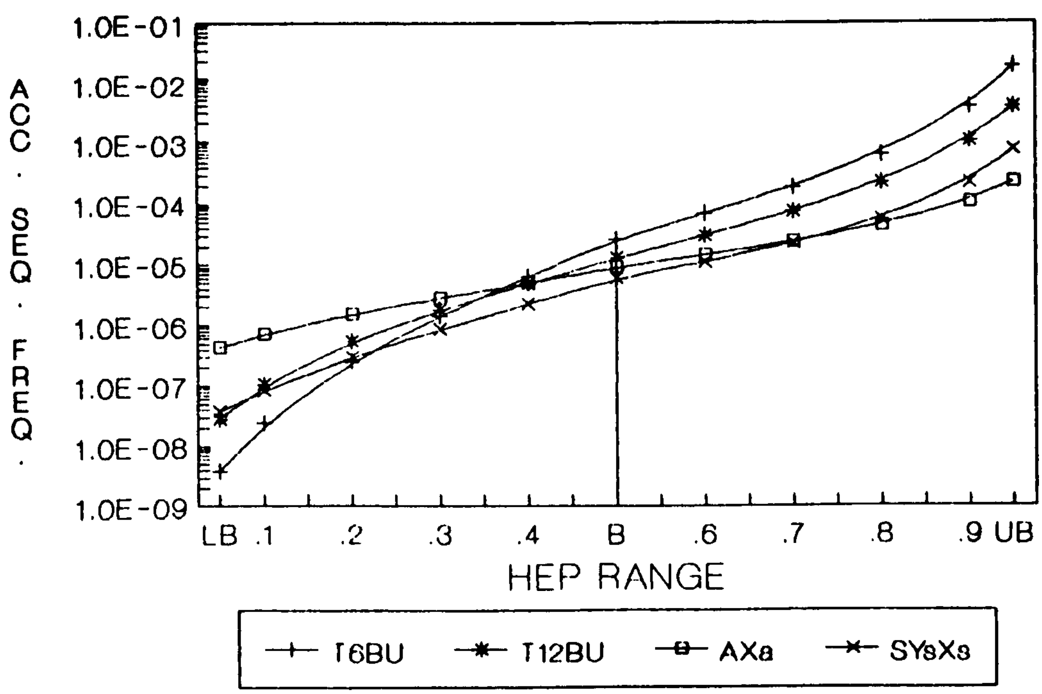


Figure 5.7. Sensitivity of accident sequence frequency to HEP variations (HEP range)

The impact of human performance in accident sequences can be examined by observing the factor by which ASF changes as HEPs are varied in steps to their upper and lower bounds. Figure 5.8 shows the variation of ASF factors due to changes in HEPs for three different accident sequences. An interesting feature of these accident sequences is that significant improvements on frequencies can be made for relatively small improvements in HEPs. For example, a factor of 5 improvement in HEPs will decrease the T_6BU sequence frequency by a factor of 120, and the SY_sX_s sequence frequency by a factor of 20. This is because multiple human errors appear in the dominant terms of the accident sequence frequency for T_6BU. One impact of multiple human errors is further highlighted by the large ASF factors for T_6BU and T_5QX_s (loss of offsite power with stuck-open safety relief valve) sequences. An interesting insight gleaned from these observations is that a less dominant sequence such as T_5QX_s, which is a transient-induced LOCA sequence, can have a dominant impact on plant risk when human performance becomes degraded. The nature of risk variation curve for T_5QX_s is due to the number of cutset-dependency recovery errors modeled in its accident sequence frequency expression. For this sequence, cutset-dependent recovery errors such as failure to recover power to the instrument air (RESUBAIR1) are modeled to account for support system dependencies.

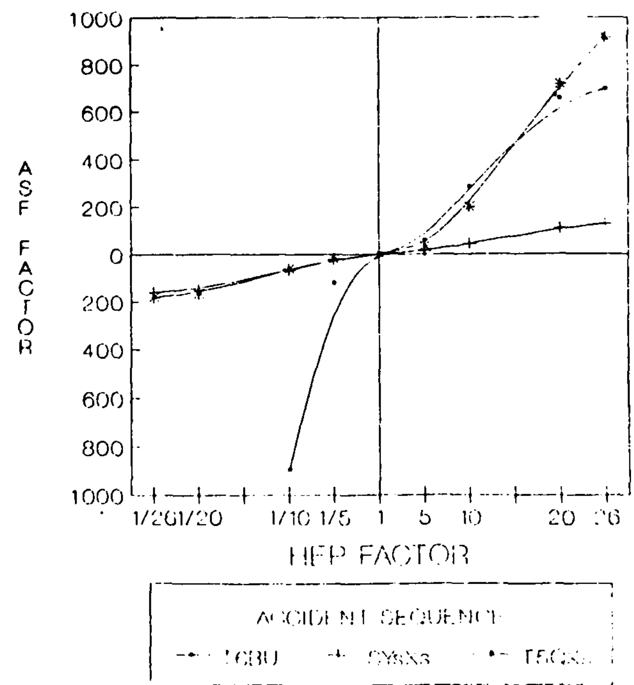


Figure 5.8. Variation of ASF factors to HEP changes

5.3.2.3 Accident Consequence Sensitivity to Human Errors

Since "core melt bins" define the nature of offsite consequences, the sensitivity of accident consequences to human errors can be evaluated by considering the changes in the frequency of these bins. In Figures 5.9 and 5.10, the sensitivities of various core melt bin frequencies are given to show the impact of human error of accident consequences. Table 5.2 summarizes the accident sequence characteristics for each core melt bin, which indicate the severity of their consequences.

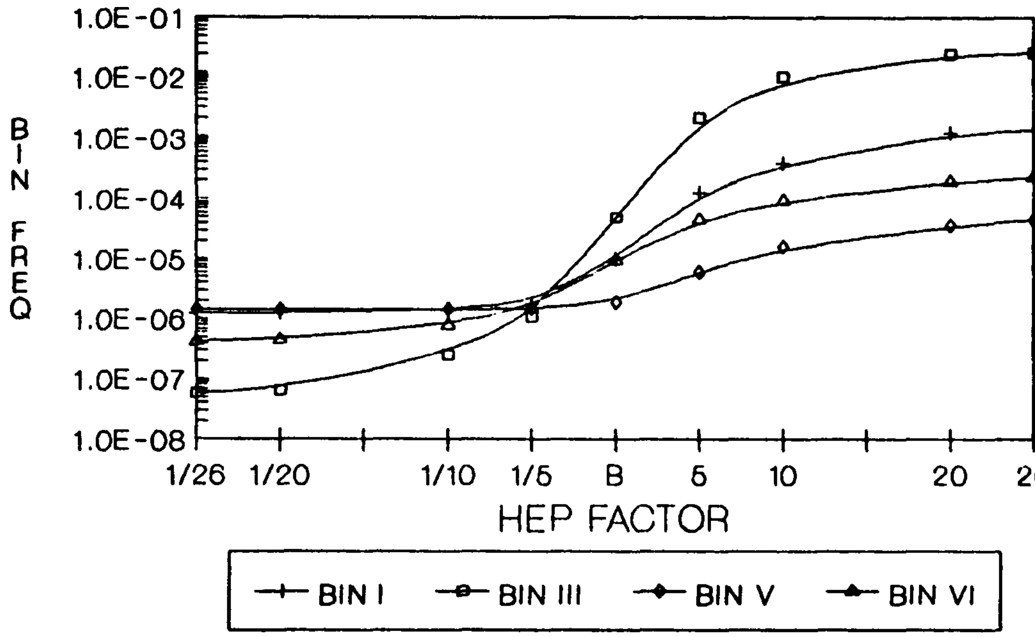


Figure 5.9. Core melt bin sensitivity - bins I, III, V, and VI

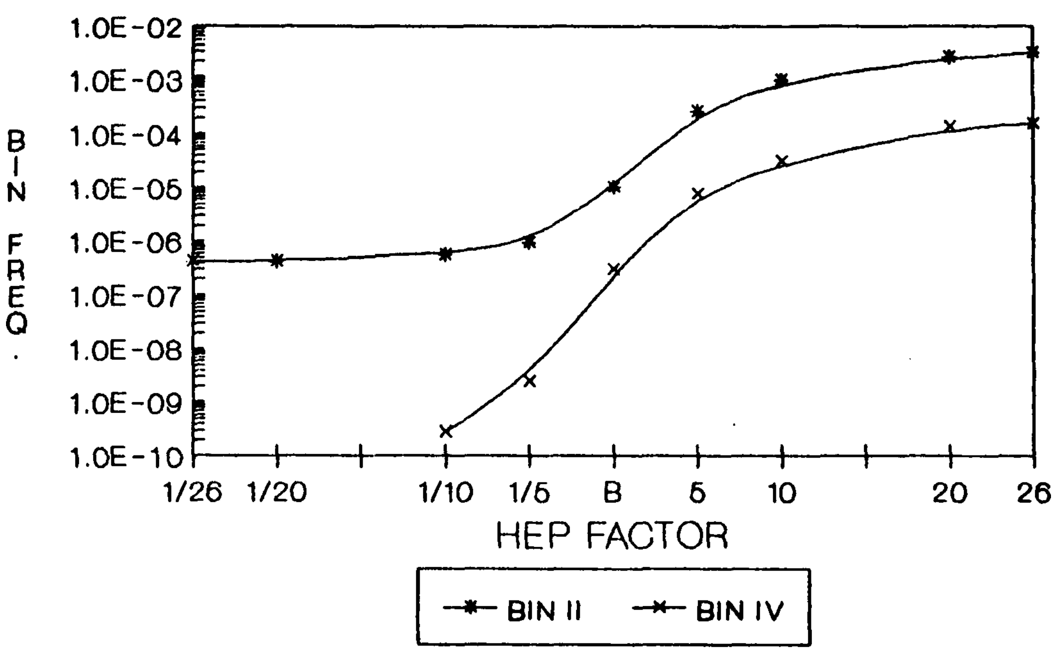


Figure 5.10. Core melt bin sensitivity - bins II and IV

Table 5.2. Summary of Core Melt Bins

Bin	Accident Sequence Characteristics
I	RCS pressure and leakage rates associated with small-break LOCAs with early melting of the core (i.e., within about two hours after the break occurs)
II	RCS pressure and leakage rates associated with small-break LOCAs with late melting of the core (i.e. after about 12 hours after the break occurs
III	High RCS pressure and leakage rates associated with boiloff of the reactor coolant through cycling pressurizer relief valves with early core melting (within about two hours)
IV	High RCS pressure and leakage rates associated with boiloff of the reactor coolant through cycling relief valves with late melting of the core
V	Large rates of leakage from the RCS and low pressures associated with large-break LOCAs with failure of core injection
VI	Large-break LOCA conditions with failure of coolant recirculation

Large sensitivity to human errors is observed for Bin III, which characterizes early core melt conditions due to transient-initiated sequences. Even though Bin III defines moderately high consequences, the large sensitivity to human errors is primarily due to multiple human errors modeled in the minimal cutsets of the loss of instrument air (T_6BU) sequence. Bin I, which characterizes early core melt conditions due to small LOCAs shows moderate sensitivity when HEPs are increased by factors greater than base values. Bin I sequences include transient-induced LOCA sequences such as a loss of offsite power with stuck-open relief valve sequence. Such sequences contain a number of cutset-dependent recovery errors which drive the sensitivity. Lesser sensitivity to human errors is observed for high consequence Bins V and VI, which characterize early core melt conditions due to large LOCAs. As for late core melt bins, reduction in HEP factors eliminate the risk of transient-initiated sequences assigned to Bin IV. In general, increased HEPs have greater impact on low consequence bins than high consequence bins. Also, low HEPs reduce the risk of transient-initiated sequence bins.

5.3.3 Sensitivity of Risk to Various Categories of Human Errors

Sensitivity evaluations of risk parameters to various categories of human errors can be used to identify the contributors to the spectrum of risk in terms of accident timing, type of activity, event type, personnel involvement, and error type. The risk impact of various categories of human errors is addressed by the relative ranking of human error aspects, such that a small subset of human errors can be identified that might reduce risk. Contributions of human error to core melt frequency are analyzed by changes in generic categories of human error probabilities. The results are presented in Figures 5.11 through 5.14.

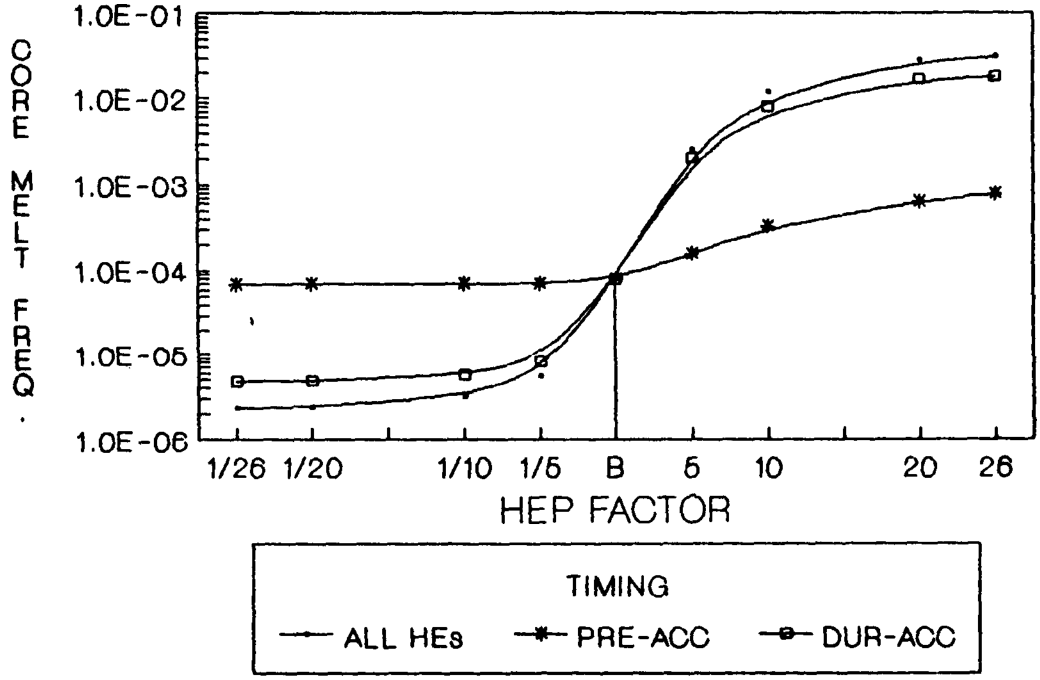


Figure 5.11. Sensitivity of CMF to pre-accident and during accident errors

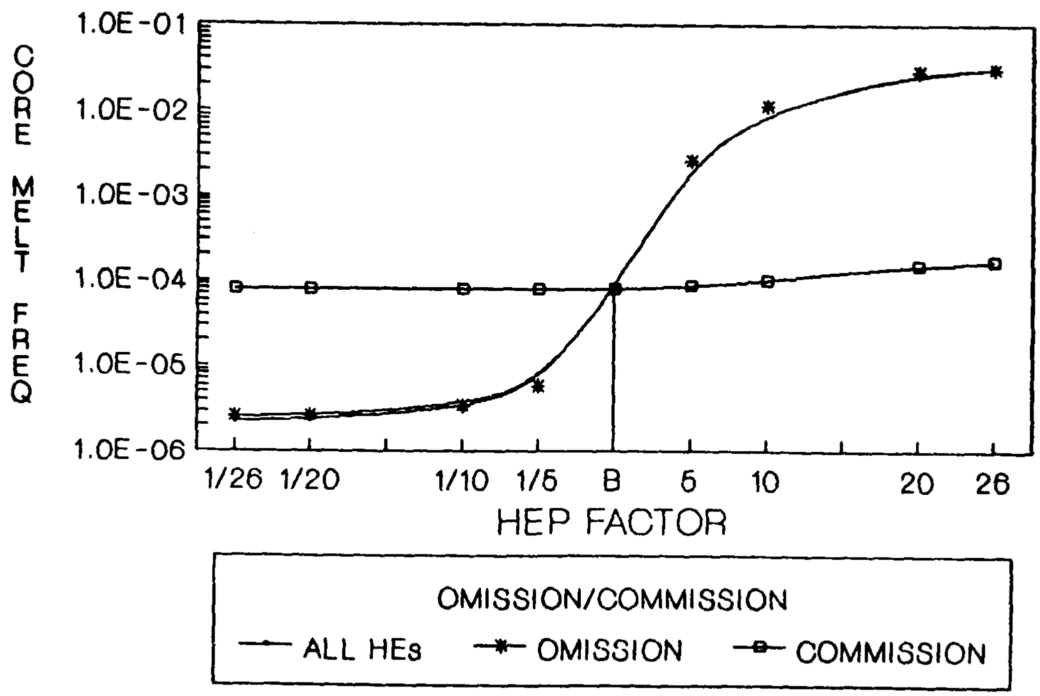


Figure 5.12. Sensitivity of CMF to omission and commission errors

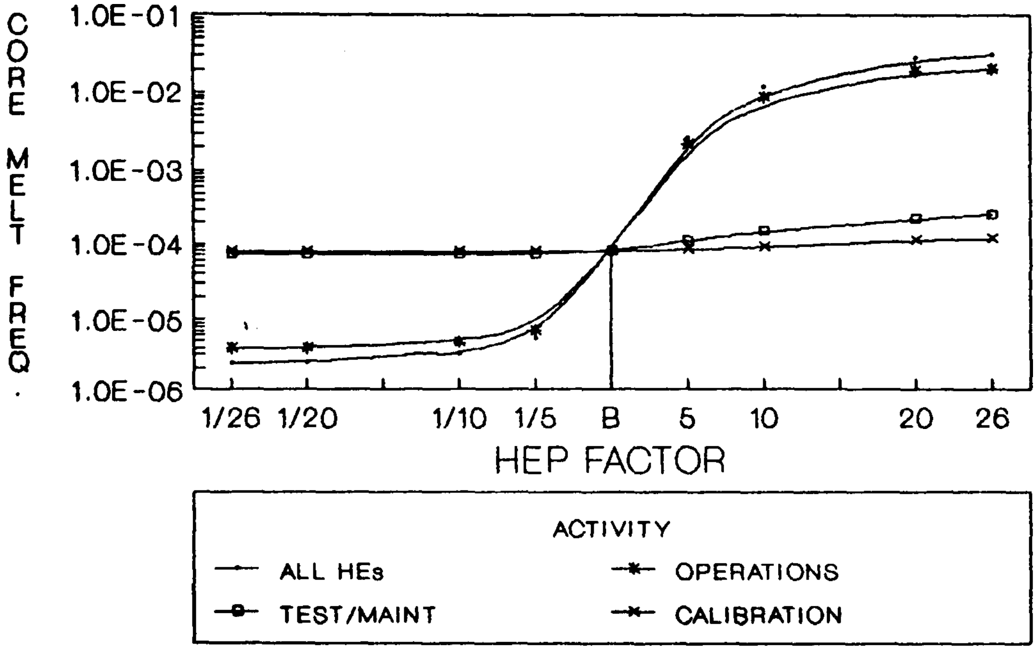


Figure 5.13. Sensitivity of CMF to utility activities

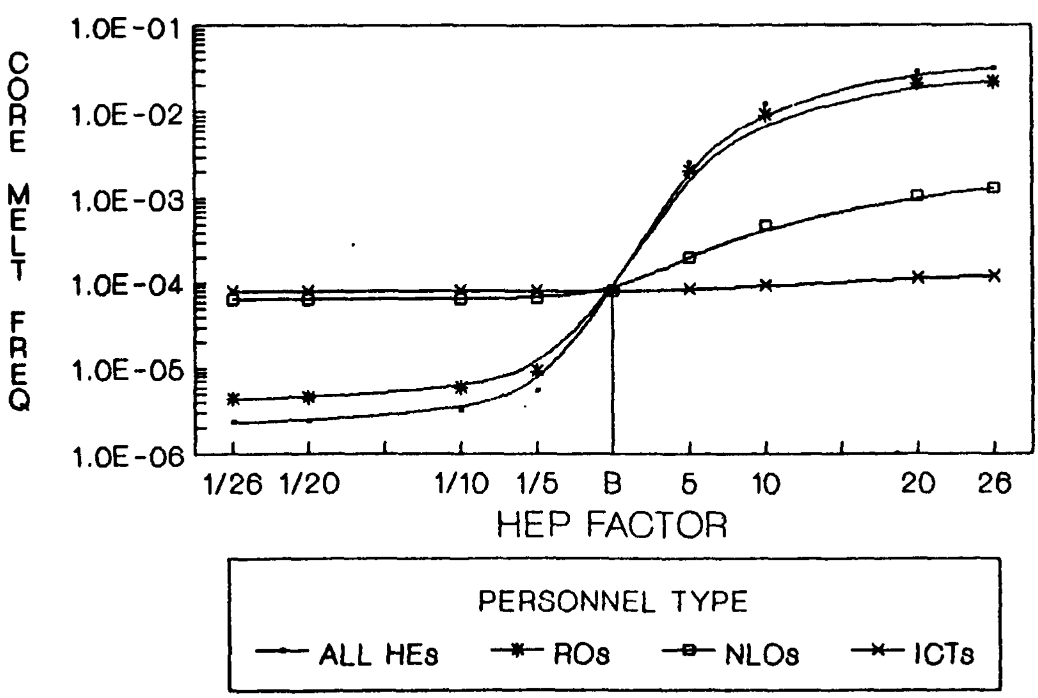


Figure 5.14. Sensitivity of CMF to Personnel Type with prime responsibility for HE

5.3.3.1 Timing

Core melt frequency is more sensitive to during-accident human errors than pre-accident errors as shown in Figure 5.11. This sensitivity is largely due to the dominant effect of recovery errors during accidents. For specific accident sequences, recovery errors such as failure to restore a system or plant function (e.g., REIA1, REFDW1, RESSFSI) have estimated probabilities greater than 1×10^{-1}.

These errors are sequence dependent and occur in all cutsets of certain dominant accident sequences. Within the dominant cutsets, the recovery errors are also combined with dynamic human errors, i.e., errors in taking actions by following procedures during an accident sequence (e.g., UTHPIH, XALPRH). Even though dynamic human errors may have probability estimates between 1×10^{-2} and 1×10^{-3}, the multiple effect of recovery errors and dynamic human errors modeled in the dominant cutsets has a large impact on core melt frequency when HEPs are increased. The various combinations of these errors that drive risk sensitivity are given in the cutset analysis as detailed in Appendix F.

Pre-accident errors are observed to have a relatively moderate effect on core melt frequency when HEPs are increased. Even though the estimated probabilities of most pre-accident errors are on the order of 1×10^{-3}, this moderate effect is attributed to latent human errors, e.g., failure to restore a component after testing or maintenance (LP4142VVH, HPCROSSH, LP15MVMH, LP16MVMH), which have probability estimates between 1×10^{-1} and 1×10^{-3}. In contrast to during-accident errors, pre-accident errors usually occur as singular events in the dominant cutsets. Also, the number of cutsets containing one or more pre-accident errors is less than those containing multiple during-accident errors. Therefore, the greater number and contribution of dominant cutsets with multiple during-accident errors are key reasons that "during accident" errors have greater influence on core melt frequency than pre-accident errors.

In the direction of decreasing HEPs, the pre-accident errors show essentially no effect on CMF. This says that while it is important to maintain these error rates, further decreasing of them does not reduce risk.

5.3.3.2 Omission/Commission

Figure 5.12 shows that omission errors rather than commission errors have a dominant effect on core melt frequency. Errors of omission are related to operator failure to perform required actions, while most errors of commission modeled for Oconee are associated with calibration activities and have little effect on core melt frequency. The marked sensitivity of core melt frequency to errors of omission is attributable primarily to the modeling of human error in the Oconee PRA where recovery events are modeled as acts of omission. Also, commission errors are difficult to model and usually have a very low probability of occurrence.

5.3.3.3 Utility Program Activity

In evaluating the impact of human error on various types of utility program or plant activity, errors during operations were found to have significant sensitivity to core melt frequency (Figure 5.13). The impact of operations-type errors is largely due to dynamic errors in the operator's response to plant upset and to recovery actions when the staff is performing actions under stress. Test and maintenance errors, especially restoration errors after maintenance, show moderate sensitivity, while errors in calibration activities have the least impact on core melt frequency.

