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Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development

Prepared by W. B. Andrews, R. H. V. Gallucci, S. W. Heaberlin, W. E. Bickford, G. J. Konzek, D. L. Strenge, R. I. Smith, S. A. Weakley

Pacific Northwest Laboratory
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ABSTRACT

Pacific Northwest Laboratory has developed a methodology, with examples, to calculate--to an approximation serviceable for prioritization purposes--the risk, dose and cost impacts of implementing resolutions to reactor safety issues. This report is an applications guide to issue-specific calculations. A description of the approach, mathematical models, worksheets and step-by-step examples are provided.

Analysis using this method are intended to provide comparable results for many issues at a cost of two staff-weeks per issue. Results will be used by the NRC to support decisions related to issue priorities in allocation of resources to complete safety issue resolutions.

PREFACE

This report was prepared by the Pacific Northwest Laboratory (PNL) to communicate results of the Prioritization of Safety Issues Project. This project has an objective to develop a methodology that can be used to quantify risk, dose and cost impacts of resolutions to reactor safety issues and apply it to issues of interest to the NRC. Results of this project will be used by the NRC to support, in part, decisions on resource allocation to resolve specific issues.

This volume of NUREG/CR-2800 contains a description of the general approach to the development of safety issue information and three example analyses. Future supplements to this volume are planned to document analyses of specific issues.

ACKNOWLEDGEMENT

This methodology to develop information on reactor safety issues benefited from "in the field" reviews of the draft instructions by contributors to issue analyses in the Energy Systems, Engineering Physics, Radiation Science and Materials departments at the Pacific Northwest Laboratory. These reviews resulted in practical improvements that have been incorporated in this final version and are gratefully acknowledged.

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1.0 SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Reactor Regulation (NRR) is implementing a plan for early resolution of safety issues related to operating reactors, reactors under construction and standard designs. The Pacific Northwest Laboratory was contracted to develop and implement a method to quantify risks, doses, and costs associated with specific safety issues in support of the plan. This information, along with other subjective factors, will be used by the NRR to rank issues for further investigation and possible implementation.

Currently, the NRC encourages the quantification of safety benefits in terms of man-rem of exposure averted, where possible, using a risk-based approach. A risk model is constructed for representative PWR and BWR plants to consider issues that reduce reactor accident frequencies or release quantities. The use of man-rem as a measure of consequences in the model will also allow the consideration of issues related to protective actions following a release of radioactive material, as well as preventive and mitigative measures.

Risk and dose are divided into public and occupational categories. Public risk reduction is defined to be the incremental reduction in expected public dose due to the implementation of a safety issue resolution (SIR). One measure of occupational dose is defined to be the incremental occupational dose due to the implementation, maintainance and operation of the SIR. The expected value of occupational dose avoided for a reduction in accident frequency is quantified so that it can be used, if needed, in decision-making.

Costs associated with implementing a SIR are divided into industry and NRC categories. Industry costs are defined to be incremental costs associated with the implementation, operation and maintenance of the SIR. The expected value of avoided accident costs due to a reduction in accident frequencies over the remaining life of affected plants is quantified for potential use in decision-making. NRC costs are defined to be future NRC costs associated with the development of a SIR, the review of industry implementation actions associated with SIR compliance, and ongoing reviews of the licensee to assure proper maintenance and operation. These costs may be positive or negative (savings).

A five-step procedure is used in the PNL methodology to develop risk, dose, and cost information on a specific issue. The first step is to obtain information on the safety issue and determine which plants are affected. Components of representative plant risk equations are examined to determine which may be affected. The second step is to obtain or postulate a SIR. This is done by consultation with the responsible component of the NRC. In addition, this step includes the review of applicable literature. Step 3 is to estimate the effect on the risk equations of the SIR and then calculate public risk reduction and occupational dose, including uncertainties. This is accomplished by estimating a change in applicable terms in the risk equation and then measuring the incremental risk reduction for the representative plant due to the change. Occupational doses are evaluated for both the decrease due to accident avoidance and the increase from SIR implementation, operation, and maintenance. A standardized approach to uncertainty estimates is provided for use, where applicable. Industry totals are calculated by multiplying this result by the number of affected plants and their remaining lifetimes. Calculation of costs is the fourth step. Engineering costs, projected industry and NRC labor levels, and incremental plant down-time are estimated for the SIR. These are used with appropriate scaling factors and accident frequency reduction estimates from step 3 to calculate industry and NRC costs, including error bounds. Step 5 is the presentation of results for use by the NRC. Work sheets are developed for each step in the calculations to facilitate documentation and consistent analyses.

The relatively large number of issues requires that the methodology emphasize development of defensible risk, dose and cost estimates at a modest cost. Each issue considered can be completed with 2-3 staff-weeks of effort using these methods. Results are intended only for use by the NRC to allocate resources for future study. Additional, more detailed, analyses are required for decisions related to actions on specific issues.

Cost and risk information is developed for three issues to provide examples of the method: Training and Qualifications of Operations Personnel--Issue I.A.2.2; Diesel Generator Reliability--Issue B-56; Steam Line Break with Consequential Small LOCA--Issue 18. Sample results are shown in Table 1.1. These results indicate potential for public risk reductions and both decreases and increases in occupational dose. NRC costs are positive, but cost savings may accrue to industry due to the accident cost avoided.

Additional analyses are planned for other safety issues. If done consistently, these analyses can provide quantitive input for use in NRC prioritization decisions regarding safety issues.

TABLE 1.1. Risk, Dose and Cost Results for Example Safety Issues

Result[a]	Issue I.A.2.2	Issue B-56	Issue #18
RISK/DOSE (man-rem)			
Public Risk Reduction	1.5E+5	5.8E+4	1500
	(0;2.3E+7)	(0;2.4E+6)	(0;5.3E+4)
Occupational Doses:			
Implementation	0	0	420
Operation/Maintenance	-2.3E+5	0	7800
Total of Above	-2.3E+5 (-6.9E+5;-7.6E+4)	0 (0;0)	8200 (2700;2.5E+4)
Accident-Avoidance	950 (0;2.9E+4)	350 (0;2900)	11 (0;80)
COSTS ($\$10^6$)			
Industry:			
Implementation	45	16	19
Operation/Maintenance	610	30	35
Total of Above	650 (350;960)	46 (29;63)	54 (34;74)
Accident-Avoidance	78 (0;2400)	29 (0;240)	0.94 (0;6.7)
NRC:			
Development	.055	0	0.17
Implementation Support	.055	0.12	0.20
Operation/Maintenance Review	2.8	0.69	2.9
Total of Above	2.9 (1.5;4.4)	0.81 (0.46;1.2)	3.3 (1.8;4.8)

(a) Best estimate is given with lower and upper bounds, respectively, in parenthesis where calculated.

2.0 INTRODUCTION

This report documents a methodology used by the Pacific Northwest Laboratory[(a)] to provide the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Reactor Regulation (NRR) with information to use in prioritizing safety issues related to nuclear power plants. The objective of this methodology is to provide users with a set of assumptions and analysis tools that, if properly applied to specific safety issues, will yield consistent quantitative estimates of safety costs and benefits. These estimates can then be compared, along with other subjective factors, by the NRC to rank issues for further investigation or possible implementation.

2.1 BACKGROUND

The TMI Action Plan (NUREG-0660, Section IV.E) called for the development of a plan for the early resolution of safety issues, including application of the plan's solutions to problems dealing with operating reactors, reactors under construction, and standard designs. The plan was to address the following objectives:

1. identify possible safety issues through evaluation of operating experience, results of safety-related research, results of risk assessment analyses, licensing reviews by the NRC staff and the Advisory Committee on Reactor Safeguards (ACRS), and public allegations;
2. identify those issues that are deemed to have substantial potential for adverse impacts on safety;
3. identify explicit time requirements for notifying boards of these issues;
4. develop a timely program for evaluating the significance of each issue and determining any appropriate resolution, including realistic evaluations of expected plant reponses to combinations and

(a) Operated by Battelle Memorial Institute.

permutations of events or potential failure sequences and the subseuent course, consequences, and probabilities of possible accidents;

5. develop recommended changes to the regulations, Standard Review Plan, review method, and/or inspection procedures to implement any necessary criteria resulting from the evaluation of the problem, including criteria for modification of standardized design;

6. develop a management and quality assurance program to assure the effective and reasonable implementation of the program and effective interaction with the industry and the public.

This document is the result of activities conducted under objective 4 to establish priorities among reactor safety issues.

The NRC plan for early resolution of safety issues requires the prioritization of issues using a numerical index for each issue. This report develops information for use in a priority index for a given issue through an evaluation of the public risk reduction and occupational doses associated with the safety issue resolution (SIR) requirements, and the predicted cost to NRC and the industry resulting from the proposed change.

2.2 PRIORITIZATION INFORMATION DEFINITION

The NRC objective in establishing priorities for safety issues is to use NRC and industry resources to produce the greatest safety benefits at a reasonable cost. Numerous subjective judgments are required to properly implement the management plan. For this reason, it was decided to develop as many pieces of information germane to the safety benefits and costs of each issue that could be completed within a several man-week effort. This will allow NRC to consider current and future prioritization criteria.

Information important to the evaluation of an issue resolution includes the potential reduction in the risk to the public and the dose to power plant site workers. Man-rem is chosen as the risk/dose measure for simplicity and for convenient relationship with most safety effects. Models used to calculate man-rem allow the consideration of issues that affect both the frequency and consequence parameters of risk.

2.2.1 Public Risk Reduction

The public risk reduction term is defined as the product of the number of plants affected by the SIR, the average remaining life of the plants and the average risk reduction due to offsite releases from accidents. This can be stated as:

$$(\Delta W)_{Total} = \begin{bmatrix}\text{affected portion of} \\ \text{public risk before} \\ \text{issue resolution}\end{bmatrix} - \begin{bmatrix}\text{affected portion of} \\ \text{public risk after} \\ \text{issue resolution}\end{bmatrix}$$

$$= N\bar{T}(\Delta W) \text{ in man-rem}$$

where N = number of rectors affected by the SIR

$\bar{T}$ = average remaining operating life of reactors afected (years)

$\Delta W = \Delta(FR)$ = change, due to he SIR, in the product of estimated time frequency of accidents in (reactor-years)$^{-1}$ and public consequences per accident in man-rem for an average plant.

2.2.2 Occupational Dose

Occupational dose has two components: the incremental dose increase from implementation and operation/maintenance (O/M) of the SIR, and the dose avoided by lowering the accident frequency. The incremental dose from SIR implementation and O/M can be stated as follows:

$$G = \text{occupatinal dose increase due to implementation and O/M of the SIR}$$

$$= N(D_o\bar{T} + D) \text{ in man-rem}$$

where N = number of reactors affected by the SIR

$\bar{T}$ = average remaining operating live of reactors affected (years)

D_o = annual incrementation dose increase due to O/M of the SIR (man-rem/reactor-year)

D = incremental dose increase due to implemenation of the SIR (man-rem/reactor).

The accident-related occupational dose reduction, like public risk reduction, has both probability and consequence components:

ΔU = change, due to the SIR, in the accident-frequency-weighted occupational dose from cleanup and repair of a reactor following an accident (man-rem)

$$= N\bar{T}\,\Delta(FD_R)$$

where N = number of reactors affected by the SIR

$\bar{T}$ = average remaining operating life of reactors affected (years)

$\Delta(FD_R)$ = change, due to the SIR, in the product of estimated time frequency of accidents in $(\text{reactor-years})^{-1}$ and occupational dose due to cleanup and repair of the reactor following an accident (man-rem).

2.2.3 Costs

Costs incurred for implementing the SIR include: 1) the cost to the NRC for developing each requirement and reviewing the utility's design to assure that the requirement is properly implemented, operated, and maintained; and 2) the utility's cost of design, procurement, installation, and testing to implement the requirement and its cost for O/M. Accident-avoidance results in cost savings to the utility. Information on both NRC and industry costs is considered since both represent costs that are paid by the public, either as taxpayers or ratepayers. Only future costs are relevant to current decisions, so sunk costs are ignored.

2.2.3.1 NRC Costs

NRC costs are divided into three components. The first two are forward-looking SIR development and implementation support costs. The third is annual O/M review costs for the issue resolution. NRC costs can be stated mathematically as:

$(S_N)_{Total}$ = Future costs to the NRC for SIR development, support of SIR implementation, and review of SIR O/M ($\$10^6$)

$$= C_D + N(\bar{T}C_o + C)$$

where N = number of plants affected by the SIR

$\bar{T}$ = average remaining operating life of reactors affected (years)

C_D = future NRC costs for SIR development ($\$10^6$)

C_o = annual incremental NRC costs for annual review of SIR O/M ($\$10^6$/reactor-year)

C = incremental NRC costs for support of SIR implementation ($\$10^6$/reactor).

2.2.3.2 Industry Costs

Industry costs are defined as follows:

S_I = future costs to the industry for SIR implementation and O/M ($\$10^6$)

$= N(\bar{T}I_o + I)$

where N = number of reactors affected

$\bar{T}$ = average remaining operating life of reactors affected (years)

I_o = annual incremental industry costs for SIR O/M ($\$10^6$/reactor-year)

I = incremental industry costs for SIR implementation ($\$10^6$/reactor).

Cost savings to industry from accident-avoidance are estimated with respect only to onsite damage since public risk is a sufficient representation of offsite consequences. This cost savings is defined as follows:

ΔH = industry savings (cost reduction) due to accident-avoidance ($\$10^6$)

$= N\bar{T}\ \Delta(FA)$

where N = number of reactors affected

$\bar{T}$ = average remaining operating life of reactors affected (years)

$\Delta(FA)$ = change, due to the SIR, in the product of estimated time frequency of affected accidents in (reactor-years)$^{-1}$ and cost of cleanup, repair and replacement power following an accident ($\$10^6$)

2.3 APPROACH TO PRIORITIZATION INFORMATION DEVELOPMENT

Results of this analysis will be used primarily to set priorities for future NRC work on safety issues. The relatively large number of issues to be analyzed requires that the methodology emphasize technically defensible estimates of the potential risk, dose and costs associated with SIRs at a relatively modest cost. The approach described in these guidelines is intended to require about 2-3 staff-weeks of effort for an analyst familiar with the method to perform the assessment. It is felt that this approach provides adequate information to the NRC for their use in prioritizing these issues. It may not be adequate for making decisions or regulatory actions for specific issues, although this level of analysis can provide useful perspective in guiding future work on the issue.

It is recognized that major simplifications have been required to produce an approach that can be implemented with the level of effort required for the prioritization process. For example, a major simplification is the use of risk estimates for one representative PWR and one representative BWR for all current and future plants. Risks for any particular plant could vary significantly from those of the representative plants, although these plants are believed to reasonably represent the industry as a whole.

Other major simplifications include the use of only dominant accident sequences. These sequences typically contribute ~90 percent of the total plant risk. Also, the risk equations used in this study do not model all issues directly. Modifications of original equations are developed on a case-by-case basis to accommodate issue-specific information. Finally, issues treated using this method are assumed to be independent. When an initial ranking has been completed, additional analyses can be performed to identify interdependences.

The remainder of this report provides guidance on developing the information described in Section 2.2 for use in prioritizing safety issues. A five-step procedure is used:

1. obtain information on each safety issue
2. obtain or develop possible SIRs

3. estimate the nominal impact and range of impacts on public risk and occupational dose from implementing, operating and maintaining the SIRs

4. estimate the nominal industry and NRC costs, and range of costs for implementing, operating and maintaining the SIRs

5. report results for use by the NRC.

Results of the first two steps are required before using the methods described in this report. Detailed information on the potential SIR is desirable but may not be required. A general understanding of the implementation process and effect on other plant systems is needed to prepare risk reduction, dose, and cost estimates. Specific data requirements are discussed in the risk, dose, and cost sections of this report.

Results of steps 1 and 2 are used in step 3 to estimate the impact on public risk and occupational dose of potential SIRs. Data used in representative plant risk analyses are modified to reflect issue resolution These data are then used to calculate a new estimate of plant risk. The incremental risk reduction is attributed to the SIR. Occupational dose estimates are based on historic data for backfit and operations activities. Details of the method and development of the representative data are discussed in Section 3.0 of this report.

Results of the first three steps are used for the cost calculations in step 4. Industry costs (engineering, labor, replacement power and accident-avoidance) and NRC costs are estimated in this step for proposed resolutions to the safety issue. Analysis methods for cost are discussed in Section 4.0 of this report.

In step 5, results of the analyses are presented for use by the NRC in prioritizing safety issues. Uncertainty analyses are performed and are presented to facilitate consideration of judgmental factors in making the final issue ranking. Additional quantitative analyses for parameter sensitivity, issue independence, capital allocation, and incremental cost may be performed based on the data but are not discussed in this report. Step 5 of the prioritization approach is described in Section 5.0 of this report.

Numerical examples of three safety issues are presented in Section 6.0 to demonstrate the prioritization methodology. These issues include Training and Qualification of Operations Personnel (Issue I.A.2.2, Section 6.1), Diesel Generator Reliability (Issue B-56, Section 6.2) and Steam Line Break with Consequential Small LOCA (Issue 18, Section 6.3).

3.0 SAFETY ISSUE RESOLUTION RISK AND DOSE

In Section 2, safety-related parameters for use in the prioritization of safety issues were identified as public risk and occupational dose. Consequences are quantified in terms of man-rem. Occupational doses are accumulated during the implementation, operation and maintenance of the SIR. Dose is avoided by reducing accident frequency or mitigating accident consequences. The remainder of this section is divided into discussions of background and methods to estimate each of these risk and dose contributors. Development of uncertainty estimates is discussed in the last subsection and Appendix F.

3.1 BACKGROUND FOR PUBLIC RISK CALCULATIONS

A risk model that includes major contributors to plant risk is needed to calculate the risk reduction for the resolution of a safety issue. The model can then be exercised to determine the change in plant risk due to the implementation of the resolution. This section provides the development of a general risk model and terminology necessary for the safety impact calculations. Details on the implementation of the risk model to safety issues are discussed in Sections 3.2 and 3.4.

Risk is generally defined as the product of accident frequency and consequences. Accident frequencies are in units of events/reactor-year and consequences are defined in terms of man-rem of exposure. For a plant where accident releases and accident sequences can be divided into distinct categories, the public risk equation can be written as:

$$\text{Public Risk } W = \sum_i \left(R_i \sum_j F_{ij} \right) \quad \frac{\text{man-rem}}{\text{plant-year}}$$

NOTE: $\bar{F} = \sum_i \sum_j F_{ij}$ is the frequency of an accident sequence occurring at the plant.
If the release categories i are restricted to those resulting from a core-melt, then $\bar{F}$ is the core-melt frequency.

where i = release category index

j = accident sequence index

R_i = public consequences (man-rem) for release category i

F_{ij} = frequency $(\text{plant-year})^{-1}$ of accident sequence j contributing to release category i.

The consequence term can be expanded to account for contributions from individual isotopes and environmental pathways. The magnitude of consequences is also related to the surrounding population. This can be described as:

$$R_i = P\left(\sum_k \sum_l B_{ikl}\, E_{kl}\, X_{kl}\right) \text{ man-rem}$$

where k = radionuclide index

l = pathway index

P = demographic function

B_{ikl} = amount of radionuclide k (Ci) released in release category i via pathway l

E_{kl} = environmental transport function for radionuclide k via pathway l

X_{kl} = exposure function for radionuclide k via pathway l (rem/Ci).

The frequency term can be expanded to account for contributions from accident initiators and separate plant systems. Mathematically this can be stated as:

$$F_{ij} = \sum_m \left(I_{jm}\, \underline{B}_n\, Q_{jmn}\right) \text{plant-year}^{-1}$$

where m = initiator index

n = system index

I_{jm} = occurrence rate (plant-year)$^{-1}$ of initiator m in accident sequence j

Q_{jmn} = failure probability of system n given initiator m and any preceding failures in accident sequence j

$\underline{B}_n$ = Boolean product of terms Q_{jmn}.

Boolean algebra must also be used when the terms of the minimal cut sets and component failure probabilities comprising Q_{jmn} are multiplied together. This is described as:

$$Q_{jmn} = \underline{Z}_o \; K_{jmno}$$

where o = component index

K_{jmno} = failure probability of component o in system n given initiator m and any preceding failures in accident sequence j

$\underline{Z}_o$ = Boolean logic operator which describes Q_{jmn} in terms of the contributing K_{jmno} (e.g., minimal cut sets).

Some of the terms in these equations can be quite complicated. These have been simply referred to as "functions" in this illustration. The Boolean logic operator, although typically quite extensive, usually consists only of the sums and products of numerous terms related to component failure probabilities. Most often, it is written as the sum of many products of terms, each product constituting a minimal cut set of a fault tree for system failure.

The public risk equation must be expanded to determine the impact of a SIR. For example, assume a risk equation with only two release categories. Associated with each category is a frequency F and a consequence R. The risk equation is then:

$$W = F_1R_1 + F_2R_2.$$

Three initiating events A, B, and C are presumed possible for the two release categories. Each has an occurrence frequency I. Furthermore, only four systems, W, X, Y, and Z, are assumed to be potentially available to prevent radionuclide release. Each has a failure probability Q. Typical expressions for the frequencies of release via the two release categories would be:

$$F_1 = I_A Q_X + I_B Q_Y Q_Z$$
$$F_2 = I_B Q_W Q_Z + I_C Q_X Q_Y.$$

Each product of terms corresponds to an accident sequence. These are often determined by event tree analysis. Note that initiator B and systems X, Y, and Z contribute to the frequencies of both release categories. Also, the failure probability of any system in an accident sequence is conditional upon the sequence's initiator and any failures preceding that of the system. For example, in accident sequence $I_C Q_X Q_Y$, Q_Y (failure of system Y) is conditional upon I_C (occurrence of initiator C) and Q_X (failure of system X, which is itself conditional upon I_C).

To complete the example, further simplification is made by assuming only components a through j comprise the four systems. Each component has a failure probability K. The contribution of these components to the failures of their respective systems is usually expressed as a Boolean logic equation consisting of a sum of products of component failure probabilities. Each product represents a minimal cut set. The following are typical examples, greatly simplified:

$$Q_W = K_a + K_b K_c$$
$$Q_X = K_d K_e$$
$$Q_Y = K_f + K_g$$
$$Q_Z = K_g K_h + K_i K_j$$

The minimal cut sets are often found by fault tree analysis. K_a represents a one-element minimal cut set, while K_bK_c represents one with two elements. Note that component g contributes to the failure probabilities of both systems Y and Z. Also, the failure probability of any component in an accident sequence is conditional upon the sequence's initiator and any failures preceding that of the component. For example, in accident sequence $I_cQ_XQ_Y$, K_f (failure of component f) is conditional upon I_c and Q_X (including failures of components d and e).

The public risk reduction associated with resolution of a safety issue can be measured by first estimating the quantitative effect of the resolution upon the values of the appropriate accident sequences. Next, the new value of the public risk is calculated using the new cut set frequency values. The difference between the base (before SIR) and the adjusted (after SIR) public risk is the public risk reduction. For the majority of issues, only the public risk resulting from core-melt release sequences and consequences will be considered. These dominate the risk spectrum.

Some issues cannot be directly, or even indirectly, related to the parameters of the public risk equations. In these cases, it may be necessary to supplement the original risk equation with new accident sequences, which could prove dominant if the failure probabilities for the appropriate components are high. If such reassessment still does not place the components/systems into dominant accident sequences, it may be possible only to bound the associated risk reduction based on the total contribution of non-dominant accident sequences to the public risk.

3.2 ESTIMATING PUBLIC RISK REDUCTION

The reduction in public risk at a representative plant due to issue resolution is estimated by calculating the difference between the public risk before and after SIR implementation. Before issue resolution, the risk to the public from accidents is presumed to have some "base-case" value determined by the "base-case" values of all the parameters in the plant's risk equation.

Implementation of the SIR will alter one or more of these parameter values to some "adjusted-case" values. Associated with these "adjusted-case" values is an "adjusted-case" risk to the public, representing the situation subsequent to issue resolution. The difference between these "adjusted" and "base-case" risk values is the reduction in the public risk due to issue resolution. For the purposes of this effort, only core-melt accidents are used to estimate public risk reductions. Previous work (Hall 1979) concluded that less severe accidents make minor contributions to public risk.

There are several steps involved in estimating the public risk reduction. These are discussed in the following subsections on issue definition, identification of affected parameters, calculation of base-case affected risk, calculation of adjusted-case affected risk and public risk reduction. A step-by-step approach and work sheet are described in Section 5.1.1

Risk reduction results from decreasing either the frequency of releases or the consequences due to a release. It is anticipated that most issues will deal with release frequency reductions. The approach taken in this section emphasizes procedures to perform these calculations. Issues that deal with reductions in consequences may require the use of computer analyses. The approach to these analyses is discussed briefly in Section 3.2.6.

3.2.1 Issue Definition

A safety issue must be clearly defined in terms of its impacts on plant systems and the plants affected. The starting point is an issue description that the analyst can translate into effects on nuclear power plants. More specifically, the analyst will need to interpret how resolution of the issue will affect certain parameters at the plant related to the public risk. A systematic procedure is described in the following sections to aid the analyst, but knowledge of plant systems is needed to utilize the procedure effectively.

A safety issue may be generic, affecting a wide range of nuclear plants, or specific, affecting only to a few plants or one plant type. An accurate estimate of all plants to which the issue affects is required.

Ideally, the (public) risk equation would be known for each plant. However, only certain plants have currently been subjected to risk studies. Furthermore, the risk equations for some of these plants are not in a convenient form (lists of minimal cut sets of dominant accident sequences) for use within the scope of this project. For example, the risk equations for the WASH-1400 plants are not reported for the most part in terms of component failures comprising minimal cut sets. To obtain such a detailed list, the WASH-1400 fault trees must be traced--a very time-consuming procedure. Some of the Reactor Safety Study Methodology Applications Program (RSSMAP) and Interim Reliability Evaluation Program (IREP) plants have conveniently reported risk equations (Garcia 1981, Hatch 1981, Kolb 1981, Hatch 1982). The analyst must select one or more of these plants to be "representative" of the entire group of affected plants. Minor modifications to the risk equation can make the plant more characteristic of the group it represents (see Section 3.2.3). However, it is implicitly assumed that the representative plant reasonably approximates its corresponding plants with respect to the issue being studied. As more plant risk studies become available, this restriction can be relaxed.

3.2.2 Affected Parameters in the Public Risk Equation

Following selection of one or more representative plants for which the risk equation is in a form convenient for analysis, the analyst determines which parameters of the risk equation can be affected by the issue via a review of the minimal cut sets for the dominant accident sequences. Results of this exercise will depend on the clarity of the issue definition, the representative plant and the definition of the risk parameters.