As explained previously, the dynamic errors and recovery actions occur as multiple events in the dominant cutsets. These effects contribute to the marked sensitivity of core melt frequency to operations-type errors. In contrast, restoration errors after test and maintenance activities occur as singular events in dominant cutsets and their probability estimates are between 10^{-3} and 10^{-1}. Therefore, the impact of test and maintenance errors on core melt frequency is relatively moderate. Calibration errors are few in number and their probability estimates are on the order of 10^{-5}. As such, their impact on core melt frequency is very small.

5.3.3.4 Personnel

Figure 5.14 identifies personnel that are dominant contributors to plant risk. The curves show that core melt frequency is markedly sensitive to operational errors committed by reactor operators (ROs). Non-licensed operator (NLO) related errors have a moderate effect, while errors due to instrumentation and calibration technicians (ICTs) have minimal influence on core melt frequency.

The RO errors are those where the Reactor Operator has prime responsibility. The following section gives a further breakdown of the errors depending on the secondary responsibility (if any) of other personnel. As discussed in Section 3, 57% of the human error database for sensitivity evaluation are categorized as RO errors. Moreover, the larger number and higher probability estimates of RO errors, which also occur mostly in dominant cutsets, contribute to the marked sensitivity of core melt frequency to these errors. The NLO errors are those where the non-licensed operator has prime responsibility. The moderate effect of NLO errors is due to their smaller number, which constitutes 23% of the human error database, and the fact that the NLO errors involve the less important restoration errors from test and maintenance activities. Core melt frequency is least affected by ICT errors because there are fewer number of these errors and the magnitude of their probability estimates is usually on the order of 1×10^{-5}.

Similar sensitivity curves were obtained for the location of human error occurrence, i.e., within the control room (CR), outside the control room (OCR), and uncertainty of whereabouts or dual location (CROCR). The interpretation of the location sensitivity curves are similar to the personnel category and provided no new insights, so they are not presented.

5.3.3.5 Personnel Interactions

The overall level of plant risk has marked sensitivity to operations-related errors committed by reactor operators (ROs) or ROs in conjuction with other personnel. Therefore, the impact of errors due to interactions between the reactor operator and other plant personnel was assessed by evaluating the sensitivity of core melt frequency to HEP variations of different sorts of personnel interactions. In this section, all of the RO prime responsibility errors shown previously on Figure 5.14 are further decomposed into: errors by ROs only, errors by ROs and NLOs (RO/NLO), and errors by ROs and maintenance technicians (RO/MT). Figure 5.15 shows that the interaction between reactor operator and non-licensed operator has more influence on core melt frequency than errors committed by reactor operators alone (ROs only). Even though there are 21 RO/NL errors compared to 80 "ROs only" errors, the large impact shown by RO/NL errors is because most of these errors involve sequence-dependent recovery errors. As discussed earlier, this set of recovery errors occurs as multiple events in dominant cutsets of important accident sequences and the magnitude of their probability estimates is high. This implies that coordination between reactor operator and non-licensed personnel, especially during recovery actions in accident situations (e.g., REIA1, RESW12), is important in limiting risk. The interaction between the reactor operator and maintenance personnel has minimal effect on the core melt frequency. There are 26 RO/MT errors and their minimal effect is due to some restoration errors after test and maintenance work (e.g., EFTDPP1H, LP16MVMH). These restoration errors occur as single events in less important cutsets and the magnitude of their probability estimates is usually between 10^{-3} and 10^{-2}. Therefore, their influence on core melt frequency is not significant as the HEPs are varied.

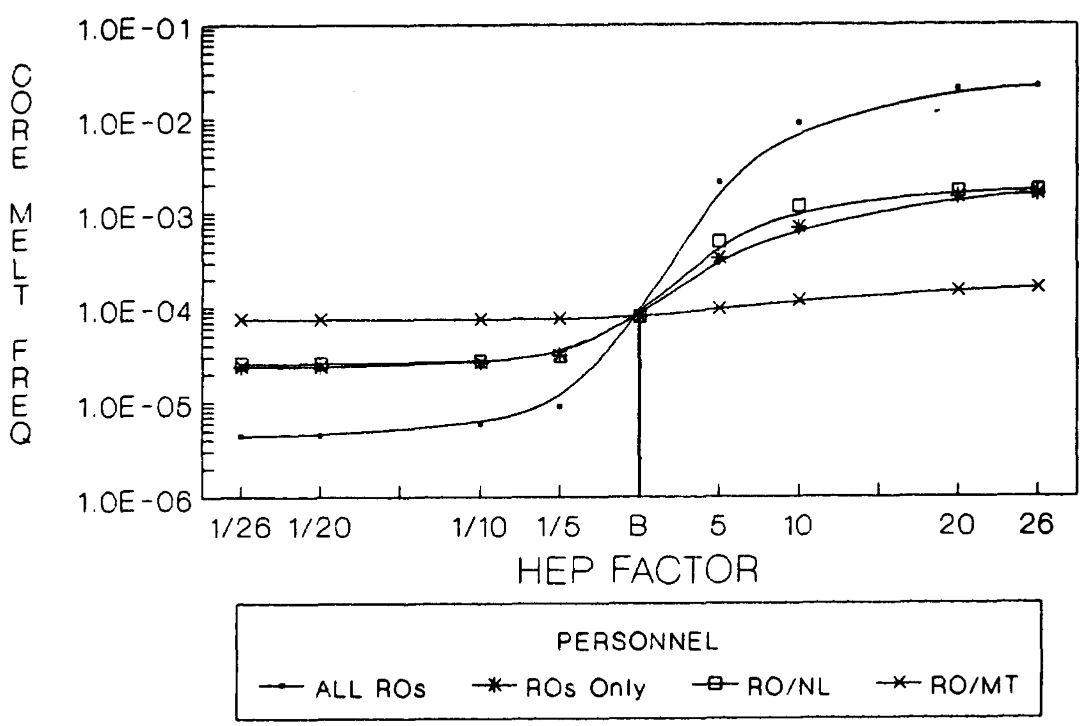


Figure 5.15. Sensitivity of core melt frequency to various categories of reactor operator interactions with other plant personnel

5.3.3.6 NRC Inspection Programs

In this subsection, the influence of NRC inspection programs for possibly controlling risk due to human errors is assessed. The categorizations of human errors for this sensitivity evaluation are not independent, i.e., a given error may be coded as affected by more than one NRC inspection program, such as operations and training. Figure 5.16 shows that HEs judged to be affected by inspections of operations programs (e.g., procedure reviews) and training programs are the most dominant. This funding corresponds with insights obtained from the sensitivity curves for types of plant activity, showing the importance of operations activities and from the sensitivity curves of during accident versus pre-accident errors showing the importance of RO and NLO actions during the accident time regime. This funding also agrees with an examination of the errors shown to be important in the cutsets (see Appendix F). Curves of NRC inspection programs for the dominant accident sequences (e.g., T_6BU) also look similar. There is a noticeable but not dominant effect from human errors that may be discovered during system walkdown inspections, that ensure equipment is properly lined up. These are pre-accident errors involving restoration from test and maintenance that show moderate sensitivity on the risk variation curves for plant activity.

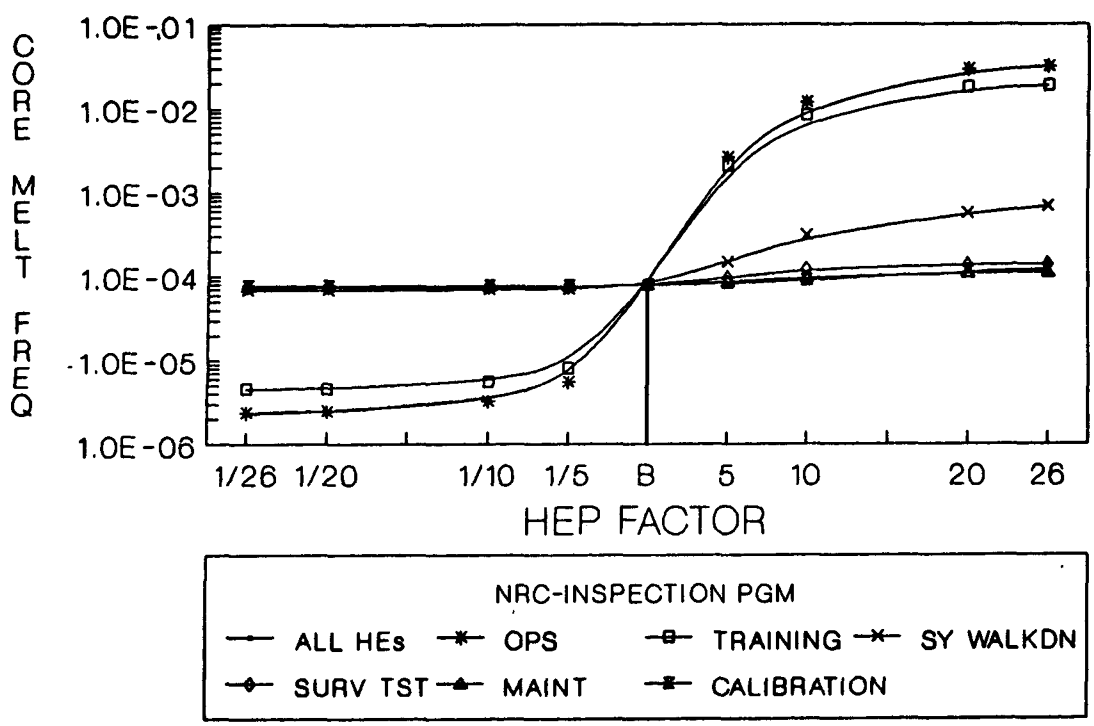


Figure 5.16. Influence of NRC inspection programs for possibly controlling risk due to human errors

5.3.4 Sensitivity of Dominant Accident Sequences to Human Errors

As discussed earlier in subsection 5.3.2.2, dominant accident sequences are sensitive to human errors and vary over seven orders of magnitude as HEPs are varied. In this study, the three most dominant accident sequences in the baseline risk model were selected to analyze the role of human errors at the accident sequence level. These three accident scenarios are: the loss of instrument air (T_6BU), loss of service water transient ($T_{12}BU$), and large-break loss of coolant accident (AX_a) sequences.

The T_6BU sequence involves a loss of instrument air, as an initiating event, or as a result of loss of offsite power, and it is responsible for 36% of the total core melt frequency in the base case. The $T_{12}BU$ sequence is characterized by failure of the low pressure service water (LPSW) system as an initiator, or failure of the 4.6-kV bus 3TC with other failures in the second LPSW pump. This sequence is responsible for 16% of the base case core melt frequency. The AX_a sequence is characterized by a large-LOCA initiating event, and it accounts for about 12% of the base case core melt frequency.

5.3.4.1 Loss of Instrument Air Transient (T_6BU)

When a loss of instrument air occurs, main feedwater (MFW) is unavailable because the air-operated control valves in the MFW lines to the steam generators fail "as is." The emergency feedwater (EFW) system becomes unavailable if the steam-driven pump is not available and air is not recovered. After the loss of MFW and EFW, the failure of operators to establish high pressure injection (HPI) cooling and to make feedwater available to the steam generators from the Standby Shutdown Facility (SSF), will result in core damage. Figure 5.17 shows that the ASF of T_6BU sequence is sensitive to both recovery and non-recovery errors when HEPs are increased. Also, the accident sequence risk is considerably reduced when recovery error HEPs are decreased. A primary reason for this behavior is that the recovery errors that impact risk sensitivity are sequence-dependent. The detailed analysis of T_6BU sequence cutsets in Appendix F shows that two sequence-dependent recovery errors, RESSFW30 (operator failure to initiate auxiliary service water from SSF within 30 minutes) and REIA1 (failure to recover IA in one hour), are the most significant contributors to the ASF sensitivity.

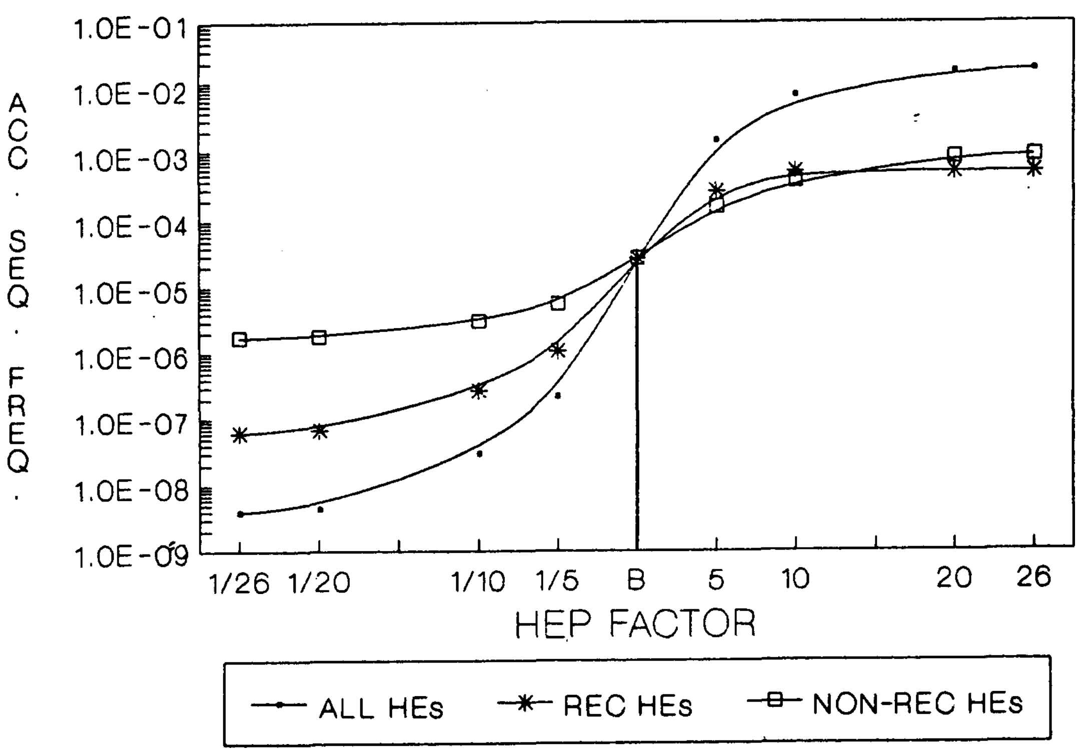


Figure 5.17. Sensitivity of ASF to recovery and non-recovery errors for T_6BU sequence

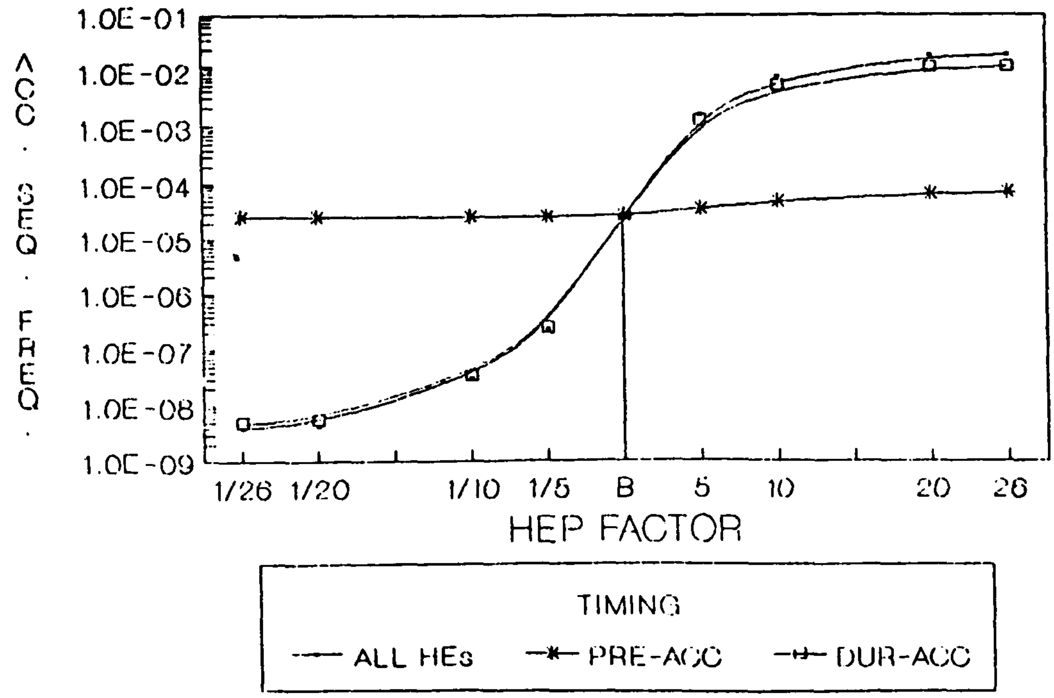


Figure 5.18. Sensitivity of ASF to pre-accident and during accident errors for T_6BU sequence

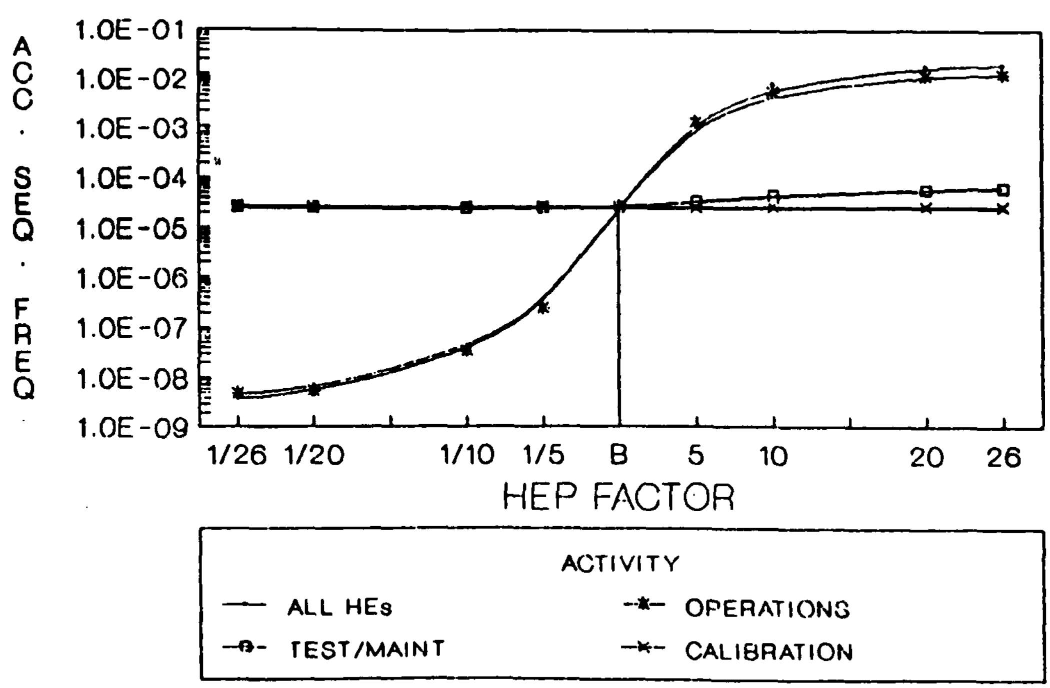


Figure 5.19. Sensitivity of ASF to plant activity-related errors for T_6BU sequence

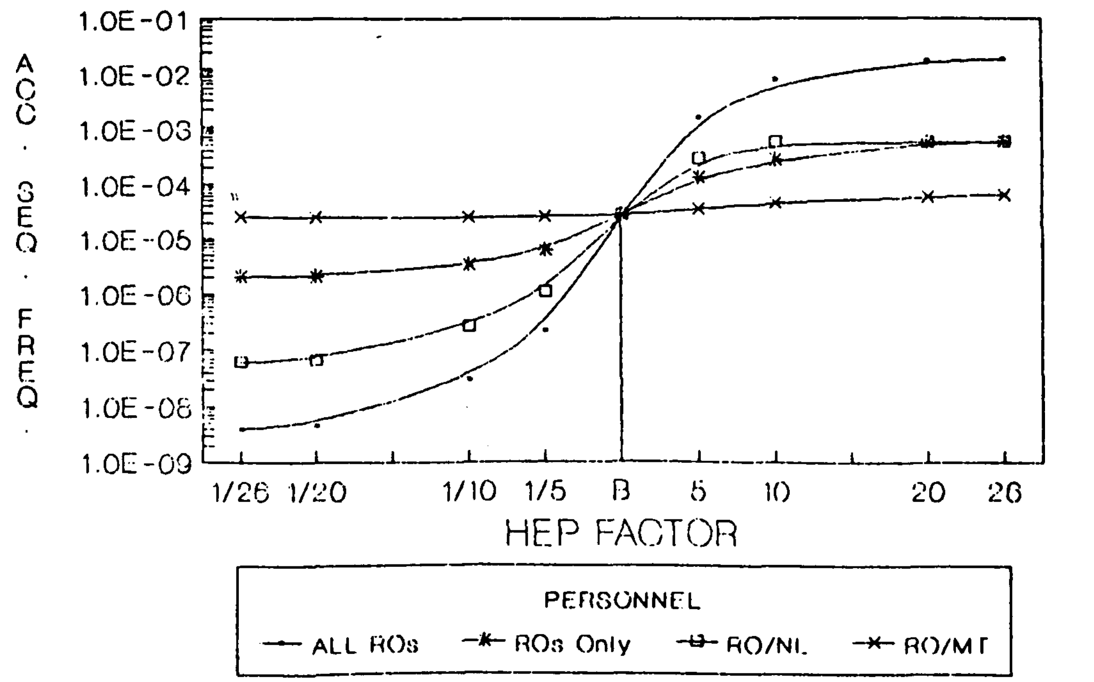


Figure 5.20. Sensitivity of ASF to categories of personnel interactions for T_6BU sequence

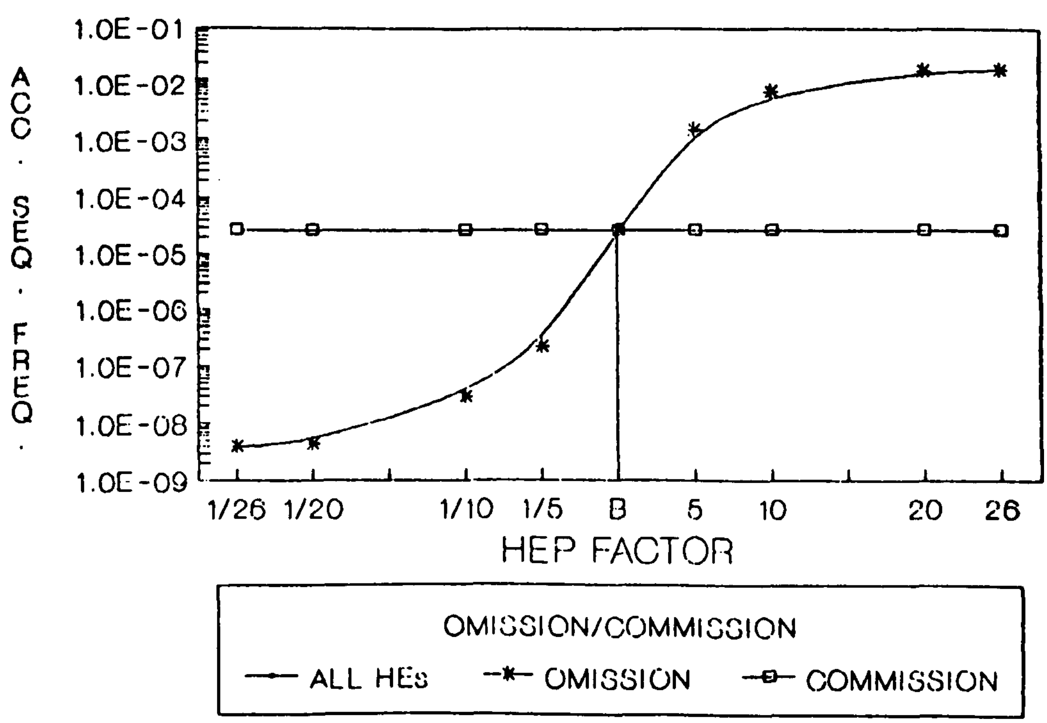


Figure 5.21. Sensitivity of ASF to omission/commission errors for T_6BU sequence

Figures 5.18 through 5.21 show the ASF sensitivity to various categories of human errors for the T_6BU sequence. The ASF of this accident sequence is dominantly sensitive to "during accident" human errors. These errors include recovery events such as REIA1 and RESSFW30, and operational errors such as UTHPIH (operator fails to achieve HPI cooling), and PLS1H (operator fails to reapply LC-3X1 following load shed). There is minimal sensitivity of ASF to pre-accident errors such as EFTDPP1H (TD EFW pump not restored after test or maintenance work), which have probability estimates on the order of 1 x 10^{-2}.

As expected, the ASF is highly sensitive to operations-related errors (Figure 5.19) because the significant "during accident" errors are also categorized as operation errors. Since the ASF of the T_6BU sequence is significantly affected by the sequence-dependent recovery errors, the sensitivity to RO/NL errors is observed (Figure 5.20). Therefore, accident recovery requires that actions by both ROs and NLOs be well coordinated to mitigate the accident risk level. Finally, the ASF is totally dominated by omission errors (Figure 5.21). Commission errors associated with test, maintenance, and calibration activities have little effect on the accident sequence. This is because all the dominant human errors are omission errors. Table 5.3 summarizes the categorization of seven dominant human errors for the T_6BU sequence which affects its ASF sensitivity. It should also be noted that several insights for this sequence are similar to those for overall CMF sensitivity, since this is the dominant sequence and thus has a large effect on overall CMF.

Table 5.3. Categorization of Seven Dominant Human Errors for T_6BU Sequence

Event	Timing	Personnel	Activity	Om/Comm
RESSFW30	During	RO/NL	Ops	Omission
REIA1	During	RO/NL	Ops	Omission
UTHPIH	During	RO	Ops	Omission
PLS1H	During	RO	Ops	Omission
HP2425MVH	Pre	RO/MT	T/M	Omission
LP28VVCH	Pre	RO/MT	T/M	Omission
EFTDPP1H	Pre	RO/MT	T/M	Omission

5.3.4.2 Loss of Service Water Transient ($T_{12}BU$)

In this accident sequence, a loss of the LPSW system causes failure of the HPI pumps due to interruption of cooling flow to the pump motor bearings. The reactor coolant pumps (RCPs) are tripped and the failure of HPI seal injection will result in a small reactor coolant system (RCS) leak with an inability to maintain make-up if the SSF seal injection is not actuated within 30 minutes. Figure 5.22 shows that the ASF of the $T_{12}BU$ is sensitive to recovery errors more than non-recovery errors when HEPs are varied. The detailed analysis of $T_{12}BU$ sequence cutsets (Appendix F) shows that the ASF is highly sensitive to a sequence-dependent recovery error, RESSFSI (operator failure to initiate SSF seal injection within 30 minutes following a loss of normal HPI seal injection). Another recovery error, RESW12 (operator failure to recover LPSW from another source before failure of HPI pumps), which is

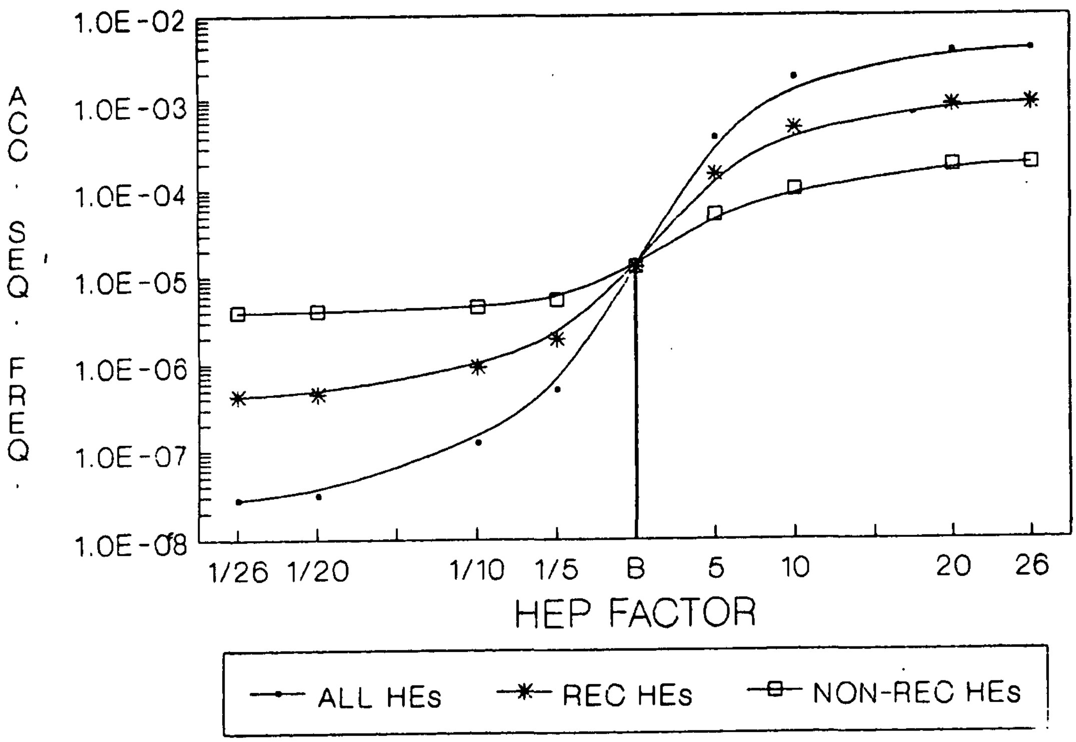


Figure 5.22. Sensitivity of ASF to recovery and non-recovery errors for $T_{12}BU$ sequence

cutset-dependent, is identified from the cutset analysis to be a significant contributor to ASF sensitivity. The multiple effect of the two recovery errors in the dominant cutsets, i.e., their combined occurrence in 8 of 21 cutsets, drives the variation of ASF sensitivity.

The ASF sensitivities to various categories of human errors for the $T_{12}BU$ sequence are similar to those observed for the T_6BU sequence. Therefore, the risk variation curves will not be reproduced here. Also, interpretation of the risk sensitivity curves is similar because both T_6BU and $T_{12}BU$ sequences are initiated by transients, even though the human errors may be unique to a particular sequence.

5.3.4.3 Large-break LOCA (AX_a)

This accident sequence is characterized by a large-break LOCA initiating event, with successful injection but failure of low-pressure recirculation. The low-pressure recirculation fails because high flow develops during the recirculation phase, and the operators fail to throttle the flow. Following this failure to throttle, pump cavitation and failure can occur.

Figure 5.23 shows that the ASF of the AX_a sequences is not sensitive to recovery errors. Instead, the ASF is influenced by other operational errors that occur during the course of the accident. This is not surprising because the large-break LOCA sequences are design-basis accidents with systems better designed to cope with them than some transient sequences. A review of the AXa sequence cutsets (Appendix F) shows that there are eight dominant human

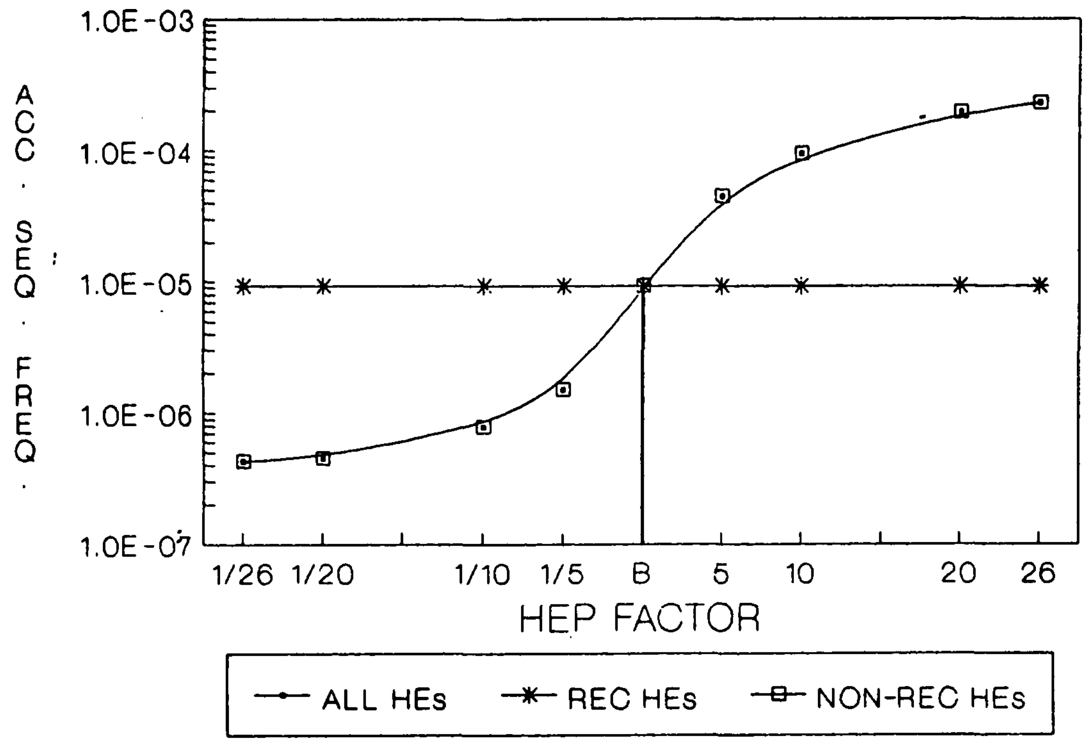


Figure 5.23. Sensitivity of ASF to recovery and non-recovery errors for AX_a sequence

errors that influence the accident sequence frequency. The three most dominant human errors are: LPFLOWH (operator fails to throttle high flow), XALPRH (operator fails to achieve LPR within 30 minutes), and LPTHROTTLE (operator fails to throttle LPR flow).

All of the dominant human errors in this sequence are categorized as "during accident" and omission errors. Therefore, the shape of sensitivity curves is similar to those obtained for the T_6BU sequence.

5.3.5 Sensitivity Evaluation of Special Situations

5.3.5.1 General Discussion

In the following subsections, the sensitivity evaluations address the role of human errors in certain plant situations such as (1) recovery efforts during accident conditions, and (2) "routine" human actions which may affect the consequence of an accident. Human errors of recovery identify the ability of operating staff to restore an interrupted function in response to accident conditions. By performing a sensitivity analysis with respect to these errors, the operator performance during abnormal plant conditions can be assessed. This assessment evaluates how a well-prepared and managed operating crew can reduce the risk, or how an error-prone operating crew can exacerbate the risk from abnormal occurrences.

Errors in "routine" human actions refer to all possible human errors during normal plant operation which can affect the risk level. These errors are essentially pre-accident errors and include restoration from test or maintenance, calibration, and pre-accident operational errors. The objective of the sensitivity evaluation with respect to these errors is to identify improvement/deterioration of risk level due to changes in performance level of routine human actions. In addition, the sensitivity analysis can help identify human errors that may impact the consequences of an accident.

5.3.5.2 Errors of Recovery

5.3.5.2.1 Definition

In this study, the term "recovery action" refers to a manual action taken to restore an interrupted function, usually by initiating alternative equipment, or sometimes by repairing or restarting the equipment that has failed. These actions are usually taken outside the control room requiring coordination between ROs and NLOs. Non-recovery errors, or operational errors, are those actions usually associated with following procedures appropriately.

5.3.5.2.2 Impact of Recovery Errors on Core Melt Frequency

The impact of recovery during accident conditions is shown by the sensitivity curves plotted on Figure 5.24. The plotted risk values are obtained when the HEPs of all during accident errors excluding recovery errors are varied simultaneously by a multiplicative factor, and with recovery HEPs set at the noted fixed value. When all recovery error probabilities are assumed to be 1.0, (representing failure of recovery actions), the core melt frequency is increased by more than an order of magnitude. The baseline core melt frequency becomes 3.80E-3 under this assumption of no recovery. If recovery error probabilities are assumed to be 1×10^{-3} to represent success, the core melt frequency is reduced by a factor of 3.0. The baseline core melt frequency is 2.41E-5 when successful recovery (.001) is assumed. If all recovery errors are assumed to be "perfect" (probability equal to zero), the core melt frequency only decreases to 2.39E-5.

The large potential for risk increase and relatively smaller potential for risk reduction, are collectively due to HEPs of some recovery actions being assigned low base values. Recovery actions with low HEPs are those expected over a longer time interval after accident initiation. This result shows that the ability of operating staff to recover from accident conditions significantly influences the core melt frequency.

5.3.5.2.3 Impact of recovery errors on accident sequence frequency

The impact of recovery during the occurrence of the most dominant accident sequence, viz, T_6BU sequence, is shown on Figure 5.25. The sensitivity curves are plotted from risk values obtained when the HEPs of all during accident errors excluding recovery errors are varied simultaneously. The values of recovery error probabilities are fixed as indicated. For this particular sequence, the baseline accident frequency is increased to 5.42E-4 when no recovery is assumed. If successful recovery (.001) is assumed, the accident frequency is reduced by five orders of magnitude to 5.42E-10.

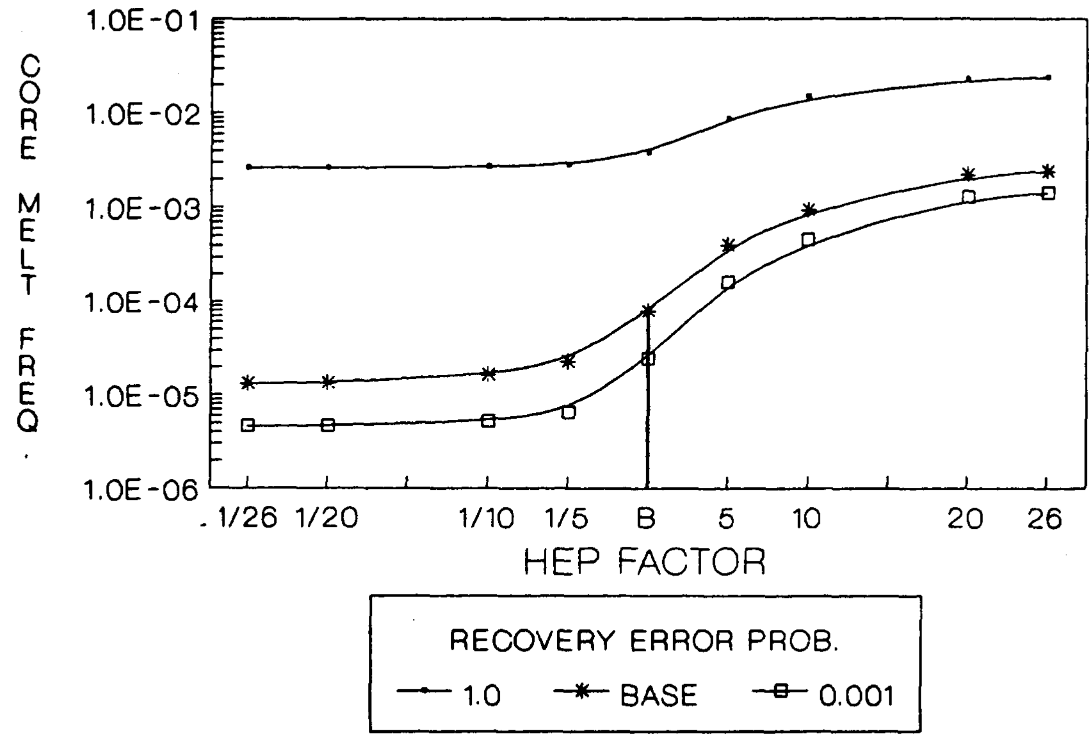


Figure 5.24. Impact of recovery on core melt frequency for during accident errors

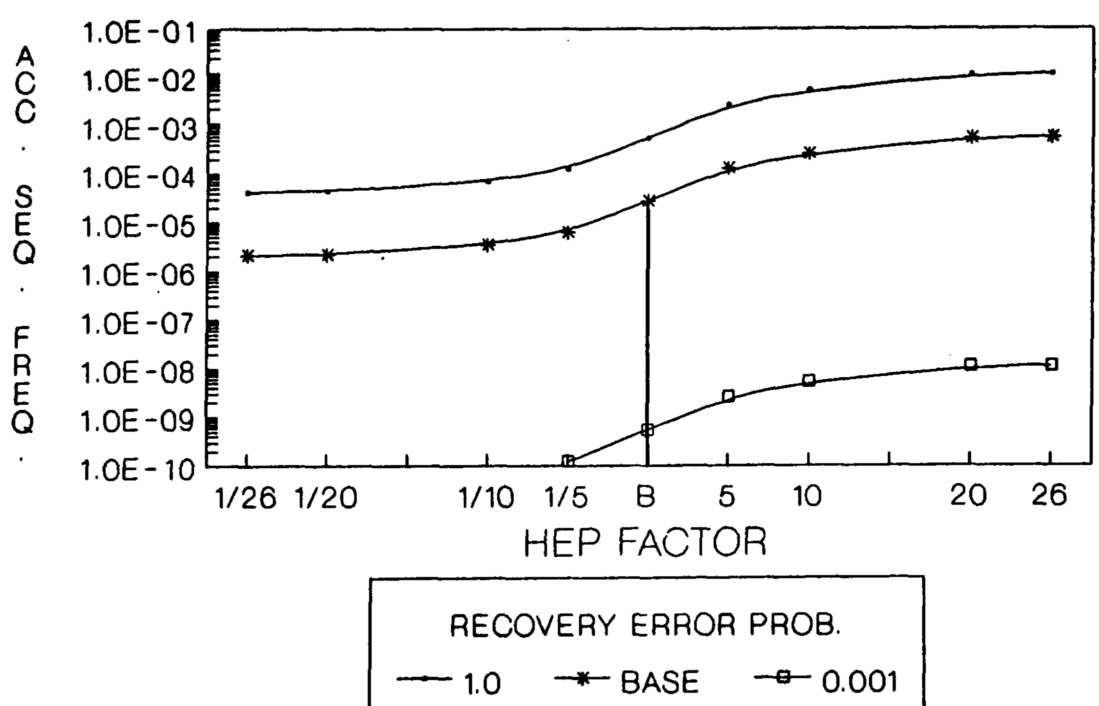


Figure 5.25. Impact of recovery on accident sequence frequency of T_6BU sequence for during accident errors

The large potential for risk reduction is due to HEPs of recovery actions, especially the sequence-dependent recovery events modeled in the dominant cutsets, having high initial values. This substantial reduction in risk characterizes the importance of successful recovery actins for the management of accident risk in a sequence which is dominated by multiple human errors of recovery. It should be noted that such recovery error probabilities may be very difficult, if not impossible, to obtain.

5.3.5.2.4 Operational Errors Relative to Recovery Errors

In addition to the ability to recover from abnormal plant conditions, other operator actions such as dynamic or latent human errors contribute to plant risk. The impact of operational errors relative to errors of recovery is analyzed by considering the sensitivity curves derived for these "non-recovery" or operational errors at defined recovery error probabilities. The curves are plotted from risk values obtained by varying the HEPs of all errors in the database excluding recovery errors simultaneously by a constant factor, and keeping the recovery HEPs at a fixed value. Figure 5.26 shows changes in core melt frequency due to variation of HEPs for operational errors at assumed recovery error probabilities of 1.0, 0.5, 0.1, and 0.001.

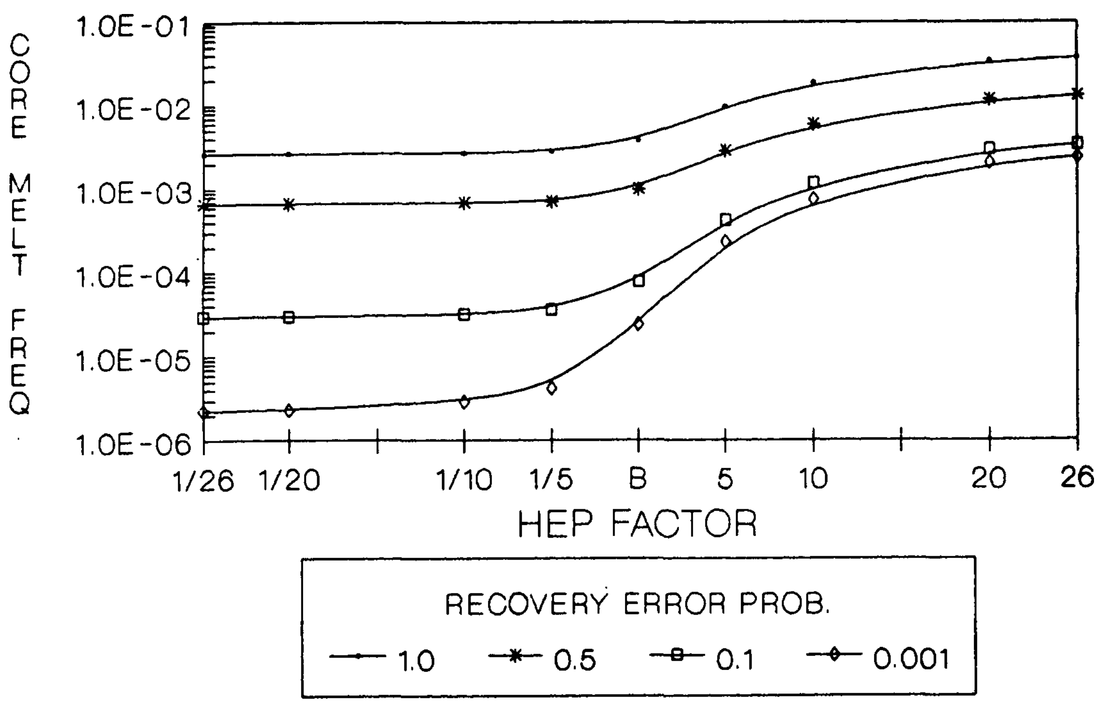


Figure 5.26. Sensitivity of operational errors with respect to recovery

When no credit is taken for recovery actions, the core melt frequency increases from 2.6E-3 to 3.8E-2, i.e., by an order of magnitude, over a range of HEP factors exemplifying good human performance to degraded performance. This trend is largely followed by the sensitivity curve for operational errors at a recovery error probability of 0.5. If credit is taken for all recovery actions, core melt frequency decreases by two to three orders of magnitude for variation of constant factors on operational error probabilities at recovery error probabilities of 0.1 and 1.0×10^{-3}. At the assumed recovery error probabilities of 0.1, the core melt frequency decreases from 3.4E-3 to 3.0E-5; and when the recovery error probabilities are fixed at 1×10^{-3}, the core melt frequency decreases from 2.4E-3 to 2.3E-6. This finding shows that operational errors have lesser impact on risk level when potential recovery from accident conditions is degraded, i.e., when recovery error probabilities are high. Conversely, as recovery error probabilities improve (decrease), the impact of operational errors increases.

5.3.5.3 "Routine" Human Actions

5.3.5.3.1 Definition

"Routine" human actions in a nuclear power plant refers to all actions performed to operate the plant. Errors in "routine" human actions during normal plant operation are essentially pre-accident errors and include recovery from test or maintenance, calibration and pre-accident operational errors. Since risk levels have been shown to have marked sensitivity to human errors committed during accident, the sensitivity evaluation of "routine" human activity were performed with HEPs for during accident human errors set at their lower bound values so that the sensitivity impact of pre-accident errors are not masked.

5.3.5.3.2 Impact of Pre-Accident Human Errors

As represented in Figure 5.27, pre-accident omission errors have more influence on core melt frequency than pre-accident commission errors. The risk impact of omission errors prior to accident initiation is largely due to unavailability contributions from valves left unrestored after test or maintenance (e.g., LP4142VVH, SWBHPIH, SW3BFPH). These errors are restoration errors after test or maintenance work, or "operations" errors, and their probability estimates range from 1×10^{-3} to 1×10^{-1}. Pre-accident commission errors, mostly calibration errors that cause unavailability of control signals to safety equipment, have little effect on core melt frequency because the magnitude of HEPs associated with these errors is usually below 1×10^{-3}.