Neither all the elements nor all the accident sequences are listed since only the dominant ones (contributing $\gtrsim$5% to their release category frequencies) are provided. Furthermore, even if all such elements and sequences could be provided, there would still be no assurance that all were included. This is an inherent limitation of risk assessment. Since the risk reduction is a measure of the change in risk, a relative rather than an absolute value, the effects of these limitations are reduced.

For most issues, it is anticipated that one or more affected parameters will be readily identifiable from among the minimal cut set elements for the dominant accident sequences. If so, the analyst proceeds to determine the base-case parameter values as discussed in Section 3.2.4.

In some cases, certain parameters may require "redefinition" to suit the issue for a representative plant. Such is the case for Diesel Generator Reliability as analyzed using the Oconee 3 PWR as a representative plant. The issue clearly affects diesel generators, of which there are none at the Oconee plant. However, one of the risk parameters is related to the Oconee hydroelectric generators and can be "redefined" in terms of diesel generators as if Oconee has them (see Section 6.2.2.1).

It is possible that for some issues no parameter will be readily identifiable as affected. This will be likely if:

1. the issue is a minor one with respect to public risk and, thus, would not be expected to affect any of the dominant accident sequences.
2. the issue is influential with respect to public risk but was not considered so either at the time of the risk study or for the specific plant. This could be the consequence of a data base much-improved since the time of the study.

Treatment of these difficulties is discussed in the next two subsections.

3.2.2.1 Bounding Effect of Minor Issues

In the case of a minor issue, it is unlikely that generating "new" minimal cut sets/accident sequences (or resurrecting "old", non-dominant ones) containing the parameters that would be affected by the issue will place these cut sets/sequences among the dominant ones. This would require that the parameter values be significantly different from those at the time of the risk study. Therefore, the issue's effect may only be boundable by assuming its affected parameters contribute to some portion of the non-dominant minimal cut sets/accident sequences.

Typically, only the dominant minimal cut sets of the dominant accident sequences are listed. Likewise, only the dominant accident sequences of each core-melt release category are listed. Thus, there is some small contribution to the dominant accident sequences and the core-melt release categories arising from non-dominant minimal cut sets and non-dominant accident sequences respectively. Such contributions amount usually to $\lesssim 10\%$.

To bound the effect of a minor issue, the analyst first postulates one or more parameters that the issue affects. Next, he assumes that some portion of the non-dominant minimal cut sets of one or more dominant accident sequences (or some portion of the non-dominant accident sequences of one or more core-melt release categories) contains each of these affected parameters. Finally, it is assumed that the contribution to the dominant accident sequences arising from each portion of their non-dominant minimal cut sets (or the contribution to the core-melt release categories arising from each portion of their non-dominant accident sequences) containing an affected parameter is directly proportional to that parameter's value.

Engineering judgment will play a role in bounding the effect of a minor issue. One possible way of apportioning the contribution from non-dominant minimal cut sets/accident sequences is to assume each postulated parameter contributes to the non-dominant minimal cut sets/accident sequences in a direct proportion to the contribution of some similar parameter in the dominant minimal cut sets/accident sequences to those dominant cut sets/sequences. For example, some minor issue is assumed to affect a postulated parameter X that is not found among the dominant minimal cut sets. X is presumed to contribute to some portion of the non-dominant minimal cut sets of dominant accident sequence A. It is also found to be similar to parameter Y which contributes to dominant minimal cut sets accounting for 25% of the frequency of sequence A. Thus, parameter X can be assumed to be responsible for 25% of the contribution to sequence A arising from the non-dominant minimal cut sets. If these cut sets contribute 4% of sequence A's frequency, then parameter X contributes (0.25)(0.04) = 0.01 or 1% of sequence A's frequency.

3.2.2.2 Generating New Minimal Cut Sets for Influential Issues

In the case of a new and influential issue, it is possible that generating "new" minimal cut sets/accident sequences (or resurrecting "old", non-dominant ones) containing the parameters that would be affected by the issue will place these cut sets/sequences among the dominant ones. Presumably, these parameter values will be significantly different from those at the time of the risk study. Their corresponding minimal cut sets/accident sequences must be evaluated (or re-evaluated if they were previously grouped with the non-dominant ones) in light of this new knowledge.

As in the bounding approach, the analyst first postulates one or more parameters that the issue affects (he may find such parameters among the non-dominant minimal cut sets/accident sequences). He then develops "new" minimal cut sets/accident sequences containing these postulated parameters. (If these minimal cut sets/accident sequences were already developed in the study but assigned non-dominant status, they should be used.) These may be similar to already existing dominant minimal cut sets/accident sequences, requiring only a slight modification.

For example, consider an influential issue which affects some postulated parameter X. Presume that this parameter would contribute to dominant accident sequence A. Sequence A has the following dominant minimal cut sets:

IDE
IFG
IHJK

The analyst determines that parameter X would contribute to sequence A via a minimal cut set involving parameters I and F. Thus, he generates a "new" minimal cut set IFX and adds it to the preceding list for sequence A.

3.2.3 Base-Case Affected Risk

The affected public risk is that portion of the public risk attributable to the affected parameters. The base-case, affected public risk is calculated by assuming values for the affected parameters characteristic of the issue before its resolution. These are then substituted into the risk equation of the representative plant. Typically, the issue will affect only a few parameters and accident sequences.

If new cut sets were not developed, the affected parameters will already have values as used in the original study. Sometimes these may be plant-specific and/or outdated with respect to the issue as currently understood. In these cases, the parameter values should be updated to reflect the current state of knowledge. These become the new base-case values and are used to calculate the base-case affected risk. This updating can usually be accomplished by substituting generic data for plant-specific data as dictated by the issue.

The analyst should also check to see if factors such as common-cause failures were incorporated into the original calculations if such factors are identified as prevalent for the issue. For example, in Diesel Generator Reliability, common-cause failure of two diesel generators was not included in the original study for the Grand Gulf 1 BWR. These failures had to be quantified and properly incorporated into the minimal cut sets containing multiple diesel terms to more accurately estimate the base-case affected risk (see Section 6.2.2.1).

Whenever "new" affected parameters are postulated for minor or influential issues (see Sections 3.2.2.1 and 3.2.2.2), they should also be assigned base-case values. Unless these parameters are already present among non-dominant minimal cut sets, they will not have any predetermined value from the time of the representative plant study. Thus, it will always be necessary to estimate some base-case value for each. The procedure is basically the same as updating, except that no prior value is available.

Once the base-case values for the affected parameters have been estimated, the frequencies of the minimal cut sets (those containing affected parameters) are requantified. These are summed within their respective accident sequences to yield the frequencies of the affected portions of the accident sequences (those portions attributable to affected parameters). When using the bounding technique, there will be no change from the original study values for the representataive plant since no "new" minimal cut sets/accident sequences have been developed.

Once the base-case frequencies for the affected accident sequences (only the portions attributable to affected parameters) have been estimated, the frequencies of the affected portions of the core-melt release categories (those portions attributable to affected parameters) are requantified. Again, there will be no change from original values for the bounding technique.

The base-case, affected public risk is calculated by multiplying the frequency of each affected release category (only the portion attributable to affected parameters) by that category's public dose factor (see Section 3.2.6) and then summing the products. The adjusted affected public risk due to issue resolution will be compared against this base affected public risk to yield the public risk reduction for issue resolution.

Appendices A and B list the release categories and their frequencies (Tables A.1 and B.1); the dominant accident sequences, minimal cut sets and their frequencies (Tables A.3 and B.3); and the minimal cut set elements (Tables A.4 and B.4) from the Oconee and Grand Gulf RSSMAP studies, respectively. Table D.1 lists the public dose factors for the release categories.

3.2.4 Adjusted-Case Affected Risk

The adjusted-case, affected public risk is calculated by changing the values for the affected parameters to ones that would be characteristic of the issue subsequent to its resolution. These are then substituted into the risk equation of the representative plant as was done for the base case.

Adjustment of the affected parameter values will primarily involve engineering judgment since the analyst is essentially projecting to a future situation for which no data currently exist. The analyst may be able to assume some goal will be attained as defined in the issue. For example, in Diesel Generator Reliability, the proposed goal of a diesel generator unreliability of 0.03 is assumed to be the adjusted-case failure probability for a diesel generator (see Section 6.2.2.1). However, any current data will already have been used to update the values of the affected parameters for the base case. Thus, only projections based primarily on engineering judgment will remain for the analyst to use in estimating adjusted-case values.

For the bounding technique, the analyst will likewise estimate adjusted-case values for the postulated, affected parameters. However, since no prescribed minimal cut sets (or possibly accident sequences) are known for these, the analyst can only presume that the contribution from each postulated, affected parameter changes in direct proportion to the change in that parameter's value. For example, if the adjusted-case value of parameter X (discussed in Section 3.2.2.1) is 50% of its base-case value, then its contribution to the frequency of sequence A will be 50% of that in the base case. Thus, X will contribute only (0.50)(0.01) = 0.005 or 0.5% of sequence A's frequency in the adjusted case (compared to 1% in the base case).

If any factors, such as common-cause failures, were incorporated into the base-case risk calculations, they must also be retained in the adjusted case. For example, in Diesel Generator Reliability, the probability of a common-cause failure of two diesel generators is adjusted from its base-case value and incorporated into the estimate of the adjusted-case, affected public risk (see Section 6.2.2.1).

Quantification of the frequencies of the affected minimal cut sets, accident sequences, and release categories for the adjusted case parallels that for the base case. The dose factors for each release category are similarly applied to yield the adjusted-case, affected public risk. The analyst is now ready to calculate the public risk reduction due to issue resolution.

3.2.5 Public Risk Reduction

The public risk reduction (ΔW) due to the SIR is the difference between the base-case (W) and the adjusted-case, affected public risk (W^*). This calculation is performed for each representative plant. The total public risk reduction is the sum of the total contribution from all affected plants of each representative type over their average remaining operating lives. In the form of an equation, this is:

$$(\Delta W)_{Total} = \sum_{x} N_x \bar{T}_x (\Delta W)_x$$

where x = the index of the representative plant-type

N_x = the number of affected plants to which representative plant-type x corresponds

$\bar{T}_x$ = the average remaining operating life of affected plant-type x (BWR, PWR) (see Appendix C)

$(\Delta W)_x$ = the public risk reduction for representative plant-type x in man-rem/plant-year.

Uncertainties on the public risk reduction for a representative plant and on the total public risk reduction are discussed in Appendix F.

Another quantity of interest is the reduction in accident frequency ($\Delta\bar{F}$) due to issue resolution, which is used in estimating the occupational dose reduction and industry cost savings due to accident-avoidance. For a representative plant, the accident frequency reduction is just the difference between the base-case ($\bar{F}$) and the adjusted-case affected frequency ($\bar{F}^*$). The affected accident frequency is that portion of the accident frequency attributable to the affected parameters. Dominant accident sequences for the representative plants used in this study all lead to core-melt accidents. Both $\bar{F}$ and $\bar{F}^*$ are found by summing the affected portions of the frequencies for all the core-melt release categories in each case (base and adjusted). Uncertainties on the core-melt frequency reduction for a representative plant are discussed in Appendix F.

It is anticipated that the approach that has been described will be feasible for estimating the public risk reduction for most issues dealing with reductions in accident frequencies. However, it is conceivable that an issue could be so defined as to not lend itself to this analysis approach involving the use of known risk equations. This could occur if an issue is so general as to influence plant safety as a whole, rather than any specific areas.

In such cases, it might be more practical to abandon the systematic technique presented here and opt for some more judgmental process. A formalized technique involving expert opinion such as the Delphi method could be used to estimate an issue's public risk reduction. Another option that could be used if several such issues exist would be to quantitatively rank these issues with respect to one another in terms of their risk reduction. For example, if three issues (A, B, and C) are being considered, expert opinion could determine that the public risk reductions associated with B and C are 50% and 25% respectively of that associated with A. If one of these issues can be "normalized" to some known value of the public risk reduction associated with a more readily quantified issue, then values are obtained for the other two. This is only an approximate technique, but it may be the only reasonable option for some issues.

3.2.6 Dose Factors for Release Categories

In estimating affected public risk, consequence factors (man-rem per occurrence) are required for each affected release category. Dose consequence factors are estimated for the 15 release categories defined in WASH-1400. The computer program CRAC2 is applied to a typical midwest site (Braidwood). Assumptions and parameters used for the calculations are as follows.

- Dose consequences are represented by the whole body population dose commitment (man-rem) received within 50 miles of the site.
- An exclusion area of 1/2 mile is assumed with uniform population density of 340 persons per square mile beyond 1/2 mile.
- Evacuation of people is not considered.
- All exposure pathways are included for non-core-melt sequences (PWR-8 and 9, and BWR-5). For core-melt sequences all exposure pathways except ingestion pathways are included.
- Farmland usage parameters for the state of Illinois are used for non-core-melt ingestion pathway calculations.
- Meteorological data is taken from the U.S. National Weather Service station at Moline, Illinois.
- The core inventory at the time of the accident is assumed to be represented by a 3412 MWt (1120 MWe) PWR as reported by Ostmeyer (1981).

Results of the dose calculations are presented in Table D.1 for the PWR and BWR release sequences.

The calculated dose factors are nearly independent of the choice of reactor site. The only site-dependent parameters are reactor power level, meteorological data, and farmland usage data. For the core-melt release sequences the dose values include crop and animal product ingestion pathways and are influenced by farmland parameters. The meteorological data base has only a moderate influence (15%) on the calculated doses (Strip 1982). Sample calculations for the first three release sequences (PWR 1A, 1B and 2) give nearly identical results (within 5%) when New York City meteorological data is substituted for Moline, Illinois data. The power level determines the radionuclide inventory in the reactor at the time of the accident. The calculated dose consequences are approximately proportional to the power level. The consequences for a reactor other than Braidwood can be estimated by the ratio of the reactor power level to that of Braidwood (1120 MWe). Because a uniform population distribution is used, the calculated dose consequences are not dependent on the Braidwood site demographic data.

For the reasons stated in the above discussion, the dose consequence factors may be considered generic. The use of generic dose calculations in this project is a convenience and is assumed to introduce only small amounts of error. Risk studies subsequent to WASH-1400 have tended to use the same release category definitions, so few problems related to models of radionuclide amounts/rates (the B terms in the risk equation of Section 3.1) are introduced. Similarly, the environmental transport and human exposure functions (terms E and X in Section 3.1) used in each risk assessment are essentially similar to those for WASH-1400, with some updating. The demographic function (the term P in Section 3.1) is highly site-dependent, varying from plant to plant. However, the use of a constant population density will streamline comparision of issues not related to siting.

Issues that influence the amount or type of nuclides that are released will require special analyses. In terms of the two-release-category risk equation from Section 3.1:

$$W = F_1R_1 + F_2R_2$$

where W = public risk (man-rem/reactor-yr)

F_1, F_2 = frequencies of release categories 1 and 2

R_1, R_2 = consequences of release categories 1 and 2

The consequences for each release category can then be expressed as follows (note: for simplicity, only one environmental pathway is assumed):

$$R_1 = P(B_{1\alpha}E_\alpha X_\alpha + B_{1\beta}E_\beta X_\beta)$$

$$R_2 = P(B_{2\alpha}E_\alpha X_\alpha + B_{2\beta}E_\beta X_\beta)$$

P is the demographic function which does not vary with release category. $B_{1\alpha}$ and $B_{2\alpha}$ are the release amounts of radionuclide α for release categories 1 and 2 respectively. $B_{1\beta}$ and $B_{2\beta}$ are the corresponding amounts for radionuclide β. E_α and E_β are the environmental transport functions for radionuclides α and β respectively. X_α and X_β are the exposure functions for the two radionuclides.

It is assumed that an issue's resolution changes the release amount of radionuclide α in release category 1. This change is manifested as a lower value of $B_{1\alpha}$, indicated by $B_{1\alpha}{}^*$. The consequences of release category 1 will decrease to the following:

$$R_1{}^* = P(B_{1\alpha}{}^*E_\alpha X_\alpha + B_{1\beta}E_\beta X_\beta)$$

Subsequently, a lower risk is calculated:

$$W^* = F_1R_1{}^* + F_2R_2$$

Changes in the consequence parameters may require additional computer-aided analyses to determine the effect on dose in each release category. It is anticipated that few issues will require this approach. A description of the analyses will be developed in the appropriate issue reports.

3.3 OCCUPATIONAL DOSE: A GENERAL DISCUSSION

Occupational doses can arise from both the implementation and operation/maintenance of SIRs and during cleanup, repair, and refurbishment of nuclear power plants following accidents. As described previously in Section 2.2.2, occupational dose has two components: 1) incremental doses due to SIR implementation, operation and maintenance and 2) the accident-related dose weighted by the reduction in accident frequency developed in Section 3.2.

To model the occupational dose consequences of accidents in PWR and BWR plants, three accident scenarios are postulated. The three scenarios, developed and analyzed in a recent NRC-sponsored study on decommissioning (Murphy 1982) are as follows:

1. A small loss-of-coolant accident (LOCA) (e.g., a small steam line break or the inadvertent opening of a safety or relief valve) in which the emergency core cooling system (ECCS) functions to cool the core and to limit the release of radioactivity. Some fuel cladding rupture is postulated but no fuel melting. The consequence scenario includes moderate contamination of the containment building but no significant physical damage to the building and equipment.
2. A small LOCA in which ECCS is delayed, resulting in 50% fuel cladding failure and a small amount of fuel melting. The consequence scenario includes extensive radioactive contamination of the containment building but only minor physical damage to the building and equipment. (The consequences of this accident in terms of radioactive contamination and physical damage are chosen to be similar in magnitude to those which resulted from the March 28, 1979 accident at Three Mile Island, Unit 2.)
3. A major LOCA (e.g., the rupture of a main coolant line) in which ECCS is delayed, resulting in 100% fuel cladding failure and significant fuel melting and core damage. The postulated consequences include extensive radioactive contamination of the containment building and major physical damage to structures and equipment. Some radioactive contamination of the auxiliary and fuel buildings is also postulated.

The parameters of interest related to these accident scenarios are listed in Table 3.1. Procedures used to calculate occupational dose reduction due to accident avoidance based on these scenarios are discussed in Section 3.4.

Occupational doses for implementation and operation/maintenance are derived from existing data on radiation dose rates in various areas of reference reactors and from an estimate of the staff labor required to complete the tasks. These factors are discussed further in Section 3.4. If specific issues require the use of more accurate dose estimates, specific time-motion and radiation field analyses may be required.

3.4 ESTIMATING OCCUPATIONAL DOSE

Occupational dose associated with a particular safety fix has accident and non-accident components. The accident component is the product of the occupational radiation dose resulting from cleanup, repair and refurbishment following a reactor accident (D_R) and the expected reduction in accident frequency ($\Delta\bar{F}$). The non-accident component is the occupational radiation dose received while implementing (D) and operating/maintaining the SIR (D_O). Contributions to these occupational doses are discussed in the following subsections.

3.4.1 Occupational Dose Reduction Due to Accident-Avoidance

The estimated occupational radiation dose resulting from the cleanup and immediate dismantlement following each of the three accident scenarios discussed in Section 3.3 are listed in Table D.2. It is assumed that the occupational radiation doses associated with repair and refurbishment will be about the same as estimated for immediate dismantlement (Murphy 1982).

These accident scenarios are assumed to be related to the WASH-1400 release categories for calculations using this methodology in the following manner:

TABLE 3.1. Reference PWR Accident Parameters (Murphy 1982)

Parameter	Parameter Value[a] Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Percent of fuel cladding failure	10	50	100
Percent of fuel melting	0	5	50
Volume of sump water (m^3)	200	1000	1600
Depth of sump water (m)	0.2	1.0	1.6
Total fission product radioactivity in sump water (Ci)	2.5×10^4	3.5×10^5	2.5×10^6
Average fission product radioactivity in sump water (Ci/m^3)	125	350	1560
Total fission product radioactivity plated out on building surfaces (Ci)[b]	5	70	500
Average fission product radioactivity on building surfaces (Ci/m^2)			
• Floors	0.001	0.014	0.1
• Walls	0.00001	0.00014	0.001
Average gamma radiation exposure rate at operating floor level (R/hr)			
• Contribution from plateout	0.01	0.15	1.0
• Contribution from sump water	0.015	0.045	0.2
• Total exposure rate	0.025	0.20	1.2
Average gamma radiation exposure rate at lowest entry level (R/hr)			
• Contribution from plateout	0.01	0.15	1.0
• Contribution from sump water	8	30	170
• Total exposure rate	8	30	170
Damage to fuel core	Slight damage to some fuel elements as a result of fuel swelling and cladding rupture.	Oxidation of fuel cladding. Melting and fusing together of stainless steel fittings on center fuel elements. Cracking and crumbling of some fuel pellets. Melting of fuel in localized areas of central core.	Cracking, crumbling, and melting of fuel pellets. Melting and fusing together of stainless steel parts on adjacent fuel assemblies. Molten fuel present over much of core radius. Fuel and cladding fragments carried throughout primary coolant system.
Damage to containment building and equipment.	No significant physical damage.	Contamination of building ventilation system. Some electrical equipment and some valves inoperable due to water damage and corrosion. Minor structural damage. Polar crane inoperable.	Ventilation ductwork damaged. Doors, catwalks, pipes, and cable conduits dented or ripped away. Loss of electrical and other services. Erosion of concrete and metal surfaces. Polar crane inoperable.
Contamination of auxiliary and fuel buildings	--(c)	--(c)	Plateout on building surfaces. CVCS contaminated with 20,000 Ci of fission product radioactivity. General area radiation exposure levels about 100 mR/hr.

(a) Values refer to conditions inside the containment building approximately 1 year after the postulated accident.
(b) Plateout values are after washdown of the walls by condensing moisture.
(c) Contamination of the auxiliary and fuel buildings is postulated only for the scenario 3 accident.

Release Categories		Accident Scenarios
PWR 1-7 BWR 1-4		3
PWR-8, 9 BWR-5	(non-core-melt)	2
Other non-core melt accidents		1

For the majority of issues analyzed using this methodology, only core-melt accidents like scenario 3 will be considered. Scenarios 1 and 2 are included in case future safety issues require their use.

The change in frequency of core-melt accidents due to the safety issue resolution is multiplied by the dose estimates to yield occupational dose reduction due to accident-avoidance. The total occupational dose reduction due to accident-avoidance (ΔU) is:

$$\Delta U = D_R \sum_x N_x \bar{T}_x (\Delta\bar{F})_x$$

where

x = the index of the plant-type (BWR, PWR)

N_x = the number of affected plants of type x

D_R = the occupational dose from reactor cleanup, repair, and refurbishment following an accident

$\bar{T}_x$ = the average remaining operating life of plant-type x

$(\Delta\bar{F})_x$ = the reduction in accident frequency for plant-type x

3.4.2 Occupational Dose Increase

A change in a reactor's systems and/or components will, in general, involve working in radiation zones, both during the implementation of the new equipment/components and during the routine operation and maintenance of the equipment. Estimation of the increase in occupational radiation dose associated with implementation and operation/maintenance is discussed in the following subsections. There is a very slight chance of a radiation release to the surrounding environs occurring during the installation work. Thus, any population dose from such an occurrence is assumed to be negligible.

3.4.2.1 Implementation Dose

During backfitting of an operating plant, implementation (involving installation/testing) of safety fixes can result in radiation doses to plant workers ranging from zero for procedural changes to many man-rem for equipment changes required inside of containment. Obviously, there would be no radiation dose associated with forward-fitting of plants not yet in operation.

Occupational doses for installation/testing are derived from existing data on radiation dose rates in various areas of reference reactors (Smith 1978 and Oak 1980) and from an estimate of the staff labor required to complete the tasks. In addition, Final Safety Analysis Reports (specifically, Chapter 12 data) are utilized where values for dose rates anticipated for various status modes--normal operation, hot standby, refueling, etc.--are needed.

3.4.2.2 Operation/Maintenance Dose

If operating actions or maintenance efforts are required in radiation zones as a result of implementing a safety issue fix, those efforts will result in occupational radiation doses. These dose rates will be highly job- and location-dependent. The estimated dose rate is multiplied by the estimated amount of staff labor in the radiation zone to determine the occupational radiation dose increase for each SIR. Unless issue-specific information is available, the data sources mentioned above are utilized in calculating these doses.

3.5 UNCERTAINTY ANALYSIS

Generic techniques for estimating the uncertainties on parameters related to the public risk and occupational dose are developed in Appendix F, together with standardized approximations on error bounds. The results are summarized here.

3.5.1 Public Risk Reduction

For the total public risk reduction $[(\Delta W)_{Total}]$, the standardized error bounds (at a 90% confidence level) are as follows (from section F.1.1):

$$[(\Delta W)_{Total}]_{upper} = 30 \sum_x N_x \bar{T}_x \hat{W}_x$$

$$[(\Delta W)_{Total}]_{lower} = 0$$

where x = the index of the representative plant-type (BWR,PWR)

N_x = the number of affected plants to which plant-type x corresponds [from step 2 of the Public Risk Reduction Work Sheet (PRRWS) in Section 5.1.1]

$\bar{T}_x$ = the average remaining operating life of plant-type x (from step 2 of the PRRWS)

$\hat{W}_x$ = the best estimate of the base-case, affected public risk for plant-type x (from step 9 of the PRRWS).

3.5.2 Occupational Dose Reduction Due to Accident-Avoidance

For the total occupational dose reduction due to accident avoidance (ΔU), the standardized error bounds (at a 90% confidence level) are as follows (from Section F.1.2):

$$(\Delta U)_{upper} = 6\hat{D}_R \sum_x N_x \bar{T}_x \hat{F}_x$$

$$(\Delta U)_{lower} = 0$$

where x, N_x and $\bar{T}_x$ are defined as before

$\hat{D}_R$ = the best estimate of the occupational dose due to reactor cleanup and repair following an accident (from Appendix D)

$\hat{F}_x$ = the best estimate of the base-case, affected core-melt frequency for plant-type x (from step 8 of the PRRWS).