Figure 5.28 shows the change of core melt frequency when HEPs of pre-accident errors for different types of plant activity are varied individually by activity. The curve labeled "Pre-Acc HEs" is plotted from risk values obtained when HEPs of pre-accident errors for all activities are varied simultaneously. Figure 5.28 also shows that core melt frequency is sensitive to both operations and test or maintenance errors occurring before the plant upset. As performance improves significantly, i.e., as HEPs decrease to lower bound values, test and maintenance errors have the most influence in reducing core melt frequency. On the other hand, as performance becomes degraded, i.e., as HEPs increase to upper bound values, core melt frequency is most sensitive to operational errors. The greater impact of pre-accident operational errors when performance degrades significantly is because operational errors are most important and calibration errors are least important. When performance improves, there is not much change in CMF, but restoration from test and maintenance errors have the largest effect. A primary reason is that the error factors of HEPs associated with pre-accident operational errors are

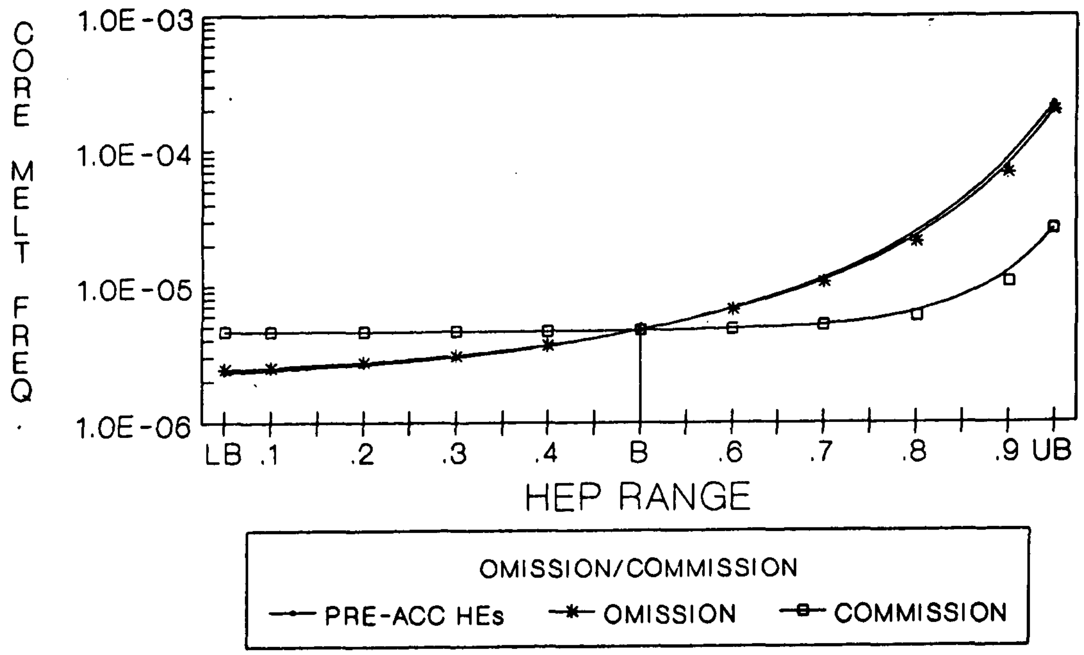


Figure 5.27. Sensitivity of core melt frequency to pre-accident errors of omission and commission

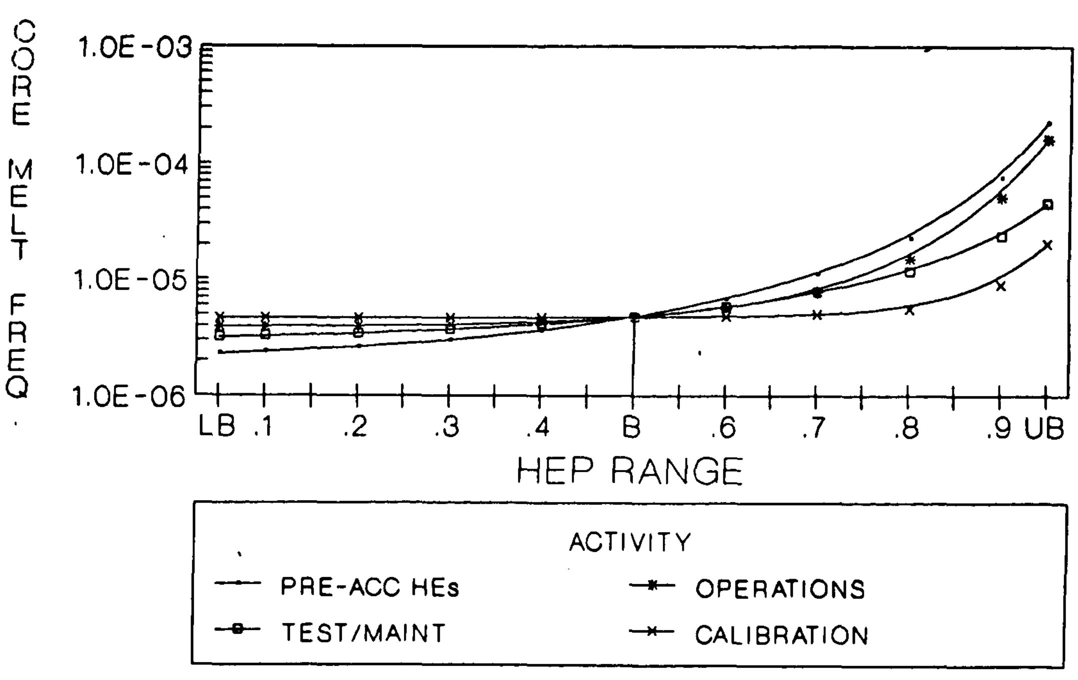


Figure 5.28. Sensitivity of core melt frequency to pre-accident human errors for various types of plant activity

generally two times or more higher than those for test and maintenance errors and calibration errors. Also, the number of calibration errors in the cutsets is much less than those of pre-accident operational errors.

To assess risk sensitivity to the performance of various personnel during normal plant operation, the changes in core melt frequency are plotted when HEPs of pre-accident errors are varied simultaneously for different types of personnel. As shown in Figure 5.29, errors committed by non-licensed operators during normal plant operation have the most influence on core melt frequency. Errors due to reactor operators before an accident state have a moderate influence, while errors committed by instrumentation and calibration

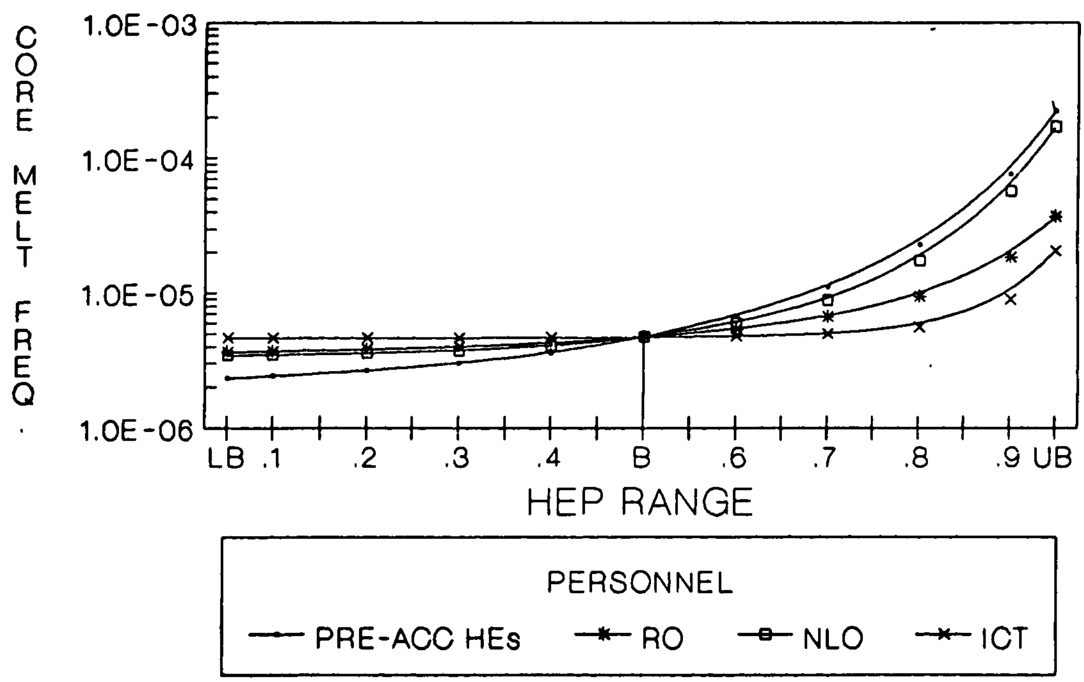


Figure 5.29. Sensitivity of core melt frequency to pre-accident errors for different types of personnel

technicians have a minimal effect on core melt frequency. The marked sensitivity of core melt frequency to errors by non-licensed operators is largely due to restoration errors after test or maintenance work, such as valves left open (e.g., LP4142VVH, LWD99103VVH) or component trains unrestored (e.g., MF3C1HXH, MF3D1HXH). These restoration errors occur in the more dominant cutsets and therefore, their contributions have a greater impact on the risk level. The moderate influence on plant risk from reactor operator errors during pre-accident state is due to the operator failure to open valves (e.g., HPCROSSH, RC417VCH), or failure to restore equipment to operable status (e.g., EFTDPP1H, LPAPPMH, ACDIATMIVH) and inadvertent closing of valves (e.g., LP5MVH, LP8MHV). These reactor operator errors occur in the less dominant cutsets and therefore, their contributions have a moderate effect o the risk level. The minimal impact of errors due to instrumentation and calibration technicians on pre-accident risk is due to the magnitude of HEPs associated with miscalibration errors (e.g., I476SUH, I8413SUH) and errors in setting manual/automatic control stations in manual mode with high output signal (e.g., ISS14ASSH, IIC35ASSH). The magnitude of these HEPs is usually on the order of 1×10^{-3}.

5.3.5.3.3 Risk-Significant Human Errors During Pre-Accident Regime

Risk-significant human errors during normal plant operation can be identified by performing single-event or pairwise importance analyses on a selected category of errors that characterize operator performance on a specific aspect of plant operations. Errors committed prior to plant upset are ranked by single-event importance analysis according to unnormalized Fussell-Vesely and Birnbaum importance measures. Table 5.4 lists 20 pre-accident human errors found to be most important in terms of the Fussell-Vesely importance measure, and Table 5.5 ranks these errors in terms of both importance measures.

Significant human errors in the pre-accident state are mostly errors in restoration of valves (e.g., LWD99103VVH, LP40VVH, LP4142VVH) or equipment to operable status (e.g., EFTDPP1H) after test or maintenance acts (Tables 5.4 and 5.5). Inadvertent closure of suction valves (e.g., LP5MVH, LP8MVH) that renders a major system flowpath or component train unavailable also is important, according to the Fussell-Vesely measure. Miscalibration errors in 15V dc power supply to engineered safety actuation channels are also found to be significant because of the vulnerability of ESAS to power supply failure: a single power supply failure renders half of the channels unavailable, while a double power supply failure will cause all ES channels to fail. In general, pre-accident human errors with small unavailability contributions to risk parameter probabilities (e.g., LWD99103VVH, LP40VVH) are ranked as most important according to the Fussell-Vesely measure.

5.4 Conclusions

The sensitivity evaluations conducted in this study help to identify the role of human errors in various aspects of plant risk. In general, plant risk parameters such as the core melt frequency and accident sequence frequencies have been shown to be quite sensitive to variations in human error probabilities, and specific categories of human errors. The risk variations over several orders of magnitude are largely due to several significant human errors that occur as multiple events in the dominant cutsets. For instance, sequence-dependent recovery errors in combination with other operational errors effect the large increase in risk when the HEPs are increased. Also, these errors are usually "during accident" errors and their probability estimates range from 1×10^{-1} to 1.0.

Because the variations of risk levels are controlled by human error influences, the sensitivity analyses show the potential to identify attributes of human performance for risk reduction and oversight control. The insights gleaned from the various sensitivity evaluations and major findings are discussed in summary fashion in the Overview Summary.

Table 5.4. List of Significant Pre-Accident Human errors in Terms of Fussell-Vesely Importance Ranking

NO.	HUMAN ERROR	HEP	DESCRIPTION
1	LWD99103VVH	6.0E-4	Drain valves not restored after test/maintenance.
2	LP40VVH	1.0E-3	Valve 3LP-40 left open.
3	LP4142VVH	1.0E-1	Both valves 3LP-41 and 3LP-42 left open.
4	HP2425MVH	5.0E-5	MOVs 3HP-24 and 3HP-25 left unavailable.
5	EFTDPP1H	1.0E-2	Turbine-driven EFW pump not restored after test or maintenance.
6	SM7778CMH	3.0E-3	Valves LPSW-77 and -78 left in wrong position.
7	LP28VVCH	2.8E-5	BWST valve left closed.
8	SW527VVH	8.5E-4	Valve LPSW-527 left closed.
9	CCW87VVH	8.5E-4	CCW-87 left closed.
10	EPS110000H	1.0E-3	+15V dc power supply 1-1 miscalibrated with low or no voltage output.
11	EPS120000H	1.0E-3	-15V dc power supply 2-1 miscalibrated with low or no voltage output.
12	EPS210000H	1.0E-3	+15V dc power supply 2-1 miscalibrated with low or no voltage output.
13	EPS220000H	1.0E-3	-15V dc power supply 2-2 miscalibrated with low or no voltage output.
14	RC417VCH	8.3E-1	PORV block valve left closed to inactivate PORV.
15	CW156MVH	1.0E-3	MOV 3C-156 not restored after test or maintenance.
16	EF88VVH	1.0E-3	MV 3FDW-88 not restored (closed) after test verification by train operator.
17	LP5MVH	1.1E-3	Valve 3LP-5 closed inadvertently.
18	LP8MVH	1.1E-3	Valve 3LP-8 closed inadvertently.
19	LP16MVMH	1.8E-3	Failure to restore valve 3LP-16 after test or maintenance.
20	LP15MVMH	1.8E-3	Failure to restore valve 3LP-15 after test or maintenance.

Table 5.5. Ranking of Significant Pre-Accident Human Errors in Terms of Fussell-Vesely and Birnbaum Importance Measures

NO.	HUMAN ERROR	UNNORMALIZED FUSSELL-VESELY IMPORTANCE	BIRNBAUM IMPORTANCE
1	LWD99103VVH	1.88E-6	3.13E-3
2	LP40VVH	1.58E-6	1.58E-3
3	LP4142VVH	1.37E-6	1.37E-7
4	HP2425MVH	1.35E-6	2.71E-2
5	EFTDPP1H	1.28E-6	1.28E-4
6	SM7778CMH	1.18E-6	3.94E-4
7	LP28VVCH	7.84E-7	2.80E-2
8	SW527VVH	1.89E-7	2.22E-4
9	CCW87VVH	1.89E-7	2.22E-4
10	EPS110000H	1.63E-7	1.63E-4
11	EPS120000H	1.63E-7	1.63E-4
12	EPS210000H	1.59E-7	1.59E-4
13	EPS220000H	1.59E-7	1.59E-4
14	RC417VCH	1.35E-7	1.63E-7
15	CW156MVH	1.28E-7	1.28E-4
16	EF88VVH	1.28E-7	1.28E-4
17	LP5MVH	1.17E-7	1.06E-4
18	LP8MVH	1.11E-7	1.01E-4
19	LP16MVMH	6.23E-8	3.46E-5
20	LP15MVMH	5.25E-8	2.91E-5

6. COMPARISON WITH NUREG/CR-1879

6.1 Introduction

In this chapter, the sensitivity results obtained in the Oconee study are compared with those obtained previously for the Surry plant in NUREG/CR-1879, "Sensitivity of Risk Parameters to Human Errors in Reactor Safety Study for a PWR." The objectives were: (a) to identify any common insights in the two studies, (b) to identify any differences and the reasoning behind such differences, (c) to identify any new insights derived from this study, and finally, (d) to identify any generic implications of these two studies.

The comparisons are made at a broad level to identify the differences or the commmonalities, and the reasonings behind them. In many cases, there is more than one reason, and all are discussed. However, it was not possible to delineate the relative contribution of each reason. For example, if both plant design features and the modeling differences are assessed to be causes, we could not delineate the portions of the difference due to each separately.

6.2 Justification and Limitation of Comparing Sensitivity Results from Different Plant-Specific PRAs

A comparison of this study which used methodology similar to that presented in NUREG/CR-1879 for the Surry plant has considerable practical significance for understanding the role of human errors in plant risk. The justification behind the comparisons is the similarities in these two studies which are summarized below:

a) In both evaluations, the output parameters, i.e., the risk parameters whose sensitivities are being assessed are the same. Namely, core melt frequency and accident sequence frequencies are analyzed for their sensitivity to human errors.

b) In both studies, the input parameters are also the same, i.e., the human errors that appear in the accident sequence models are varied to observe the changes in the risk parameters.

c) The categorization of the human errors followed similar structures, even though more categories were studied for the Oconee plant. The various categories studied in the Surry plant are all included in the Oconee study. There are differences in the actual application of the categorization schemes in these studies, but on the broad level they have similar characteristics.

d) In both cases, plant-specific PRAs with a similar modeling approach of event/fault trees are used.

Along with strong similarities, significant differences between these studies can be identified which also should be taken into consideration in the comparison. Ideally, if the modeling approaches and methodologies in the respective studies are the same, then the results can be directly compared to understand one plant's sensitivity to human errors as opposed to the other.

However, direct comparison is not possible due to differences in human error modeling in these PRAs. The major differences that have implications on the sensitivity results are presented below.

a) Inclusion of a Larger Number of Human Errors in Oconee PRA

The Oconee PRA carried out a very extensive analysis of human errors and the approximately 223 errors that remained in the accident sequence models after trucation was a factor of two larger than that included in Surry evaluation (approximately 110). The larger number of significant human errors in the Oconee PRA could be due to the features in the plant, but based on the review of the respective human reliability analyses, it is clear that a large portion is the result of more thorough analysis.

b) Extended Treatment of During-Accident Errors in Oconee PRA

The Oconee PRA included in reasonable detail the role of the operators during an accident, i.e., following an accident initiating event. It modeled operator failure to perform the required actions, important inadvertent actions by the operating crew and also, the intentional defeating of the function of a system due to misdiagnosis. These errors are largely included in the event trees, whereas the pre-initiating event errors are included in the system fault trees. In the Surry PRA detailed human error modeling was performed in the system fault tree, but no human errors were considered in the event trees. Overall, the Oconee PRA included about 101 (45% of the total human errors) and the Surry PRA contained about 35 (32% of the total) during-accident human errors.

c) Incorporation of Recovery Errors in Oconee PRA

Current PRAs, including the Oconee PRA, take into account recovery actions which are manual actions taken to restore an interrupted function and are sometimes not called for by procedures. The Surry PRA, used in NUREG/CR-1879 study, did not account for these type of actions.

The treatment of recovery errors in the PRAs had an interesting effect on the sensitivity evaluations. These errors are added on as an additional error in the combination of events that result in core damage, and hence, increase the number of human errors that appear in one "cutset" for the core damage equation.In the sensitivity evaluation, when the human errors are changed together, multiple human error probabilities are changed in one term resulting in increased sensitivity. Appendix F provides additional discussion and examples of the importance of recovery errors.

d) Reduction in the Contribution of Hardware Failure

Another important factor is that since the Surry PRA (WASH-1400) which identified dominant contributors to hardware failure, significant efforts were concentrated in reducing these failures. The results of these efforts are reflected in the reduced contribution of hardware failures in recent PRAs. At the same time, significant efforts were undertaken to understand the human role, resulting in better treatment of human errors which is discussed above.

6.3 Comparison of Sensitivity Results of Surry and Oconee Plants

6.3.1 Comparison of CMF Sensitivity to Human Errors

Figures 6.1 and 6.2, respectively, present the CMF sensitivity to human errors in the Oconee and Surry plants. The curves were obtained in a similar manner, i.e., the core melt frequency was obtained by increasing or decreasing the human error probabilities from their base case values by multiplicative factors. The general behavior of the curves are the same, i.e., CMF increases when HEPS are increased, and decreases when HEPs are decreased, but there are characteristic differences that are discussed below.

a) Stronger sensitivity of CMF to human errors in the Oconee PRA

Comparison of the two curves show stronger sensitivity of Oconee CMF to human errors compared to the Surry plant, which can be explained by the following features of the curves. First, over the range of HEP variation (comparing a factor of 26 increase/decrease in both cases) the change in CMF for Oconee is higher than that observed for the Surry plant in NUREG/CR-1879. In Oconee, four orders of magnitude variation is observed as opposed to less than two orders of magnitude variation in the Surry plant. Second, the delta change in CMF divided by the factor change in HEPs is higher for Oconee. A factor of 10 change in HEPs in the Oconee plant resulted in a factor of 150 change in CMF, whereas only a factor of 6 change in Surry CMF was observed for similar changes in the HEPs.

This stronger sensitivity is attributable to (a) the design features of the Oconee plant, and (b) the more extensive modeling of human errors in the Oconee plant. Regarding (a), Oconee is a B+W reactor with a high systems dependence on the Instrument System. The dominant sequence in the Oconee PRA is Loss of Instrument Air (T6BU), which is quite sensitive to human errors. Regarding (b), a stronger sensitivity results from larger numbers of risk-significant human errors and the associated high error probabilities. The inclusion of a larger number of human errors in a PRA is manifest in the accident sequence model in two ways. One, more terms in the accident sequence expression contain human errors, and second, each term in the expression contains more human errors. When all error probabilities are changed together, both phenomenon result in stronger sensitivity. The associated error probabilities also can result in stronger sensitivity because of high values. When human error probabilities are high, the terms containing the human errors become dominating, thus controlling the sensitivity results. It can be argued that the larger number of human errors and the high error probabilities in the Oconee PRA are due to the design features in the plant. At the same time, the modeling of human errors in the PRA are significantly different, which also influences the sensitivity curves. Particularly, two aspects of the human reliability analysis in Oconee PRA are significantly different from the Surry WASH-1400 PRA which contribute to the stronger Oconee CMF sensitivity to human errors. First, improved modeling of operator's role during an accident increased the number of human errors. Second, without actual evidence, these errors were assigned high estimates resulting in stronger sensitivity. This results in these errors

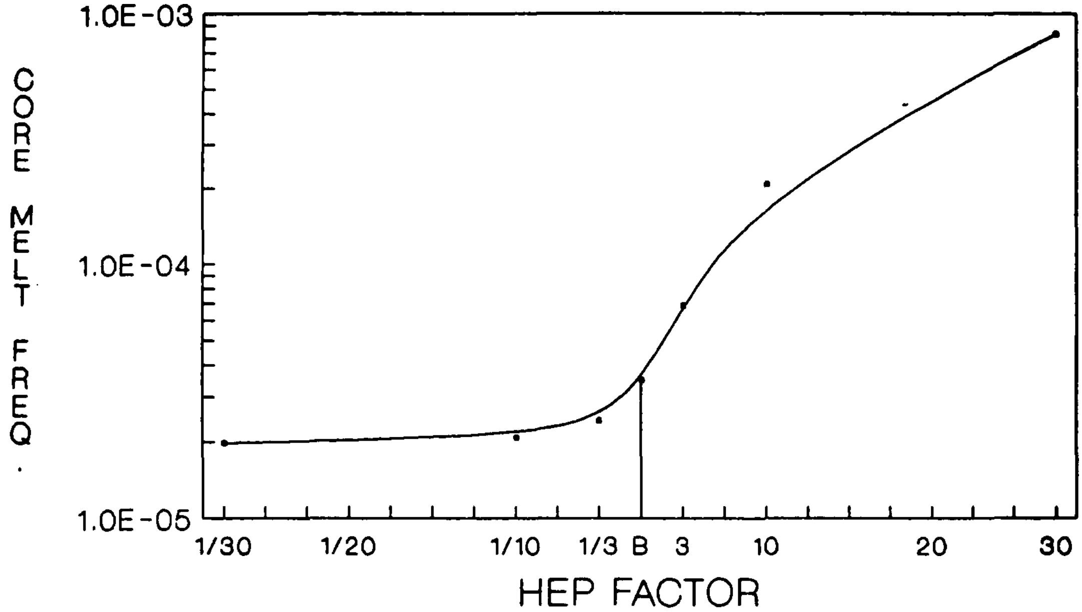


Figure 6.1. Sensitivity of core melt frequency in the Surry plant (Note the difference in scale from Figure 6.2)

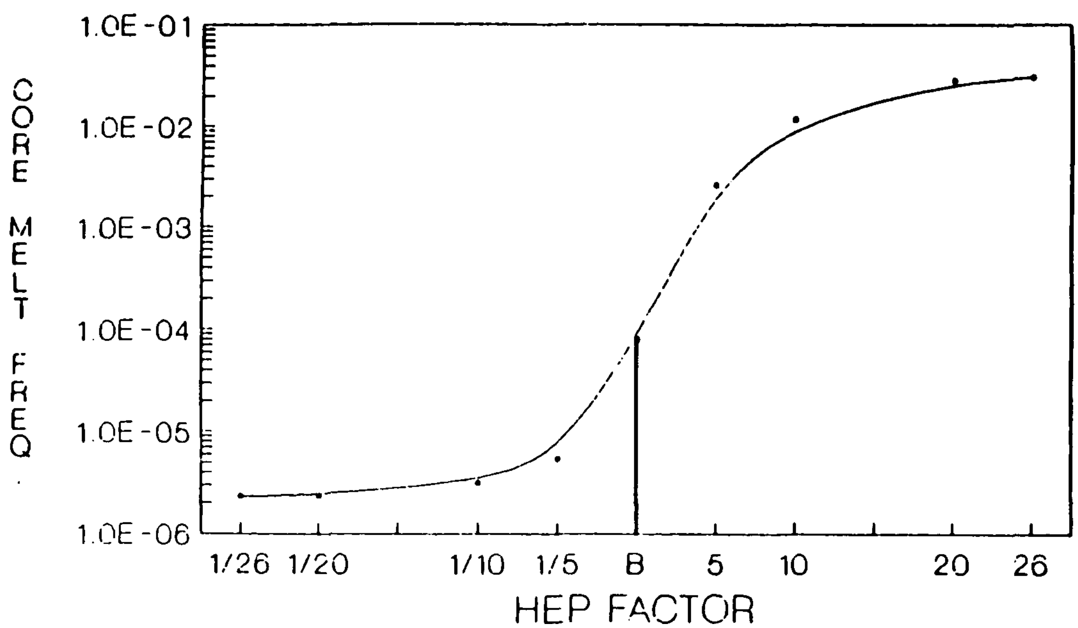


Figure 6.2. Sensitivity of core melt frequency in the Oconee plant

appearing in dominant cutsets and hence, creates a strong sensitivity to their variation. This approach may be closer to reality, but was not the modeling approach in the Surry PRA due to the less advanced state of PRA techniques. The other aspect relates to the incorporation of recovery errors which reduced the base case core melt frequency, but also introduced another human error in the combination of events that result in core melt. Treating the recovery errors as pure human errors, additional multiplicative factors are introduced in the CMF calculation when the human error probabilities are changed together, resulting in a much faster increase in the Oconee CMF.

We could not ascertain how much of the differences in core melt sensitivity curves between the two plants is due to the human error modeling differences in core melt sensitivity curves between the two plants is due to the human error modeling differences. Nevertheless, even if all recovery errors are removed, the Oconee CMF still shows stronger sensitivity to human errors.

b) Greater potential for risk reduction with improved human performance

In a sensitivity evaluation, when a group of basic event probabilities are decreased, saturation of the risk parameters occurs. For human error sensitivity evaluations, this saturation takes place when the terms containing the human errors are no longer dominant, or in other words, when terms related to hardware failure become dominant. By observing the saturation behavior of the curve for lower HEPs, i.e., improved human performance, one can make inferences about the relative dominance of hardware and human failures in the plant.

The Oconee CMF sensitivity curve saturates at much lower HEPs than the curve for the Surry plant, signifying stronger dominance of human errors in the Oconee plant. In the Surry plant, the CMF saturated much faster (at higher HEPs) showing the relative dominance of hardware failures. This later saturation also indicates that substantial improvement in plant risk is achievable through reduction in HEPs (improved human performance). Since a significant portion of this difference between the studies is due to improved modeling, which should apply to all plants, the potential for risk reduction with improved human performance is likely not Oconee specific.

c) Significant variation in CMF around the base HEPs in Oconee

The Oconee CMF showed much stronger sensitivity to human errors around the base value compared to the Surry results, signifying that considerable improvement (lowering) of core melt frequency can be achieved by improving (lowering) HEPs. At the same time, increased HEPs showed a large factor increase in CMF. The Surry CMF showed only improvement by about a factor of two for decrease in HEPs, and the factor increase in CMF due to change in HEPs was relatively small.

These differences in the curves are primarily due to: (a) the dominance of human errors in Oconee, (b) multiple human errors in the combination of events that contribute to the CMF, and (c) large base probabilities assigned to human errors, which places them in dominant cutsets. As discussed, these conditions are attributable both to the plant design features and to the modeling of human errors.

6.3.2. Comparison of Dominance of Categories of Human Errors

a) Timing of Error (During-Accident versus Pre-Accident Errors)

The Oconee CMF shows stronger sensitivity to during-accident errors than pre-accident errors, whereas the Surry CMF shows the opposite. This difference is largely attributed to the improvement in during-accident human error modeling in the Oconee PRA compared to the Surry PRA. In the Surry PRA, many risk-important during-accident human errors were not modeled, and accordingly, the CMF showed stronger sensitivity to pre-accident errors.

b) Types of Activity (Operational error, restoration error, etc.)

The sensitivity evaluation of the types of activity differs for the Oconee plant and plants. Due to improved modeling of during-accident errors in the Oconee PRA, which are also operational errors, the sensitivity results show much stronger sensitivity to operational errors. The human errors in restoration following test and maintenance (called test and maintenance errors in NUREG/CR-1879) is found to be the next important category, which is a reversal of the results obtained in the Surry plant. Human errors of calibration were the least important category in both studies.

Further evaluations were made for Oconee, focussing on the pre-accident errors. This was performed by fixing the during-accident error probabilities at their lower bounds, and then conducting the sensitivity evaluations for the various pre-accident errors. The justification for taking such an approach was to identify the importance of various categories in pre-accident conditions, and also to obtain a basis for comparison comparable with the Surry evaluation (NUREG/CR-1879), where the human error modeling largely focussed on pre-accident errors. The results still show the importance of operational errors at Oconee; however, the sensitivity to restoration errors is almost comparable to operational errors. In comparing to the Surry sensitivity results, the factor increase in CMF due to restoration errors is similar in both cases. This tends to support a conclusion that the major difference in overall sensitivity results is due to improved "during accident" human error modeling and not plant design differences.

c) Omission/Commission Category

The sensitivity results in both plants show that omission errors are noticeably more important than commission type errors. This finding signifies the importance of assuring that the plant's crew does not fail to perform the required actions, and that the errors of commission do not on the average have much influence on the risk parameters of the plant. It is conceivable that in certain situations the conditional risk resulting from an act of commission can be significantly high; however, such situations were not analyzed in this study.

A number of questions remain regarding the adequate treatment of commission type of errors. In general, the argument is that PRA modeling techniques (fault/event trees) are adequate for modeling omission type errors, but not those of commission errors and hence, commission errors are not sufficiently treated in PRAs. Additionally, experience (e.g., the Three Mile Island and Chernobyl accidents) has shown that certain specific commission errors can have very serious effects. The Oconee PRA attempted to model operator commission errors and included many more than the Surry PRA. An interesting point is that there was no noticeable increase in sensitivity to these errors, the reason being that commission errors are unlikely, and their probabilities are comparatively small. Accordingly, commission errors do not become a dominating influence in the sensitivity evaluations.

d) Location of errors (control room versus outside control room)

The sensitivity of this error categorization, as expected, followed the pattern observed for the timing and activity categories. In Oconee, during-accident errors and operational errors are primarily control room errors, and the sensitivity of risk parameters to such errors was more significant than outside control room errors. In the Surry evaluation, outside control room errors were more significant, which was expected because of the dominance there of pre-accident errors and restoration errors (called test and maintenance errors) in that study. The reason behind this difference is discussed earlier for the timing and activity categories.

Additional sensitivity evaluations were conducted for Oconee, focussing on during accident errors to study the relative importance of the various categories. In this regard, an additional sub-category of errors, called Control Room/Outside Control Room (CR/OCR) errors, was defined, identifying errors made by personnel both in and outside of the control room. Many of the recovery errors, and valve line-up errors requiring operator checking belong to the CR/OCR group of errors. Among the during-accident errors, the CR/OCR errors are found to be as significant as the control room errors. This is due to the importance of recovery errors requiring coordination of personnel both in and out of the control room.

When the sensitivity evaluations focus on pre-accident errors, it was observed that sensitivity to outside control room errors were slightly more significant than either control room errors or combined CR/OCR errors in the Oconee study. This is expected, since in pre-accident situations, more activities are conducted outside of the control room.

6.3.3 Comparison of Accident Sequence Frequencies

Sensitivity of individual accident sequence frequencies to human errors were analyzed in this study and in NUREG/CR-1879. The way accident sequences were defined in these PRAs is different and accordingly, direct comparison of accident sequences are not possible, nevertheless, several general observations can be made.

1. Sensitivity of dominant accident sequences

The dominant accident sequences in both the plants are strongly sensitive to human errors. For the Surry plant, the very small LOCA sequence S_2C, and the transient sequence TMLB' were among the sensitive sequences. Loss of Instrument Air, Loss of Service Water, and Very Small LOCA are among the dominant sequences in the Oconee plant, and are also strongly sensitive to human errors. In all these sequences, significant risk reduction can be achieved if human error probabilities can be improved. Two possible reasons for this high sensitivity of the dominant accident sequences were identified. One reason is that greater attention is paid to the HRA in the dominant sequences, resulting in more human errors and hence, in greater sensitivity. Secondly, it is clear that transient sequences will have a higher sensitivity to human error. Also, transient sequences generally have been dominant in many PRAs, due to the significant amount of design attention paid to large LOCAs, resulting in lower accident sequence frequencies for the large LOCAs.

2. Dominance of transient-initiated sequences for increased HEPs

The transient initiated accident sequences are strongly sensitive to human errors in both plants. The large LOCA initiated sequences, Vessel Rupture sequence, are among the least human error sensitive sequences in both. When the HEPs are increased, the dominance of transient-initiated events increases in both plants. This finding is expected, since the transient-initiated events have significant human role compared to large LOCA or Vessel Rupture sequences. The dominance of transient-initiated events in the Oconee plant when the HEPs are increased is comparable to that observed for the Surry plant.

3. Reduction in accident sequence frequencies for improved HEPs

In both plants, accident sequence frequencies can be reduced through reduction of the HEPs. Specific accident sequences are identified in both studies where large reductions can be obtained through relatively small reductions in HEPs. A significantly higher reduction can be obtained in the Oconee accident sequence frequencies for relatively small reduction in HEPs than can be obtained in the Surry plant. Also, a larger number of accident sequences show this phenomenon in the Oconee plant compared to the Surry plant. Reasons for this are discussed in 6.3.1(b) above.

7. FUTURE RESEARCH

This project identified a number of insights relative to human performance and risk and also highlighted areas where additional research would be beneficial. Future research could help to extend and amplify the conclusions herein, improve the understanding of the relationship of human performance to risk, and aid in eventual actions to improve performance and hence limit risk. Recommended areas for future research are summarized in Table 7.1 and are discussed below.

Table 7.1. Recommended Areas of Future Research

I.	**SENSITIVITY EVALUATIONS**
	1. Generic Implications from Sensitivity Evaluation
	2. Maximum Plant Risk Level due to Human Performance
II.	**HUMAN RELIABILITY ANALYSIS**
	1. Evaluation of the Contribution of Human Performance in Maintenance to Plant Risk
III.	**HUMAN FACTOR STUDIES**
	1. Accident Management Role
	2. Adequacy of Procedures, Training, etc.
IV.	**DATA BASED EVALUATIONS**
	1. Reality Check of PRA Data in Human Performance Area

I. SENSITIVITY EVALUATION

1. Generic Implications

This study performed sensitivity evaluations for a Babcock and Wilcox (B and W) designed PWR (Oconee Unit 3), while the earlier study evaluated a Westinghouse designed PWR (Surry). The general conclusions from these two studies support each other, with some differences as discussed in Section 6. It is felt that certain insights and conclusions of these two studies are generically applicable. A similar study is underway for a BWR (LaSalle), and when it is completed, a further generalization to other plants should be feasible.

2. Maximum Plant Risk Level

This study established upper bounds on each human error and then used these ranges or upper bounds to determine risk sensitivity. However, it is likely that there are interactive factors between personnel and errors which would further limit the ranges of the human error probabilities (HEPs). By determining these factors and obtaining more realistic upper bounds on HEPs, a more realistic Maximum Plant Risk level could be obtained.

II. HUMAN RELIABILITY ANALYSIS

1. Evaluation of the Contribution of Human Performance in Maintenance to Plant Risk

This study evaluated the sensitivity of risk to all HEs explicitly modeled in the PRA. However, as is typical with all current PRAs, the only maintenance related HEs explicitly included in the Oconee PRA were the errors of failure to properly restore components to their normal lineup after maintenance. Errors committed during maintenance, which would result in equipment failing at a later date, when called on to operate, are only included implicitly in the data on hardware failure rates. Also, some portion of the initiator frequency is due to maintenance errors, such as mistakes during calibration. By accounting for all of the different portions of human performance during maintenance, a clearer picture of the importance of maintenance relative to risk may be obtained.

III. HUMAN FACTOR STUDIES

1. Accident Management Role

The role of plant management in reducing risk both pre-accident and during-accident is very important. Work is underway to develop a plant management/organizational model, which would then have many applications, including its use to determine quantitatively management's effect on risk.

2. Adequacy of Procedures and Training

This study has identified a number of HEs (such as recovery actions) which are very risk sensitive, which have high HEPs and are also quite complex. Good procedures and training at individual plants could ensure that these HEPs are kept low and hence, reduce risk. It is likely that many plants, to which these results apply, have few procedures or training for some of these high risk recovery actions since they are outside the normal design basis.

IV. DATA BASED EVALUATIONS

1. Reality Check of PRA Data in Human Performance Area

There are ongoing concerns about how valid and representative the human performance data is that is used in PRAs and sensitivity studies. Additional work is needed to verify the baseline HEPs and to compare HE modeling with real plant data, such as Licensee Event Reports (LERs), NRC Inspection Reports, and in-plant data.

8. REFERENCES

Hanan, N. et al., "A Review of the Oconee-3 Probabilistic Risk Assessment," NUREG/CR-4374, 2 volumes, March 1986.

Harris, R.J., "A Primer of Multivariate Statistics," pp. 156-199, Academic Press, Inc., 1975.

Kelly, J.E., Leverenz, Jr., F.L., McCormick, N.J., and Erdmann, R.C., "Sensitivity Assessment in Reactor Safety Analysis," Nuclear Technology, 32, pp. 155-166, February 1977.

Letter from P. Samanta, Brookhaven National Laboratory to R. Palla, U.S. Nuclear Regulatory Commission, "A Strategy for Performing Human Error Sensitivity Evaluation to Obtain Insights on Variations in Plant Risk," January 29, 1988.

NSAC-60, "Oconee PRA: A Probabilistic Risk Assessment of Oconee Unit 3," NSAC-EPRI, June 1984.

O'Brien, J. and Spettell, C., "Uses of Human Reliability Analysis, Probabilistic Risk Assessment Results to Resolve Personnel Performance Issues that Could Affect Safety," NUREG/CR-4103, October 1985.

PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, NUREG/CR-2300, January 1983.

Parkinson, W.J., "Sensitivity Analysis of the Reactor Safety Study," M.S. Thesis, Massachusetts Institute of Technology, February 1979.

Samanta, P.K., and Mitra, S.P., "Modeling of Multiple Sequential Failures During Testing, Maintenance, and Calibration," NUREG/CR-2211, December 1981.

Samanta, P.K., Hall, R.E. and Swoboda, A., "Sensitivity of Risk Parameters to Human Errors in Reactor Safety Study for a PWR," NUREG/CR-1879, June 1981.

Swain, A.D., and Guttmann, H.E., "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278, October 1983.

Trager, Jr., E.A., "Case Study Report on Loss of Safety System Function Events," Office for Analysis and Evaluation of Operational Data (AEOD), AEOD/C504, December 1985.

WASH-1400, "Reactor Safety Study (RSS), An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," 1975.

Wechsler, D., Range of Human Capacities, Williams and Wilkins, Baltimore, Maryland.

APPENDIX A

DETAILS OF CATEGORY CODES NOT PRESENTED IN SECTION 3

SYSTEMS - BNL IDENTIFIED OCONEE 3 PRA

ACPS - (or AC) - AC Power System

CFS - Core Flood System

DCPS - DC Power System

EFW - (or EF) - Emergency Feedwater

EFW/M - EFW/Main Feedwater

ES - Engineered Safeguards

HPI - High Pressure Injection

HPI/R - HPI Recirculation

HVAC - Heating, Ventilation and Air Conditioning

IA - Instrument Air

IA/OF - IA/Offsite

IA/SA - IA/Station Air

ICS - Integrated Control System

LPI - Low Pressure Injection

LPI/D - LPI/Decay Heat Removal

LPI/H - LPI/High Pressure Injection

LPI/R - LPI Recirculation

MFW - Main Feedwater (subset of PCS)

OFPWR - Offsite Power

PCS - Power Conversion System (includes Condensate but excludes MFW)

PCS/O - PCS/Oil Purification

PCS/S - PCS/Other Steam Source

SYSTEMS - BNL IDENTIFIED OCONEE 3 SYSTEMS (Cont'd)

RBC - Reactor Building Cooling

RBS - Reactor Building Spray

RCS - Reactor Coolant System

RCSPZ - RCS Pressurizer

SSF - Standby Shutdown Facility

SSFAS - SSF Auxiliary Service Water

SSFCS - SSF Reactor Coolant Volume Control System

SSFDG - SSF Diesel Generator

SSFEP - SSF Electric Power

SSFFW - SSF Feedwater

SW - Service Water

SWCCW - SW/Component Cooling Water

SW/HP - SW/High Pressure

SW/LP - SW/Low Pressure

COMPONENTS - BNL IDENTIFIED FROM OCONEE 3 PRA

AHU - Air Handling Unit (Emergency Pump Room)

ANNUN/LAMP - Annunciator Lamp (Light)

AOV - Air Operated Valve

BATCHRG - Battery Charger

BATTERY - Battery - DC Power

BI - Bistable

BI/COMMON - Bistable - Common

BI/RB Pres - Reactor Building Pressure Bistable

BRKR - Circuit Breaker

BUFF AMPLF - Buffer Amplifier

BUFF COMMN - Buffer Amplifier - Common

BUS - Electrical Bus

BUS LEE - Bus named Lee

BUS KEOWE12 - Bus named Keowee 1 or 2

BWST - Borated Water Storage Tank

CHANNEL - Engineered Safeguards Channel

CIRCUIT - DC Power Circuit

CRT/TST/PR - Power Range Test Circuit

COMPR - Air Compressor

COMPR/DSL - Diesel Compressor - Station Air

COMPR/LDSD - Load Shed Air Compressor

CONTRLR/PI - Controller/Pressure Indicator

DIESEL GEN - Diesel Generator

H/A STATION - Hand (Manual)/Automatic Station

H/A RX/CRD - Reactor Demand H/A or CRD (in Manual)

COMPONENTS - BNL IDENTIFIED FROM OCONEE 3 SYSTEMS (Cont'd)

HEAT EXCHG - Heat Exchanger (Cooler)

HTR TRAIN - Heater Train

INTLK/PERM - Interlocks and Permissives

INVTR - Inverter

LD CTR - Load Center

MOV - Motor Operated Valve

PMP - Pump

PORV - Power Operated Relief Valve

PS - Power Supply

RB VENT UNT - Reactor Building Ventilation Units

RCP - Reactor Coolant Pump

RLAY LD BR - Relay for Load Breakers MFB1 and MFB2

RLAY LD BU - Relay for Load Buses 3TC, 3TD, and 3TE

S - System

- S/1 SO AVL - S/One Source Available
- S/2 SO AVL - S/Two Sources Available
- S/3 SO AVL - S/Three Sources Available
- S/AIR COMP - S/Air Compressor
- S/HPI PPS - S/HPI Pumps
- S/MAIN STM - S/Main Steam
- S/POWDEX - S/Powdex Column
- S/RX DEMAD - S/Reactor Demand
- S/SEAL INJ - S/SSF Seal Injection
- S/SU HDR - S/Startup Header
- S/TURB PRE - S/Turbine Header Pressure Setpoint
- S/WGT - S/Weighted Averages - Events

COMPONENTS - BNL IDENTIFIED FROM OCONEE 3 SYSTEMS (Cont'd)

SIG GEN - Signal Generator (Adjustable)*

SIM PLUG - Simulator Plug Connection*

SUMMER - Summer Bias*

SW - Switch

- SW/PP CB - Condensate Booster Pump Control Switch
- SW/PP HW - Hotwell Pump Control Switch
- SW/PRES - Pressure Switch
- SW/SELECT - Selector Switch
- SW/SGWR - Control Switch or Switchgear
- SW/TRANS - Transfer Switch

TANK - Core Flood Tank

TANK/STRG - Storage Tank - Elevated

TAVE/SETPT - Tave Setpoint

TRANS CT5 - Transformer CT5

TRANS/LVL - Level Transmitter

TRANS/PRES - Pressure Transmitter

TRM SIG/PP - Trim Signal for Pump

VV - Manual Valve

- VVS/HPI PP - VVs for HPI Pumps

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* in the Integrated Control System (ICS)

APPENDIX B

LISTING OF UPPER AND LOWER BOUNDS FOR EACH HUMAN ERROR

APPENDIX B

LISTING OF UPPER AND LOWER BOUNDS FOR EACH HUMAN ERROR

This appendix provides the results of the ranges developed for each human error event. The HEs are listed in the five groups that were developed in accordance with the range methodology of Section 3 and that are listed in Table 3.6. Each HE is listed, along with its Lower Bound Human Error Probability (HEP), its base case HEP, as used in the PRA, and its upper bound or high HEP.