3.5.3 Occupational Dose Increase for SIR Implementation, Operation, and Maintenance

For the total occupational dose increase for SIR implementation, operation, and maintenance (G), the standardized error bounds (at a 90% confidence level) are as follows (from Section F.2.1):

$$G_{upper} = 3\hat{G}$$

$$G_{lower} = \hat{G}/3$$

where $\hat{G}$ = the best estimate of G (from step 12 of the Occupational Dose Work Sheet in Section 5.1.2)

If $\hat{G}<0$, the error bounds are modified as follows:

$$G_{upper} = \hat{G}/3$$

$$G_{lower} = 3\hat{G}$$

4.0 SAFETY ISSUE RESOLUTION COSTS

Implementation of any safety issue resolution (SIR) will incur costs. Some of the costs are incurred by the nuclear industry in performing the engineering, procurement, installation, testing, operation, and maintenance of the SIR. They include efforts required for making license, technical specification, or facility design change submittals, and the cost of replacement power if an extended plant outage is required. The nuclear industry may also avoid costs by reducing the frequency of a postulated reactor accident. Costs are incurred by NRC in the process of developing the SIR, supporting SIR implementation, and reviewing the operation/maintenance of the SIR. These cost terms are discussed in subsequent subsections.

4.1 INDUSTRY COST

Industry costs involve both non-accident (I and I_0, the SIR implementation and operation/maintenance costs respectively) and accident-related components (ΔH, the cost savings due to accident-avoidance), as presented in Section 2.2.3.2. These parameters are discussed in this section.

4.1.1 Industry Implementation Cost

The cost to the nuclear industry of implementing a SIR (I) involves utility (or consultant) staff to develop the design changes, process the planned changes through the approval chain (including NRC approval and any license amendments), procure the necessary equipment (if any), plan the implementation effort, train the staff, make the necessary changes in plant equipment and procedures, and conduct final tests to ensure proper operation following completion of the work. In addition, if plant outage days are required, and if the utility must purchase replacement power from outside its system, the replacement power costs must also be included. These costs are discussed below.

4.1.1.1 Utility Staff Labor Cost

The labor cost for a specific SIR is proportional to the amount of staff labor required to accomplish the work. Industry labor cost factors are based on data from an operating utility and are listed in Appendix E. The cost includes management, operations personnel, and craftsmen, but excludes security personnel.

The use of speciality contractors by the utility is issue-specific. The calculations outlining the details of these costs (including labor and material) are normally included in an attachment to the cost work sheet for that issue.

4.1.1.2 Equipment Cost

Equipment costs are estimated on a case-by-case basis using published information and/or contacts with equipment vendors, as appropriate. In general, it is anticipated that these costs will be small in comparison with the costs of industry staff labor and replacement power for issues where these costs apply.

4.1.1.3 Replacement Power Cost

The value assumed for the purchase of replacement power during each outage day attributable to the implementation of the SIR is listed in Appendix E. The actual cost of replacement power for a specific plant will depend on many factors, including the capacity of the plant, the capacity of the utility's total system, and the size of the system margin.

4.1.1.4 License Amendment Fees

Consideration is given to license amendment fees in those cases where the licensee's SIR effort is anticipated to include proposed changes in plant equipment and/or procedures that change the technical specifications of the plant. Schedule of Fees for Facility License Amendments is contained in 10CFR170.22 and is not repeated here.

4.1.2 Annual Industry Cost for SIR Operation/Maintenance

This cost (I_o) is estimated on a case-by-case basis and includes the annual staff labor for performing the additional surveillance, maintenance, and training necessitated by the SIR. The average labor rate used is given in Appendix E.

4.1.3. Industry Cost Savings Due to Accident-Avoidance

For each plant-type, the accident-avoidance cost savings, $\Delta(FA)$, is the change in the product of the accident frequency and cost for cleanup, repair/refurbishment, and replacement power following an accident,. For most issues, this change will result from a reduction in accident frequency ($\Delta\bar{F}$). Thus, the total industry cost savings due to accident-avoidance (ΔH) is:

$$\Delta H = A \sum_x N_x \bar{T}_x (\Delta\bar{F})_x$$

where
- x = the index of the plant-type (BWR, PWR)
- N_x = the number of affected plants of type x
- A = the industry cost for reactor cleanup, repair, and replacement power following an accident
- $\bar{T}_x$ = the average remaining operating life of plant-type x
- $(\Delta\bar{F})_x$ = the reduction in accident frequency for plant-type x

For these analyses, the three events described in Section 3.3 are considered, having costs A_1, A_2, and A_3 listed in Appendix E. These costs include cleanup, repair/refurbishment, and replacement power. The cost of repair/refurbishment is assumed to be the same as the cost of decommissioning by immediate dismantlement.

For safety issue analysis, the accident scenarios are interpreted as being reasonably applicable for the various release categories defined in WASH-1400. These release categories are shown below.

Accident Scenario	Release Categories
1	Other non-core-melt release categories
2	PWR-8,9; BWR-5 (non-core-melt)
3	PWR-1A, 1B, 2, 3, 4, 5, 6, 7; BWR-1, 2, 3, 4. (core-melt)

Most issues in the present analyses are limited to core-melt accidents modeled by accident scenario 3. The frequency of these accidents can be evaluated using methods described in Section 3.2. Frequencies of accidents in these categories are evaluated, as needed, on a case-by-case basis. The other two scenarios are included for potential use in future issues related to non-core-melt accidents.

4.2 NRC COST

NRC costs involve parameters related to SIR development (C_D), support of SIR implementation (C), and review of SIR operation/maintenance (C_O), as presented in Section 2.2.3.1. These parameters are discussed in this section.

4.2.1 NRC Cost for SIR Development

For purposes of continuity, NRC cost of developing the SIR (C_D) is assumed to involve both NRC staff labor and contractor costs (where applicable) expended after a specific reference date (October 1982 is assumed here). Therefore, sunk costs incurred before October 1982 are not included. Since the NRC status of SIR development varies with each issue, the cost to complete development of each issue will vary.

4.2.2 NRC Cost for Support of SIR Implementation

The NRC cost to support implementation of the SIR (C) is comprised principally of staff labor utilized for reviewing the proposed changes in the reactor systems, the safety analysis report, and other associated documentation prior to implementation of the SIR, and with surveillance of the ongoing activities during the implementation of the SIR. The NRC staff labor cost factor is listed in Appendix E.

4.2.3 NRC Cost for Review of SIR Operation/Maintenance

The NRC cost for review of SIR operation/maintenance (C_O) is comprised primarily of staff labor requirements associated with annual inspections subsequent to the licensee's implementation of the SIR. Normally, the NRC labor cost factor given in Appendix E is applicable for most issues under consideration.

4.3 UNCERTAINTY ANALYSIS

Generic techniques for estimating the uncertainties on parameters related to the industry and NRC costs are developed in Appendix F, together with standardized approximations on error bounds. The results are summarized here.

4.3.1 Industry Cost Savings Due to Accident-Avoidance

For the total industry cost savings due to accident-avoidance (ΔH), the standardized error bounds (at a 90% confidence level) are as follows (from Section F.1.3):

$$(\Delta H)_{upper} = 6\hat{A} \sum_x N_x \bar{T}_x \hat{\bar{F}}_x$$

$$(\Delta H)_{lower} = 0$$

where x = the index of the representative plant-type (BWR,PWR)

N_x = the number of affected plants to which plant-type x corresponds [from step 2 of the Safety Issue Cost Work Sheet (SICWS) in Section 5.2]

$\bar{T}_x$ = the average remaining operating life of plant-type x (from step 3 of the SICWS)

$\hat{A}$ = the best estimate of the industry cost for reactor cleanup, repair, and replacement power following a core-melt accident (from Appendix E)

$\hat{\bar{F}}_x$ = the best estimate of the base-case, affected core-melt frequency for plant-type x [from step 8 of the Public Risk Reduction Work Sheet (PRRWS) in Section 5.1.1].

4.3.2 Industry Cost for SIR Implementation, Operation, and Maintenance

For the total industry cost for SIR implementation, operation and maintenance (S_I), the standardized error bounds (at a 90% confidence level) are as follows (from Section F.2.2):

$$(S_I)_{upper} = \hat{S}_I + d_{S_I}$$

$$(S_I)_{lower} = \hat{S}_I - d_{S_I}$$

where $\hat{S}_I$ = the best estimate of S_I (from step 12 of the SICWS)

$d_{S_I} = \frac{1}{2}\sqrt{(N\bar{T}\hat{I}_o)^2 + (N\hat{I})^2}$

$N\bar{T}\hat{I}_o$ = the best estimate of the total industry cost for SIR operation/maintenance (from step 11 of the SICWS)

$N\hat{I}$ = the best estimate of the total industry cost for SIR implementation (from step 8 of the SICWS).

4.3.3 NRC Cost for SIR Development, Implementation Support, and Operation/Maintenance Review

For the total NRC cost related to SIR development, implementation, operation, and maintenance (S_N), the standardized error bounds (at a 90% confidence level) are as follows (from Section F.2.3):

$$(S_N)_{upper} = \hat{S}_N + d_{S_N}$$

$$(S_N)_{lower} = \hat{S}_N - d_{S_N}$$

where $\hat{S}_N$ = the best estimate of S_N (from step 21 of the SICWS)

$d_{S_N} = \frac{1}{2}\sqrt{\hat{C}_D^2 + (N\bar{T}\hat{C}_o)^2 + (N\hat{C})^2}$

$\hat{C}_D$ = the best estimate of the total NRC cost for SIR development (from step 14 of the SICWS)

$N\bar{T}\hat{C}_o$ = the best estimate of the total NRC cost to review SIR operation/maintenance (from step 20 of the SICWS)

$N\hat{C}$ = the best estimate of the total NRC cost to support SIR implementation (from step 17 of the SICWS).

5.0 PRIORITIZATION INFORMATION RESULTS

This section contains detailed work sheets to facilitate development of risk, dose and cost information for use by the NRC to prioritize safety issues. It is recommended that the five-step process outlined in Section 2.3 be followed:

1. Obtain information on each safety issue.
2. Obtain or develop the potential SIR for each safety issue.
3. Use work sheets in Section 5.1 to estimate impacts on public risk and occupational dose of the SIR.
4. Use work sheets in Section 5.2 to estimate impacts on industry and NRC costs of the SIR.
5. Use work sheets in Section 5.3 to present a summary of the results for NRC use.

Example problems using this approach are presented in Section 6.0 of this report.

5.1 SAFETY ISSUE RESOLUTION RISK AND DOSE

Separate work sheets are provided for public risk and occupational dose calculations and are discussed in the following two subsections.

5.1.1 Public Risk Reduction Work Sheet

The discussion in Section 3.2 has been systematized into an outline of steps used to calculate the public risk reduction. These steps are summarized in a "Public Risk Reduction Work Sheet" such as that presented in Table 5.1. The steps on the work sheet correspond to the ones given here in the text. Any detailed calculations or deviations from the standard procedure need to be documented on separate pages and referenced on the work sheet. A typical Public Risk Reduction Work Sheet is shown as Table 6.3.2 for Steam Line Break with Consequential Small LOCA. Additional detail to demonstrate the steps in completing the work sheet is shown in Section 6.2.2.1 for Diesel Generator Reliability.

TABLE 5.1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:
2. Affected Plants (N) and Average Remaining Lives ($\bar{T}$):
   (include total number of each plant-type - BWR, PWR)
3. Plants Selected for Analysis:
   (must have known risk equations, e.g., Oconee 3 in Appendix A)
4. Parameters Affected by SIR:
   (from Table A.4 or B.4 in the appendices; document any modifications)
5. Base-Case Values for Affected Parameters:
   (if these differ from those values given in Table A.4 or B.4, document the calculations)
6. Affected Accident Sequences and Base-Case Frequencies:
   (from Table A.3 or B.3 in the appendices; also list the release categories to which they contribute)
7. Affected Release Categories and Base-Case Frequencies:
   (from Table A.1 or B.1 in the appendices)
8. Base-Case, Affected Core-Melt Frequency ($\bar{F}$):
9. Base-Case, Affected Public Risk (W):
10. Adjusted-Case, Affected Values for Affected Parameters:
    (document the assumptions and calculations)
11. Affected Accident Sequences and Adjusted-Case Frequencies:
    (relist the sequences and the release categories to which they contribute from step 6, but with the adjusted-case frequencies)
12. Affected Release Categories and Adjusted-Case Frequencies:
    (relist the categories from step 7, but with the adjusted-case frequencies)
13. Adjusted-Case, Affected Core-Melt Frequency ($\bar{F}^*$):
14. Adjusted-Case, Affected Public Risk (W^*):
15. Reduction in Core-Melt Frequency ($\Delta\bar{F}$):
16. Per-Plant Reduction in Public Risk (ΔW):
17. Total Public Risk Reduction, $(\Delta W)_{Total}$:
    (also list the upper and lower bounds)

The analyses for public risk reduction are performed for BWRs and PWRs corresponding to the representative plant risk models in Appendices A and B. To implement the work sheet, steps 4 through 16 must be repeated for each representative plant. The remainder of the steps need be completed only once.

Outline

1. Define the safety issue and understand it sufficiently to postulate a SIR.

2. Determine which plants the issue affects. If the issue is plant-specific, it should be so stated along with the appropriate plants. If it is generic, this should be declared along with the appropriate plant-types. The number of affected plants (of each type, if so distinguished) must also be determined. List their average remaining lives using the tables in Appendix C.

3. Select the plants for which the issue will be analyzed. These will normally be chosen from among the respresentative plants for which the plant risk equations are known. For convenience, the plant risk equations, in terms of radioactive release categories, dominant accident sequences, and dominant minimal cut sets, have been provided for Oconee 3 and Grand Gulf 1 in Appendices A and B. Additional representative plants may be used (e.g., other BWRs and PWRs) if appropriate to the issue, and information comparable to that in Appendices A and B can be developed.

4. For each representative plant, determine which parameters of the risk equation may be affected (subject to a change in likelihood) by the SIR. These parameters come from the elements of the dominant minimal cut sets for the plant (Tables A.4 and B.4). If no effect seems possible upon any of these parameters, consider generating new minimal cut sets or bounding via the non-dominant minimal cut sets as discussed in Secton 3.2.2.

5. For the affected parameters determined above, estimate their "base-case" values (before issue resolution) against which any changes due to the SIR implementation will be measured. These base-case values can be the ones used in the original risk assessment (reproduced in Tables A.4 and B.4) if they are representative of the parameter values associated with the

issue. However, since the values as originally used may be now out-of-date with the current data base, as well as being perhaps plant-specific, it may be necessary to alter them to more accurately represent the base-case values for a representative plant. In either case, the end result is a set of base-case values for the parameters affected by the issue.

6. For the affected parameters, determine to which accident sequences they contribute (as indicated by the minimal cut sets) and estimate the base-case frequencies of the affected portions of these sequences (by summing the frequencies of the affected minimal cut sets) using the parameters' base-case values. These sequences and cut sets are listed in Tables A.3 and B.3.

7. For the affected accident sequences, determine to which radioactive release categories they contribute and estimate the base-case frequencies of the affected portions of these categories using the base-case frequencies determined above for the affected sequences. These release categories and accident sequences are listed in Tables A.1 and B.1.

8. Estimate the base-case frequency of an affected accident occurring which leads to a radioactive release by summing the base-case frequencies of the affected portions of the release categories. This sum ($\bar{F}$) is the plant's affected core-melt frequency since only core-melt accidents are treated in this methodology.

9. Estimate the base-case, affected public risk from an affected core-melt accident as follows:

   - For each affected release category, multiply its base-case frequency (affected portion) by its public dose factor in Appendix D.
   - Sum all of the above products. This sum (W, see Section 3.1) is the base-case, affected public risk from an affected core-melt accident.

10. For the affected parameters determined in step 4, estimate their "adjusted-case" values (after SIR implementation). The techniques used to obtain these estimates may vary for each parameter and issue, with an

emphasis placed on engineering judgment. Some general approaches for adjusting parameter values are discussed in Section 3.2.4.

11-14. Repeat steps 6 to 9 using the adjusted-case parameter values determined above.

15. Calculate the reduction in core-melt frequency associated with issue resolution ($\Delta\bar{F}$) by subtracting the adjusted-case, affected core-melt frequency ($\bar{F}^*$, from step 13) from that of the base case ($\bar{F}$, from step 8).

16. Calculate the reduction in the public risk associated with the SIR (ΔW) by subtracting the adjusted-case, affected public risk (W^*, from step 14) from that of the base case (W, from step 9). ΔW must be estimated for each representative plant.

17. Calculate the total public risk reduction for all affected plants by summing the products of the following terms for each representative plant-type:

    - the public risk reduction (ΔW, from step 16)
    - the number of affected plants to which the representative plant corresponds (N, from step 2)
    - the average remaining operating life ($\bar{T}$, from step 2).

    The upper and lower bounds on this total public risk reduction are calculated using the formulae in Section 3.5.1.

5.1.2 Occupational Dose Work Sheet

Discussions in Sections 3.3 and 3.4 have been systematized into an outline of steps to calculate the occupational dose parameters. These steps are summarized in an "Occupational Dose Work Sheet" presented in Table 5.2. The following text describes the procedures in the work sheet. Any detailed calculations or deviations from the standard procedure need to be documented on separate pages and referenced in the work sheet. A typical Occupational Dose Work Sheet is shown as Table 6.3.3.

Like public risk, occupational dose calculations are performed for PWRs and BWRs. In addition, each of these is further divided into backfit and forward-fit classes. Calculations need to be performed for both BWRs and PWRs

TABLE 5.2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

2. Affected Plants (N):
   (include total number of each plant-type - BWR, PWR. Divide each type into backfit and forward-fit classes)

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

4. Per-Plant Occupational Dose Reduction Due to Accident-Avoidance, $\Delta(\bar{F}D_R)$:

5. Total Occupational Dose Reduction Due to Accident-Avoidance (ΔU):
   (also list upper and lower bounds)

6. Per-Plant Utility Labor in Radiation Zones for SIR Implementation:

7. Per-Plant Occupational Dose Increase for SIR Implementation (D):

8. Total Occupational Dose Increase for SIR Implementation (ND):

9. Per-Plant Utility Labor in Radiation Zones for SIR Operation and Maintenance:

10. Per-Plant Occupational Dose Increase for SIR Operation and Maintenance (D_o):

11. Total Occupational Dose Increase for SIR Operation and Maintenance ($N\bar{T}D_o$):

12. Total Occupational Dose Increase (G):
    (also list upper and lower bounds)

for occupational dose reduction due to accident-avoidance. Implementation dose calculations apply only to backfit plants. Operation/maintenance dose calculations are applicable to both backfit and forward-fit plants. Certain steps, as indicated in the following procedure, must be repeated for each of these calculations.

Outline

1. Define the safety issue and understand the resolution postulated as part of the public risk reduction calculations.

2. Determine which plants the issue affects. This corresponds to step 2 of the Public Risk Reduction Work Sheet (PRRWS). Break the plant-types into backfit and forward-fit classes.

3. Estimate the average remaining life ($\bar{T}$) in each of the four classes of affected plants. See Appendix C for a plant listing and plant characteristics.

4. Calculate the per-plant reduction in the occupational dose due to accident-avoidance associated with the SIR [$\Delta(\bar{F}D_R)$, see Section 3.4] by multiplying the following terms:

   - the occupational dose associated with cleanup, repair, and refurbishment of a facility following a major core-melt accident (see Section 3.4.1 and Appendix D).
   - the reduction in core-melt frequency ($\Delta\bar{F}$, from step 15 of the PRRWS).

   This product is the reduction in the occupational dose from a core-melt accident. $\Delta(\bar{F}D_R)$ must be estimated for BWR and PWR plants.

5. Calculate the total occupational dose reduction due to accident-avoidance (ΔU) by summing the products of the following terms for each plant-type:
   - the occupational dose reduction [$\Delta(\bar{F}D_R)$, from step 4]
   - the number of affected plants (N, from step 2)
   - the average remaining operating life ($\bar{T}$, from step 3).

   The upper and lower bounds on ΔU are calculated using the formulae in Section 3.5.2.

6. Estimate the amount of labor to be spent in radiation zones during implementation of the SIR for PWR and BWR backfit plants.

7. Calculate the incremental occupational dose increase per plant for implementation of the SIR (D, see Section 3.4.2.1) by multiplying the labor estimates from step 6 by the occupational dose-rate factors (discussed in Section 3.4.2.1).

8. Calculate the total occupational dose increase for implementation of the SIR (ND) by summing the products of the following terms for each plant-type:

   - The per-plant occupational dose increase for SIR implementation (D, from step 7)
   - The number of affected plants (N, from step 2).

9. Estimate the annual amount of labor to be spent in radiation zones for operation and maintenance of the SIR for each plant-type.

10. Calculate the incremental occupational dose increase per plant for operation and maintenance of the SIR (D_o, see Section 3.4.2.2) by multiplying the labor estimates in step 9 by the occupational dose-rate factors (discussed in Section 3.4.2.1).

11. Calculate the total occupational dose increase for operation and maintenance of the SIR ($N\bar{T}D_o$) by summing the products of the following terms for each plant-type:

    - The per-plant occupational dose increase for SIR operation and maintenance (D_o, from step 10)
    - the number of affected plants (N, as above)
    - the average remaining operating life ($\bar{T}$, from step 3).

12. Sum ND and $N\bar{T}D_o$ from steps 8 and 11 to obtain the total occupational dose increase due to SIR (G). Its upper and lower bounds are calculated using the formulae in Section 3.5.3.

5.2 SAFETY ISSUE COSTS

The calculations of industry and NRC costs due to resolution of a safety issue have been combined into a single work sheet. The procedure is similar to those used for public risk and occupational dose in that information from Sections 4.1 and 4.2 have been systematized into a "Safety Issue Cost Work Sheet" presented in Table 5.3.

The following text describe this procedure in the work sheet. Any detailed calculations or deviations from the standard procedure need to be documented on separate pages and referenced in the work sheet. It is anticipated that these supporting analyses will be more voluminous for cost than for risk/dose. A typical Safety Issue Cost Work Sheet is shown as Table 6.3.4.

Like public risk, cost calculations are performed for PWRs and BWRs. In addition, each of these is further divided into backfit and forward-fit classes. Calculations need to be performed for both BWRs and PWRs for industry cost savings due to accident-avoidance. Implementation-cost-related calculations may need separate treatment for backfit and forward-fit plants. Operation/maintenance-cost-related calculations are applicable to all BWR and PWR plants. NRC SIR development costs typically apply to the nuclear industry as a whole. Certain steps, as indicated in the following procedure, must be repeated for each of these calculations.

<u>Outline</u>

1. Define the safety issue and understand the resolution postulated as part of the public risk reduction calculations.
2. Determine which plants the issue affects. This corresponds to step 2 of the occupational dose work sheet (ODWS).
3. Estimate the average remaining life ($\bar{T}$) in each of the four classes of affected plants. See step 3 of the ODWS.
4. Calculate the per-plant industry cost savings due to accident-avoidance associated with the SIR [$\Delta(\bar{F}A)$, see Section 4.1.3] by multiplying the following terms:
   - the cost associated with cleanup, repair, and refurbishment of a facility (plus replacement power) following a core-melt accident (A, see Section 4.1.3 and Appendix E)

TABLE 5.3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:
2. Affected Plants (N):
   (see step 2, Table 5.2)
3. Average Remaining Lives of Affected Plants ($\bar{T}$):

Industry Costs (steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident-Avoidance, $\Delta(\bar{F}A)$:
5. Total Industry Cost Savings Due to Accident-Avoidance (ΔH):
   (also list upper and lower bounds)
6. Per-Plant Industry Resources for SIR Implementation:
7. Per-Plant Industry Cost for SIR Implementation (I):
8. Total Industry Cost for SIR Implementation (NI):
9. Per-Plant Industry Labor for SIR Operation and Maintenance:
10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_o):
11. Total Industry Cost for SIR Operation and Maintenance ($N\bar{T}I_o$):
12. Total Industry Cost (S_I):
    (also list upper and lower bounds)

NRC Costs (steps 13 through 21)

13. NRC Resources for SIR Development:
14. Total NRC Cost for SIR Development (C_D):
15. Per-Plant NRC Labor for Support of SIR Implementation:
16. Per-Plant NRC Cost for Support of SIR Implementation (C):
17. Total NRC Cost for Support of SIR Implementation (NC):
18. Per-Plant NRC Labor for Review of SIR Operation and Maintenance:
19. Per-Plant NRC Cost for Review of SIR Operation and Maintenance (C_o):
20. Total NRC Cost for Review of SIR Operation and Maintenance ($N\bar{T}C_o$):
21. Total NRC Cost (S_N):
    (also list upper and lower bounds)

- the reduction in core-melt frequency ($\Delta\bar{F}$, from step 15 of the PRRWS).

This product is the reduction in the expected cost from a core-melt accident. $\Delta(\bar{F}A)$ must be estimated for BWR and PWR plants.

5. Calculate the total industry cost savings due to accident-avoidance (ΔH) by summing the products of the following terms for each plant-type:

   - the industry cost savings due to accident-avoidance [$\Delta(\bar{F}A)$, from step 4]
   - the number of affected plants (N, from step 2)
   - the average remaining operating life ($\bar{T}$, from step 3).

   The upper and lower bounds on ΔH are calculated using the formulae in Section 4.3.1.

6. Estimate the amounts of the following resources needed by industry to implement the SIR in PWRs and BWRs (backfit and forward-fit):
   - labor
   - additional down-time requiring purchase of replacement power
   - equipment.

7. Calculate the incremental industry cost per plant for SIR implementation (I, see Section 4.1.1). Labor and down-time estimates are multiplied by labor and replacement power cost rates, respectively, from Appendix E. Equipment costs are estimated on a case-by-case basis. These three factors are summed to obtain per-plant implementation costs in each of the four plant classes.

8. Calculate the total industry cost for implementation of the SIR (NI) by summing the products of the following terms for each plant-type:

   - the per-plant industry cost for SIR implementation (I, from step 7)
   - the number of affected plants (N, from step 2)

9. Estimate the annual amount of labor to be spent for operation and maintenance of the SIR for each plant-type (PWR, BWR).

10. Calculate the incremental industry cost per plant for operation and maintenance of the SIR (I_0, Section 4.1.2) by multiplying the labor estimates in step 9 by the industry labor cost rate from Appendix E.

11. Calculate the total industry cost for operation and maintenance of the SIR ($N\bar{T}I_0$) by summing the products of the following terms for each plant-type:

    - the per-plant industry cost for SIR operation/maintenance (I_0, from step 10)
    - the number of affected plants (N, as above)
    - the average remaining operating life ($\bar{T}$, from step 3)

12. Calculate the total industry cost due to the SIR (S_I) by summing NI and $N\bar{T}I_0$ from steps 8 and 11. The upper and lower bounds are calculated using the formulae in Section 4.3.2.