HUMAN ERRORS THAT ARE ACTS OF TESTING, MAINTENANCE, OR CALIBRATION
WITH HEP > 1E-3, ERROR FACTOR = 13

Human Error	Page in Oconee PRA	Low HEP	HEP	High HEP
LP15MVMH	A2-53	0.0001385	0.001800	0.023400
LP16MVMH	A2-53	0.0001385	0.001800	0.023400
MF3C1HXH	A8-90	0.0007692	0.010000	0.130000
MF3C2HXH	A8-90	0.0007692	0.010000	0.130000
MF3E1HXH	A8-90	0.0007692	0.010000	0.130000
MF3D1HXH	A8-90	0.0007692	0.010000	0.130000
MF3D2HXH	A8-90	0.0007692	0.010000	0.130000
EFTDPP1H	A10-41	0.0007692	0.010000	0.130000
SW3BFPH	A14-48	0.0006667	0.014000	0.294000
SW71VVH	A14-48	0.0001539	0.002000	0.026000
I476SUH	A9-217	0.0001462	0.001900	0.024700
I553SUH	A9-217	0.0001462	0.001900	0.024700
I8413SUH	A9-217	0.0001462	0.001900	0.024700
I8513SUH	A9-217	0.0001462	0.001900	0.024700
ISS14ASSH	A9-218	0.0002308	0.003000	0.039000
IIC35ASSH	A9-218	0.0002308	0.003000	0.039000
ISS14BSSH	A9-218	0.0002308	0.003000	0.039000
IIC35BSSH	A9-218	0.0002308	0.003000	0.039000
IIC10SSH	A9-218	0.0002308	0.003000	0.039000
ES578CM	A11-39	0.0001154	0.001500	0.019500

HUMAN ERRORS OF OMISSION THAT ARE ACTS OF OPERATIONS
ERROR FACTOR = 21

Human Error	Page in Oconee PRA	Low HEP	HEP	High HEP
BMUH	3-46	0.0000476	0.001000	0.021000
BEFWH	3-46	0.0000476	0.001000	0.021000
OLTCH	3-70	0.0001429	0.003000	0.063000
UTHPIH	3-50	0.0004762	0.010000	0.210000
XALPRH	3-62	0.0002381	0.005000	0.105000
XHPR2H	3-53	0.0001429	0.003000	0.063000
XHPR12H	3-53	0.0000143	0.000300	0.006300
XRDHRH	3-68	0.0000143	0.000300	0.006300
XRRCPH	3-69	0.0004762	0.010000	0.210000
XRSPCH	3-69	0.0047620	0.100000	1.000000
RC46MVH	A7-20	0.0002857	0.006000	0.126000
YRBSH	3-51	0.0238100	0.500000	1.000000
HPRCPH	A3-71	0.0004762	0.010000	0.210000
HPLDSTH	A3-71	0.0004762	0.010000	0.210000
HPCPPH	A3-71	0.0476200	1.000000	1.000000
HPSEGOH	A3-71	0.0000105	0.000220	0.004620
HP148VVH	A3-71	0.0000291	0.000610	0.012810
HP24MVH	A3-71	0.0004762	0.010000	0.210000
HP25MVH	A3-71	0.0004762	0.010000	0.210000
HP26MVH	A3-71	0.0476200	1.000000	1.000000
HP26MVCH	A3-71	0.0000071	0.000150	0.003150
HP27MVH	A3-71	0.0000071	0.000150	0.003150
LPBPPH	A2-52	0.0476200	1.000000	1.000000
LPAPPH	A2-52	0.0476200	1.000000	1.000000
LP12MVOH	A2-52	0.0476200	1.000000	1.000000
LP14MVOH	A2-52	0.0476200	1.000000	1.000000
LP20MVRH	A2-53	0.0001429	0.003000	0.063000
LP18MVRH	A2-53	0.0476200	1.000000	1.000000
LP17MVRH	A2-53	0.0476200	1.000000	1.000000
LP67MVH	A2-53	0.0004762	0.010000	0.210000
LP15MVH	A2-53	0.0000476	0.001000	0.021000
LP16MVH	A2-53	0.0000476	0.001000	0.021000
LPPPSTOPH	NA2-53	0.0000381	0.000800	0.016800
LP13VVH	A2-52	0.0000052	0.000110	0.002310
LP11VVH	A2-52	0.0000052	0.000110	0.002310
LP34VVH	A2-52	0.0000052	0.000110	0.002310
LP32VVH	A2-52	0.0000052	0.000110	0.002310
RC418VCH	A7-20	0.0476200	1.000000	1.000000
RC417VCH	SA7-20	0.0395200	0.830000	1.000000
RC665SVH	A7-20	0.0004762	0.010000	0.210000
RC660SVH	A7-20	0.0004762	0.010000	0.210000
MFSNGLH	A8-90	0.0476200	1.000000	1.000000
MFOLTALOH	A8-90	0.0001429	0.003000	0.063000
OBWSTH	D-125	0.0238100	0.500000	1.000000
QFDWR	D-125	0.0333333	0.700000	1.000000
REFEEDAIR12	ND-126	0.0000571	0.001200	0.025200
REFEEDAIR2	D-127	0.0005714	0.012000	0.252000

HUMAN ERRORS OF OMISSION THAT ARE ACTS OF OPERATIONS
ERROR FACTOR = 21

Human Error	Page in Oconee PRA	Low HEP	HEP	High HEP
REFDW1	D-127	0.0238100	0.500000	1.000000
REFDW2	D-127	0.0142860	0.300000	1.000000
REHPPPCS	D-128	0.0023810	0.050000	1.000000
REIA12	D-128	0.0001143	0.002400	0.050400
REIA2	D-128	0.0142860	0.300000	1.000000
RESSFSI	D-129	0.0047620	0.100000	1.000000
RESSFW12	ND-129	0.0001667	0.003500	0.073500
RESSFW30	D-129	0.0047620	0.100000	1.000000
RESUBAIR12	ND-130	0.0001905	0.004000	0.084000
RESUBAIR2	SD-130	0.0010476	0.022000	0.462000
RESUMPMF	D-131	0.0047620	0.100000	1.000000
RESW12	ND-131	0.0006667	0.014000	0.294000
RESWLPI	D-131	0.0095238	0.200000	1.000000
EF88VVH	A10-41	0.0000476	0.001000	0.021000
CSC10AVH	A10-42	0.0001429	0.003000	0.063000
EFSUH	A10-42	0.0476200	1.000000	1.000000
P3XS1SWH	A12-87	0.0476200	1.000000	1.000000
PXS2F3ASWH	A12-87	0.0476200	1.000000	1.000000
P3XS33ASWH	A12-87	0.0476200	1.000000	1.000000
P13XS1SWH	A12-87	0.0476200	1.000000	1.000000
PS29ARSWH	A12-87	0.0476200	1.000000	1.000000
PX334ARSWH	A12-88	0.0476200	1.000000	1.000000
P3X84CSWH	A12-88	0.0476200	1.000000	1.000000
P3X94CSWH	A12-88	0.0476200	1.000000	1.000000
PLS1H	A12-88	0.0476200	1.000000	1.000000
ACDIAIVSWH	A13-50	0.0047620	0.100000	1.000000
ACDIBIVSWH	A13-50	0.0047620	0.100000	1.000000
ACDIATMIVH	A13-50	0.0000476	0.001000	0.021000
ACDIBTMIVH	A13-50	0.0000476	0.001000	0.021000
ACDICTMIVH	A13-50	0.0000476	0.001000	0.021000
AC3KITMIVH	A13-50	0.0000476	0.001000	0.021000
SWH247VVH	A14-48	0.0008571	0.018000	0.378000
SWAHPIH	A14-48	0.0000000	0.000001	0.000021
SWBHPIH	A14-48	0.0001429	0.003000	0.063000
SWCHPIH	A14-48	0.0001429	0.003000	0.063000
SWAFPH	A14-48	0.0000143	0.000300	0.006300
SWBFPH	A14-48	0.0001429	0.003000	0.063000
SWCFPH	A14-48	0.0001429	0.003000	0.063000
SW3BPPCH	A14-48	0.0000190	0.000400	0.008400
SW405AVH	A14-48	0.0000476	0.001000	0.021000
SW3BPPSH	NA14-48	0.0003810	0.008000	0.168000
SW404AVH	A14-48	0.0000476	0.001000	0.021000
SW169VVH	NA14-49*	0.0000001	0.000003	0.000063
SWEXCESSH	A14-49	0.0476200	1.000000	1.000000
SWC88VVH	A14-49	0.0000381	0.000800	0.016800
SWC89VVH	A14-49	0.0000381	0.000800	0.016800
HCAH1FNBH	A17-6	0.0000476	0.001000	0.021000
CCW87VVH	NA-32	0.0000405	0.000850	0.017850

HUMAN ERRORS OF OMISSION THAT ARE ACTS OF OPERATIONS
ERROR FACTOR = 21

Human Error	Page in Oconee PRA	Low HEP	HEP	High HEP
SW527VVH	NA-34	0.0000405	0.000850	0.017850
HP111VVH	NA3-50*	0.0000290	0.000610	0.012810
XSFDWR12H	3-59	0.0000095	0.000200	0.004200
LPIPUMPC	NA-32	0.0047620	0.100000	1.000000
RESW78	NA-34	0.0023810	0.050000	1.000000
XOLP1034H	3-68	0.0047620	0.100000	1.000000
RELPD16	NA-33	0.0047620	0.100000	1.000000
REDHRSUC	NA-33	0.0047620	0.100000	1.000000
XYBRSH	3-51+X	0.0238100	0.500000	1.000000
ES34MT	A11-53	0.0476200	1.000000	1.000000
ES56MT	A11-54	0.0476200	1.000000	1.000000
HP114VVH	NA3-50*	0.0000290	0.000610	0.012810
LPFLOWH	A2-51	0.0476200	1.000000	1.000000
LPDHRSUC	NA-32	0.0009524	0.020000	0.420000
RESUBAIR90	NA-33	0.0011905	0.025000	0.525000
REIA1	NA-33	0.0238100	0.500000	1.000000
SWEXCESSLPR	NA-34	0.0047620	0.100000	1.000000

HUMAN ERRORS THAT ARE ACTS OF TESTING, MAINTENANCE, OR CALIBRATION
WITH HEPs <= 1E-3, ERROR FACTOR = 22

Human Error	Page in Oconee PRA	Low HEP	HEP	High HEP
CFALTAH	A4-15	0.0000045	0.000100	0.002200
CFAPTAH	A4-15	0.0000045	0.000100	0.002200
CFALTLH	A4-15	0.0000045	0.000100	0.002200
CFBPTAH	A4-15	0.0000045	0.000100	0.002200
CFBLTLH	A4-15	0.0000045	0.000100	0.002200
CFBLTAH	A4-15	0.0000045	0.000100	0.002200
IIC13SSH	A9-217	0.0000455	0.001000	0.022000
IIC36ASSH	A9-217	0.0000009	0.000020	0.000440
IIC36BSSH	A9-217	0.0000009	0.000020	0.000440
ISIMPLGH	A9-217	0.0000013	0.000029	0.000638
IIC36BASH	A9-218	0.0000005	0.000010	0.000220
EBI159000H	A11-57	0.0000455	0.001000	0.022000
EBI259000H	A11-57	0.0000455	0.001000	0.022000
EBI359000H	A11-57	0.0000455	0.001000	0.022000
EBA156000H	A11-57	0.0000455	0.001000	0.022000
EBA256000H	A11-57	0.0000455	0.001000	0.022000
EBA356000H	A11-57	0.0000455	0.001000	0.022000
BS41PT000H	A11-57	0.0000455	0.001000	0.022000
BS42PT000H	A11-57	0.0000455	0.001000	0.022000
BS43PT000H	A11-57	0.0000455	0.001000	0.022000
EPS110000H	A11-58	0.0000455	0.001000	0.022000
EPS210000H	A11-58	0.0000455	0.001000	0.022000
EPS310000H	A11-58	0.0000455	0.001000	0.022000
EPS120000H	A11-58	0.0000455	0.001000	0.022000
EPS220000H	A11-59	0.0000455	0.001000	0.022000
EPS320000H	A11-59	0.0000455	0.001000	0.022000
EPS164000H	A11-59	0.0000455	0.001000	0.022000
EPS264000H	A11-59	0.0000455	0.001000	0.022000
EPS364000H	A11-59	0.0000455	0.001000	0.022000
P27CALH	A12-86	0.0000045	0.000100	0.002200
CFABLTAH	NA4-7*	0.0000005	0.000010	0.000220
CFABPTAH	NA4-7*	0.0000005	0.000010	0.000220
ES13CM	A11-39	0.0000024	0.000050	0.001050
ES3CM	A11-39	0.0000068	0.000150	0.003300
LPBPPMH	A2-53	0.0000205	0.000450	0.009900
LPAPPMH	A2-53	0.0000205	0.000450	0.009900
LP18MVMH	A2-52	0.0000068	0.000150	0.003300
LP17MVMH	A2-52	0.0000068	0.000150	0.003300
LP12MVMH	A2-52	0.0000068	0.000150	0.003300
LP14MVMH	A2-52	0.0000068	0.000150	0.003300
LP9MVMH	A2-52	0.0000068	0.000150	0.003300
LP10MVMH	A2-52	0.0000068	0.000150	0.003300
LP6MVMH	A2-53	0.0000159	0.000350	0.007700
LP7MVMH	A2-53	0.0000159	0.000350	0.007700
LP19MVMH	A2-53	0.0000136	0.000300	0.006600

HUMAN ERRORS THAT ARE ACTS OF TESTING, MAINTENANCE, OR CALIBRATION
WITH HEPs <= 1E-3, ERROR FACTOR = 22

Human Error	Page in Oconee PRA	Low HEP	HEP	High HEP
LP20MVMH	A2-53	0.0000136	0.000300	0.006600
LP28VVCH	A2-53	0.0000013	0.000028	0.000616
MS90VVH	A10-41	0.0000455	0.001000	0.022000
MS89VVH	A10-41	0.0000455	0.001000	0.022000
MS86VVH	A10-41	0.0000455	0.001000	0.022000
CW157VVH	NA10-41	0.0000045	0.000100	0.002200
CW156MVH	NA10-41	0.0000455	0.001000	0.022000
CW166VVH	NA10-41	0.0000045	0.000100	0.002200
CW391MV1H	A10-41	0.0000455	0.001000	0.022000
CW180VVH	NA10-42	0.0000045	0.000100	0.002200
CW573VVH	NA10-42	0.0000045	0.000100	0.002200
DCADA1MDIH	A13-50	0.0000455	0.001000	0.022000
DCADA2MDIH	A13-50	0.0000455	0.001000	0.022000
DCADB1MDIH	A13-50	0.0000455	0.001000	0.022000
DCADB2MDIH	A13-50	0.0000455	0.001000	0.022000
EFPSIVH	N10-32*	0.0000000	0.000000	0.000000

DEPENDENT HUMAN ERRORS
ERROR FACTOR = 26

Human Error	Page in Oconee PRA	Low HEP	HEP	High HEP
HPCROSSH	A3-71	0.0003846	0.010000	0.260000
HPBCPPH	A3-71	0.0000058	0.000150	0.003900
HP2425MVH	A3-71	0.0000019	0.000050	0.001300
LP910MVRH	A2-53	0.0003846	0.010000	0.260000
LP19MVRH	A2-53	0.0001154	0.003000	0.078000
LPTHROTTLE	A2-53	0.0001154	0.003000	0.078000
LP40VVH	A2-52	0.0000385	0.001000	0.026000
LP4142VVH	A2-52	0.0038460	0.100000	1.000000
LWD99103VVH	A2-53	0.0000231	0.000600	0.015600
CW157VV1H	A10-41	0.0038462	0.100000	1.000000
EFP13XCH	A10-41	0.0000000	0.000001	0.000026
SWEFCCH	A14-48	0.0000077	0.000200	0.005200
SW7778CMH	A14-48	0.0001154	0.003000	0.078000

HUMAN ERRORS OF COMMISSION THAT ARE ACTS OF OPERATIONS
ERROR FACTOR = 24

Human Error	Page in Oconee PRA	Low HEP	HEP	High HEP
QHPIH	3-39	0.0020830	0.050000	1.000000
CF1MV2H	A4-15	0.0000000	0.000001	0.000024
CF2MV2H	A4-15	0.0000000	0.000001	0.000024
LPABH	A2-52	0.0000042	0.000100	0.002400
LP12MVCH	A2-52	0.0001250	0.003000	0.072000
LP14MVCH	A2-52	0.0001250	0.003000	0.072000
LP5MVH	A2-53	0.0000458	0.001100	0.026400
LP8MVH	A2-53	0.0000458	0.001100	0.026400
CSC10AVH	A8-90	0.0001250	0.003000	0.072000
CSPWDXH	A8-91	0.0001250	0.003000	0.072000
CSHWAPPH	A8-91	0.0001250	0.003000	0.072000
MFSSH1	SA8-91	0.0208300	0.500000	1.000000
MFESUH1	A8-91	0.0001042	0.002500	0.060000
MFSSH2	A8-91	0.0104200	0.250000	1.000000
MFESUH2	A9-47	0.0002083	0.005000	0.120000
RC415VCH	A7-20	0.0001250	0.003000	0.072000

APPENDIX C

OCONEE-3 PRA COMPUTER MODEL

APPENDIX C

OCONEE-3 PRA COMPUTER MODEL

The computational model of the "baseline" risk plane for human error sensitivity analyses is defined by dominant accident sequences that were identified in the Oconee-3 Probabilistic Risk Assessment (PRA) study[1] and its full-scope review by Brookhaven National Laboratory (BNL)[2]. The accident sequences considered in this baseline risk model are initiated by internal events (accidents initiated by a functional equipment failure or an external loss of power) that lead to core damage. As such, the risk model and the risk impact of human errors is enveloped by the internal event analysis of the Oconee-3 nuclear plant that was considered under the BNL review.

The accident sequences included in the baseline risk model are presented in Table C-1 with the following breakdown: ten transients, four small LOCAs (loss-of-coolant accidents), two large LOCAs, two SGTRs (steam generator tube ruptures), seven transient-induced LOCAs, one interfacing system LOCA, one pressure vessel rupture, and 14 ATWS (anticipated transient without scram) sequences. A single block file containing the cutset equations for each of 25 dominant accident sequences was created on the mainframe computer (AMD Cyber 830) using the SETS computer code. These sequences, viz., the transients, small-break LOCAs, large-break LOCAs, SGTRs and transient-induced LOCAs, primarily constitute the computational model on which sensitivity calculations were performed. The interfacing system LOCA and pressure vessel rupture sequences were treated as constants in the sensitivity analyses due to no variation in the impact of human error. Due to the limited scope of analysis, the impact of human error contributions within the ATWS sequences is assumed to have proportionate effects in the risk model.

For all accident sequences within the envelope of the risk model, sequence-dependent as well as cutset-dependent recovery acts have been incorporated to calculate the "base case" mean annual accident frequency. The "base case" accident frequencies are shown in Table C-1, and they are generally in close agreement with those obtained in the BNL review study. Also, the cutsets used in the calculations for each accident sequence are similar to those presented in the BNL review.

The truncation level for the accident sequences that are considered in the risk model is 10^{-7}. Accident sequences below this truncation (e.g., $T_{5,6}$BYLX) are considered whenever human interactions may have a significant impact. The maximum cutoff for minimal cutset (MC) terms derived for all accident sequences is seven variables per cutset. The number of MC terms in each accident sequence range from 15 to 700, with an average of 100 cutsets per sequence. The number of human errors which impact the risk parameters during sensitivity calculations is about 220. This includes 20 recovery events involving operator response.

Due to extended sensitivity analyses of human error impact on the plant risk model, the Oconee-3 risk model was "downloaded" from the mainframe computer on to an IBM PC diskette. This diskette contains the computational algorithms to perform the sensitivity calculations in a convenient manner as well

Table C-1. Oconee-3 Accident Sequences for Human Error Analysis

		Mean Annual Frequency	
Sequence	Initiator Frequency	BNL Review (NUREG/CR-4374)	BNL Base Case
SY_sX_s	3×10^{-3}	5.4E-6	6.04E-6
V_sX_s	3×10^{-3}	1.9E-6	2.16E-6
SX_s	3×10^{-3}	9.0E-7	1.15E-6
SU_s	3×10^{-3}	4.9E-7	5.93E-7
AU_A	9.3×10^{-4}	4.3E-7	6.51E-7
AX_A	9.3×10^{-4}	3.6E-6	9.33E-6
		4.8E-6	
$RU_{RA,RB}$	8.6×10^{-3}	4.1E-7	1.62E-6
		8.1E-7	
$RX_{RA,RB}O$	8.6×10^{-3}	6.0E-7	2.76E-6
		1.5E-6	
T_2BU	0.5	1.3E-6	1.18E-6
T_4BU	0.21	4.8E-7	4.94E-7
$T_{5\,SUBF,FEEDF}$BU	8.0×10^{-2}	2.5E-7	1.28E-6
	4.0×10^{-2}		
T_6BU	0.21	2.9E-5	2.71E-5
T_{10}BU	9.3×10^{-4}	4.8E-6	4.74E-6
T_{11}BU	5.0×10^{-2}	1.8E-7	1.99E-7
T_{12}BU	4.9×10^{-3}	1.8E-5	1.31E-5
T_xBU	5.7	7.6E-7	4.88E-7
$T_{5,6}$BLX		2.3E-7	2.80E-7
$T_{5,6}$BYLX			4.84E-9

Table C-1 (Continued)

		Mean Annual Frequency	
Sequence	Initiator Frequency	BNL Review (NUREG/CR-4374)	BNL Base Case
$T_6Q_{SEAL}U_s$	0.21	2.9E-7	3.59E-7
$T_{12}Q_{SEAL}U_s$	$4.9x10^{-3}$	7.3E-7	7.29E-7
$T_5Q_{PORV}U_s$	$1.7x10^{-5}$ $4.2x10^{-5}$	4.8E-7	4.14E-7
$T_{8,13}Q_{PORV}U_s$	$1.0x10^{-2}$ $4.4x10^{-2}$	3.7E-7	3.86E-7
$T_{12,14}Q_{PORV}U_s$	$4.9x10^{-3}$ $5.4x10^{-3}$	5.5E-8	1.24E-7
$T_5Q_{PORV}X_s$	$8.0x10^{-2}$ $4.0x10^{-2}$	1.0E-7	1.05E-7
$T_6Q_{PORV}X_s$	0.21	3.2E-6	3.42E-6
I	1.4E-7	1.4E-7	1.4E-7
VR	1.1E-6	1.1E-6	1.1E-6
ATW			
TWS 8	1.44E-4	1.9E-6	1.90E-6
TWS 9	1.44E-4	2.1E-6	2.12E-6
TWS 11	1.44E-4	4.7E-7	4.67E-7
TWS 12	1.44E-4	5.2E-7	5.18E-7
TWS 15	1.44E-4	1.9E-7	1.86E-7
TWS 20	1.44E-4	1.9E-7	1.90E-7
TWS 21	1.44E-4	2.1E-7	2.12E-7
TWS 26	1.44E-4	1.3E-7	1.30E-7
TWS 27	1.44E-4	1.4E-7	1.44E-7
TWS 61	1.29E-5	1.7E-7	1.66E-7
TWS 66	1.29E-5	1.7E-7	1.70E-7
TWS 67	1.29E-5	1.9E-7	1.89E-7
TWS 72	1.29E-5	1.2E-7	1.16E-7
TWS 73	1.29E-5	1.3E-7	1.28E-7

TOTAL CORE-MELT FREQUENCY: 8.89E-5 8.66E-5

as a "PAIRWISE" program that permits the performance of "pairwise importance" analyses. The capabilities of the "PAIRWISE" code are fully described in Appendix D.

In summary, the accident sequences considered here in this baseline risk model for human error sensitivity analyses account for 96% of overall plant core-melt frequency due to internal events. The "base case" estimate of the mean annual core-melt frequency due to internal events for the Oconee-3 PRA computer model used in this study is 8.66E-5.

APPENDIX D

DESCRIPTION OF THE PAIRWISE CODE

APPENDIX D

DESCRIPTION OF THE PAIRWISE CODE

D.1 Introduction

Many of the calculations appearing in the main body of this report were performed with a computer code called Pairwise. The purpose of this appendix is to provide a general introduction to Pairwise, and show how it applies to studies of this type. Full documentation of the program will be provided elsewhere.

Pairwise is essentially an importance code. Some results of Pairwise can be obtained by other importance codes. However, Pairwise differs from many other importance codes in several ways: (1) it computes certain quantities which other codes do not compute, (2) it computes importances by a process akin to numerical differentiation of top event probability, rather than by interpreting a large file of minimal cut sets; (3) it works with a compact, factored form of the Boolean expression for the top event, so that a very large Boolean expression can be accommodated; (4) use of the program is interactive, and based on event classes, which can be defined so as to permit very efficient sensitivity studies.

D.2 Background

Although its use in this project has so far been primarily for sensitivity studies, Pairwise was originally developed for very different reasons, which may apply to future stages of this project. For this reason, and in order to motivate the discussion of the computational strategy, its background is sketched below.

Pairwise was initiated as part of a methodology study, carried out at BNL under the sponsorship of the NRC. That study, which was carried out in support of the resolution of USI A-17, was concerned with the identification of systems interactions; the issue leading to the development of Pairwise was how to set up the priorities for the search for system interactions, given a functional logic model of the plant. Pairwise can contribute to setting priorities by calculating the importance of conjunctions of events. For instance, how risk-significant is a coupling (some sort of dependence) between a given hardware failure and a given human error? If these two events occur together in one or more cut sets, and if they are coupled, then the usual approximation of independence of basic events may seriously underestimate the top event probability. Pairwise computes importance measures for such conjunctions of events which are analogous to importance measures of single events. Two events whose conjunction is important are candidates for examination to see whether there is any possible coupling between them; on the other hand, coupling between pairs of events whose conjunction is unimportant is not generally significant (unless it significantly boosts the single-event probability). Originally, then, Pairwise was developed so that a search for system interactions could be focussed on risk-significant pairs of events, rather than having to proceed in an ad hoc fashion by first finding physical interactions, and then assessing whether they matter. Pairwise gets its name from its ability to calculate pair importances.

Pairwise is applicable beyond system interactions; issues of configuration management are addressed naturally within this framework. For example, if a particular component (Pump A) is out for maintenance, other components which contribute to the top event in conjunction with Pump A are relatively more important while the maintenance is in progress. Pairwise is an efficient tool for shedding light on these combinations. This is less easily studied with single-event importances, because components whose importance is enhanced by Pump A maintenance are intermingled in the printout with components which are important anyhow, for reasons having nothing to do with Pump A maintenance.

Essentially, pair importances are the next logical refinement beyond single-event importances, which are one step more complex than the single number giving top event probability. Given a reason to consider triplet importance, it could straightforwardly be computed as well. These numbers collectively provide a way of understanding the structure of the top event expression in successively finer layers of detail.

Much of the code's present usefulness stems from its ability to handle large expressions quickly. This capability stems from the present approach to the calculations, which was motivated by the need to consider pair importances, but turns out to be worthwhile even for single-event applications.

3. Calculations Performed by Pairwise

Pairwise obtains importances by a process akin to numerical differentiation of the top event expression. In this, it differs from many other importance codes; in Pairwise, the top event expression is an integral part of the program, rather than being a file which must be read and interpreted. In order to illustrate the principle, a single calculation will be described here. Full documentation will be provided elsewhere.