13. Estimate the future amount of NRC resources to develop the SIR.

14. Multiply the NRC resource estimates and NRC cost rates (for labor cost rates, see Appendix E) to obtain the total NRC SIR development cost (C_D, see Section 4.2.1).

15. Estimate the amount of NRC labor per plant needed to support SIR implementation.

16. Multiply the NRC labor estimates and cost rates (see Appendix E) to obtain the incremental NRC cost per plant to support SIR implementation (C, see Section 4.2.2).

17. Calculate the total NRC cost for support of the SIR implementation by summing the products of the following terms for each plant-type:

    - the per-plant NRC cost to support SIR implementation (C, from step 16)
    - the number of affected plants (N, from step 2).

18. Estimate the annual amount of NRC labor to be spent in the review of ongoing maintenance and operation of the SIR per plant.

19. Calculate the incremental NRC cost per plant to review the operation and maintenance of the SIR (C_o, see Section 4.2.3) by multiplying the labor estimates in step 16 by the NRC labor cost rate from Appendix E.

20. Calculate the total NRC cost to review the operation and maintenance of the SIR ($N\bar{T}C_o$) by summing the products of the following terms for each plant-type:

    - the per-plant NRC cost to review SIR operation/maintenance (C_o, from step 19)
    - the number of affected plants (N, as above)
    - the average remaining operating life ($\bar{T}$, from step 3)

21. Calculate the total NRC cost due to the SIR (S_N) by summing C_D, NC, and $N\bar{T}C_o$ from steps 14, 17 and 20. The upper and lower bounds are calculated using the formulae in Section 4.3.3.

5.3 SAFETY ISSUE SUMMARY WORK SHEET

This section presents a work sheet that is utilized to summarize the results of the previous risk, dose and cost analyses in a standardized format. The work sheet is a single page and is intended only for summary purposes. Persons interested in additional detail should refer to the appropriate supporting work sheets described previously. The remainder of this section gives specific instructions for each entry in the work sheet. The format is shown in Table 5.4. See Section 6.0 for examples of completed work sheets.

The first entry of Issue Number and Title is identical to that on the previous work sheets. The Summary of the Problem is intended to be a very brief statement of the safety issue and the proposed resolution. For brevity, this description should not exceed the space allowed. The Numbers of Plants Affected by the SIR are listed next.

The entries in the Risk/Dose Results section summarize results from the Public Risk Reduction and Occupational Dose Work Sheets. The entry in the Public Risk section is from step 17 of Table 5.1. Entries in the Occupational Dose section are from steps 8, 11, 12, and 5 of Table 5.2, respectively.

TABLE 5.4. Issue Summary Work Sheet

ISSUE NO./TITLE:

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

AFFECTED PLANTS	BWR: Operating =	Planned =
	PWR: Operating =	Planned =

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =

OCCUPATIONAL DOSES:

SIR Implementation =

SIR Operation/Maintenance =

Total of Above =

Accident-Avoidance =

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation =

SIR Operation/Maintenance =

Total of Above =

Accident-Avoidance =

NRC COSTS:

SIR Development =

SIR Implementation Support =

SIR Operation/Maintenance Review =

Total of Above =

The entries in the Cost Results section summarize results from the Safety Issue Cost Work Sheet (Table 5.3). Entries in the Industry Cost section correspond to steps 8, 11, 12, and 5 respectively. Entries in the NRC Cost section correspond to steps 14, 17, 20, and 21, respectively.

5.4 UNCERTAINTIES ON COMBINATIONS OF SAFETY ISSUE RANKING PARAMETERS

The six parameters $[(\Delta W)_{Total}$, ΔU, ΔH, G, S_I and $S_N]$ developed in this project for use as input in ranking NRC safety issues can be combined in various ways to yield ranking measures. Both best estimates and error bounds will be calculated for these parameters using the techniques developed here. Several options exist for combining these best estimates and error bounds, one of which is an arithmetic combination. Section F.3 of Appendix F discusses some approximate procedures for arithmetically combining uncertainties.

6.0 ANALYSES OF EXAMPLE SAFETY ISSUES

This section presents analyses for three example safety issues: Training and Qualifications of Operations Personnel (Issue I.A.2.2, Section 6.1), Diesel Generator Reliability (Issue B-56, Section 6.2) and Steam Line Break with Consequential Small LOCA (Issue 18, Section 6.3).

The purpose of this section is twofold: 1) to provide further clarification on the risk, dose and cost analysis methods and 2) to indicate the standard format for reporting issue analyses. All subsections in Section 6.0 provide unique guidance on the application of the methods developed in Sections 2.0 through 5.0 to specific issues. Section 6.1 presents the analysis of an issue resolution dealing exclusively with human factors. Section 6.2 details the step-by-step completion of the work sheets for a specific issue. Section 6.3 illustrates the level of detail presumed appropriate for most issues whose analyses require use only of the standard techniques discussed previously.

6.1 TRAINING AND QUALIFICATIONS OF OPERATIONS PERSONNEL: TMI ACTION PLAN ITEM I.A.2.2.

The training and qualifications of operations personnel are covered under TMI Action Plan (TAP) Item I.A.2.2 (NUREG-0660 1980). This issue is chosen to demonstrate the methodology developed in this report because it is representative of many training-related issues. The results of the analysis for the issue are summarized in Table 6.1.1.

6.1.1 Safety Issue Description

Under the TAP, the NRC may require reactor licensees to review their training and qualifications programs for all operations personnel. This is interpreted to include licensed and auxiliary operators, technicians, maintenance personnel and supervisors. The review will examine current practices in light of the safety significance of the duties of the operations staff. If the review determines that the current practices adequately assure proper

TABLE 6.1.1. Issue Summary Work Sheet

ISSUE NO./TITLE: I.A.2.2, Training and Qualifications of Operations Personnel

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

This TMI Action Plan Item recognizes a need to improve the safety-related performance of operations personnel through improvement in training and qualifications programs at all plants.

AFFECTED PLANTS	BWR:	Operating = 24	Planned = 20
	PWR:	Operating = 47	Planned = 43

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	1.5E+5
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	-2.3E+5
Total of Above =	-2.3E+5
Accident-Avoidance =	950

COST RESULTS ($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	45
SIR Operation/Maintenance =	610
Total of Above =	650
Accident-Avoidance =	78
NRC COSTS:	
SIR Development =	0.055
SIR Implementation Support =	0.055
SIR Operation/Maintenance Review =	2.8
Total of Above =	2.9

safety-related staff conduct, then the justification for this determination must be documented. The documentation need not be submitted to the NRC but must be maintained on site. If the review uncovers inadequacies, the licensee is required to upgrade the training and qualifications practices to ensure adequate performance of operations personnel.

Guidance from the NRC to the utilities on this issue is not yet completed. The TAP, however, does suggest the use of position task analysis. The Institute for Nuclear Power Operations (INPO) has completed such analyses for a portion of the operations staff positions. Furthermore, INPO has surveyed utilities on their current training practices and their plans for improvement.

The risk, dose and cost of resolving this issue are difficult to quantify because the issue does not relate specifically to plant systems. The incomplete nature of NRC guidance to the utilities further compounds the difficulty. The analysis approach employed by PNL utilizes expert opinion to estimate the effect of training reviews on human error contributions to plant risk.

The first step in the development of the opinion was the assembly of a panel of experts from the PNL staff. This panel possessed considerable experience in reactor operations, utility training programs and reactor plant systems. The panel included reactor operator licensing examiners and members with utility field experience.

The judgments of the panel, are based on two major insights.

(1) The potential effect of this issue is limited by its semi-voluntary nature. That is, the judgment of adequacy is in the hands of the individual utilities. Furthermore, the current INPO and NRC research work in task analysis deals generically with routine operations. Plant-specific operation and operation under upset conditions are left to the individual utilities. This dilutes the effectiveness of the task analysis efforts in providing the bases for the training and qualifications review.

Related activities which are supported by and in turn support this issue are the conduct of plant drills and the accreditation of training programs. While neither of these is directly required by the training and qualifications review, both could be a part of the response, and both would have a positive effect on personnel performance.

(2) There is a wide variation among utilities in both the training programs and the performance of operations staff. In many facilities there is much room for improvement. Therefore, while the potential effect of the training and qualifications review effort is limited, a significant overall reduction in safety-related human error for operations personnel is expected because of the wide margin available for improvement.

Affected Plants

In estimating the risk, dose and costs, and PNL panel divided licensees into three groups. This division and the assignment of the fractions of the affected reactor population to each group are somewhat arbitrary. However, they reflect the panel's best engineering judgment based on its experience. These groups are as follows.

(1) Minimally-affected group. These utilities currently have a good effective training and qualifications program and good operations personnel performance. They would be minimally affected by resolution of this safety issue. The fraction of affected reactors in this group is estimated to be 15% of the total population.

(2) Intermediately-affected group. These utilities' training and qualifications programs and/or operations personnel performance have room for improvement. This group, estimated to be 60% of the population, would undergo improvements and therefore be affected more than the first group.

(3) Maximally-affected group. These utilities have deficiencies in their training and qualifications programs and in operations personnel performance. They would be significantly affected by resolution of this safety issue, and major restructuring of programs would be expected. This group is estimated to contain 25% of all reactor licensees.

It is important to emphasize that any implication of nuclear reactors being operated in an unsafe manner is not intended. The standards by which performance at nuclear facilities are measured are, and should be, high. There are facilities which meet or exceed these standards. The performance at other facilities is judged to be less desirable.

6.1.2 Safety Issue Risk and Dose

The panel's judgment regarding potential decreases in human error probabilities and annual occupational dose is quantified below.

	Group			
	Minimally Affected	Intermediately Affected	Maximally Affected	Weighted Average
Fraction of Total Reactor Population	0.15	0.60	0.25	1.0
% Decrease in Human Error Probability for[a]				
1. Licensed Operators (RO, SRO)	5 (0, 10)	15 (10, 30)	30 (20, 50)	17 (11, 32)
2. Other Operations Staff (Technicians, Maintenance, etc.)	10 (5, 20)	25 (10, 40)	45 (25, 60)	28 (13, 42)
% Decrease in Annual Occupational Dose[a,b]	NE (5, 10)	NE (10, 15)	NE (20, 25)	NE (12, 17)

(a) Best estimates are given with lower and upper bounds, respectively, in parenthesis.
(b) NE = not estimated.

The table shows an increasing improvement in human error probability from the minimally to the maximally-affected groups. The error bounds show potential overlap between groups. The greater improvement in the "Other Operations Staff" category as compared to that of "Licensed Operators" recognizes that the former group is exposed to relatively less extensive training. It is postulated that, if training programs are improved, the performance of maintenance personnel and technicians could be improved more.

Also shown in the above table is the decrease in annual occupational dose associated with issue resolution. With the potential for improved training, this SIR is likely to cause a decrease in occupational dose. Improved performance of maintenance personnel and technicians is expected to reduce their time in radiaion zones and thereby decrease the overall exposure.

The values given above are in terms of percent changes. The reductions in human error probability must be transformed into the resulting reduction in public risk. The decreases in annual occupational exposure must be transformed from percents into man-rem. The estimates of public risk reduction and occupational dose are discussed in the following subsections. Analysis results are summarized in Tables 6.1.2 and 6.1.3, respectively.

6.1.2.1 Public Risk Reduction

The proposed resolution of TAP Item I.A.2.2 has been discussed. As indicated, this issue resolution centers around operator and maintenance staff training programs to improve personnel performance. This relates generically to both BWRs and PWRs, and ideally the risk reduction would be estimated by selecting a representative plant of each type. However, operator and maintenance staff performance impact essentially all accident sequences in the risk equations. To keep the analysis tractable, the calculations are performed for one representative PWR, and inferences are drawn for BWRs. Oconee 3 is selected as the representative PWR.

Resolution of I.A.2.2 deals with improvement in operator and operations staff performance. It is assumed that all parameters directly or indirectly related to operator or operations staff errors in the Oconee 3 risk equation will be affected by resolution of this issue. The values of these affected parameters will be altered by the weighted averages of the percent decreases in human error probabilities given earlier, i.e. 17% for operator errors and 28% for maintenance-related errors.

TABLE 6.1.2. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Training and Qualifications of Operations Personnel (I.A.2.2)

2. Affected Plants (N) and Average Remaining Lives ($\bar{T}$):

   All plants are assumed to be affected.

	N	$\bar{T}$
PWRs	90	28.8 yr
BWRs	44	27.4 yr

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR
   Grand Gulf 1 - representative BWR
   (The analysis is conducted for Oconee 3, and the results are scaled for Grand Gulf 1, as discussed in Attachment 1).

4. Parameters Affected by SIR:

   Oconee: B, C, D, E, CH1, CH2, CH3, CH4, CONST1, CONST2, A1, B1, C1, (B_3), K, HHMAN, HPMAN, HPMAN1, LPISCM, HPRSCM, RCSRBCM, WXCM, D•E, W•X, B•W, C•X, D•X, E•W, B•D, E•C, G1.

5. Base-Case Values for Affected Parameters:

   Original values from Appendix A are assumed.

6. Affected Accident Sequences and Base-Case Frequencies:

   All accident sequences, with the exception of V, are affected by issue resolution. Original frequencies are assumed for the base case.

7. Affected Release Categories and Base-Case Frequencies:

   All PWR release categories are affected by issue resolution. The original frequencies are assumed for the base case with the exception of PWR-2, from which the contribution of sequence V must be removed. Thus, PWR-2 = 6.0E-6/ry (reactor-year).

TABLE 6.1.2. (contd)

8. Base-Case, Affected Core-Melt Frequency ($\bar{F}$):

$\bar{F}_{PWR}$ = 7.8E-5/ry $\bar{F}_{BWR}$ = 3.5E-5/ry[a]

9. Base-Case, Affected Public Risk (W):

W_{PWR} = 188 man-rem/ry W_{BWR} = 225 man-rem/ry[a]

10. Adjusted-Case Values for Affected Parameters:

B = C	=	0.0025
D = E	=	0.021
CH1	= CH2 = CH3 = CH4 =	0.0044
CONST1	=	1.4E-4
CONST2	=	4.5E-4
A1 = C1	=	0.0092
B1	=	0.034
(B_3)	=	3.7E-4
K	=	2.0E-5
G1	=	0.012
HHMAN	= HPMAN1 =	0.083
HPMAN	=	0.012
LPISCM	=	0.0022
HPRSCM	= WXCM =	0.0025
RCSRBCM	=	2.3E-5
D•E	=	4.2E-4
W•X	=	7.9E-5
B•W	= C•X =	2.0E-5
D•X	= E•W =	1.8E-4
B•D	= E•C =	4.6E-5

11. Affected Accident Sequences and Adjusted-Case Frequencies:

$$T_2MLU - \begin{cases} \gamma \text{ (PWR-3)} = 3.2\text{E-7/ry} \\ \beta \text{ (PWR-5)} = 4.7\text{E-9/ry} \\ \varepsilon \text{ (PWR-7)} = 3.2\text{E-7/ry} \end{cases}$$

TABLE 6.1.2. (contd)

T_1MLU -	γ	(PWR-3)	= 5.3E-7/ry
	β	(PWR-5)	= 7.8E-9/ry
	ε	(PWR-7)	= 5.3E-7/ry
$T_1(B_3)MLU$ -	γ	(PWR-3)	= 6.5E-7/ry
	β	(PWR-5)	= 9.5E-9/ry
	ε	(PWR-7)	= 6.5E-7/ry
T_2MQH -	γ	(PWR-3)	= 4.5E-6/ry
	β	(PWR-5)	= 6.6E-8/ry
	ε	(PWR-7)	= 4.5E-6/ry
S_3H -	γ	(PWR-3)	= 4.0E-6/ry
	β	(PWR-5)	= 5.8E-8/ry
	ε	(PWR-7)	= 4.0E-6/ry
S_1D -	α	(PWR-1)	= 5.9E-8/ry
	γ	(PWR-3)	= 1.2E-6/ry
	β	(PWR-5)	= 4.3E-8/ry
	ε	(PWR-7)	= 4.7E-6/ry
T_2MQFH -	γ	(PWR-2)	= 2.1E-6/ry
	β	(PWR-4)	= 3.1E-8/ry
	ε	(PWR-6)	= 2.1E-6/ry
S_3FH -	γ	(PWR-2)	= 1.7E-6/ry
	β	(PWR-4)	= 2.5E-8/ry
	ε	(PWR-6)	= 1.7E-6/ry
S_2FH -	α	(PWR-1)	= 1.0E-8/ry
	β	(PWR-4)	= 7.6E-9/ry
	ε	(PWR-6)	= 8.4E-7/ry

TABLE 6.1.2. (contd)

T_2MLUO -	γ (PWR-3)	=	3.5E-6/ry
	β (PWR-5)	=	5.1E-8/ry
	ϵ (PWR-7)	=	3.5E-6/ry
T_2KMU -	γ (PWR-3)	=	2.5E-6/ry
	β (PWR-5)	=	3.6E-8/ry
	ϵ (PWR-7)	=	2.5E-6/ry
S_2D -	α (PWR-1)	=	1.5E-8/ry
	γ (PWR-3)	=	3.1E-7/ry
	β (PWR-5)	=	1.1E-8/ry
	ϵ (PWR-7)	=	1.2E-6/ry
S_3D -	γ (PWR-3)	=	6.3E-7/ry
	β (PWR-5)	=	9.4E-9/ry
	ϵ (PWR-7)	=	6.3E-7/ry
T_1MLUO -	γ (PWR-3)	=	2.3E-6/ry
	β (PWR-5)	=	3.4E-8/ry
	ϵ (PWR-7)	=	2.3E-6/ry
T_3MLUO -	γ (PWR-3)	=	4.6E-7/ry
	β (PWR-5)	=	6.7E-9/ry
	ϵ (PWR-7)	=	4.6E-7/ry
T_2MQD -	γ (PWR-3)	=	6.7E-7/ry
	β (PWR-5)	=	9.8E-9/ry
	ϵ (PWR-7)	=	6.7E-7/ry

(Note: the contributions from the non-dominant minimal cut sets are assumed to decrease in the same proportions as those from the dominant minimal cut sets in all affected accident sequences.)

TABLE 6.1.2. (contd)

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-1 = 9.2E-8/ry

PWR-2 = 5.0E-6/ry

PWR-3 = 2.3E-5/ry

PWR-4 = 8.0E-8/ry

PWR-5 = 3.7E-7/ry

PWR-6 = 6.0E-6/ry

PWR-7 = 2.8E-5/ry

(Note: the contributions from the non-dominant accident sequences are assumed to decrease in the same proportions as those from the dominant accident sequences in all affected release categories, with sequence V excluded.)

13. Adjusted-Case, Affected Core-Melt Frequency ($\bar{F}^*$):

$\bar{F}^* = 6.3E-5/ry$

14. Adjusted-Case, Affected Public Risk (W^*):

$W^*_{PWR} = 150$ man-rem/ry

15. Core-Melt Frequency Reduction ($\Delta\bar{F}$):

$(\Delta\bar{F})_{PWR} = 1.5E-5/ry$ $(\Delta\bar{F})_{BWR} = 6.8E-6/ry$[a]

16. Per-Plant Public Risk Reduction (ΔW):

$(\Delta W)_{PWR} = 38$ man-rem/ry $(\Delta W)_{BWR} = 46$ man-rem/ry[a]

17. Total Public Risk Reduction [$(\Delta W)_{Total}$]:

Best Estimate	Upper Bound	Lower Bound
1.5E+5 man-rem	2.3E+7 man-rem	0

(a) See Attachment 1.

ATTACHMENT 1

The RSSMAP studies for Oconee 3 and Grand Gulf 1 give total core-melt frequencies ($\bar{F}_o$) of 8.2E-5/ry and 3.7E-5/ry, respectively, for these plants. Using the original release category frequencies and the public dose factors (Appendix D), one obtains total public risks (W_o) of 207 man-rem/ry and 250 man-rem/ry, respectively, for Oconee and Grand Gulf. For the purposes of scaling the base-case, affected core-melt frequency ($\bar{F}$) and public risk (W), and the reductions in the core-melt frequency ($\Delta\bar{F}$) and public risk (ΔW) from Oconee to Grand Gulf, the following are assumed:

$$\left.\begin{array}{l} \bar{F}_{BWR}/\bar{F}_{PWR} \\ \\ (\Delta\bar{F})_{BWR}/(\Delta\bar{F})_{PWR} \end{array}\right\} = (\bar{F}_o)_{BWR}/(\bar{F}_o)_{PWR}$$

$$\left.\begin{array}{l} W_{BWR}/W_{PWR} \\ \\ (\Delta W)_{BWR}/(\Delta W)_{PWR} \end{array}\right\} = (W_o)_{BWR}/(W_o)_{PWR}$$

Using the original values of $\bar{F}_o$ and W_o for Oconee and Grand Gulf, the scaling equations become:

$$\bar{F}_{BWR} = 0.45\ \bar{F}_{PWR}$$

$$(\Delta\bar{F})_{BWR} = 0.45\ (\Delta\bar{F})_{PWR}$$

$$W_{BWR} = 1.2\ W_{PWR}$$

$$(\Delta W)_{BWR} = 1.2\ (\Delta W)_{PWR}$$

TABLE 6.1.3. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

Training and Qualifications of Operations Personnel (I.A.2.2)

2. Affectd Plants (N):

All plants are assumed to be affected.

	N
PWRs	90
BWRs	44

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

	$\bar{T}$
PWRs	28.8 yr
BWRs	27.4 yr

4. Per-Plant Occupational Dose Reduction Due to Accident-Avoidance [$\Delta(\bar{F}D_R)$]:

PWR: (19,900 man-rem)(1.5E-5/ry) = 0.30 man-rem/ry

BWR: (19,900 man-rem)(6.8E-6/ry) = 0.14 man-rem/ry

5. Total Occupational Dose Reduction Due to Accident-Avoidance (ΔU):

Best Estimate	Upper Bound	Lower Bound
950 man-rem	2.9E+4 man-rem	0

6-8. Steps Related to Occupational Dose Increase for SIR Implementation:

Since SIR implementation involves improving training programs, no occupational dose will be accrued. Thus, D = 0.

9. Per-Plant Utility Labor in Radiation Zones for SIR Operation/Maintenance:

Dose increase is estimated directly in next step.

10. Per-Plant Occupational Dose Increase for SIR Operation/Maintenance (D_O):

D_O = -60 man-rem/ry (negative sign indicates reduction)

This applies to all plants.

TABLE 6.1.3. (contd)

11. Total Occupational Dose Increase for SIR Operation/Maintenance ($N\bar{T}D_O$):

$N\bar{T}D_O$ = -2.3E+5 man-rem

12. Total Occupational Dose Increase (G):

Best Estimate	Upper Bound	Lower Bound
-2.3E+5 man-rem	-7.6E+4 man-rem	-6.9E+5 man-rem

(negative signs indicate reductions)

It will be assumed that the 17% reduction in operator error can be applied directly to minimal cut set elements involving operator errors and the 28% can be applied directly to minimal cut set elements involving maintenance errors. This assumption introduces some error in the maintenance contribution since some maintenance operations on nuclear systems have fixed times associated with cool-down, preparation, etc., in addition to somewhat variable staff labor time that would be subject to improvement through training. Maintenance performed properly the first time also reduces the frequency of maintenance outages and the down-time for proper repairs at some future date. Thus, fixed time periods in maintenance outages are indirectly reduced over the long run by improved performance simply because the need for maintenance may be reduced for all systems except those that undergo preventative maintenance at set intervals.

Multiple Maintenance Contributions

The list of elements in the dominant minimal cut sets for Oconee 3 are examined for operator and maintenance terms (see Table A.4). It is pointed out in Appendix A that contributions frm multiple maintenance terms must be removed when calculating some variable and sequence values. This is due primarily to the presence of redundant success pathways in a system's maintenance procedures which would not preclude more than one pathway at one time. An example would be two parallel pump and valve trains for a feedwater system. Maintenance procedures would not allow both pumps to be down for maintenance at the same time. Thus, the probability of the event "maintenance on pump A and pump B" would be zero. To eliminate the contributions from multiple maintenance terms the following approach is used.

Two variables A and B containing maintenance and non-maintenance terms can be broken into $A = \bar{A} + m$, $B = \bar{B} + m$, where m (the maintenance contribution) has an equivalent value in both variables. The probability of the event "A and B" would then be written as:

$$
\begin{aligned}
P(AB) &= P(\bar{A}\bar{B} + \bar{A}m + \bar{B}m + m^2) \\
&= P(\bar{A}\bar{B}) + P(\bar{A}m + \bar{B}m) + P(m^2)
\end{aligned}
$$

Recognizing that $P(m^2) = 0$, the expression becomes:

$$P(AB) = P(A) \cdot P(B) - P^2(m)$$

Triple maintenance terms can arise with variables containing two or more maintenance events. For example, a variable C containing two equivalent maintenance terms would be expressed as $C = \bar{C} + 2m$. The probability of an event such as "A and C" would then be estimated as:

$$\begin{aligned} P(AC) &= P\,[(\bar{A} + m) \cdot (\bar{C} + 2m)] \\ &= P\,[\bar{A}\bar{C} + 2\bar{A}m + \bar{C}m + 2m^2] \\ &= P(A) \cdot P(C) - 2P^2(m) \end{aligned}$$

CONST1 AND CONST2

The Boolean expansion of the terms CONST1 and CONST2, which deal with the failure of the emergency feedwater system due to failure of the turbine pump train, electric pump trains, and blockage of flow through steam generator discharge lines, is given in Addendum A.I. The terms comprising these two elements as listed in Table A.I-1 are examined for maintenance terms, breaking them up into maintenance and non-maintenance components. The terms are then expressed as

$A3 = A + m$

$B3 = B + m$

$E3 = E + 2m$

$G3 = G + 2m$

$F3 = F + 3m$

$P3 = P + 4m.$

The equations developed previously for multiple maintenance contributions are used to derive correction factors by which the original values given for the products of terms comprising CONST1 and CONST2 (see Table A.I-1) can be obtained from the original values given for the individual terms in these products (see Table A.I-2). The values for the products of the above terms are corrected by subtracting the multiple maintenance contributions from the products of the individual values. The products with their correction factors are given below. Note that the probability notation has been dropped, i.e., $P(m) = m$.

Correction Factors for CONST1 and CONST2

Product	Correction Factor[a]
A3 • B3	m^2
E3 • F3 • G3	$(6E + 4F + 6G)\ m^2$
A3 • G3 • F3	$(6A + 3G + 2F)\ m^2$
E3 • F3 • B3	$(3E + 2F + 6B)\ m^2$
E3 • P3 • G3	$(8E + 4P + 8G)\ m^2$
E3 • P3 • B3	$(4E + 2P + 8B)\ m^2$
A3 • G3 • P3	$(8A + 4G + 2P)\ m^2$
E3 • G3	$4m^2 - 0.054(E3 \cdot G3)$ (b)
E3 • B3	$2m^2 - 0.028(E3 \cdot B3)$ (b)
A3 • G3	$2m^2 - 0.028(A3 \cdot G3)$ (b)

(a) Must be subtracted from product of individual terms, e.g.,

$$E3 \cdot P3 \cdot G3 = (0.017)^2(0.036) - [8(0.0054) + 4(0.013) + 8(0.0054)](0.0058)^2 = 5.8E\text{-}6$$

where:

$E3 = G3 = 0.017$
$P3 = 0.036$
$m = 0.0058$
$E = E3 - 2m = 0.0054$
$G = G3 - 2m = 0.0054$
$P = P3 - 4m = 0.013$

(b) Subtractive terms are minor corrections to account for round-of errors.

No attempt is made to eliminate common maintenance terms above the system level. This was the approach used in calculating the values in the original RSSMAP study. Accident sequence frequencies are therefore calculated as the sum of the products of the variable strings comprising the minimal cut sets. Note also that no attempt is made to eliminate multiple operator error terms. These are covered by variables addressing common-cause failures.

It is assumed that issue resolution would apply to all plants existing and planned, as given in Appendix C. The base-case, affected core-melt frequency and public risk for a representative BWR (Grand Gulf 1) are scaled from the corresponding values estimated for Oconee 3. Likewise, the reductions in core-melt frequency and public risk for Grand Gulf 1 are scaled from the corresponding values estimated for Oconee 3. These calculations are discussed in Attachment 1 to Table 6.1.2.

6.1.2.2 Occupational Dose

No increase in occupational dose will result from implementation of the issue resolution since this involves improving training programs. However, the PNL panel felt that occupational dose accrued during annual operation and maintenance might decrease as a result of SIR. Based on the PNL panel's estimates, a weighted-average decrease of 12%-17% in annual occupational dose is estimated to result from SIR (see table at beginning of Section 6.1.2).

It is estimated that workers at a nuclear plant currently accumulate an average of 300-500 man-rem/ry of routine exposure. Applying the PNL panel's estimates, a decrease of 36-85 man-rem/ry in occupational exposure appears feasible. A value of 60 man-rem/ry is assumed to be the potential decrease in occupational dose resulting from the SIR.

6.1.3 Safety Issue Costs

Costs to the industry for SIR implementation, operation and maintenance were estimated by the PNL panel. The results are given below.

	Group			
	Minimally Affected	Intermediately Affected	Maximally Affected	Weighted Average
Fraction of Total Reactor Population	0.15	0.60	0.25	1.0
Implementation Cost ($\$10^3$/plant)	100	325	500	335
Operational Cost ($\$10^3$/ry)	50	150	250	160

NRC labor to develop the SIR and support its implementation are taken from the TAP to be 1.1 man-yr. Assuming an equal division between development and implementation support gives an estimate of 0.55 man-yr for each. NRC labor to review documentation and new training programs resulting from the SIR is estimated to require one man-yr/yr. These estimates apply over the industry as a whole. Results of the cost analysis are summarized in Table 6.1.4.

TABLE 6.1.4. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Training and Qualifications of Operations Personnel (I.A.2.2)

2. Affected Plants (N):

   All plants are assumed to be affected.

	N
PWRs	90
BWRs	44
All	134

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

	$\bar{T}$
PWRs	28.8 yr
BWRs	27.4 yr
All	28.3 yr

Industry Costs (steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident-Avoidance [$\Delta(\bar{F}A)$]:

   PWR: (\$1.65E+9)(1.5E-5/ry) = \$2.5E+4/ry

   BWR: (\$1.65E+9)(6.8E-6/ry) = \$1.1E+4/ry

5. Total Industry Cost Savings Due to Accident-Avoidance (ΔH):

Best Estimate	Upper Bound	Lower Bound
\$7.8E+7	\$2.4E+9	0

6. Per-Plant Industry Resources for SIR Implementation:

   Cost is estimated directly in next step.

7. Per-Plant Industry Cost for SIR Implementation (I):

   I = \$3.35E+5/plant

   This applies to all plants.

TABLE 6.1.4. (contd)

8. Total Industry Cost for SIR Implementation (NI):

   NI = \$4.49E+7

9. Per-Plant Industry Labor for SIR Operation/Maintenance:

   Cost is estimated directly in next step.

10. Per-Plant Industry Cost for SIR Operation/Maintenance (I_0):

   I_0 = \$1.60E+5/ry
   This applies to all plants.

11. Total Industry Cost for SIR Operation/Maintenance ($N\bar{T}I_0$):

   $N\bar{T}I_0$ = \$6.08E+8

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$6.5E+8	\$9.6E+8	\$3.5E+8

NRC Costs (steps 13 through 21)

13. NRC Resources for SIR Development:

   0.55 man-yr

14. Total NRC Cost for SIR Development (C_D):

   C_D = \$5.5E+4

15. Per-Plant NRC Labor for Support of SIR Implementation:

   Cost is estimated directly in step 17.

16. Per-Plant NRC Cost for Support of SIR Implementation (C):

   Cost is estimated directly in step 17.

17. Total NRC Cost for Support of SIR Implementation (NC):

   NC = (0.55 man-yr)(\$1.0E+5/man-yr) = \$5.5E+4

TABLE 6.1.4. (contd)

18. Per-Plant NRC Labor for Review of SIR Operation/Maintenance:

Cost is estimated directly in step 20.

19. Per-Plant NRC Cost for Review of SIR Operation/Maintenance (C_O):

Cost is estimated directly in step 20.

20. Total NRC Cost for Review of SIR Operation/Maintenance ($N\bar{T}C_O$):

$N\bar{T}C_O$ = (1 man-yr/yr)(28.3 yr)($1.0E+5/man-yr) = $2.83E+6

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$2.9E+6	$4.4E+6	$1.5E+6

6.2 DIESEL GENERATOR RELIABILITY: GENERIC SAFETY ISSUE B-56

In the third quarter of 1977, NRC initiated Generic Safety Issue B-56 "Diesel Reliability" (Clemenson 1977). This safety issue was instigated by examination of Licensee Event Reports (LERs) on experience with diesel generators from 1969 through 1975 which indicated that emergency onsite diesel generators at operating plants were demonstrating an average starting reliability of about 0.94/demand compared with the NRC's goal for new plants of 0.99/demand as expressed in Regulatory Guide 1.108. The NRC awarded a contract to the University of Dayton Research Institute to identify the more significant causes of diesel generator unreliability. The significant causes of diesel generator failures at operating plants and the recommended corrective measures are reported in NUREG/CR-0660, Enhancement of Onsite Emergency Diesel Generator Reliability (Boner 1979).

In a memorandum from P. Check to T. Novak dated July 30, 1980, the Division of Systems Integration (DSI) recommended the backfitting of Regulatory Guide 1.108 diesel generator testing frequency and associated failure reporting requirements to all operating plants. In two memorandums from D. Ross to D. Eisenhut dated September 24, 1980 and October 6, 1980, DSI also recommended the implementation of the NUREG/CR-0660 recommended remedial actions at all operating plants as the final actions of Generic Safety Issue B-56. In November of 1980, the Division of Licensing recognized some inconsistency between the recommendations of NUREG/CR-0660 and the Regulatory Guide 1.108 testing frequency requirements, and requested that the Division of Safety Technology (DST) develop a comprehensive program to address the necessity and urgency of the DSI recommended actions at operating plants.

6.2.1 Safety Issue Description

Resolution of this safety issue involves two components:

1. Implementing the Diesel Generator Interim Reliability Program to determine which diesel generators require reliability improvement
2. Implementing hardware and/or procedural fixes to improve the reliability of those diesel generators.

These are discussed in the following two sections.

6.2.1.1 Diesel Generator Interim Reliability Program

The Diesel Generator Interim Reliability Program provides criteria for surveillance test frequency, reporting, and remedial action as a function of diesel generator failure experience. The requirement for reliability improvement programs at the nuclear plants and the possibility of a requalification testing program should induce licensees to maintain acceptably high diesel generator reliability.

Each diesel generator unit in service at a nuclear power plant may be subject to failures which may be attributed to nongeneric weaknesses in the unit's manufacture, installation or previous maintenance history. Similar individual units may, therefore, have quite different failure rates at different plant sites or even at the same site. The reliability should be established for each diesel generator unit at a site. Each unit should meet minimum reliability requirements to be considered operable, i.e., to continue to be utilized in the onsite emergency power system. A record of demands and failures should therfore be kept for each individual unit in the power system.

Under the normal test frequency each diesel generator unit should be subjected to a surveillance test no less frequently than once every 31 days or at a more frequent interval if deemed necessary or advisable by the diesel generator manufacturer to maintain high reliability. The 31-day maximum test interval is consistent with the maximum recommended test interval for most other active components of emergency safety feature equipment.

However, to achieve a balanced sensitivity to abrupt degradation in a diesel generator unit's reliability in a timely fashion, an increased test frequency criterion is established. The increased test frequency requirement will reduce the normal surveillance test interval for an individual diesel generator unit to no greater than 7 days whenever two or more failures have been experienced in the last 20 valid demands performed on the unit. Two failures in 20 demands could be a point indication of a failure rate of about 0.1, or the threshold of acceptable diesel generator performance. Reducing the test interval will allow for a more timely accumulation of additional tests upon which to base a judgment of the reliability of the unit.

During requalification testing, the natural incentive is to minimize the requalification test period. A 24-hour minimum time interval is required in order to allow the diesel generator to return to an ambient (cold) temperature condition prior to each attempted start. Cold starting is recognized as the most severe expected starting condition in an emergency situation.

The two reliability levels at which definite actions are prescribed (0.95/demand and 0.90/demand) were selected by inspection of 1) the diesel generator failure/demand data used in completion of Task 1 of Unresolved Safety Issue (USI) A-44, "Station Blackout," 2) LER failure data for diesel generators for the period of 1978 through 1980, and 3) the results of Task 1 of USI A-44.

In evaluating the diesel generator contribution to the probability of a station blackout, USI A-44 evaluated LER data on diesel generator failures from 1976 through 1978. In general USI A-44 found the median value of diesel generator reliability to be 0.98/demand with about 75% of all units having an estimated reliability of 0.95/demand or greater. The DST has evaluated LER data from the period of 1978 through 1980. They found the median value of diesel generator reliability to be 0.97/demand and the distribution for those data to compare closely with the assessment made in USI A-44.

This program is developed around the concept of improving the reliability of those diesel generators which have demonstrated the poorest performance. The program will probably require no special actions for diesel generators with a reliability of 0.97/demand or greater (about 50% of those currently operating). The program will almost guarantee that diesel generators with a reliability of less than 0.95/demand will be required to be improved or eventually be removed from service. In addition, plants which utilize a diesel generator in their onsite emergency power system with a reliability in the range of 0.95 to 0.96/demand will also have a significant chance of being required to initiate reliability improvement efforts although the reliability of the unit is slightly greater than or equal to the minimum desired reliability level (0.95/demand).

By inspection of the LER data utilized in Task 1 of USI A-44 and the LER data evaluated by DST, two reliability limits were selected. A reliability of

0.95/demand was selected as the minimum desirable diesel generator reliability. A reliability of 0.90/demand was selected as the minimum acceptable diesel generator reliability. For the purpose of estimating the risk, dose and costs associated with resolution of this issue, it is assumed that 30% of all operating plants will have to implement a diesel generator reliability improvement program. A small fraction of these (5% of the operating plants) will presumably have to requalify their diesel generators.

6.2.1.2 Hardware/Procedural Fixes for Diesel Generators

For the 5% of the operating plants requiring diesel generator requalification, it is assumed that major repair of a diesel generator will be necessary. This may necessitate some additional plant down-time. For all 30% of the operating plants needing diesel generator reliability improvement, it is assumed that several of the changes proposed in NUREG/CR-0660 to improve the diesel generator reliability will be implemented. These changes are quite straightforward and do not impact other portions of the plant in any significant way. The proposed fixes are discussed in order of perceived value for improving reliability.

In NUREG/CR-0660 it is concluded that the principal cause of a diesel generator's failure to perform is a failure to start upon demand due to problems with the air-driven starters. It is proposed that placing air dryers on the compressed air system used for starting the diesel engines will greatly reduce the incidence of failures due to fouling of the starting motors by rust and scale deposits.

The second most likely cause of failure to perform is found to be failure of electrical contactors to close properly due to dust/dirt on the contact surfaces. Two remedies are proposed for this condition: 1) replacement of unsealed contactors with units having sealed dust-tight enclosures, and placing all switchgear inside enclosures with dust-tight seals on the openings; 2) installation of ventilation ducting to deliver outside air to the diesel generators, with appropriate filtration on the air intakes, and installation of diesel exhaust ducting to vent the exhaust gases to the outside of the building. The diesel generator room can then be made more dust-tight since air-inleakage is no longer required for cooling and combustion air, and air-outleakage is no longer the mechanism for escape of the diesel exhaust gases.

Another significant cause of diesel generator failure is failure of the drive gears for the turbocharger. It is recommended that the existig gear set be replaced with a heavy duty set with a slightly different gear ratio. A wide ranging set of recommendations are also made in NUREG/CR-0660 for changes in operating procedures for diesel generator units that should help to reduce failures to perform upon demand.

6.2.2 Safety Issue Risk and Dose

The public risk reduction and occupational dose parameters are estimated for the proposed issue resolution. A step-by-step description of the analysis is provided.

6.2.2.1 Public Risk Reduction

The procedure used to estimate the reduction in public risk follows that presented in Section 5.1.1. For demonstrative purposes, the analysis is detailed in a text format, with a summary work sheet (Table 6.2.1) provided at the end of this section. Generally, only the work sheet (with supplemental detail as necessary) will be needed, shortening the overall length of the presentation.

Issue Definition

The proposed resolution of Generic Safety Issue B-56, Diesel Generator Reliability, is the implementation of the Diesel Generator Interim Reliability Program and the hardware and/or procedural fixes discussed in Section 6.2.1.

Affected Plants and Average Remaining Lives

While the program is intended for implementation at all operating plants, its thrust is aimed at those plants with diesel generator reliabilities below 0.95/demand. As mentioned earlier, 30% of the operating plants are assumed to require diesel generator reliability improvement, including the 5% needing requalification. Given the number of operating plants from Appendix C (47 PWRS + 24 BWRs = 71 total), the numbers of affected plants become:

	Backfit BWR	Backfit PWR
Diesel Generator Reliability Improvement	7	14
Diesel Generator Requalification	1	2

The average remaining operating lives are 25.2 yr (BWR) and 27.7 yr (PWR), also taken from Appendix C.

Selected Analysis

Since this issue is presumably generic to the affected PWRs and BWRs, two representative plants are selected for which the plant risk equations are known--a PWR (Oconee 3) and a BWR (Grand Gulf 1). Their risk equations are described in Appendices A and B.

Affected Parameters

The parameters in the plant risk equations which will be affected by implementation of the Diesel Generator Interim Reliability Program are those related to diesel generator failure. Tables A.4 and B.4 list the risk parameters. For the two representative plants, these parameters are DIESEL1, DIESEL2, and DIESEL3 for Grand Gulf 1 and (B_3) for Oconee 3.

Upon closer inspection of event (B_3) in Table A.4 one finds that it refers to failures associated with hydroelectic rather than diesel generators. However, for the purposes of estimating the public risk reduction associated with this issue at a representative PWR, (B_3) can be redefined as if it referred to diesel generators. This is done as follows.

Originally, (B_3) was comprised of three failure contributions:

1. dual failure of two hydroelectric generators
2. failure of either hydroelectric generator while the other is down for maintenance
3. failure of both emergency DC batteries needed for generator startup.

If one assumes that Oconee has two diesel rather than two hydroelectric generators, (B_3) can be redefined as follows:

1. dual failure of two diesel generators

2. failure of either diesel generator while the other is down for maintenance.

The third contribution is judged inappropriate for diesel generators and is thus omitted.

Affected Parameters' Base-Case Values

This issue has been assumed to apply to seven operating BWRs and 14 operating PWRs. Of these, one BWR and two PWRs will have to requalify their diesel generators. Given the minimum acceptable reliability level of 0.90, it will be assumed that the base-case failure probabilities of the diesel generators at these three plants correspond to unreliabilities of 0.10, the complement of the minimum acceptable reliability level. It will be assumed that the base-case failure probabilities of the diesel generators at the remaining six BWRs and 12 PWRs correspond to unreliabilities of 0.07, the complement of an assumed reliability level of 0.93. (Somewhat below the minimum desirable reliability level of 0.95.) These represent the assumed situations at the affected plants prior to issue resolution (implementation of the Interim Reliability Program).

These values presumably represent the probabilities of diesel generator failure from all causes, independent and common. Thus, they are taken to be the total failure probabilities. As much as 7% of the diesel generator failures can be attributed to common cause (Baranowsky 1981). From this, one can estimate both the independent and common-cause failure probabilities for diesel generators for both sets of plants using the β-factor method.

For the plants needing diesel generator requalification:

$$P_{total} = P_{ind} + P_{cc} = .10$$

$$P_{cc} = \beta P_{total} = .07(.10) = .007$$

(β = fraction of total failures due to common cause)

$$P_{ind} = P_{total} - P_{cc} = .10 - .007 = .09$$

For the plants needing diesel generator reliability improvement only:

$$P_{total} = .07$$

$$P_{cc} = \beta P_{total} = (0.7)^2 = .005$$

$$P_{ind} = .07 - .005 = .065$$

In their original risk studies, Grand Gulf and Oconee used the following failure probabilities for the affected parameters:

Grand Gulf	Oconee
$\left.\begin{array}{l}\text{DIESEL1}\\ \text{DIESEL2}\\ \text{DIESEL3}\end{array}\right\} = .036$	$(B_3) = 5E-4$

Considering the base-case values assumed above and the redefinition of (B_3), it is necessary to define new base-case values for these affected parameters. Because they are defined differently for each reactor, these parameters enter into the risk equations in slightly different ways. Each reactor is discussed separately.

Grand Gulf

Review of the calculations of the Grand Gulf minimal cut set frequencies indicates that the common-cause contribution was not modelled. Thus, besides the changes in the failure probabilities of DIESEL1, DIESEL2, and DIESEL3 to 0.1 (requalification) and 0.07 (improvement), it is also necessary to include the contributions from common-cause failure. This is done as follows (the three diesels are referred to as #'s 1, 2, and 3):

1. Designate the total failure (independent plus common-cause) of an individual diesel generator as D_1, D_2, or D_3.
2. Resolve D_1 into its constituents - an independent failure D_1 and one or more common-cause failures (D_{12} for failure with diesel #2; D_{13} for failure with diesel #3). In fault-tree terminology, failure event

D_1 has been developed via an OR gate into its constituents. Do likewise for D_2 and D_3.

3. Perform the Boolean multiplication of the appropriate diesel generator failures as indicated by the cut set. For example, if the cut set contains D_1 and D_2:

$$\begin{aligned} D_1D_2 &= (D_1' + D_{12} + D_{13})(D_2' + D_{12} + D_{23}) \\ &= D_1'D_2' + D_1'D_{23} + D_2'D_{13} + D_{12} + D_{13}D_{23} \end{aligned}$$

If the cut set contains D_1, D_2, and D_3:

$$\begin{aligned} D_1D_2D_3 &= (D_1' + D_{12} + D_{13})(D_2' + D_{12} + D_{23})(D_3' + D_{13} + D_{23}) \\ &= D_1'D_2'D_3' + D_1'D_{23} + D_2'D_{13} + D_3'D_{12} + D_{12}D_{13} \\ &\quad + D_{12}D_{23} + D_{13}D_{23} \end{aligned}$$

4. Since each diesel generator has the same failure probability, reduce the Boolean equation as follows:

   i. $D_1' = D_2' = D_3' = D'$ (independent)

   ii. $D_{12} = D_{13} = D_{23} = D_c$ (common cause)

   iii. $$\begin{aligned} D_1D_2 &= (D')^2 + 2D'D_c + D_c + D_c{}^2 \\ &= (D')^2 + D_c(1 + D_c + 2D') \end{aligned}$$

   [Note, if there are only two diesel generators at a plant, the D_{13} and D_{23} terms would drop out for D_1D_2 in step 3 and the above equation would be:

$$D_1D_2 = (D')^2 + D_c]$$

iv. $$D_1D_2D_3 = (D')^3 + 3D'D_c + 3D_c^{\ 2}$$

$$= (D')^3 + 3D_c(D' + D_c)$$

$$= (D')^3 + 3D_cD$$

Thus, the base-case parameter values will be (for the plants undergoing requalification):

$$\left.\begin{array}{l}\text{DIESEL1}\\ \text{DIESEL2}\\ \text{DIESEL3}\end{array}\right\} = D = 0.1$$

$$\left.\begin{array}{l}\text{DIESEL1} \cdot \text{DIESEL2}\\ \text{DIESEL1} \cdot \text{DIESEL3}\\ \text{DIESEL2} \cdot \text{DIESEL3}\end{array}\right\} = (D')^2 + D_c(1 + D_c + 2D')$$

$$= (.09)^2 + (.007)(1 + .007 + 2[.09])$$
$$= .02$$

$$\text{DIESEL1} \cdot \text{DIESEL2} \cdot \text{DIESEL3} = (D')^3 + 3D_cD$$

$$= (.09)^3 + 3(.007)(0.1)$$
$$= .003$$

For the plants requiring reliability improvement only:

$$\left.\begin{array}{l}\text{DIESEL1}\\ \text{DIESEL2}\\ \text{DIESEL3}\end{array}\right\} = .07$$

$$\left.\begin{array}{l}\text{DIESEL1} \cdot \text{DIESEL2} \\ \text{DIESEL1} \cdot \text{DIESEL3} \\ \text{DIESEL2} \cdot \text{DIESEL3}\end{array}\right\} = (.065)^2 + (.005)[1 + .005 + 2(.065)]$$

$$= .01$$

$$\text{DIESEL1} \cdot \text{DIESEL2} \cdot \text{DIESEL3} = (.065)^3 + 3(.005)(.007)$$

$$= .001$$

Oconee

Since Oconee is presumed to have only two diesel generators, the contribution of their dual failure to event (B_3) is as follows (for requalification):

$$\begin{aligned} D_1D_2 &= (D')^2 + D_c \\ &= (.09)^2 + .007 \\ &= .02 \end{aligned}$$

For failure of either diesel generator while its mate is down for maintenance, the contribution to (B_3) will be:

$$\begin{aligned} 2D(Q_{\text{maint. outage}}) &= 2(0.1)(.0058) \\ &= .001 \end{aligned}$$

where it has been assumed that the unavailability due to a maintenance outage is the same as that for the hydroelectric generators. Thus, the base-case failure probability of event (B_3) will be (for requalification):

$$\begin{aligned} (B_3) &= .02 + .001 \\ &= .02 \end{aligned}$$

For reliability improvement, the base-case value of (B_3) becomes:

$$D_1D_2 = (.065)^2 + .005$$
$$= .01$$

$$2DQ = 2(.07)(.0058)$$
$$= 8E-4$$

$$(B_3) = .01 + 8E-4$$
$$= .01$$

Affected Accident Sequences' Base-Case Frequencies

Tables A.3 and B.3 indicate to which minimal cut sets and accident sequences the affected parameters belong. The base-case frequencies of the affected minimal cut sets are obtained by substituting in the base-case values of the affected parameters (along with the original values of the non-affected parameters). These are summed to yield the frequencies of the affected portions of the accident sequences in the base case.

Grand Gulf

The accident sequences to which the affected parameters DIESEL1, DIESEL2, and DIESEL3 belong and the base-case frequencies of these affected sequences (found via the affected parameters' contributions to the minimal cut sets) are the following:

	Frequency (ry^{-1})	
Sequence	Requalification	Improvement
$T_1PQI-\alpha$	3.6E-8	2.2E-8
$T_1PQI-\delta$	3.6E-6	2.2E-6
$T_1QW-\delta$	2.5E-5	1.4E-5
$T_1PQE-\gamma$	8.0E-7	3.7E-7
$T_1QUV-\gamma$	8.1E-6	3.8E-6
$T_1PQE-\delta$	8.0E-7	3.7E-7
$T_1QUV-\delta$	8.1E-6	3.8E-6

Oconee

The accident sequences to which the affected parameter (B_3) belongs and the base-case frequencies of these affected sequences are the following:

Sequence	Frequency (ry^{-1}) Requalification	Improvement
$T_1(B_3)MLU-\gamma$	4.3E-5	2.2E-5
$T_1(B_3)MLU-\beta$	6.3E-7	3.2E-7
$T_1(B_3)MLU-\varepsilon$	4.3E-5	2.2E-5

Affected Release Categories' Base-Case Frequencies

Tables A.1/B.1 and A.3/B.3 indicate to which release categories the affected accident sequences belong. The base-case frequencies of the affected portions of the release categories are obtained by summing the base-case frequencies of the affected accident sequences.

Grand Gulf

The affected accident sequences previously listed contribute to the following BWR release categories (based on WASH-1400):

1. BWR-1: $T_1PQI-\alpha$
2. BWR-2: $T_1QW-\delta$ and $T_1PQI-\delta$
3. BWR-3: $T_1PQE-\gamma$ and $T_1QUV-\gamma$
4. BWR-4: $T_1PQE-\delta$ and $T_1QUV-\delta$

The base-case frequencies of these affected categories become:

Category	Frequency (ry^{-1}) Requalification	Improvement
BWR-1	3.6E-8	2.2E-8
BWR-2	2.9E-5	1.6E-5
BWR-3	8.9E-6	4.2E-6
BWR-4	8.9E-6	4.2E-6

Oconee

The affected accident sequences previously listed contribute to the following PWR release categories (based on WASH-1400):

1. PWR-3: $T_1(B_3)MLU-\gamma$
2. PWR-5: $T_1(B_3)MLU-\beta$
3. PWR-7: $T_1(B_3)MLU-\varepsilon$

The base-case frequencies of these affected categories become:

Category	Frequency (ry^{-1}) Requalification	Improvement
PWR-3	4.3E-5	2.2E-5
PWR-5	6.3E-7	3.2E-7
PWR-7	4.3E-5	2.2E-5

Base-Case, Affected Core-Melt Frequency

The base-case, affected core-melt frequency ($\bar{F}$) for each representative plant is found by summing the base-case frequencies of the affected release categories. The base-case, affected core-melt frequencies become:

	Requalification	Improvement
Grand Gulf	4.7E-5/ry	2.4E-5/ry
Oconee	8.7E-5/ry	4.4E-5/ry

Base-Case, Affected Public Risk

The base-case, affected public risk (W) for each representative plant is found by summing the products of each affected release category's base-case frequency and dose factor (from Appendix D). This is done only for affected release categories. For the two representative plants, the base-case, affected public risks become:

	Requalification	Improvement
Grand Gulf	257 man-rem/ry	152 man-rem/ry
Oconee	233 man-rem/ry	132 man-rem/ry

Affected Parameters' Adjusted-Case Values

Previously, it was assumed that the base-case failure probabilities of diesel generators correspond to unreliabilities of 0.10 and 0.07 for the requalification and improvement programs respectively. The review of 1978-1980 LER data indicated a median diesel generator reliability of 0.97/demand, above the minimum acceptable level of 0.95. Thus, it will be assumed that the adjusted-case failure probability of a diesel generator corresponds to an unreliability of 0.03, the complement of the 0.97 median reliability. This is the assumed adjusted-case value for both the requalification and improvement programs.