For illustration, consider the problem of computing the contribution to top event probability due to the conjunction of events X_i and X_j, or, more colloquially, what is the probability of the union of all those minimal cutsets which include both X_i and X_j? Let x_i represent the probability of X_i, and let F(x) represent the top event function. In our application, F is an arithmetic function corresponding to the rare event approximation to top event probability. Although F is actually in factored form, it is useful to think in terms of the expanded (minimal cutset) form; for any given pair i,j, we can rewrite this expanded form as:

$$F(x) = A_{ij} + B_{ij} * x_i + C_{ij} * x_j + D_{ij} * x_i * x_j.$$

That is, we can collect all terms continaing both x_i and x_j together into the D term, all the terms containing neither into the A term, all the terms containing x_i but not x_j into the B term, and all terms containing x_j but not x_i into the C term. In this example, what we want is to compute the value of just the D term. We can do this, for example, by evaluating (1) top event probability with x_i and x_j both set to zero, which returns simply the value of A_{ij}, and (2) the top event "probability" with x_i and x_j artificially set to $-x_i$ and $-x_j$, which returns $A_{ij} - B_{ij} * x_i - C_{ij} * x_j + D_{ij} * x_i * x_j$. Adding this latter quantity to top event probability gives 2 * (A_{ij} + D_{ij} * x_i * x_j); dividing by two and subtracting A_{ij} gives the desired result, D_{ij} * x_i * x_j. Thus, given the top event probability, we must evaluate the top event expression twice more to calculate each pair importance. A normal importance code would need to read and process a large file of cutsets to compute a single importance value, which Pairwise does by arithmetic; on the other hand, a normal importance code could calculate many importances at once, as it sorted through the large file a single time. Thus, part of Pairwise's speed is derived from its match to the task of doing targeted importance calculations.

4. Classes

Pairwise works with classes of events. A class can be any subset of the events appearing in the event table. Examples of potentially useful classes are "all human errors," "all failures of service water pumps," and "all emergency feedwater system component failures."

There are several reasons for working with event classes. One reason is that it facilitates the conduct of sensitivity studies; being able to adjust the probabilities of an entire class with a single command is convenient. Another reason is that class importance is interesting in itself (if classes are defined with suitable insight). One frequently hears such claims as "human error contributes 40% to system unavailability" without a clear indication of how this was computed or what it means. If this is computed by summing the importances of all human errors, then it is incorrect, unless no two human errors ever appear in conjunction. A better definition is the collective importance of all minimal cutsets containing one or more human errors. Pairwise computes class importance in the latter way. It may also be interesting to know what fraction of top event probability involves the conjuction of any human error with any DC bus failure. With suitable class definitions, this is easily calculated. Finally, of course, it is useful to be able to compute the individual importances of "all recovery acts," and do this a series of times for different assignments of other basic event probabilities.

Class definitions can be read from disk by Pairwise, or entered at the keyboard.

5. Preparation of Pairwise Diskettes

The information flow in a Pairwise application is illustrated in Figure E.1. First, of course, a logic model is developed and "solved," that is, its minimal cutsets are obtained. In the present case, this was done using SETS. For present purposes, a key feature of SETS is that it stores equations in factored form, which is much more compact that the disjunctive normal form (the usual minimal cutset representation). That is, rather than an expression of the form:

A * B * C + A * B * D + ...,

one works with,

(A + ..) * (B + ..) * (C + D + ..),

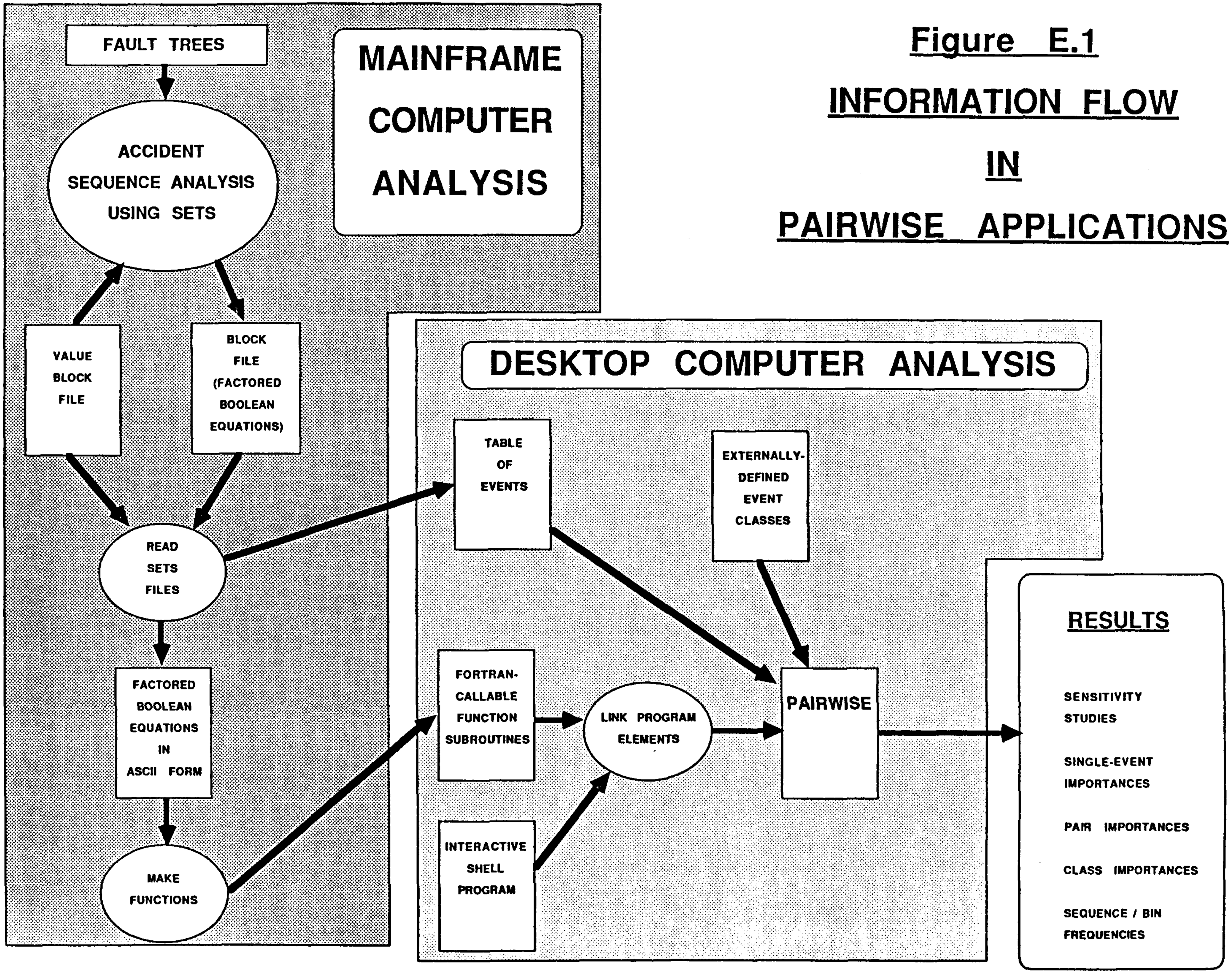

Figure E.1
INFORMATION FLOW IN PAIRWISE APPLICATIONS

which can be much shorter than the fully expanded representation. (As usual, multiplication means "AND" and addition means "OR"). The scratch file (the "Block File") on which SETS routinely stores its results contains equations in this factored form. Preparation of a Pairwise diskette consists essentially of accessing this Block File, transforming the Boolean equations into corresponding arithmetic functions, and linking these arithmetic functions with a standard interactive program shell. (The program used to access the Block File was developed for BNL by R.B. Worrell, of Logic Analysts, Alburquerque, New Mexico. Mr. Worrell also wrote SETS while on the staff of Sandia National Laboratory.) Thus, the accident sequence expressions end up as compiled code, which can be called and executed very quickly, rather than as a file which must be read and processed every time an importance calculation is to be performed. In the present application, the core damage expression consists of some two dozen sequences, totalling nearly thirty thousand minimal cutsets; this expression amount to less than ninety kilobytes of object code, and can be accessed (e.g., the importance of a given basic event within the entire core damage expression can be calculated) in a fraction of a second.

STEPS IN PREPARATION OF PAIRWISE

(Refer to Figure E.1)

1) Complete the accident sequence analysis using SETS. A Block File should be kept, containing an equation block which contains all equations of interest, and only the equations of interest. For inconvenience, the Value File should also be kept.

2) Prepare ASCII versions of the desired Boolean equations. The program which does this works directly on the two SETS files mentioned in Step 1: the Block File and the Value File. It also produces an ASCII file containing all pertinent event names, along with the probabilities which were assigned to these events in the Value File. This event table is to be downloaded to the desktop computer and used as an input file for Pairwise.

3) Make Fortran-callable function subroutines for the accident sequences. From the one-line example sketched above, one sees that if multiplication is AND and addition is OR, then a Boolean equation can be though of as a rare-event approximation to an arithmetic formula for the left-hand-side-event probability. In an extremely simple case, then, one can trivially transform a Boolean equation into a function subroutine. In most cases of practical interest, however, such a simple rewrite will not lead to a legal Fortran function, because the expressions are too big. Therefore, a program has been developed to recast the expressions by breaking them down into pieces which are individually small enough for Fortran, and reassembling them again. This program was developed collaboratively at BNL by D. Xue and R. Youngblood.

4) On a desktop computer, link the accident sequence functions to the generic interactive program shell. (This requires that the expressions be downloaded from the mainframe to the desktop.) The resulting executable module is a Boolean-expression-scientific program, which, in the present case, uses about 400k of memory. Much of this is consumed by rather

large arrays which are used for storing and sorting results. The present interactive shell was developed at BNL by R. Youngblood. A previous mainframe version of Pairwise was developed at BNL in collaboration with D. Zue and N. Cho; this version computed pair importances, but did not work with classes, and was not interactive.

The time necessary to execute steps 2 through 4 is relatively nominal. The accident sequence analysis can easily consume weeks or months; the labor involved in preparing a Pairwise module is somewhat less than a working day. Several minutes of mainframe time were used in steps 2 and 3. Step 3 could be done on a desktop, but since steps 1 and 2 cannot presently be done on a desktop, there is no incentive to move step 3 to a desktop. Downloading and compilation on the desktop each required over an hour. Once all this is done, of course, sensitivity results can be produced in seconds.

6. Limitations and Approximations

Pairwise uses the rare event approximation

Pairwise does not perform Boolean algebra. It works with the Boolean expression given to it. The trunction process applied to the expression during step 1 must be understood by persons interpreting the results of Pairwise computations.

If an event probability is set to 1, the rare event approximation naturally deteriorates (events are no longer "rare"). Boosting the probabilities of selected events in sensitivity studies can interact somewhat with the truncation process applied in step 1. The rare event approximation always gives an upper bound for the probability of the expression it is working with; as event probabilities are boosted, the overestimate simply gets higher. However, the effect of boosting event probabilities would also have an effect on terms which were truncated away, and this could theoretically affect the conclusions. This was foreseen in the present project, so the Boolean expression was derived using artificially high probabilities for events which were candidates for sensitivity studies; thus, the expression incorporated into Pairwise included all minimal cutsets which would contribute appreciably to top event probability in any sensitivity studies being contemplated.

APPENDIX E

SENSITIVITY CALCULATION DATA

Table E.1. Changes in Core Melt Frequency Due to Human Error Probability Variation by Multiplicative Factors

HUMAN ERRORS	HEP FACTOR								
	1/26	1/20	1/10	1/5	Base Case	x5	x10	x20	x26
All HEs	2.32E-6	2.37E-6	3.18E-6	5.47E-6	7.87E-5	2.60E-3	1.18E-2	2.85E-2	3.10E-2
Recovery HEs	2.51E-5	2.51E-6	2.65E-5	2.98E-5	7.87E-5	5.03E-4	1.19E-3	1.74E-3	1.79E-3
Non-Recovery HEs	9.06E-6	9.23E-6	1.21E-5	1.81E-5	7.87E-5	5.50E-4	1.55E-3	4.08E-3	4.64E-3

Table E.2. Changes in Core Melt Frequency Due to Human Error Probability Variation Over the Range

HUMAN ERRORS	HEP RANGE										
	LB	0.1	0.2	0.3	0.4	BASE	0.6	0.7	0.8	0.9	UB
All HEs	2.32E-6	3.02E-6	5.70E-6	1.22E-5	2.91E-5	7.87E-5	1.71E-4	4.15E-4	1.27E-3	6.92E-3	3.10E-2
Recovery HEs	2.51E-5	2.63E-5	3.02E-5	3.72E-5	5.06E-5	7.87E-5	1.08E-4	1.62E-4	2.85E-4	7.31E-4	1.79E-3
Non-Recovery HEs	9.06E-6	1.16E-5	1.87E-5	2.95E-5	4.71E-5	7.87E-5	1.27E-4	2.20E-4	4.51E-4	1.46E-3	4.64E-3

Table E.3. Changes in Individual Accident Sequence Frequencies Due to HEP Variation by Multiplicative Forces

DOMINANT ACCIDENT SEQUENCE	HEP FACTOR								
	1/26	1/20	1/10	1/5	Base Case	x5	x10	x20	x26
T_6BU	3.91E-9	4.46E-9	3.03E-8	2.22E-7	2.71E-5	1.59E-3	7.70E-3	1.78E-2	1.89E-2
$T_{12}BU$	2.81E-8	3.10E-8	1.25E-7	5.01E-7	1.31E-5	3.97E-4	1.94E-3	3.90E-3	4.10E-3
AX_a	4.29E-7	4.52E-7	7.85E-7	1.50E-6	9.33E-6	4.51E-5	9.53E-5	1.95E-4	2.29E-4
SY_sX_s	3.81E-8	4.01E-8	9.28E-8	2.76E-7	6.04E-6	9.09E-5	2.61E-4	6.63E-4	7.87E-4

Table E.4. Changes in Individual Accident Sequence Frequencies Due to Variation Over the HEP Range

DOMINANT ACCIDENT SEQUENCE	HEP RANGE									
	LB	0.1	0.2	0.3	0.4	BASE	0.6	0.7	0.8	0.9	UB
T_6BU	3.91E-9	2.45E-8	2.58E-7	1.46E-6	6.50E-6	2.71E-5	7.02E-5	1.96E-4	6.72E-4	4.04E-3	1.89E-2
$T_{12}BU$	2.81E-8	1.07E-7	5.53E-7	1.81E-6	4.98E-6	1.31E-5	3.06E-5	7.62E-5	2.27E-4	1.09E-3	4.10E-3
AX_a	4.29E-7	7.23E-7	1.55E-6	2.87E-6	5.12E-6	9.33E-6	1.46E-5	2.41E-5	4.42E-5	1.07E-4	2.29E-4
SY_sX_s	3.81E-8	8.36E-8	3.01E-7	8.74E-7	2.30E-6	6.04E-6	1.12E-5	2.26E-5	5.52E-5	2.21E-4	7.87E-4

Table E.5. Changes in Core Melt Bin Frequencies Due to HEP Variation by Multiplicative Forces

CORE MELT BIN	HEP FACTOR								
	1/26	1/20	1/10	1/5	Base Case	x5	x10	x20	x26
I	1.25E-6	1.26E-6	1.40E-6	1.77E-6	9.88E-6	1.27E-4	3.95E-4	1.22E-3	1.43E-3
II	4.19E-7	4.27E-7	5.61E-7	9.18E-7	9.98E-6	2.59E-4	9.69E-4	2.58E-3	3.12E-3
III	5.87E-8	6.44E-8	2.51E-7	1.07E-6	4.86E-5	2.15E-3	1.03E-2	2.43E-2	2.60E-2
IV	0.0E-0	0.0E-0	2.79E-10	2.42E-9	2.85E-7	7.37E-6	3.09E-5	1.35E-4	1.54E-4
V	1.41E-6	1.41E-6	1.43E-6	1.46E-6	1.89E-6	5.90E-6	1.62E-5	3.61E-5	4.64E-5
VI	4.29E-7	4.52E-7	7.85E-7	1.50E-6	9.33E-6	4.51E-5	9.53E-5	1.95E-4	2.29E-4

Table E.6. Changes in Core Melt Frequency During Accident Conditions

RECOVERY EVENT PROBABILITY	HEP FACTOR								
	1/26	1/20	1/10	1/5	Base Case	x5	x10	x20	x26
0.001	4.61E-6	4.64E-6	5.21E-6	6.48E-6	2.41E-5	1.61E-4	4.56E-4	1.28E-3	1.41E-3
BASE	1.32E-5	1.34E-5	1.63E-5	2.23E-5	7.84E-5	4.00E-4	9.31E-4	2.22E-3	2.40E-3
1.0	2.64E-3	2.65E-3	2.71E-3	2.84E-3	3.80E-3	8.60E-3	1.47E-2	2.27E-2	2.36E-2

Table E.7. Changes in Accident Frequency for T_6BU Sequence During Accident

RECOVERY EVENT PROBABILITY	HEP FACTOR								
	1/26	1/20	1/10	1/5	Base Case	x5	x10	x20	x26
0.001	0.0	0.0	0.0	1.22E-10	5.42E-10	2.61E-9	5.24E-9	1.07E-8	1.13E-8
BASE	2.12E-6	2.18E-6	3.49E-6	6.11E-6	2.71E-5	1.30E-4	2.62E-4	5.35E-4	5.63E-4
1.0	4.24E-5	4.36E-5	6.98E-5	1.22E-4	5.42E-4	2.61E-3	5.24E-3	1.07E-2	1.13E-2

Table E.8. Changes in Core Melt Frequency Due to Operational Errors for Various Assumptions of Recovery Error Probabilities

RECOVERY EVENT PROBABILITY	HEP FACTOR								
	1/26	1/20	1/10	1/5	Base Case	x5	x10	x20	x26
0.001	2.26E-6	2.30E-6	2.89E-6	4.22E-6	2.41E-5	2.36E-4	7.57E-4	2.05E-3	2.40E-3
0.1	2.95E-5	2.96E-5	3.14E-5	3.57E-5	7.80E-5	4.23E-4	1.16E-3	2.93E-3	3.37E-3
0.5	6.64E-4	6.64E-4	6.80E-4	7.10E-4	1.02E-3	2.83E-3	5.88E-3	1.16E-2	1.28E-2
1.0	2.62E-3	2.63E-3	2.68E-3	2.80E-3	3.80E-3	9.60E-3	1.87E-2	3.42E-2	3.67E-2

Table E.9. Sensitivity of Core Melt Frequency to Changes in Categories of Human Error Probabilities

CATEGORIES OF HUMAN ERROR	HEP FACTOR								
	1/26	1/20	1/10	1/5	Base Case	x5	x10	x20	x26
Pre Accident	6.90E-5	6.91E-5	6.95E-5	7.04E-5	7.87E-5	1.54E-4	3.28E-4	6.40E-4	7.92E-4
During Accident	4.74E-6	4.78E-6	5.64E-6	8.04E-6	7.87E-5	2.06E-3	8.15E-3	1.71E-2	1.81E-2
Omission	2.62E-6	2.67E-6	3.48E-6	5.78E-6	7.87E-5	2.58E-3	1.17E-2	2.82E-2	3.07E-2
Commission	7.77E-5	7.77E-5	7.77E-5	7.78E-5	7.87E-5	8.53E-5	9.95E-5	1.48E-4	1.65E-4
Operations	3.94E-6	3.98E-6	4.80E-6	7.13E-6	7.87E-5	2.17E-3	8.73E-3	1.91E-2	2.04E-2
Test/Maintenance	7.19E-5	7.20E-5	7.23E-5	7.30E-5	7.87E-5	1.09E-4	1.50E-4	2.24E-4	2.58E-4
Calibration	7.80E-5	7.80E-5	7.81E-5	7.81E-5	7.87E-5	8.24E-5	8.97E-5	1.12E-4	1.18E-4
Reactor Operators	4.38E-6	4.45E-6	5.79E-6	8.97E-6	7.87E-5	2.07E-3	8.95E-3	2.03E-2	2.16E-2
Non-Lic. Operators	6.31E-5	6.32E-5	6.39E-5	6.53E-5	7.87E-5	1.97E-4	4.68E-4	1.05E-3	1.29E-3
Instru. & Calib. Technicians	7.80E-5	7.80E-5	7.81E-5	7.81E-5	7.87E-5	8.24E-5	8.97E-5	1.12E-4	1.18E-4

Table E.10. Sensitivity of Core Melt Frequency to Changes in Subset of a Category of Human Error Probabilities

SORTS OF HUMAN ERRORS	HEP FACTOR								
	1/26	1/20	1/10	1/5	Base Case	x5	x10	x20	x26
ROs[a] Only	2.36E-5	2.38E-5	2.62E-5	3.12E-5	7.87E-5	3.38E-4	6.98E-4	1.45E-3	1.55E-3
RO/NL[b]	2.55E-5	2.56E-5	2.69E-5	3.01E-5	7.87E-5	4.92E-4	1.15E-3	1.66E-3	1.71E-3
RO/MT[c]	7.49E-5	7.50E-5	7.51E-5	7.55E-5	7.87E-5	9.49E-5	1.16E-4	1.49E-4	1.60E-4

Note:
a) Wholly reactor operator.
b) Reactor operator - non-licensed operator interaction.
c) Reactor operator - maintenance/test personnel interaction.

Table E.11. Sensitivity of Core Melt Frequency to Changes in Pre-Accident Human Error Categories Over the HEP Range

PRE-ACCIDENT HUMAN ERROR CATEGORIES	HEP RANGE										
	LB	0.1	0.2	0.3	0.4	BASE	0.6	0.7	0.8	0.9	UB
Omission	2.44E-6	2.52E-6	2.74E-6	3.08E-6	3.67E-6	4.74E-6	6.65E-6	1.07E-5	2.14E-5	6.85E-5	1.97E-4
Commission	4.61E-6	4.61E-6	4.62E-6	4.64E-6	4.68E-6	4.74E-6	4.86E-6	5.14E-6	6.01E-6	1.07E-5	2.63E-5
Operations	3.94E-6	3.95E-6	3.99E-6	4.08E-6	4.28E-6	4.74E-6	5.65E-6	7.91E-6	1.50E-5	5.13E-5	1.62E-4
Test/Maintenance	3.18E-6	3.25E-6	3.43E-6	3.70E-6	4.10E-6	4.74E-6	5.79E-6	7.65E-6	1.16E-5	2.35E-5	4.51E-5
Calibration	4.67E-6	4.67E-6	4.67E-6	4.68E-6	4.70E-6	4.74E-6	4.81E-6	5.00E-6	5.59E-6	8.96E-6	2.04E-5
Reactor Operators	3.66E-6	3.71E-6	3.84E-6	4.02E-6	4.30E-6	4.74E-6	5.43E-6	6.69E-6	9.43E-6	1.84E-5	3.71E-5
Non-Licensed Operators	3.46E-6	3.49E-6	3.59E-6	3.76E-6	4.08E-6	4.74E-6	6.00E-6	8.87E-6	1.72E-5	5.66E-5	1.71E-4
Instrumentation and Calibration Technicians	4.67E-6	4.67E-6	4.67E-6	4.68E-6	4.70E-6	4.74E-6	4.81E-6	5.00E-6	5.59E-6	8.96E-6	2.04E-5

Table E.12. Relative Contribution to Core Melt Frequency for Various Accident Sequence Types

ACCIDENT SEQUENCE TYPE	HEP RANGE										
	LB	0.1	0.2	0.3	0.4	BASE	0.6	0.7	0.8	0.9	UB
Small LOCA (3)	3.20E-7 8.7	3.88E-7 8.4	6.79E-7 8.4	1.40E-6 8.9	3.17E-6 9.3	7.78E-6 9.0	1.42E-5 7.8	2.87E-5 6.6	7.11E-5 5.4	2.90E-4 4.1	1.04E-3 3.3
Large LOCA (2)	5.97E-7 16.2	9.04E-7 19.5	1.77E-6 22.0	3.15E-6 20.1	5.52E-6 16.2	9.98E-6 11.6	1.56E-5 8.5	2.59E-5 6.0	4.82E-5 3.7	1.21E-4 1.7	2.74E-4 0.9
Very Small LOCA (1)	3.34E-7 9.1	3.89E-7 8.4	5.45E-7 6.8	8.03E-7 5.1	1.26E-6 3.7	2.16E-6 2.5	3.68E-6 2.0	7.07E-6 1.6	1.68E-5 1.3	6.54E-5 0.9	2.23E-4 0.7
SGTR (2)	7.81E-7 21.2	8.37E-7 18.0	1.01E-6 12.5	1.35E-6 8.6	2.13E-6 6.3	4.38E-6 5.1	8.37E-6 4.6	1.96E-5 4.5	6.30E-5 4.8	3.85E-4 5.5	1.88E-3 6.0
LOOP (3)	1.73E-7 4.7	1.96E-7 4.2	2.77E-7 3.4	4.49E-7 2.9	8.32E-7 2.4	1.80E-6 2.1	3.38E-6 1.9	7.55E-6 1.7	2.32E-5 1.8	1.49E-4 2.1	8.26E-4 2.7
LOIA (5)	2.25E-8 0.6	7.33E-8 1.6	4.48E-7 5.6	2.02E-6 12.9	8.01E-6 23.5	3.12E-5 36.1	7.85E-5 42.9	2.15E-4 49.4	7.23E-4 55.2	4.27E-3 60.9	1.98E-2 63.5
FW Line Break (1)	1.27E-8 0.4	4.44E-8 1.0	2.17E-7 2.7	6.90E-7 4.4	1.86E-6 5.5	4.74E-6 5.5	8.44E-6 4.6	1.55E-5 3.6	3.20E-5 2.4	8.76E-5 1.3	2.01E-4 0.6
LOSW (3)	2.99E-8 0.8	1.14E-7 2.5	5.88E-7 7.3	1.92E-6 12.2	5.30E-6 15.5	1.40E-5 16.2	3.27E-5 17.9	8.20E-5 18.9	2.46E-4 18.8	1.20E-3 17.1	4.60E-3 14.7
LOFW (1)	2.95E-9 0.1	9.42E-9 0.2	4.42E-8 0.6	1.44E-7 0.9	4.13E-7 1.2	1.18E-6 1.4	2.69E-6 1.5	7.09E-6 1.6	2.57E-5 2.0	2.05E-4 2.9	1.35E-3 4.3
Other	1.4	1.2	1.5	1.9	1.8	1.5	1.5	1.5	1.8	2.3	2.7
Vessel Rupture (1)	1.1 E-6 29.9	1.1 E-6 23.7	1.1 E-6 13.7	1.1 E-6 7.0	1.1 E-6 3.2	1.1 E-6 1.3	1.1 E-6 0.6	1.1 E-6 0.3	1.1 E-6 0.1	1.1 E-6 0.02	1.1 E-6 0.004
ATWS	2.55E-7 6.9	5.22E-7 11.3	1.25E-6 15.5	2.37E-6 15.1	3.90E-6 11.4	6.63E-6 7.7	1.13E-5 6.2	1.86E-5 4.3	3.51E-5 2.7	8.42E-5 1.2	1.72E-4 0.6

APPENDIX F

CUTSET ANALYSIS

F.1 Introduction

A detailed analysis was performed to characterize the human errors in the cutsets of the dominant accident sequences. This cutset analysis was performed for two reasons:

1) To obtain an understanding of which types of human errors contributed to which sequences. This understanding then could be used to help guide the sensitivity calculation strategy.

2) To obtain a set of results, independent from the sensitivity calculations, which would define the types of important human errors and the specific individual errors that were very important.

In general, the cutset analysis was found to be beneficial. The initial results helped to guide the sensitivity calculations. The final cutset analysis agreed quite well with the final sensitivity evaluations in describing important types and groups of human errors. This provided two different techniques which produced consistent results. Also, the cutset analysis provided the specific individual human errors which tended to dominate risk in each sequence.

F.2 Cutset Analysis for Top Three Sequences

F.2.1 Loss of Instrument Air (T_6BU)

Sequence T6BU has a total base case frequency of 2.71E-5 events per year (events/year). Unlike most sequences, every cutset in T6BU has human error. Two recovery errors, RESSFW30 and REIA1, are sequence dependent and occur in all 178 cutsets for this sequence. An examination of the cutsets for this sequence revealed that the top 20 cutsets containing human error events account for a total cutset frequency of 2.64E-5/year. The total frequency for all cutsets, in the top 20, containing double human errors is 3.18E-7/year. The total frequency for all cutsets in the top 20 containing triple human errors is 2.44E-5/year. All the cutsets in the top 20, containing quadruple human errors, have a frequency of 1.67E-6/year.

Table F.1 lists which human errors occur in the top 178 cutsets of sequence T6BU, as well as how many times each error occurs and which cutset it first appears in. Table F.2 lists the various combinations of these errors within the top 20 cutsets. Finally, Table F.3 provides a brief description of those human errors that appear to be the driving forces in the sequence of T6BU. While this sequence has an exceptional amount of human error terms, the other transient sequences at Oconee also appear very sensitive human error probability variation.

F.2.2 Loss of Low Pressure Service Water (T_{12}BU)

Sequence T12BU has a base case frequency of 1.31E-5. An examination of the cutsets for this sequence revealed that all 21 dominant cutsets contain human errors. A recovery error, RESSFSI, is sequence dependent and occurs in all 21 cutset terms for this sequence. The total frequency for all cutsets in the top 21 terms containing double human errors is 1.24E-5. The total frequency for all cutsets in the 21 terms containing triple human errors is 6.99E-7.

Table F.1. Human Errors in T6BU

Human Errors in T6BU	# of Times Error Occurs in T6BU	# of First Cutset Term Error Appears
RESSFW30	178	1
REIA1	178	1
HP2425MVH	28	7
LP28VVCH	24	14
UTHPIH	22	1
EFTDPP1H	13	4
PLS1H	12	7
SW169VVH	8	38
HPLDSTH	6	78
BMUH	5	26
EF88VVH	4	17
CW156MVH	4	15
SWH246VVH	2	101
CW157VVH	1	41
CW157VV1H	1	92

Table F.2. Combinations of Human Errors in Top 20 Cutsets of T6BU

Combinations of Doubles in Top 20 Cutsets

RESSFW30, REIA1 (Occurs Twice)

Combinations of Triples in Top 20 Cutsets

RESSFW30, UTHPIH, REIA1 (Occurs Nine Times)

RESSFW30, RESSFW30, PLS1H, REIA1 (Occurs Twice)

HP2425MVH, RESSFW30, REIA1 (Occurs Once)

Combinations of Quadruples in Top 20 Cutsets

RESSFW30, UTHPIH, REIA1, EFTDPP1H, (Occurs Once)

HP2425MVH, RESSFW30, PLS1H, REIA1 (Occurs Twice)

RESSFW30, PLS1H, REIA1, LP28VVCH (Occurs Once)

RESSFW30, UTHPIH, REIA1, CW156MVH (Occurs Once)

RESSFW30, UTHPIH, REIA1, EF88VVH (Occurs Once)

Table F.3. Description of 7 Human Errors in T6BU That Dominate Sequence Frequency

RESSFW30 — Failure of the operating staff to initiate the Safe Shutdown Facility (SSF) to provide feedwater (FW) within 30 minutes (after loss of all FW due to power loss and failure of emergency feedwater turbine-driven pump).

Oconee has defined this error as important. It has an HEP of 1E-1. This is a recovery event. Activity is operations. Event type is operator fails to recover. Timing is during. Personnel is RO/NL. Error of omission.

REIA1 — Failure to recover Instrument Air (IA) in one hour. This is a BNL review of Oconee PRA addition. It has an HEP of 5E-5 and is also a recovery event. Activity is operations. Event type is operator fails to recover (implicit). Timing is during. Personnel is RO/NL. Error of omission.

HP2425MVH — MOVs 3HP-24 and 3HP-25 left unavailable. Oconee has defined this error as important as well as dependent. It has an HEP of 5E-5. Activity is Test/Maintenance. Event type is Unavailable. Timing is Pre-accident. Personnel is RO/MT. Error of omission.

LP28VVCH — BWST valve left closed. Oconee has defined this error as important. It has an HEP of 2.8E-5. Activity is Test/Maintenance. Event type is unavailable. Timing is Pre and Personnel is RO/MT. Error of omission.

UTHPIH — Operator fails to attempt HPI cooling. Oconee has defined this error as important. It has an HEP of 2.0E-2. Activity is operations. Event type is operator fails (implicit). Timing is during. Personnel is RO. Error of omission.

PLS1H — Operator fails to reapply LC 3X1 following load shed. This error has an HEP of 1E. Activity is operations. Event type is operator fails to recover. Timing is during. Personnel is RO. Error of omission.

EFTDPP1H — TD EFW pump not restored after T or M. This error has an HEP of 1E-2. Activity is Test/Maintenance. Event type is unavailable. Timing is Pre. Personnel is RO/MT. Error of omission.

Table F.4 lists which human errors occur in the top 21 cutset terms of sequence T12BU, as well as how many times each error occurs and which cutset it first appears in. Table F.5 gives a list of the various combinations these errors exist in within the top 21 cutset terms. Finally, Table F.6 provides a brief description of those human errors that appear to be the driving forces in the sequence T12BU.

Table F.4. Human Errors in T12BU

Human Errors in T12BU	# of Times Error Occurs in T12BU	# of First Cutset Term Error Appears
RESSFSI	21	1
RESW12	11	2
SW3BPPSH	7	7
CW157VVIH	4	3
SWH247VVH	1	1
CW391MVIH	1	8
SWL89VVH	1	10
HPBCPPH	1	17
HPCROSSH	1	21

Table F.5. Combinations of Human Errors in 21 Cutset Terms of T12BU

Combinations of Doubles in 21 Cutset Terms

SWH247VVH, RESSFSI (Occurs Once)
RESW12, RESSFSI (Occurs Eight Times)
SW3BPPSH, RESSFSI (Occurs Four Times)
HPBCPPH, RESSFSI (Occurs Once)
HPCROSSH, RESSFSI (Occurs Once)

Combinations of Triples in 21 Cutset Terms

RESW12, CW157VVIH, RESSFSI (Occurs Once)
RESW12, RESSFSI, CW391MVIH (Occurs Once)
SW689VVH, RESW12, RESSFSI (Occurs Once)
SW3BPPSH, CW157VVIH, RESSFSI (Occurs Three Times)

Table F.6. Description of 5 Errors in T12BU That Seem To Drive Sequence

RESSFSI — Failure of the operating staff to initiate SSF seal injection in approximately 30 minutes following a loss of normal seal injection (HPI pumps).

Oconee has defined this error as important. It has an HEP of 1E-1. This is a recovery event. Activity is Operations, Event type is operator fails to recover. Timing is during. Personnel is RO/NL. Error of omission. NRCPGM = OPS/P/TR,OPP

RESW12 — Failure of operating staff to recover LPSW from another source before failure of all HPI pumps. Failure probability includes two contributions: P(Operator fails to properly cycle HPI pumps to prevent overheating) and P(Operator fails to get service water including other units and HPSW) = $(8.0 \times 10^{-3}) + (5.0 \times 10^{-3})$.

Table F.6. Continued

	This value assumes the ES signal is not present to start the HPI pumps. This recovery does not apply to the failure of LPSW due to LPSW108, or to the cutset [T14 * SW3BPPSH] which describes a misdiagnosis of the event. Oconee has defined this error as important. It has an HEP of 1.4E-2. This is a recovery event. Activity is Operations. Event type is operator fails to recover. Timing is during. Personnel is RO/NL. Error of omission. NRCPGM=OPS/P/TR.
SW3BPPSH	Operator fails to start pump B. Oconee has defined this error as important. It has an HEP of 8E-3. Activity is Operations, Event type is operator fails, Timing is during. Personnel is RO. Error of omission. NRCPGM=-OPS/P/TR.
CW157VVIH	MV 3C-157 not closed. This error has an HEP of 1E-4. Activity is Test and/or Maintenance. Event type is unavailable. Timing is pre. Personnel is NL/MT. Error of omission. NRCPGM=-TR/OPS/P.
SWH247VVH	Manual valve HPSW-247 not opened by operator. This error has an HEP of 1.8E-2. Activity is operations. Event type is Operator fails to recover. Timing is during. Personnel is NLO, Error of omission. NRCPGM=OPS/P/TR.

F.2.3 Large Break LOCA (AXA)

Sequence AXa has a base case frequency of 9.32E-6. An examination of the cutsets for this sequence revealed that the top 40 cutsets containing human errors have a total frequency of 9.13E-6. The total frequency for all cutsets in the top 40 containing double human errors is 9.33E-6. The total frequency for all cutsets in the top 40 containing triple human errors is 4.19E-9.

Table F.7 lists which human errors occur in the top 124 cutsets of sequence AXa, as well as how many times each error occurs and which cutsets it first appears in. Table F.8 gives a list of the various combinations these errors exist in within the top 40 cutset terms. Finally, Table F.9 provides a brief description of those human errors that appear to be the driving forces in the sequence AXa.

Table F.7. Human Errors in AXa

Human Errors in AXa	# of Times Error Occurs in AXa	# of First Cutset Term Error Appears
LPFLOWH	16	2
LPBPPH	10	50
LPAPPH	10	52

Table F.7. Continued

EPS220000H	9	23
EPS210000H	9	24
EPS120000H	9	25
EPS110000H	9	22
LP18MVRH	8	23
LP8MVH	8	20
LP5MVH	8	19
LP17MVRH	8	22
LPBPPMH	7	38
LPAPPMH	7	39
LP19MVRH	7	12
LP20MVRH	7	13
PS29ARSWH	6	31
PX334ARSWH	6	32
LP40VVH	5	7
LP20MVMH	4	42
LP19MVMH	4	43
LP17MVMH	4	73
LP18MVMH	3	74
SW3BFPH	1	75
PXS2F3ASWH	1	33
P3XSISWH	1	34
XALPRH	1	1
SW7778CMH	1	28
LWD99103VVH	1	4
LPTHROTTLE	1	2
LP4142VVH	1	7
LPABH	1	5
ES56MT	1	28
ES578CM	1	28

Table F.8. Combinations of Human Errors in Top 40 Cutset Terms of AXa

Combinations of Doubles in Top 40 Cutset Terms

LPFLOWH, LPTHROTTLE (Occurs Once)
LP40VVH, LP4142VVH (Occurs Once)
LP19MVRH, LP20MVRH (Occurs Once)
LP5MVH, LPFLOWH (Occurs Once)
LP8MVH, LPFLOWH (Occurs Once)
LP17MVRH, EPS110000H (Occurs Once)
LP18MVRH, EPS220000H (Occurs Once)
LP17MVRH, EPS120000H (Occurs Once)
LPFLOWH, LPAPPMH (Occurs Once)

Combinations of Triples in Top 40 Cutset Terms

ES578CM, ES56MT, SW7778CMH (Occurs Once)

Table F.9. Descriptions of 8 Errors in AXa that Seem to Drive Sequence

LPFLOWH	High flow (>4200 gpm) in A loop (large LOCA). This errors has an HEP of 1E. Activity is Operations; event type is operator fails. Timing is during. Personnel is RO. Error of omission. NRCPGM=TR,OPS/P.
LP19MVRH	Operator fails to open for recirculation. This error is dependent. It has an HEP of 3E-3. Activity is Operations; timing is during. Personnel is RO. Error of omission. NRCPGM = OPS/P/TR.
LP20MVRH	Operator fails to open valve for recirculation. This error has an HEP of 3E-3. Activity is operations. Event type is operator fails. Timing is during. Personnel is RO. Error of omission. NRCPGM=OPS/P/TR.
LP40VVH	Valve left open. This error is dependent. It has an HEP of 1E-3. Activity is operations. Event type is unavailable. Personnel is NLO. Error of Omission. NRCPGM=OPS/P/SRO.
	Oconee has defined this error as important. It has an HEP of 1.4E-2. This is a recovery event. Activity is Operations. Event type is operator fails to recover. Timing is during. Personnel is RO/NL. Error of omission. NRCPGM=OPS/P/TR.
XALPRH	Operator fails to attempt LPR in 30 minutes. Oconee has defined this error as important. It has an HEP of 5E-3. Activity is operations. Event type is Operator fails (implicit). Timing is during. Personnel is RO. Error of Omission. NRCPGM=OPS/P/TR.
LWD99103VVH	Drain valve not restored. Oconee has defined this error as important and it is also dependent. It has an HEP of 6E-4. Activity is Test/Maintenance. Event type is unavailable. Timing is Pre. Personnel is NL/MT. Error of omission. NRCPGM=OPS/P/SW.
LPTHROTTLE	Operator fails to throttle flow. Oconee has defined this error as important and it is also dependent. It has an HEP of 3E-3. Activity is operations. Event type is operator fails. Timing is during. Personnel is RO. Error of omission. NRCPGM=OPS/P/TR.
LPABH	Operator inhibits/fails system. This error has an HEP of 1E-4. Activity is operations. Event type is operator inadvertent. Timing is during. Personnel is RO. Error of commission. NRCPGM=OPS/P/TR.

F.3 Recovery Error Analysis

Since recovery errors were found to play such a dominant role in risk at Oconee, a separate analysis was performed of the recovery errors in the cutsets. Each of the dominant cutsets in all 25 analyzed accident sequences was reviewed to determine which cutsets and which human events were important.

Each was also reviewed to see specifically what human actions were involved and if the HEP could reasonably be increased together. No examples were found where HEPs of HEs in the same cutsets could not be increased together. That is, it is judged generally acceptable to increase all the HEPs together for the sensitivity study.

Based on an analysis and review of dominant cutsets in all 25 accident sequences, the 20 Recovery Errors used in the sensitivity analysis can be arranged as shown in Table F.10.

Table F.10. Relative Importance of Recovery Errors

Most Important	Moderate Importance	Least Important
RESSFW30	RESW78	RESWLPI
REIA1	RESUMPMF	REFEEDAIR2
REIA2	REDHRSUC	REFEEDAIR12
REHPPPCS	RELPD16	RESUBAIR12
RESUBAIR90	RESUBAIR2	
RESSFSI	REFDW1	
RESW12	REFDW2	
	REIA12	
	RESSFW12	

Table F.11 provides a short description of each recovery error, along with its base case unavailability value or HEP.

Table F.11. 20 Dominant Recovery Errors Used in Sensitivity Study

Event or Error	Unavailability	(HEP) Description
REFDW1	0.5	Failure of the operating staff to recover FW in 30 minutes; one source available for recovery.
REIA1	0.5	Failure to recover IA in one hour.
REIA2	0.3	Failure of the operating staff to recover IA in 2 hours. Based on analysis of potential failure mode and operator actions required to recover.
REFDW2	0.3	Failure of the operating staff to recover FW in 30 minutes; 2 sources available for recovery.
RESWLPI	0.2	Failure of the operating staff to recover failures that lead to isolation of LPSW to LPI coolers. Value based on essentially perfect recovery of failures that can be recovered.

Table F.11. Continued

Event or Error	Unavailability	Description
REDHRSUC	0.1	Failure of the operating staff to open LPI suction MOVs for DHR, given failure of remote operation. Value is based on 5% of failures being non-recoverable.
RESSFW30	0.1	Failure of the operating staff to initiate SSF to provide FW within 30 minutes (after loss of all FW due to power loss and failure of EFW TD pump)
RESUMPMF	0.1	Failure of operating staff to find and isolate leakage from sump via LWD99 and 103 before HPI pump motors flooded (does not apply in cases where IA is lost, due to loss of indication and alarm on HAWT level)
RESSFSI	0.1	Failure of the operating staff to initiate SSF seal injection in approximately 30 minutes following a loss of normal seal injection (HPI pumps)
RELPD16	0.1	Failure to recover suction of RHR pumps.
RESW78	5.0E-2	Failure to open valve SW-78 locally.
REHPPPCS	5.0E-2	Failure of operators to protect standby HPI pumps by allowing them to remain idle when suction unavailable.
RESUBAIR90	2.5E-2	Failure to recover offsite power in 90 minutes, and to reload the load-shed IA air compressors following a loss of offsite power due to substation failure.
RESUBAIR2	2.2E-2	Failure to recover offsite power in two hours, and to reload load-shed air compressors following a loss of offsite power due to substation failure.
RESW12	1.3E-2	Failure of operating staff to recover LPSW from another source before failure of all HPI pumps. Failure probability includes two contributions: P (operator fails to properly cycle HPI pumps to prevent overheating) + P (operator fails to get service water including other units and HPSW) = $(8.0 \times 10^{-3}) + (5.0 \times 10^{-3})$

Table F.10. Continued

Event or Error	Unavailability	Description
		This value assumes the ES signal is not present to start the HPI pumps. This recovery does not apply to the failure of LPSW due to LPSW108, or to the cut set [T14 * SW3BPPSH] which describes a misdiagnosis of the event.
REFEEDAIR2	1.2E-2	Failure of the operating staff to recover power, and to reload the air compressors in 2 hours following a loss of offsite power due to failure of the feeders. Power through the Keowee overhead is a successful recovery. Failure is defined by: P (Keowee overhead unavailable) *P (nonrecovery of offsite power) *P (failure to load compressors on CT4 or CT5) + P (failure to reload compressors given adequate power) = (3.6×10^{-2}) (0.44) (0.1) + $(1.0 \times 10^{-2}) = 1.2 \times 10^{-2}$ Keowee overhead unavailability is dominated by maintenance on one or both Keowee units
RESUBAIR12	4.0E-3	Failure to recover offsite power in 12 hours, and to reload load-shed air compressors following a loss of offsite power due to substation failure.
RESSFW12	3.5E-3	Failure of the operating staff to initiate FW from the SSF within 12 hours following the initiating event = P (operator fails to initiate) + P (SSF fails due to hardware) = $(1.0 \times 10^{-3}) + (2.5 \times 10^{-3})$
REIA12	2.4E-3	Failure of the operating staff to recover IA in 12 hours. Based on review of potential failure modes and operator actions required to recover each.
REFEEDAIR12	1.2E-3	Failure of the operating staff to recover offsite power and reload IA within 12 hours of power failure caused by grid loss or feeder failure. The failure probability is determined by: P (Keowee overhead unavailable) *P (nonrecovery of offsite power) *P (failure to load compressors on CT4 or CT5)

Table F.11. Continued

Event or Error	Unavailability	Description
		+ P (failure to reload compressors given adequate power) = (3.6×10^{-2}) (0.22) (0.1) + $(1.0 \times 10^{-3}) = 1.8 \times 10^{-3}$

NOTES

1. $5.0E-2 = 5 \times 10^{-2}$

2. Recovery unavailability is the probability of failing to recover. Thus, a recovery unavailability of 1.0 means no recovery, and a recovery unavailability of 0 means perfect recovery.

3. IA = Instrument Air
   LPSW = Low Pressure Service Water
   FW = Feedwater
   LPI = Low Pressure Injection
   DHR = Decay Heat Removal
   SSF = Standby Shutdown Facility

An additional area of interesting results produced from this recovery error cutset analysis is the appearance of multiple errors in a single cutset. Six sequences contained cutsets above 1E-7 that had multiple human errors in them (including at least one recovery error). Table F.12 lists these sequences.

Table F.12. Sequences with Cutsets Containing Multiple Human Errors (HEs) (Including at least one recovery error (RE))

Sequence Name	# Cutsets With Double REs	# Cutsets with Triple HEs	# Cutsets with Quadruple HEs
SYSXS	0	1	0
RXRO	0	1	0
T5BU	0	1	0
T6BU	18	10	6
T12BU	2	1	0
TBLX	1	1	0
TOTAL:	21	16	6

where:

SYSXS - Small Break LOCA Sequence
RXRO - Steam Generator Tube Rupture Sequence
T5BU - Loss of Offsite Power Sequence
T6BU - Loss of Instrument Air Sequence
T12BU - Loss of Service Water Sequence
TBLX - Transient Sequence

One can see from this table that certain individual cutsets with multiple HEs, will be extremely sensitive to increased in the HEPs, since all HEPs are increased together in some of the sensitivity calculations. As an example of one of the cutsets from T6BU is T6*RESSFW30*REIA1*UTHPIH*EFTDPPIH = 1.05E-6 with base case values as follows: (.21) * (.1) * (.5) * (.01) * (.01) = 1.05E-6. If each of the HEs is increased by 20 times (not to exceed 1.0), this cutset value alone increases to 8.4E-3, which is about 100 times the base case total core melt frequency.

In addition to the recovery errors found important, the non-recovery HEs (listed in Table F.13), which appeared in cutsets along with recovery errors, were found to be important. UTHPIH was found to be particularly important.

Table F.13. Important Non-recovery Human Errors

CW156MVH	PLSIH
CW157VVIH	QHPIH
CW391MVIH	OBWSTH
EF88VVH	RC417VCH
EFTDPPIH	RC418VCH
HP2425MVH	SW7778CMH
LP28VVCH	SWEXCESSH
LWD99103VVH	UTHPIH
	XHPR12H

Event UTHPIH represents failure of the operators to make the decision to initiate HPI cooling in the event that feedwater is unavailable to the steam generators.