As before, this value is presumed to be that for total failure, both independent and common-cause. Previously, 7% of the diesel generator failures were attributed to common cause. Following issue resolution, a decrease in the common-cause failure contribution would be expected. A drop from 7% to 5% seems reasonable. Again, the independent and common-cause failure probabilities are estimated via the β-factor method:

$$P_{total} = P_{ind} + P_{cc} = .03$$

$$P_{cc} = \beta P_{total} = .05(.03) = .002$$

$$P_{ind} = P_{total} - P_{cc} = .03 - .002 = .03$$

Grand Gulf

The previously derived expressions for the DIESEL terms are re-evaluated to obtain the adjusted-case failure probabilities:

$$\left.\begin{array}{l}\text{DIESEL1}\\ \text{DIESEL2}\\ \text{DIESEL3}\end{array}\right\} = D^{*} = .03$$

$$\left.\begin{array}{l} \text{DIESEL1} \cdot \text{DIESEL2} \\ \text{DIESEL1} \cdot \text{DIESEL3} \\ \text{DIESEL2} \cdot \text{DIESEL3} \end{array}\right\} = (D'^{*})^2 + D_c^{*}(1 + D_c^{*} + 2D'^{*})$$

$$= (.03)^2 + (.002)(1 + .002 + 2[.03])$$

$$= .003$$

$$\text{DIESEL1} \cdot \text{DIESEL2} \cdot \text{DIESEL3} = (D'^{*})^3 + 3D_c^{*}D^{*}$$

$$= (.03)^3 + 3(.002)(.03)$$

$$= 2E{-}4$$

Oconee

The previously derived expression for the (B_3) term is re-evaluated to obtain the adjusted-case failure probability:

$$(B_3) = (D'^{*})^2 + D_c^{*} + 2D^{*}(Q_{\text{maint. outage}})$$

$$= (.03)^2 + .002 + 2(.03)(.0058)$$

$$= .003$$

Affected Accident Sequences' Adjusted-Case Frequencies

The affected accident sequences and minimal cut sets are the same as for the base case. Only their frequencies change due to the change in the affected parameter values from the base to the adjusted case. The calculational procedure is equivalent.

Grand Gulf

Sequence	Frequency (ry^{-1})
$T_1PQI\text{-}\alpha$	8.0E-9
$T_1PQI\text{-}\delta$	8.0E-7
$T_1QW\text{-}\delta$	5.0E-6
$T_1PQE\text{-}\gamma$	1.1E-7
$T_1QUV\text{-}\gamma$	1.1E-6
$T_1PQE\text{-}\delta$	1.1E-7
$T_1QUV\text{-}\delta$	1.1E-6

Oconee

Sequence	Frequency (ry^{-1})
$T_1(B_3)MLU-\gamma$	6.5E-6
$T_1(B_3)MLU-\beta$	9.4E-8
$T_1(B_3)MLU-\varepsilon$	6.5E-6

Affected Release Categories' Adjusted-Case Frequencies

The affected release categories are the same as for the base case. Only their frequencies change due to the change in the frequencies of the affected accident sequences from the base to the adjusted case. The calculational procedure is equivalent.

Grand Gulf

Category	Frequency (ry^{-1})
BWR-1	8.0E-9
BWR-2	5.8E-6
BWR-3	1.2E-6
BWR-4	1.2E-6

Oconee

Category	Frequency (ry^{-1})
PWR-3	6.5E-6
PWR-5	9.4E-8
PWR-7	6.5E-6

Adjusted-Case, Affected Core-Melt Frequency

The adjusted-case, affected core-melt frequency ($\bar{F}^*$) for each representative plant is calculated as before, except that the adjusted rather than the base-case frequencies are used for the affected release categories. The adjusted-case, affected core-melt frequencies become:

Grand Gulf: 8.2E-6/ry

Oconee: 1.3E-5/ry

Adjusted-Case, Affected Public Risk

The adjusted-case, affected public risk ($W^{\star}$) for each representative plant is calculated as before, except that the adjusted rather than the base-case frequencies are used for the affected release categories. For the two representative plants, the adjusted-case, affected public risks become:

Grand Gulf: 48 man-rem/ry

Oconee: 35 man-rem/ry

Core-Melt Frequency Reduction

The core-melt frequency reduction ($\Delta\bar{F}$) for each representative plant is just the difference between its base and adjusted-case, affected core-melt frequencies. This represents the decrease attributable to resolution of the safety issue.

	$\Delta\bar{F}(ry^{-1})$	
Rep. Plant	Requalification	Improvement
Grand Gulf	3.9E-5	1.6E-5
Oconee	7.4E-5	3.1E-5

Per-Plant Public Risk Reduction

The public risk reduction (ΔW) for each representative plant is just the difference between its base and adjusted-case, affected public risks. This represents the decrease attributable to resolution of the safety issue.

	ΔW(man-rem/ry)	
Rep. Plant	Requalification	Improvement
Grand Gulf	209	90
Oconee	198	84

Total Public Risk Reduction

The total public risk reduction for all affected plants [$(\Delta W)_{Total}$] is calculated by summing the products of the following terms for each representative plant-type x:

1. The public risk reduction [$(\Delta W)_x$]
2. The number of affected plants to which the representative plant-type corresponds (N_x)
3. The average remaining operating life ($\bar{T}_x$).

Although there are only two representative plant-types (BWR and PWR), each contains plants undergoing diesel generator requalification as well as reliability improvement programs. Since each group has its unique value of ΔW, in effect there are four plant-types:

1. BWR-requalification
2. BWR-improvement
3. PWR-requalification
4. PWR-improvement

Thus, the total public risk reduction will be:

$$(\Delta W)_{Total} = \sum_{x=1}^{4} N_x \bar{T}_x (\Delta W)_x$$

where $N_1 = 1$, $N_2 = 6$, $N_3 = 2$, $N_4 = 12$

$\bar{T}_1 = \bar{T}_2 = 25.2$ yr

$\bar{T}_3 = \bar{T}_4 = 27.7$ yr

$(\Delta W)_1 = 209$ man-rem/ry

$(\Delta W)_2 = 90$ man-rem/ry

$(\Delta W)_3 = 198$ man-rem/ry

$(\Delta W)_4 = 84$ man-rem/ry

The best estimate of $(\Delta W)_{Total}$ is calculated to be 5.8E+4 man-rem.

The error bounds on $(\Delta W)_{Total}$ are calculated using the formulae in Section 3.5.1.

$$[(\Delta W)_{Total}]_u = 30 \sum_x N_x \bar{T}_x \hat{W}_x$$

$$[(\Delta W)_{Total}]_\ell = 0$$

where x, N_x and $\bar{T}_x$ are given as above

$$\left.\begin{aligned} \hat{W}_1 &= 257 \text{ man-rem/ry} \\ \hat{W}_2 &= 138 \text{ man-rem/ry} \\ \hat{W}_3 &= 233 \text{ man-rem/ry} \\ \hat{W}_4 &= 119 \text{ man-rem/ry} \end{aligned}\right\} \text{base-case best estimates}$$

The upper bound on $(\Delta W)_{Total}$ becomes 2.4E+6 man-rem

6.2.2.2 Occupational Doses

The procedure used to estimate the occupational doses follows that presented in Section 5.1.2. For demonstrative purposes, the analysis is detailed in a text format, with a summary work sheet (Table 6.2.2) provided at the end of this section. Generally, only the work sheet (with supplemental detail as necessary) will be needed, shortening the overall length of the presentation.

Issue Definition

The discussion is the same as that in Section 6.2.2.1 under the above heading.

Affected Plants

The discussion is the same as that in Section 6.2.2.1 under the above heading. All affected plants fall into the backfit class.

TABLE 6.2.1. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue: Diesel Generator Reliability (B-56)

2. Affected Plants (N) and Average Remaining Lives ($\bar{T}$): Seven operating BWRs and 14 operating PWRs are assumed to implement diesel generator reliability improvement programs. Of these, one BWR and two PWRs are assumed to require diesel generator requalification. The BWRs have an average remaining life of 25.2 yr; the PWRs have an average remaining life of 27.7 yr. For more detail, see discussion under the above heading in Section 6.2.2.1.

3. Selected Analysis Plants:

   Grand Gulf 1 – representative BWR
   Oconee 3 – representative PWR

4. Affected Parameters:

   Grand Gulf – DIESEL1, DIESEL2, DIESEL3
   Oconee – (B_3); see discussion under above heading in Section 6.2.2.1 for parameter redefinition.

5. Affected Parameters' Base-Case Values:

	Requalification	Improvement
Grand Gulf:		
DIESEL1, DIESEL2, DIESEL3 =	0.1	.07
DIESEL1 · DIESEL2, DIESEL1 · DIESEL3, DIESEL2 · DIESEL3 =	.02	.01
DIESEL1 · DIESEL2 · DIESEL3 =	.003	.001
Oconee:		
(B_3) =	.02	.01

See discussion under above heading in Section 6.2.2.1 for calculations.

TABLE 6.2.1. (contd)

6. Affected Accident Sequences and Base-Case Frequencies:

		Requalification	Improvement
Grand Gulf:	$T_1PQI-\alpha$ (BWR-1) =	3.6E-8/ry	2.2E-8/ry
	$T_1PQI-\delta$ (BWR-2) =	3.6E-6/ry	2.2E-6/ry
	$T_1QW-\delta$ (BWR-2) =	2.5E-5/ry	1.4E-5/ry
	$T_1PQE-\gamma$ (BWR-3) =	8.0E-7/ry	3.7E-7/ry
	$T_1QUV-\gamma$ (BWR-3) =	8.1E-6/ry	3.8E-6/ry
	$T_1PQE-\delta$ (BWR-4) =	8.0E-7/ry	3.7E-7/ry
	$T_1QUV-\delta$ (BWR-4) =	8.1E-6/ry	3.8E-6/ry
Oconee:	$T_1(B_3)MLU-\gamma$ (PWR-3) =	4.3E-5/ry	2.2E-5/ry
	$T_1(B_3)MLU-\beta$ (PWR-5) =	6.3E-7/ry	3.2E-7/ry
	$T_1(B_3)MLU-\varepsilon$ (PWR-7) =	4.3E-5/ry	2.2E-5/ry

7. Affected Release Categories and Base-Case Frequencies:

		Requalification	Improvement
Grand Gulf:	BWR-1 =	3.6E-8/ry	2.2E-8/ry
	BWR-2 =	2.9E-5/ry	1.6E-5/ry
	BWR-3 =	8.9E-6/ry	4.2E-6/ry
	BWR-4 =	8.9E-6/ry	4.2E-6/ry
Oconee:	PWR-3 =	4.3E-5/ry	2.2E-5/ry
	PWR-5 =	6.3E-7/ry	3.2E-7/ry
	PWR-7 =	4.3E-5/ry	2.2E-5/ry

8. Base-Case, Affected Core-Melt Frequency ($\bar{F}$):

	Requalification	Improvement
Grand Gulf	4.7E-5/ry	2.4E-5/ry
Oconee	8.7E-5/ry	4.4E-5/ry

9 Base-Case, Affected Public Risk (W):

	Requalification	Improvement
Grand Gulf	257 man-rem/ry	138 man-rem/ry
Oconee	233 man-rem/ry	119 man-rem/ry

TABLE 6.2.1. (contd)

10. Affected Parameters' Adjusted-Case Values:

Grand Gulf:

$$\left.\begin{array}{l}\text{DIESEL1}\\ \text{DIESEL2}\\ \text{DIESEL3}\end{array}\right\} = .03$$

$$\left.\begin{array}{l}\text{DIESEL1} \cdot \text{DIESEL2}\\ \text{DIESEL1} \cdot \text{DIESEL3}\\ \text{DIESEL2} \cdot \text{DIESEL3}\end{array}\right\} = .003$$

$$\text{DIESEL1} \cdot \text{DIESEL2} \cdot \text{DIESEL3} = 2E{-}4$$

Oconee: $(B_3) = .003$

See discussion under above heading in Section 6.2.2.1 for calculations.

11. Adjusted-Case Frequencies of Affected Accident Sequences:

Grand Gulf:	$T_1PQI\text{-}\alpha$	= 8.0E-9/ry
	$T_1PQI\text{-}\delta$	= 8.0E-7/ry
	$T_1QW\text{-}\delta$	= 5.0E-6/ry
	$T_1PQE\text{-}\gamma$	= 1.1E-7/ry
	$T_1QUV\text{-}\gamma$	= 1.1E-6/ry
	$T_1PQE\text{-}\delta$	= 1.1E-7/ry
	$T_1QUV\text{-}\delta$	= 1.1E-6/ry
Oconee:	$T_1(B_3)MLU\text{-}\gamma$	= 6.5E-6/ry
	$T_1(B_3)MLU\text{-}\beta$	= 9.4E-8/ry
	$T_1(B_3)MLU\text{-}\varepsilon$	= 6.5E-6/ry

12. Adjusted-Case Frequencies of Affected Release Categories:

Grand Gulf:	BWR-1 = 8.0E-9/ry
	BWR-2 = 5.8E-6/ry
	BWR-3 = 1.2E-6/ry
	BWR-4 = 1.2E-6/ry
Oconee:	PWR-3 = 6.5E-6/ry
	PWR-5 = 9.4E-8/ry
	BWR-7 = 6.5E-6/ry

TABLE 6.2.1. (contd)

13. Adjusted-Case, Affected Core-Melt Frequency ($\bar{F}$*):

Grand Gulf: 8.2E-6/ry

Oconee: 1.3E-5/ry

14. Adjusted-Case, Affected Public Risk (W*):

Grand Gulf: 48 man-rem/ry

Oconee: 35 man-rem/ry

15. Core-Melt Frequency Reduction ($\Delta\bar{F}$):

	Requalification	Improvement
Grand Gulf	3.9E-5/ry	1.6E-5/ry
Oconee	7.4E-5/ry	3.1E-5/ry

16. Per-Plant Public Risk Reduction (ΔW):

	Requalification	Improvement
Grand Gulf	209 man-rem/ry	90 man-rem/ry
Oconee	198 man-rem/ry	84 man-rem/ry

17. Total Public Risk Reduction [$(\Delta W)_{Total}$]:

Best Estimate	Upper Bound	Lower Bound
5.8E+4 man-rem	2.4E+6 man-rem	0

(For these calculations, the numbers of affected plants in the four plant-type categories are:

BWR-requalification = 1
BWR-improvement = 6
PWR-requalification = 2
PWR-improvement = 12)

Average Remaining Life

The discussion is the same as that in Section 6.2.2.1 under the above heading.

Per-Plant Occupational Dose Reduction Due to Accident-Avoidance

The occupational dose reduction due to accident-avoidance for each representative plant [$\Delta(\bar{F}D_R)$] is the product of the occupational dose due to reactor cleanup, repair, and refurbishment following a major core-melt (D_R) and the core-melt frequency reduction ($\Delta\bar{F}$). The representative plants are the same as those assumed in estimating the public risk reduction. D_R has a value of 19,900 man-rem (see Appendix D); ΔF has values of 3.9E-5/ry (BWR) and 7.4E-5/ry (PWR) for diesel generator requalification and 1.6E-5/ry (BWR) and 3.1E-5/ry (PWR) for diesel generator reliability improvement (from step 15 of the Public Risk Reduction Work Sheet). Thus, the occupational dose reductions due to accident-avoidance at each representative plant are:

	$\Delta(\bar{F}D_R)$ (man-rem/ry)	
Rep. Plant	Requalification	Improvement
BWR (Grand Gulf)	0.78	0.32
PWR (Oconee)	1.5	0.62

Total Occupational Dose Reduction Due to Accident-Avoidance

The total occupational dose reduction due to accident-avoidance (ΔU) is calculated by summing the products of the following terms for each representative plant-type x:

1. The occupational dose reduction due to accident-avoidance [$\Delta(\bar{F}D_R)_x$]
2. The number of affected plants to which the representative plant-type corresponds (N_x)
3. The average remaining operating life ($\bar{T}_x$).

As for the total public risk reduction, there are effectively four representative plant-types (see discussion under heading "Total Public Risk Reduction" in Section 6.2.2.1). Thus, the total occupational dose reduction due to accident-avoidance will be:

$$\Delta U = \sum_{x=1}^{4} N_x \bar{T}_x \Delta(\bar{F}D_R)_x$$

where x, N_x and $\bar{T}_x$ are defined as in Section 6.2.2.1 under the heading "Total Public Risk Reduction"

$\Delta(\bar{F}D_R)_1 = 0.78$ man-rem/ry

$\Delta(\bar{F}D_R)_2 = 0.32$ man-rem/ry

$\Delta(\bar{F}D_R)_3 = 1.5$ man-rem/ry

$\Delta(\bar{F}D_R)_4 = 0.62$ man-rem/ry

The best estimate of ΔU is calculated to be 350 man-rem.

The error bounds on ΔU are calculated using the formulae in Section 3.5.2.

$$(\Delta U)_u = 6\hat{D}_R \sum_x N_x \bar{T}_x \hat{\bar{F}}_x$$

$$(\Delta U)_\ell = 0$$

where x, N_x and $\bar{T}_x$ are given as before

$\hat{D}_R = 19{,}900$ man-rem

$$\left.\begin{aligned} \hat{\bar{F}}_1 &= 4.7E{-}5/ry \\ \hat{\bar{F}}_2 &= 2.4E{-}5/ry \\ \hat{\bar{F}}_3 &= 8.7E{-}5/ry \\ \hat{\bar{F}}_4 &= 4.4E{-}5/ry \end{aligned}\right\}$$ base-case best estimates of affected core-melt frequency (from step 8 of the Public Risk Reduction Work Sheet)

The upper bound on ΔU becomes 2,900 man-rem.

<u>Steps Related to Occupational Dose Increase Due to Implementation, Operation, and Maintenance of SIR</u>

Diesel generators are not located in radiation zones of the plant. Thus, there will be no occupational dose from implementation, operation, and maintenance of the proposed issue resolution. Steps 6 through 11 outlined in Section 5.1.2 can be skipped.

Total Occupational Dose Increase

The total occupational dose increase (G) is the sum of the total occupational dose increases due to implementation (ND) and operation/maintenance of the SIR ($N\bar{T}D_o$). Since these latter two are zero, G is also zero. The error bounds would normally be calculated using the formulae in Section 3.5.3:

$$G_u = 3\hat{G}$$
$$G_\ell = \hat{G}/3$$

where $\hat{G}$ = best estimate of G.

Since $\hat{G}$ = 0, both the upper and lower bounds on G are zero.

6.2.3 Safety Issue Costs

The procedure used to estimate the industry and NRC costs follows that presented in Section 5.2. For demonstrative purposes, the analysis is detailed in a text format, with a summary work sheet (Table 6.2.3) provided at the end of this section. Generally, only the work sheet (with supplemental detail as necessary) will be needed, shortening the overall length of the presentation.

Issue Definition

The discussion is the same as that in Section 6.2.2.1 under the above heading.

Affected Plants

The discussion is the same as that in Section 6.2.2.1 under the above heading. All affected plants fall into the backfit class.

Average Remaining Life

The discussion is the same as that in Section 6.2.2.1 under the above heading.

TABLE 6.2.2. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue: Diesel Generator Reliability (B-56)

2. Affected Plants (N): Seven operating BWRs and 14 operating PWRs are assumed to implement diesel generator reliability improvement programs. Of these, one BWR and two PWRs are assumed to require diesel generator requalification. All fall into the backfit class.

3. Averaging Remaining Life ($\bar{T}$):

   7 backfit BWRs = 25.2 yr
   14 backfit PWRs = 27.7 yr

4. Per-Plant Occupational Dose Reduction Due to Accident-Avoidance [$\Delta(\bar{F}D_R)$]:

	Requalification	Improvement
BWR	0.78 man-rem/ry	0.32 man-rem/ry
PWR	1.5 man-rem/ry	0.62 man-rem/ry

5. Total Occupational Dose Reduction Due to Accident-Avoidance (ΔU):

Best Estimate	Upper Bound	Lower Bound
350 man-rem	2,900 man-rem	0

   (For these calculations, the numbers of affected plants in the four plant-type categories are:

   BWR-requalification = 1
   BWR-improvement = 6
   PWR-requalification = 2
   PWR-improvement = 12)

6-12. Steps Related to Occupational Dose Increase Due to Implementation, Operation, and Maintenance of SIR:

   Since diesel generators are not located in radiation zones, no occupational dose will be accrued during SIR implementation, operation, and maintenance. Thus,

$$D = D_o = 0$$

$$G = 0 \text{ (best estimate and error bounds)}$$

Per-Plant Industry Cost Savings Due to Accident-Avoidance

The industry cost savings due to accident-avoidance for each representative plant [$\Delta(\bar{F}A)$] is the product of the cost associated with reactor cleanup, repair, and refurbishment (plus replacement power) following a core-melt (A) and the core-melt frequency reduction ($\Delta\bar{F}$). The representative plants are the same as those assumed in estimating the public risk reduction. A has a value of \$1.65E+9 (see Appendix E); $\Delta\bar{F}$ has values of 3.9E-5/ry (BWR) and 7.4E-5/ry (PWR) for diesel generator requalification and 1.6E-5/ry (BWR) and 3.1E-5/ry (PWR) for diesel generator reliability improvement (from step 15 of the Public Risk Reduction Work Sheet).

Rep. Plant	$\Delta(\bar{F}A)$ (\$/ry) Requalification	Improvement
BWR (Grand Gulf)	6.4E+4	2.6E+4
PWR (Oconee)	1.2E+5	5.1E+4

Total Industry Cost Savings Due to Accident-Avoidance

The total industry savings due to accident-avoidance (ΔH) is calculated by summing the products of the following terms for each representative plant-type x:

1. The industry cost-savings due to accident-avoidance [$\Delta(\bar{F}A)_x$]
2. The number of affected plants to which the representative plant-type corresponds (N_x)
3. The average remaining operating life ($\bar{T}_x$).

As for the total public risk reduction, there are effectively four representative plant-types (see discussion under heading "Total Public Risk Reduction" in Section 6.2.2.1). Thus, the total industry cost savings due to accident avoidance will be:

$$\Delta H = \sum_{x=1}^{4} N_x \bar{T}_x \Delta(\bar{F}A)_x$$

where x, N_x and $\bar{T}_x$ are defined as in Section 6.2.2.1 under the heading

"Total Public Risk Reduction"

$\Delta(\bar{F}A)_1$ = \$6.4E+4/ry

$\Delta(\bar{F}A)_2$ = \$2.6E+4/ry

$\Delta(\bar{F}A)_3$ = \$1.2E+5/ry

$\Delta(\bar{F}A)_4$ = \$5.1E+4/ry

The best estimate of ΔH is calculated to be \$2.9E+7.

The error bounds on ΔH are calculated using the formulae in Section 4.3.1.

$$(\Delta H)_u = 6\hat{A} \sum_x N_x \bar{T}_x \hat{\bar{F}}_x$$

$$(\Delta H)_\ell = 0$$

where x, N_x and $\bar{T}_x$ are given as before

$\hat{A}$ = \$1.65E+9

$\hat{\bar{F}}_1$, $\hat{\bar{F}}_2$, $\hat{\bar{F}}_3$, and $\hat{\bar{F}}_4$ are given as in Section 6.2.2.2 under the heading "Total Occupational Oose Reduction Due to Accident-Avoidance."

The upper bound on ΔH becomes \$2.4E+8.

Per-Plant Industry Resources for SIR Implementation.

The resources needed to implement the SIR are labor, equipment and additional down-time requiring purchase of replacement power. It is assumed that the proposed hardware and procedural fixes discussed in Section 6.2.1.2 will all be implemented at each of the affected plants. Thus, 21 operating plants (seven BWRs and 14 PWRs) will implement a diesel generator reliability improvement program. Three of these (one BWR and two PWRs) will perform diesel requalification.

The resources required for diesel generator reliability improvement and requalification are presented below. The two cases are treated separately. Each plant is assumed to have two diesel generators. The BWRs and PWRs are treated equivalently.

Diesel Generator Reliability Improvement

Several hardware/procedural fixes for diesel generator reliability improvement were presented in Section 6.2.1.2. The resources for these are discussed below. No additional down-time (requiring purchase of replacement power) is anticipated for any of these fixes.

1. Air Dryer Installation in Compressed Air Starting Systems. It is assumed that two air dryers will be installed, with eight man-weeks of labor required for their engineering and six man-weeks of labor for installation and testing. These amount to a total of 14 man-weeks of labor. There will also be some miscellaneous material needed.

2. Installation of Dust-Tight Enclosures for Electrical Contactors. It is assumed that dust-tight enclosures will be installed for electrical contactors, with four man-weeks of labor for their engineering and twelve man-days of labor for installation and testing. These amount to a total of 6.4 man-weeks of labor.

3. Installation of Diesel Generator Room Ventilation Ducting. It is assumed that both intake and exhaust ventilation ducting will be installed for the diesel generator room, with six man-weeks of labor for engineering and four man-weeks of labor for installation and testing. These amount to a total of ten man-weeks of labor.

4. Replacement of Existing Turbocharger Gear Sets with Heavy-Duty Sets. The existing two turbocharger gear sets will presumably be replaced with two heavy-duty sets, requiring replacement of the two gear packages as units. Eight man-weeks of labor are estimated for installation and testing.

5. Revision of Operating Procedures and Personnel Training. It is assumed that operating procedures for the diesel generators will be revised and updated, requiring twenty man-weeks of labor. Training of the operating staff on the new equipment and for the new procedures will require an additional ten man-weeks. Thus, thirty man-weeks of labor are needed to revise procedures and train personnel.

The resources needed to implement these fixes for diesel generator reliability improvement are summarized below:

Fix	Equipment	Labor (Man-Weeks)	
Air Dryers	2 Air Dryers Miscellaneous	8 6 14	Engineering Inst. and Testing Total
Contactor Enclosures	Enclosures	4 2.4 6.4	Engineering Inst. and Testing Total
Ventilation Ducting	Ducting	6 4 10	Engineering Inst. and Testing Total
Gear Replacement	2 Gear Packages	8	Inst. and Testing
Procedures and Training	--	20 10 30	Proc. Revision Personnel Train. Total

Diesel Generator Requalification

Diesel generator requalification will require major repair of diesel generators, with associated labor of approximately 25 man-weeks. One week of additional down-time will presumably be incurred, requiring purchase of replacement power.

Per-Plant Industry Cost for SIR Implementation

The industry cost for implementing the issue resolution for each affected plant (I) is the sum of the labor, equipment, and replacement power costs. Equipment costs are estimated specifically for each issue, while labor and replacement power costs are calculated by multiplying their resource estimates by the standardized cost rates from Appendix E ($2,270/man-wk and $3.0E+5/day, respectively). Equipment costs are based on manufacturer's prices, where available, and engineering judgment. The cost calculations are summarized below for diesel generator reliability improvement and requalification.

Diesel Generator Reliability Improvement

For diesel generator reliability improvement, the costs per plant are as follows:

Fix	Equipment Cost			Labor Cost	Total
Air Dryers	2 Dryers ($4K each)	$8K	$10K	$32K	$42K
	Miscellaneous	$2K			
Contactor Enclosures	Enclosures	$5K		$15K	$20K
Ventilation Ducting	Ducting	$10K		$23K	$33K
Gear Replacement	2 Gear Pkgs. ($15K each)	$30K		$18K	$48K
Procedures and Training	--			$68K	$68K
Total	$55K			$156K	$211K

No license amendment is anticipated for reliability improvement. Thus, no additional fee is incurred.

Diesel Generator Requalification

For diesel generator requalification, the costs per plant are as follows:

Equipment	$1,500K
Labor	$ 57K
Replacement Power	$2,100K
License Amendment*	$ 4K
Total =	$3,661K

*Assumes a class III license amendment (10CFR170.22) due to increased test frequency for diesel generator requalification.

Thus, the three operating plants which must institute both diesel generator reliability improvement and requalification will have an implementation cost I of \$3.87E+6/plant (\$2.1E+5/plant + \$3.66E+6/plant). The remaining 18 operating plants requiring only diesel generator reliability improvement will have a much smaller I of \$2.11E+5/plant.

Total Industry Cost for SIR Implementation

Since BWRs and PWRs are treated equivalently for implementation cost analysis in this issue, there are effectively only two affected plant-types: 1) backfit plants implementing diesel generator reliability improvement and 2) backfit plants implementing both diesel generator reliability improvement and requalification. There are 18 plants in the former category ($N_1 = 18$) and three in the latter ($N_2 = 3$). Thus, the total industry cost for SIR implementation becomes:

$$NI = \sum_{x=1}^{2} N_x I_x$$

where N_x is given as above

I_1 = \$2.11E+5/plant (improvement only)

I_2 = \$3.87E+6/plant (improvement plus requalification)

The best estimate of NI is calculated to be \$1.54E+7.

Per-Plant Industry Labor for SIR Operation/Maintenance

Each of the 21 operating plants which institutes diesel generator reliability improvement will presumably expend 10 man-weeks/year for operation/maintenance of the SIR. This includes reviewing operating procedures and retraining personnel. No additional labor above these 10 man-weeks/ry is foreseen for the three operating plants which must requalify their diesel generators. All 71 operating plants will expend approximately four man-weeks/year for record-keeping and reporting as part of the Diesel Generator Interim Reliability Program whether or not they require diesel generator reliability improvement and/or requalification. These labor estimates apply equally to BWRs and PWRs.

Per-Plant Industry Cost for SIR Operation/Maintenance

The industry cost for SIR operation/maintenance for each affected plant (I_o) is calculated by multiplying the labor estimate by the standardized labor cost rate ($2,270/man-week) from Appendix E. For the 21 operating plants which implement diesel generator reliability improvement and/or requalification, this cost becomes:

$$I_o = (14 \text{ man-wk/ry})(\$2{,}270/\text{man-wk})$$
$$= \$3.18\text{E+}4/\text{ry}$$

For the remaining 50 operating plants which merely must keep records and report for the Diesel Generator Reliability Improvement Program, this cost is:

$$I_o = (4 \text{ man-wk/ry})(\$2{,}270/\text{man-wk})$$
$$= \$9{,}090/\text{ry}$$

Total Industry Cost for SIR Operation/Maintenance

Since BWRs and PWRs are treated equivalently for operation/maintenance cost analysis, there are effectively only two affected plant types: 1) backfit plants which only keep records and report for the Diesel Generator Interim Reliability Program ($N_1 = 50$) and 2) backfit plants which not only do the former but also improve the reliability of and/or requalify their diesel generators ($N_2 = 21$). For each type, the average remaining operating life is that for all backfit plants shown in Appendix C ($\bar{T}_1 = \bar{T}_2 = 26.9$ yr). Thus, the total industry cost for SIR operation/maintenance ($N\bar{T}I_o$) becomes:

$$N\bar{T}I_o = \sum_{x=1}^{2} N_x \bar{T}_x (I_o)_x$$

where N_x and $\bar{T}_x$ are given as above

$(I_o)_1$ = $9,090/ry (Program only)

$(I_o)_2$ = $3.18E+4/ry (Program plus improvement and/or requalification).

The best estimate of $N\bar{T}I_o$ is calculated to be \$3.02E+7.

Total Industry Cost

The total industry cost (S_I) is the sum of the total industry costs for SIR implementation (NI) and operation/maintenance ($N\bar{T}I_o$):

$$S_I = \$1.54E+7 + \$3.02E+7 = \$4.56E+7 \text{ (best estimate)}$$

The error bounds on S_I are calculated using the formulae in Section 4.3.2.

$$(S_I)_u = \hat{S}_I + d_{S_I}$$

$$(S_I)_\ell = \hat{S}_I - d_{S_I}$$

where $\hat{S}_I$ = best estimate of S_I (\$4.56E+7)

$$d_{S_I} = \frac{1}{2}\sqrt{(N\bar{T}\hat{I}_o)^2 + (N\hat{I})^2}$$

$N\bar{T}\hat{I}_o$ = best estimate of $N\bar{T}I_o$ (\$3.02E+7)

$N\hat{I}$ = best estimate of NI (\$1.54E+7)

With d_{S_I} = \$1.69E+7, the error bounds become:

$$(S_I)_u = \$6.25E+7$$

$$(S_I)_\ell = \$2.87E+7.$$

NRC Resources for SIR Development

NRC development of the SIR has been completed, culminating in the Diesel Generator Interim Reliability Program. Thus, no additional NRC resources will be expended for SIR development.

Total NRC Cost for SIR Development

The total NRC cost for SIR development (C_D) is zero based on the above discussion.

Per-Plant NRC Labor to Support SIR Implementation

To improve diesel generator reliability, it is assumed that 2 man-weeks of NRC labor are needed to support this implementation at each plant. To requalify diesel generators, it is assumed that 4 man-weeks of NRC labor are needed to support this implementation at each plant. There is no difference between BWRs and PWRs.

Per-Plant NRC Cost to Support SIR Implementation

The NRC cost to support SIR implementation for each affected plant (C) is the product of the labor amount and cost rate. The latter has a value of $2,270/man-wk, taken from Appendix E. For plants improving diesel generator reliability,

$$C = (2 \text{ man-wk/plant})(\$2{,}270/\text{man-wk}) = \$4{,}540/\text{plant}.$$

For plants both improving the reliability of and requalifying diesel generators,

$$C = (6 \text{ man-wk/plant})(\$2{,}270/\text{man-wk}) = \$1.36E{+}4/\text{plant}.$$

Total NRC Cost to Support SIR Implementation

The total NRC cost to support SIR implementation is:

$$NC = \sum_{x=1}^{2} N_x C_x$$

where the plant-types x are defined as for SIR implementation cost analysis (see discussion under heading "Total Industry Cost for SIR Implementation")

$N_1 = 18$

$N_2 = 3$

$C_1 = \$4,540/\text{plant}$

$C_2 = \$1.36E{+}4/\text{plant}$.

The best estimate of NC is calculated to be $1.23E+5.

Per-Plant NRC Labor to Review SIR Operation/Maintenance

For the 21 operating plants improving diesel generator reliability and/or requalifying diesel generators, NRC labor to review SIR operation/maintenance will be the additional inspection time alloted to these modifications. A small annual increase of 0.2 man-wk/plant is assumed. For all 71 operating plants keeping records and reporting for the Diesel Generator Interim Reliability Program, 0.1 man-wk/ry of NRC labor will presumably be expended in reviewing these records.

Per-Plant NRC Cost to Review SIR Operation/Maintenance

The NRC cost to review SIR operation/maintenance for each affected plant (C_o) is the product of the labor amount and cost rate. The latter has a value of $2,270/man-wk, taken from Appendix E. For the 21 operating plants which improve diesel generator reliability and/or requalify diesel generators, this cost becomes:

$$C_o = (0.3\ \text{man-wk/ry})(\$2,270/\text{man-wk}) = \$681/\text{ry}$$

For the remaining 50 operating plants which merely must keep records and report for the Diesel Generator Interim Reliability Program, this cost is:

$$C_o = (0.1\ \text{man-wk/ry})(\$2,270/\text{man-wk}) = \$227/\text{ry}$$

Total NRC Cost to Review SIR Operation/Maintenance

The total NRC cost to review SIR operation/maintenance is:

$$N\bar{T}C_o = \sum_{x=1}^{2} N_x \bar{T}_x (C_o)_x$$

where the plant-types x are defined as for SIR operation/maintenance cost analysis (see discussion under heading "Total Industry Cost for SIR Operation/Maintenance")

$N_1 = 50$
$N_2 = 21$
$\bar{T}_1 = \bar{T}_2 = 26.9$ yr
$(C_o)_1 = \$227$/ry (Program only)
$(C_o)_2 = \$681$/ry (Program plus improvement and/or requalification)

The best estimate of $N\bar{T}C_o$ is calculated to be \$6.90E+5.

Total NRC Cost

The total NRC cost (S_N) is the sum of the total NRC costs for SIR development (C_D), support of SIR implementation (NC), and review of SIR operation/maintenance ($N\bar{T}C_o$):

$$S_N = 0 + \$1.23E+5 + \$6.90E+5 = \$8.13E+5 \text{ (best estimate)}$$

The error bounds on S_N are calculated using the formulae in Section 4.3.3.

$$(S_N)_u = \hat{S}_N + d_{S_N}$$

$$(S_N)_\ell = \hat{S}_N - d_{S_N}$$

where $\hat{S}_N$ = best estimate of S_N (\$8.13E+5)

$$d_{S_N} = \frac{1}{2}\sqrt{\hat{C}_D^{\,2} + (N\bar{T}\hat{C}_o)^2 + (N\hat{C})^2}$$

$\hat{C}_D$ = best estimate of C_D (zero)

$N\bar{T}\hat{C}_o$ = best estimate of $N\bar{T}C_o$ (\$6.90E+5)

$N\hat{C}$ = best estimate of NC (\$1.23E+5)

With d_{S_N} = \$3.50E+5, the error bounds become:

$$(S_N)_u = \$1.16E{+}6$$

$$(S_N)_\ell = \$4.63E{+}5$$

6.2.4 Summary of Results

The important results from the estimates for public risk reduction, occupational dose, industry cost, and NRC cost are summarized in the "Issue Summary Work Sheet" (Table 6.2.4). This work sheet normally comes at the beginning of an issue report package, as shown for the other two example issues. It is placed at the end of this issue analysis to demonstrate the process and information required to complete it. The results presented on the work sheet are taken directly from the individual work sheets as follows:

ISSUE NO./TITLE - from step 1 of any of the work sheets (Table 6.2.1, 6.2.2, or 6.2.3)

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION - from the safety issue description (Section 6.2.1)

AFFECTED PLANTS - from step 2 of any of the work sheets

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION - from step 17 (best estimate) of the Public Risk Reduction Work Sheet (PRRWS, Table 6.2.1)

TABLE 6.2.3. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue: Diesel Generator Reliability (B-56)

2. Affected Plants (N): Seven operating BWRs and 14 operating PWRs are assumed to implement diesel generator reliability improvement programs. Of these, one BWR and two PWRs are assumed to require diesel generator requalification. All fall into the backfit class.

3. Average Remaining Life ($\bar{T}$):

 7 backfit BWRs = 25.2 yr
14 backfit PWRs = 27.7 yr

Industry Costs (steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident-Avoidance [$\Delta(\bar{F}A)$]:

	Requalification	Improvement
BWR	$6.4E+4/ry	$2.6E+4/ry
PWR	$1.2E+5/ry	$5.1E+4/ry

5. Total Industry Cost Savings Due to Accident-Avoidance (ΔH):

Best Estimate	Upper Bound	Lower Bound
$2.9E+7	$2.4E+8	0

(For these calculations, the numbers of affected plants in the four plant-type categories are:

BWR-requalification = 1
BWR-improvement = 6
PWR-requalification = 2
PWR-improvement = 12)

6. Per-Plant Industry Resources for SIR Implementation:

Requalification

Equipment: estimate not needed for next step (cost estimated directly)
Labor: 25 man-wk/plant
Additional Down-time: 7 days/plant

TABLE 6.2.3. (contd)

Improvement

Equipment (per plant): 2 Air Dryers
Enclosures for Electrical Contactors
Ventilation Ducting
2 Turbocharger Gear Packages
Miscellaneous

Labor: 68.4 man-wk/plant

Additional Down-time: none

(These values apply equally to BWRs and PWRs. For more detail, see discussion under above heading in this section.)

7. Per-Plant Industry Cost for SIR Implementation (I):

Requalification and Improvement	Improvement Only
$3.87E+6/plant	$2.11E+5/plant

(For more detail, see discussion under above heading in this section.)

8. Total Industry Cost for SIR Implementation (NI):

$1.54E+7

(For this calculation, the affected plant-types are redefined to remove the BWR-PWR distinction, i.e.:

1. Three operating plants which both improve the reliability of and requalify their diesel generators
2. Eighteen operating plants which only improve the reliability of their diesel generators.)

9. Per-Plant Industry Labor for SIR Operation/Maintenance:

As discussed in this section under the above heading, labor estimates are given for two activities:

TABLE 6.2.3. (contd)

Activity	Labor
Diesel generator reliability improvement and/or requalification	10 man-wk/ry
Diesel Generator Interim Reliability Program record-keeping and reporting	4 man-wk/ry

These values apply equally to BWRs and PWRs.

10. Per-Plant Industry Cost for SIR Operation/Maintenance (I_O):

Program plus Improvement and/or Requalification	Program Only
\$3.18+4/ry	\$9,090/ry

11. Total Industry Cost for SIR Operation/Maintenance ($N\bar{T}I_O$):

    \$3.02E+7

    (For this calculation, the affected plant-types are redefined to correspond to the activities given in step 9. The numbers of affected plants in the two plant-type categories are:

    1. 21 operating plants which not only keep records and reports for the Program but also improve the reliability of and/or requalify their diesel generators.

    2. 50 operating plants which only keep records and report for the Program.)

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$4.6E+7	\$6.3E+7	\$2.9E+7

NRC Costs (steps 13 through 21)

13. NRC Resources for SIR Development:

    SIR development is complete. No further resources are needed.

TABLE 6.2.3. (contd)

14. Total NRC Cost for SIR Development (C_D):

Zero.

15. Per-Plant NRC Labor to Support SIR Implementation:

Requalification	Improvement
4 man-wk/plant	2 man-wk/plant

(These values apply equally to BWRs and PWRs.)

16. Per-Plant NRC Cost to Support SIR Implementation (C):

Requalification and Improvement	Improvement Only
\$1.36E+4/plant	\$4,540/plant

(For more detail, see discussion under above heading in this section.)

17. Total NRC Cost to Support SIR Implementation (NC):

\$1.23E+5

(For this calculation, the numbers of affected plants and the plant-types are the same as shown in step 8.)

18. Per-Plant NRC Labor to Review SIR Operation/Maintenance:

Labor estimates are given for review of the two activities specified in step 9:

Activity	Labor
Review reliability improvement and/or requalification	0.2 man-wk/ry
Review records and reporting for Interim Reliability Program	0.1 man-wk/ry

19. Per-Plant NRC Cost to Review SIR Operation/Maintenance (C_o):

Program Review plus Review of Improvement and/or Requalification	Prgram Review Only
\$681/ry	\$227/ry

TABLE 6.2.3. (contd)

20. Total NRC Cost to Review SIR Operation/Maintenance ($N\bar{T}C_o$):

$6.90E+5

(For this calculation, the numbers of affected plants and the plant types are the same as shown in step 11.)

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
$8.1E+5	$1.2E+6	$4.6E+5

OCCUPATIONAL DOSES:

SIR Implementation - from step 8 of the Occupational Dose Work Sheet (ODWS, Table 6.2.2)

SIR Operation/Maintenance - from step 11 of the ODWS

Total of Above - from step 12 (best estimate) of the ODWS

Accident-Avoidance - from step 5 (best estimate) of the ODWS.

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:

SIR Implementation - from step 8 of the Safety Issue Cost Work Sheet (SICWS, Table 6.2.3)

SIR Operation/Maintenance - from step 11 of the SICWS

Total of Above - from step 12 (best estimate) of the SICWS

Accident-Avoidance - from step 5 (best estimate) of the SICWS.

NRC COSTS:

SIR Development - from step 14 of the SICWS

SIR Implementation Support - from step 17 of the SICWS

SIR Operation/Maintenance Review - from step 20 of the SICWS

Total of Above - from step 21 (best estimate) of the SICWS.

TABLE 6.2.4. Issue Summary Work Sheet

ISSUE NO./TITLE: B-56, Diesel Generator Reliability

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION: Diesel generator reliability at certain operating plants has been found to be below the minimum desired value of 0.95/demand. An interim reliability program is proposed to determine which diesel generators require reliability improvement, with possible requalification. Several hardware and procedural fixes can be implemented for those diesel generators.

AFFECTED PLANTS BWR: Operating = 7 Planned = 0
PWR: Operating = 14 Planned = 0

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	5.8E+4
OCCUPATIONAL DOSES:	
SIR Implementation =	0
SIR Operation/Maintenance =	0
Total of Above =	0
Accident-Avoidance =	350

COST RESULTS ($\$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	16
SIR Operation/Maintenance =	30
Total of Above =	46
Accident-Avoidance =	29
NRC COSTS:	
SIR Development =	0
SIR Implementation Support =	0.12
SIR Operation/Maintenance Review =	0.69
Total of Above =	0.81

6.3 STEAM LINE BREAK WITH CONSEQUENTIAL SMALL LOCA: GENERIC SAFETY ISSUE 18

The format and level of detail presented for this issue are intended to be representative of those required for most issue analyses. The results of the analysis for this issue are summarized in Table 6.3.1.

6.3.1 Safety Issue Description

The issue as described (EDO 1980, Kniel 1981, Denton 1981) concerns postulated accidents resulting from a coincident steam line break, steam generator tube rupture, and small LOCA in the primary system in PWRs (combined LOCAs). Analysis performed by the Office of Nuclear Reactor Regulation (NRR) indicates that the primary pressure and the pressurizer level may change qualitatively in the same way during a combined LOCA compared to a primary break, a steam line break, or a steam generator tube rupture (Denton 1981). For the primary temperature and secondary pressure, a combined LOCA behaves qualitatively like a steam line break. For these latter two parameters, a primary rupture or steam generator tube rupture (SGTR) appears clearly distinct from the behavior of a combined LOCA. However, it appears that the potential exists for misdiagnosis of combined LOCA events as a main steam line break alone.

As addressed here, issue 18 is divided into two sub-issues: 1) steam line break with a subsequent small LOCA resulting from failure of partially degraded steam generator tube(s); and 2) steam line break with a subsequent small LOCA, other than an SGTR, resulting from a stuck-open PORV or safety valve actuated during the primary system transient or resulting from pipe whip or jet impingement from the broken steam line (Hanauer 1982). The steam generator overfill transient and potential for steam line rupture resulting from filling the steam lines with water are not considered in this issue analysis.

Section 4.2 of NUREG-0844 evaluated the consequences of main steam line break (MSLB) with concurrent SGTR. Section 4.3 of NUREG-0844 evaluated the LOCA with concurrent SGTR failures. The Section 4.2 evaluation bounded the containment response to a postulated LOCA with concurrent SGTR as well as a postulated MSLB with concurrent SGTR. The general conclusion reached

TABLE 6.3.1. Issue Summary Work Sheet

ISSUE NO./TITLE: No. 18, Steam Line Break with Consequential Small LOCA

SUMMARY OF PROBLEM AND PROPOSED RESOLUTION:

In PWRs, the potential exists for steam line breaks, consequentially leading to a small primary system LOCA. The combined event could produce conditions which tend to mask the primary LOCA, thus increasing the potential for operator misinterpretation and error. Suggested SIRs emphasize operator training. Hardware fixes are a second priority to decrease the potential for steam line breaks leading to primary system LOCAs in steam generator enclosures.

AFFECTED PLANTS	BWR:	Operating = 0	Planned = 0
	PWR:	Operating = 47	Planned = 43

RISK/DOSE RESULTS (man-rem)

PUBLIC RISK REDUCTION =	1,500
OCCUPATIONAL DOSES:	
SIR Implementation =	420
SIR Operation/Maintenance =	7,800
Total of Above =	8,200
Accident-Avoidance =	11

COST RESULTS ($$10^6$)

INDUSTRY COSTS:	
SIR Implementation =	19
SIR Operation/Maintenance =	35
Total of Above =	54
Accident-Avoidance =	0.94
NRC COSTS:	
SIR Development =	0.17
SIR Implementation Support =	0.20
SIR Operation/Maintenance Review =	2.9
Total of Above =	3.3

was that actual SGTR events had not resulted in unacceptable consequences, but the potential for more significant consequences did exist and procedural and equipment changes should be made to ensure that subsequent SGTR events would not result in unacceptable consequences.

The SGTR event at the Ginna reactor on January 25, 1982 focused additional attention on the combined LOCA issue and potential new initiating mechanisms, including operator and system responses. As a result of the Ginna SGTR, NRR review and development of generic recommendations were requested for items related to: 1) plant system response, 2) human factors, 3) radiological consequences, 4) organizational responses, and 5) post-event activities (Denton 1982).

6.3.1.1 Safety Issue Resolution

Two concerns have been identified which could increase the risk associated with this issue: 1) the possibility of primary side LOCAs may be increased through the consideration of new initiating mechanisms, and 2) the symptoms of a combined primary and secondary blowdown may increase the possibility for operator error through misinterpretation and improper action.

No specific resolution has been proposed for this issue. However, resolutions will likely center around identification of new initiating mechanisms for primary breaks in the steam generator enclosure caused by a steam line break and potential operator misinterpretation of combined primary and secondary LOCAs. Thus, even if some hardware fix is implemented as a result of the NUREG-0844 evaluation and the ongoing NRC steam generator confirmatory research program, it would still be necessary to address the issue of proper operator interpretation and action. The final solution to this issue should recognize this potential condition and put emphasis on operator training. Information from TMI Action Items I.C.1(4) and I.C.9 should aid in development of the proper operator training (NUREG-0660 1980). In addition, recommendations to be derived may provide better insight to a generic solution to this issue (Denton 1982).

This issue affects all PWRs. At the present time, NRR analysis has been completed for Combustion Engineering (CE) and Westinghouse plants, considered representative of "U" type steam generator plants. Analysis has not been completed for once-through Babcock and Wilcox (B&W) plants.

6.3.2 Safety Issue Risk and Dose

The public risk reduction and occupational dose associated with resolution of issue 18 are estimated in this section using the procedures outlined in Section 5.0. For the public risk, a LOCA initiator and two operator errors are assumed to be affected. For the occupational dose, it is assumed that part of this issue resolution will require placement of pipe shielding or restraints and possibly some instrumentation in the steam generator enclosures at operating plants. The remainder of the resolution will center on operator training and thus not impact occupational exposure. Results of the analyses for public risk reduction and occupational dose are summarized in Tables 6.3.2 and 6.3.3, respectively.

6.3.3 Safety Issue Costs

Proposed resolutions for issue 18 are poorly defined at present, making cost estimation quite difficult. For this analysis, it is assumed that each utility will expend 25 man-wk/plant of labor and $150,000/plant for equipment to implement the SIR. Equipment installation is assumed to take place during normally scheduled outages, necessitating no additional down-time. No license amendment is anticipated as a result of SIR. Recurring requirements for operator training and equipment maintenance will presumably involve 3 man-wk/ry each from the utility.

Generic issue resolution will presumably require 80 man-weeks of NRC staff labor and $150,000 for contractor support. NRC labor to support SIR implementation should be minimal, about 1 man-wk/plant. To review SIR operation and maintenance, 0.5 man-wk/ry of NRC labor is assumed. The results of the industry and NRC cost analyses are summarized in Table 6.3.4.

TABLE 6.3.2. Public Risk Reduction Work Sheet

1. Title and Identification Number of Safety Issue:

   Steam Line Break with Consequential Small LOCA (No. 18)

2. Affected Plants (N) and Average Remaining Lives ($\bar{T}$):

   All 90 PWRs are assumed to be affected ($\bar{T}$ = 28.8 yr)

3. Plants Selected for Analysis:

   Oconee 3 - representative PWR

4. Parameters Affected by SIR:

   Based upon the redefinition of the parameters S_3, HPRSCM, and WXCM discussed in Attachment 1, the following parameters are designated as affected:

   $(S_3)_1$ = occurrence of combined primary and secondary system LOCA

   $(HPRSCM)_1$ = common-cause failure of the operator to align suction of the high pressure recirculation system (HPRS) to the discharge of the low pressure recirculation system (LPRS) during a combined LOCA sequence, i.e., conditional upon $(S_3)_1$

   $(WXCM)_1$ = common-cause failure of the operator to open both containment sump suction valves in the low pressure/containment spray recirculation system (LP/CSRS) at the start of recirculation during a combined LOCA sequence, i.e., conditional upon $(S_3)_1$.

5. Affected Parameters' Base-Case Values:

   $(S_3)_1$ = 4E-6/ry

   $(HPRSCM)_1$ = 0.03

   $(WXCM)_1$ = 0.03

   See Attachment 1.

TABLE 6.3.2. (contd)

6. Affected Accident Sequences and Base-Case Frequencies:

$$(S_3)_1H - \left\{ \begin{array}{lcl} \gamma\ (\text{PWR-3}) & = & 6.9\text{E-8/ry} \\ \beta\ (\text{PWR-5}) & = & 1.0\text{E-9/ry} \\ \varepsilon\ (\text{PWR-7}) & = & 6.9\text{E-8/ry} \end{array} \right.$$

$$(S_3)_1FH - \left\{ \begin{array}{lcl} \gamma\ (\text{PWR-2}) & = & 6.0\text{E-8/ry} \\ \beta\ (\text{PWR-4}) & = & 8.8\text{E-10/ry} \\ \varepsilon\ (\text{PWR-6}) & = & 6.0\text{E-8/ry} \end{array} \right.$$

$$(S_3)_1D - \left\{ \begin{array}{lcl} \gamma\ (\text{PWR-3}) & = & 2.2\text{E-9/ry} \\ \beta\ (\text{PWR-5}) & = & 3.1\text{E-11/ry} \\ \varepsilon\ (\text{PWR-7}) & = & 2.2\text{E-9/ry} \end{array} \right.$$

See Attachment 1. Also, note that the non-dominant minimal cut sets are assumed to be affected since they too contain the initiator parameter $(S_3)_1$. The containment failure mode likelihoods are assumed to be the same as for the sequences prior to redefinition.

7. Affected Release Categories and Base-Case Frequencies:

PWR-2 = 6.0E-8/ry

PWR-3 = 7.1E-8/ry

PWR-4 = 8.8E-10/ry

PWR-5 = 1.0E-9/ry

PWR-6 = 6.0E-8/ry

PWR-7 = 7.1E-8/ry

8. Base-Case, Affected Core-Melt Frequency ($\bar{F}$):

$\bar{F}$ = 2.6E-7/ry

9. Base-Case, Affected Public Risk (W):

W = 0.68 man-rem/ry

TABLE 6.3.2. (contd)

10. Affected Parameters' Adjusted-Case Values:

SIR is assumed to reduce the likelihood of operator misinterpretation during a combined LOCA, but not to the point where the operator is as reliable as during a primary LOCA only. Thus, the base-case values for $(HPRSCM)_1$ and $(WXCM)_1$ are reduced by a factor of five in the adjusted case. SIR is also assumed to have less of an effect on the frequency of a combined LOCA than on the operator errors. Thus, the base-case value for $(S_3)_1$ is reduced only by a factor of two in the adjusted case.

$$(S_3)_1 = (4E\text{-}6/ry)/2 = 2E\text{-}6/ry$$
$$(HPRSCM)_1 = (.03)/5 = 0.006$$
$$(WXCM)_1 = (.003)/5 = 0.006$$

11. Affected Accident Sequences and Adjusted-Case Values:

$$(S_3)_1H - \begin{cases} \gamma\ (PWR\text{-}3) = 1.1E\text{-}8/ry \\ \beta\ (PWR\text{-}5) = 1.6E\text{-}10/ry \\ \varepsilon\ (PWR\text{-}7) = 1.1E\text{-}8/ry \end{cases}$$

$$(S_3)_1FH - \begin{cases} \gamma\ (PWR\text{-}2) = 6.2E\text{-}9/ry \\ \beta\ (PWR\text{-}4) = 9.1E\text{-}11/ry \\ \varepsilon\ (PWR\text{-}6) = 6.2E\text{-}9/ry \end{cases}$$

$$(S_3)_1D - \begin{cases} \gamma\ (PWR\text{-}3) = 1.1E\text{-}9/ry \\ \beta\ (PWR\text{-}5) = 1.6E\text{-}11/ry \\ \varepsilon\ (PWR\text{-}7) = 1.1E\text{-}9/ry \end{cases}$$

Again, the non-dominant cut sets are also presumed to be affected, as in Step 6.

TABLE 6.3.2. (contd)

12. Affected Release Categories and Adjusted-Case Frequencies:

PWR-2 = 6.2E-9/ry

PWR-3 = 1.2E-8/ry

PWR-4= 9.1E-11/ry

PWR-5 = 1.8E-10/ry

PWR-6 = 6.2E-9/ry

PWR-7 = 1.2E-8/ry

13. Adjusted-Case, Affected Core-Melt Frequency ($\bar{F}^*$):

$\bar{F}^*$ = 3.7E-8/ry

14. Adjusted-Case, Affected Public Risk (W^*):

W^* = 0.096 man-rem/ry

15. Core-Melt Frequency Reduction ($\Delta\bar{F}$):

$\Delta\bar{F}$ = 2.2E-7/ry

16. Per-Plant Public Risk Reduction (ΔW):

ΔW = 0.58 man-rem/ry

17. Total Public Risk Reduction [$(\Delta W)_{Total}$]:

Best Estimate	Upper Bound	Lower Bound
1,500 man-rem	5.3E+4 man-rem	0

ATTACHMENT 1

Two concerns have been identified which could increase the risk associated with this issue: 1) the possibility of primary side LOCAs may be increased through the consideration of new initiating mechanisms, and 2) the symptoms of a combined primary and secondary blowdown may increase the possibility for operator error through misinterpretation and improper action.

To translate these concerns into effects upon Oconee risk parameters, two assumptions are made:

1. The accident sequences for a combined LOCA, whether it arises from an MSLB, SGTR, or stuck-open valve, will parallel those for primary side small LOCAs (i.e., those sequences initiated by rupture of primary coolant system piping with diameter $\leq$ four inches or by transient-induced failure of a pressurizer safety/relief valve to reclose). In other words, the combined LOCA accident sequences will parallel those for the S_3 and T_2MQ initiators in the Oconee dominant accident sequences. A review of Table A.3 indicates that, for corresponding S_3 and T_2MQ accident sequences (i.e., sequences whose failures subsequent to the initiating events are the same, such as S_3H and T_2MQH), the dominant minimal cut sets are the same except for the initiators S_3 and $T_2 \cdot M \cdot \bar{P}_1 \cdot Q$. For example, for both sequences S_3FH and T_2MQFH, the dominant minimal cut sets are as follows:

   (initiator) • WXCM
   (initiator) • W•X
   (initiator) • B•W
   (initiator) • C•X

   where the initiator is S_3 or $T_2 \cdot M \cdot \bar{P}_1 \cdot Q$. Thus, it is assumed that only one combined LOCA initiator need be designated to simulate accident sequences for combined LOCAs whatever their source. For simplicity, this combined LOCA initiator is designated through a redefinition of the parameter S_3.

2. Only direct operator action required during a combined LOCA sequence may be adversely affected by confusion arising from symptoms of the combined LOCA. Review of the Oconee dominant accident sequences reveals that only the parameters HPRSCM (common-cause failure of the operator to align suction of the HPRS to the discharge of the LPRS) and WXCM (common-cause failure of the operator to open both containment sump suction valves in the LP/CSRS at the start of recirculation) involve direct operation action during a LOCA. Thus, only these terms are redefined to include the possibility of operator confusion arising during a combined LOCA sequence.

Thus, it is assumed that a reasonable estimate of the public risk reduction associated with resolution of issue 18 can be obtained by redefining S_3, HPRSCM, and WXCM to include failures related to the combined LOCA and then treating their redefined portions as the affected parameters.

$\underline{S_3}$

As originally defined in the Oconee RSSMAP study, S_3 presumably does not include the possibility of a combined LOCA. To include this combined LOCA initiator, S_3 is redefined as follows:

$$S_3 = (S_3)_0 + (S_3)_1$$

where $(S_3)_0$ represents S_3 as originally defined
$(S_3)_1$ represents the combined LOCA initiator.

Thus, since issue 18 addresses only the combined LOCA, $(S_3)_1$ and not $(S_3)_0$ is treated as an affected parameter.

Since $(S_3)_1$ is not part of the original Oconee assessment, its base-case frequency cannot be estimated directly from that study. An alternative procedure is used.

Data exist for the following transients (McClymont 1982):

	Frequency (1/ry)
steam generator leakage	0.08
condenser leakage	0.04
miscellaneous leakage in secondary system	0.08

However, these leakage terms are not representative of catastrophic rupture of the pipe. As a first estimate of steam line rupture, LOCA data for the primary system are used. Due to the lower operating pressure and temperature, the frequency of rupture for the next larger category of RCS piping (S_2, 4" < d < 10") is deemed appropriate for estimating the base-case frequency of $(S_3)_1$:

$$S_2 = 4E\text{-}4/ry$$

The base-case frequency of $(S_3)_1$ is assumed to be 1% of the S_2 frequency, i.e.:

$$(S_3)_1 = (0.01)(4E\text{-}4/ry) = 4E\text{-}6/ry$$

HPRSCM AND WXCM

The parameters HPRSCM and WXCM must be redefined to reflect the potential for misinterpretation of the accident during a combined LOCA. This is done as follows in a manner similar to that for S_3:

$$\mathrm{HPRSCM} = (\mathrm{HPRSCM})_o + (\mathrm{HPRSCM})_1$$
$$\mathrm{WXCM} = (\mathrm{WXCM})_o + (\mathrm{WXCM})_1$$

where the terms with the "o" subscripts represent the parameters as originally defined, while those with the "1" subscripts represent the operator errors HPRSCM and WXCM only during a combined LOCA sequence [i.e., conditional upon $(S_3)_1$]. Thus, since issue 18 addresses only the combined LOCA, $(\mathrm{HPRSCM})_1$ and $(\mathrm{WXCM})_1$ [and not $(\mathrm{HPRSCM})_o$ and $(\mathrm{WXCM})_o$] are treated as affected parameters.

As with $(S_3)_1$, neither $(HPRSCM)_1$ nor $(WXCM)_1$ are part of the original Oconee assessment. Thus, estimation of their base-case probabilities requires an alternative procedure.

As originally assessed, the terms $(HPRSCM)_0$ and $(WXCM)_0$ each have a probability of 0.003. This value represents an operator error during a primary system LOCA sequence. It is assumed that the chance for operator error during a combined LOCA sequence will be increased above this value due to the greater possibility for operator confusion and misinterpretation of the combined LOCA symptoms (discussed earlier in assumption #2). Increasing this error likelihood by a factor of 10 is presumed to be reasonable, but conservative, for this issue analysis. Thus, the terms $(HPRSCM)_1$ and $(WXCM)_1$ each are assumed to have a base-case probability of (10)(0.003) = 0.03.

AFFECTED ACCIDENT SEQUENCES

Table A.3 lists the following dominant accident sequences initiated by S_3:

S_3H - γ, β, ε
S_3FH - γ, β, ε
S_3D - γ, β, ε

Following substitution of the redefined parameters into the minimal cut sets for these sequences, the following affected sequences and minimal cut sets result:

Affected Sequence[a]	Affected Minimal Cut Sets[b]
$(S_3)_1H$	$(S_3)_1 \cdot (HPRSCM)_1$
	$(S_3)_1 \cdot LPISCM$
	$(S_3)_1 \cdot D \cdot E$
	$(S_3)_1 \cdot E \cdot W$
	$(S_3)_1 \cdot D \cdot X$

Affected Sequence[a]	Affected Minimal Cut Sets[b]
$(S_3)_1FH$	$(S_3)_1 \cdot (WXCM)_1$
	$(S_3)_1 \cdot W \cdot X$
	$(S_3)_1 \cdot B \cdot W$
	$(S_3)_1 \cdot C \cdot X$
$(S_3)_1D$	$(S_3)_1 \cdot C1 \cdot B1$
	$(S_3)_1 \cdot A1 \cdot B1$
	$(S_3)_1 \cdot CH1 \cdot B1$
	$(S_3)_1 \cdot C1 \cdot CH2$
	$(S_3)_1 \cdot RCSRBCM$
	$(S_3)_1 \cdot CH1 \cdot CH2$

(a) Containment failure mode designators left off for simplicity.

(b) For redefined parmeters, terms as originally assessed (i.e., those with "o" subscripts) do not appear in affected minimal cut sets. Consider the following example. Sequence S_3FH originally contained the following cut set:

$$S_3 \cdot WXCM$$

Upon substituting the redefined parameters, this cut set is expanded by Boolean algebra as follows:

$$\begin{aligned} S_3 \cdot WXCM &= [(S_3)_o + (S_3)_1] \cdot [(WXCM)_o + (WXCM)_1] \\ &= \underbrace{(S_3)_o \cdot (WXCM)_o}_{\substack{\text{Original cut set} \\ \text{(unaffected)}}} + \underbrace{(S_3)_1 \cdot (WXCM)_1}_{\substack{\text{New cut set} \\ \text{(affected)}}} \end{aligned}$$

The cross-product cut sets, $(S_3)_o \cdot (WXCM)_1$ and $(S_3)_1 \cdot (WXCM)_o$, are defined to be zero since $(WXCM)_o$ is conditional only upon $(S_3)_o$ and $(WXCM)_1$ is conditional only upon $(S_3)_1$.

TABLE 6.3.3. Occupational Dose Work Sheet

1. Title and Identification Number of Safety Issue:

   Steam Line Break with Consequential Small LOCA (No. 18)

2. Affected Plants (N):

   All 90 PWRs (47 backfit and 43 forward-fit)

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

   47 backfit PWRs, $\bar{T}$ = 27.7 yr
   43 forward-fit PWRs, $\bar{T}$ = 30 yr
   all 90 PWRs, $\bar{T}$ = 28.8 yr

4. Per-Plant Occupational Dose Reduction Due to Accident-Avoidance [$\Delta(\bar{F}D_R)$]:

   $$\Delta(\bar{F}D_R) = (19{,}900 \text{ man-rem})(2.2E\text{-}7/\text{ry}) = 0.0044 \text{ man-rem/ry}$$

5. Total Occupational Dose Reduction Due to Accident-Avoidance (ΔH):

Best Estimate	Upper Bound	Lower Bound
11 man-rem	80 man-rem	0

6. Per-Plant Utility Labor in Radiation Zones for SIR Implementation:

   It is assumed that 3 man-wk/plant will be required for installation of equipment in the steam generator enclosures. Only constructed plants are assumed to have activated generator structures. Assuming a 75% utilization factor for manpower in the radiation zone gives

   (0.75)(3 man-wk/plant)(40 man-hr/man-wk) = 90 man-hr/plant

7. Per-Plant Occupational Dose Increase for SIR Implementation (D):

   It is assumed here that radiation fields of 100 mR/hr exist in the steam generator enclosures.

   D = (90 man-hr/plant)(0.10 R/hr) = 9.0 man-rem/plant

TABLE 6.3.3. (contd)

8. Total Occupational Dose Increase for SIR Implementation (ND):

$$ND = (47 \text{ backfit PWRs})(9.0 \text{ man-rem/plant})$$
$$= 420 \text{ man-rem}$$

9. Per-Plant Utility Labor in Radiation Zones for SIR Operation and Maintenance:

It is assumed that one additional man-week per reactor-year will be required for examination of equipment installed in the steam generator enclosures as part of the routine maintenance program. This applies to all 90 PWRs. Again assuming a 75% utilization factor for actual work in the radiation fields gives

$$(0.75)(1 \text{ man-wk/ry})(40 \text{ man-hr/man-wk}) = 30 \text{ man-hr/ry}$$

10. Per-Plant Occupational Dose Increase for SIR Operation and Maintenance (D_O):

Again a 100 mR/hr radiation field is assumed.

$$D_O = (30 \text{ man-hr/ry})(0.10 \text{ R/hr})$$
$$= 3.0 \text{ man-rem/ry}$$

11. Total Occupational Dose Increase for SIR Operation and Maintenance ($N\bar{T}D_O$):

$$N\bar{T}D_O = (90 \text{ PWRs})(28.8 \text{ yr})(3.0 \text{ man-rem/ry})$$
$$= 7{,}780 \text{ man-rem}$$

12. Total Occupational Dose Increase (G):

Best Estimate	Upper Bound	Lower Bound
8,200 man-rem	2.5E+4 man-rem	2,700 man-rem

TABLE 6.3.4. Safety Issue Cost Work Sheet

1. Title and Identification Number of Safety Issue:

   Steam Line Break with Consequential Small LOCA (No. 18)

2. Affected Plants (N):

   All 90 PWRs (47 backfit and 43 forward-fit)

3. Average Remaining Lives of Affected Plants ($\bar{T}$):

   47 backfit PWRs, $\bar{T}$ = 27.7 yr

   43 forward-fit PWRs, $\bar{T}$ = 30 yr

   all 90 PWRs, $\bar{T}$ = 28.8 yr

Industry Costs (steps 4 through 12)

4. Per-Plant Industry Cost Savings Due to Accident-Avoidance [$\Delta(\bar{F}A)$]:

   $\Delta(\bar{F}A)$ = ($1.65E+9)(2.2E-7/ry)

   = $360/ry

5. Total Industry Cost Savings Due to Accident-Avoidance (ΔH):

Best Estimate	Upper Bound	Lower Bound
$9.4E+5	$6.7E+6	0

6. Per-Plant Industry Resources for SIR Implementation:

   Labor (engineering, craft services, etc.) = 25 man-wk/plant

   Equipment (cost estimated directly in next step)

   Additional Down-time = none

   These apply to all PWRs.

7. Per-Plant Industry Cost for SIR Implementation (I):

   Labor = (25 man-wk/plant)($2270/man-wk) = $5.7E+4/plant

   Equipment = $1.5E+5/plant

   I = $2.07E+5/plant

TABLE 6.3.4. (contd)

8. Total Industry Cost for SIR Implementation (NI):

$$NI = (90\ \text{PWRs})(\$2.07E+5/\text{plant}) = \$1.86E+7$$

9. Per-Plant Industry Labor for SIR Operation and Maintenance:

Operator Training =	3 man-wk/ry
Equipment Maintenance =	3 man-wk/ry
	6 man-wk/ry

This applies to all PWRs.

10. Per-Plant Industry Cost for SIR Operation and Maintenance (I_O):

$$I_O = (6\ \text{man-wk/ry})(\$2270/\text{man-wk}) = \$1.36E+4/\text{ry}$$

11. Total Industry Cost for SIR Operation and Maintenance ($N\bar{T}I_O$):

$$N\bar{T}I_O = (90\ \text{PWRs})(28.8\ \text{yr})(\$1.36E+4/\text{ry}) = \$3.53E+7$$

12. Total Industry Cost (S_I):

Best Estimate	Upper Bound	Lower Bound
\$5.4E+7	\$7.4E+7	\$3.4E+7

NRC Costs (steps 13 through 21)

13. NRC Resources for SIR Development:

NRC Staff Labor = 8 man-wk

Contractor Support (cost estimated directly in next step)

14. Total NRC Cost for SIR Development (C_D):

Labor = (8 man-wk)(\$2270/man-wk) =	\$1.8E+4
Contractor Support =	\$1.5E+5
C_D =	\$1.68E+5

TABLE 6.3.4. (contd)

15. Per-Plant NRC Labor for Support of SIR Implementation:

1 man-wk/plant

16. Per-Plant NRC Cost for Support of SIR Implementation (C):

$$C = (1 \text{ man-wk/plant})(\$2270/\text{man-wk})$$
$$= \$2,270/\text{plant}$$

17. Total NRC Cost for Support of SIR Implementation (NC):

$$NC = (90 \text{ PWRs})(\$2270/\text{plant})$$
$$= \$2.04E+5$$

18. Per-Plant NRC Labor for Review of SIR Operation and Maintenance:

0.5 man-wk/ry

19. Per-Plant NRC Cost for Review of SIR Operation and Maintenance (C_o):

$$C_o = (0.5 \text{ man-wk/ry})(\$2270/\text{man-wk})$$
$$= \$1,140/\text{ry}$$

20. Total NRC Cost for Review of SIR Operation and Maintenance ($N\bar{T}C_o$):

$$N\bar{T}C_o = (90 \text{ PWRs})(28.8 \text{ yr})(\$1140/\text{ry})$$
$$= \$2.94E+6$$

21. Total NRC Cost (S_N):

Best Estimate	Upper Bound	Lower Bound
\$3.3E+6	\$4.8E+6	\$1.8E+6

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APPENDIX A

RISK PARAMETERS FOR OCONEE 3 PWR

The risk equation for Oconee 3 (Babcock & Wilcox PWR with dry containment) has been summarized for the dominant accident sequences contributing to the seven PWR core-melt release categories as defined in WASH-1400. The Oconee results have been extracted from its RSSMAP study, NUREG/CR-1659/2 (Kolb 1981) and are provided here in Tables A.1 through A.4, with Addenda A.I through A.III. The information is presented so as to be compatible with the technique described in Section 3.0 for estimating the risk reduction.

Table A.1 lists the dominant accident sequences for each PWR core-melt release category. The frequencies (reactor-year^{-1}) are given for each sequence along with the category totals. Also provided are the frequencies for the aggregates of non-dominant accident sequences per release category. Table A.2 defines the symbols used in Table A.1.

Table A.3 presents the dominant minimal cut sets for each dominant accident sequence listed in Table A.1. Also provided are the cut set frequencies, the containment failure modes for each sequence, the mode probabilities and corresponding release categories, and the frequencies of the sequences excluding the containment failure probabilities. Where appropriate, the contribution to the sequence frequency from the aggregate of non-dominant minimal cut sets is provided.

Table A.4 lists the elements of the dominant minimal cut sets given in Table A.3. A brief description of each element is provided, with extended resolution into contributory failures where appropriate. The level of resolution is limited to that provided in the RSSMAP report. Probabilities are listed for each element. These can be viewed as unavailabilities unless otherwise specified. (Note that initiating event probabilities are occurrence rates in terms of reactor-year^{-1}.)

Three elements in Table A.4 have somewhat detailed resolutions. For these, Addenda A.I through A.III have been provided. In some cases, additional detail can be found in the Oconee RSSMAP report (Kolb 1981). The analyst is referred to this for any further information that he may need.

TABLE A.1. Oconee Dominant Accident Sequences and Frequencies (Reactor-Year^{-1})

Accident Sequence	PWR Release Category (based on WASH-1400) 1	2	3	4	5	6	7
T_2MLU			γ 6.0E-7		β 8.8E-9		ϵ 6.0E-7
T_1MLU			γ 1.0E-6		β 1.5E-8		ϵ 1.0E-6
V		4.0E-6					
$T_1(B_3)MLU$			γ 1.1E-6		β 1.6E-8		ϵ 1.1E-6
T_2MQH			γ 5.5E-6		β 8.0E-8		ϵ 5.5E-6
S_3H			γ 5.0E-6		β 7.3E-8		ϵ 5.0E-6
S_1D	α 6.7E-8		γ 1.3E-6		β 4.9E-8		ϵ 5.4E-6
T_2MQFH		γ 2.5E-6		β 3.7E-8		ϵ 2.5E-6	
S_3FH		γ 2.1E-6		β 3.1E-8		ϵ 2.1E-6	
S_2FH	α 1.3E-8			β 9.5E-9		ϵ 1.0E-6	
T_2MLUO			γ 4.1E-6		β 5.9E-8		ϵ 4.1E-6
T_2KMU			γ 3.9E-6		β 5.7E-8		ϵ 3.9E-6
S_2D	α 2.0E-8		γ 4.0E-7		β 1.5E-8		ϵ 1.6E-6
S_3D			γ 7.0E-7		β 1.0E-8		ϵ 7.0E-7
T_1MLUO			γ 2.7E-6		β 3.9E-8		ϵ 2.7E-6
T_3MLUO			γ 5.5E-7		β 8.0E-9		ϵ 5.5E-7
T_2MQD			γ 7.5E-7		β 1.1E-8		ϵ 7.5E-7
Non-Dominant	1E-8	1.4E-6	1E-6	1.9E-8	2E-8	1.7E-6	2E-6
Total	1.1E-7	1.0E-5	2.9E-5	9.7E-8	4.6E-7	7.3E-6	3.5E-5

TABLE A.2. Symbols Used in Table A.1

Initiating Events

T_1 - Loss of Offsite Power Transient

T_2 - Loss of Power Conversion System Transient Caused by Other than a Loss of Offsite Power

T_3 - Transients with the Power Conversion System Initially Available

S_1 - Intermediate LOCA (10"<D≤13.5", D = pipe diameter)

S_2 - Small LOCA (4"<D≤10")

S_3 - Small-Small LDCA (D≤4")

V - Interfacing Systems LOCA

System Failures

(B_3) - Emergency Power System

D - Emergency Coolant Injection System

F - Containment Spray Recirculation System

H - Emergency Coolant Recirculation System

K - Reactor Protection System

L - Emergency Feedwater System, Recovery of Power Conversion System and High Head Auxiliary Feedwater System

M - Power Conversion System (Normal Operation)

O - Reactor Building Cooling System

Q - Reclosure of Pressurizer Safety/Relief Valves

U - High Pressure Injection System

Containment Failure Modes

α - Vessel Steam Explosion

β - Penetration Leakage

γ - Overpressure Due to Hydrogen Burning

ϵ - Base Mat Melt Through

TABLE A.3. Dominant Minimal Cut Sets of Oconee Dominant Accident Sequences

Accident Sequence	Sequence Frequency (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_2MLU	1.2E-6	γ	.5	3	T_2·M·CONST1·PCSNR·HPMAN	9.5E-7
(contribution from non-dominant minimal cut sets = 1E-7)		β	.0073	5	T_2·M·F1·G1·PCSNR·HPMAN	8.8E-8
		ε	.5	7	T_2·M·F1·CH4·PCSNR·HPMAN	3.2E-8
T_1MLU	2.0E-6	γ	.5	3	T_1·M·CONST2·HPMAN	1.9E-6
		β	.0073	5	T_1·M·F1·G1·HPMAN	5.9E-8
		ε	.5	7	T_1·M·F1·CH4·HPMAN	2.1E-8
V	4.0E-6	NA	NA	2	V	4.0E-6
$T_1(B_3)$MLU	2.2E-6	γ	.5	3	T_1·(B_3)·M·HHMAN·LOPNRE	2.0E-6
		β	.0073	5	T_1·(B_3)·M·HHMAN·HPMAN	1.5E-7
		ε	.5	7		
T_2MQH	1.1E-5	γ	.5	3	T_2·M·$\overline{P}_1$·Q·HPRSCM	4.5E-6
(contribution from non-dominant minimal cut sets = 1E-6)		β	.0073	5	T_2·M·$\overline{P}_1$·Q·LPISCM	4.5E-6
		ε	.5	7	T_2·M·$\overline{P}_1$·Q·D·E	7.4E-7
					T_2·M·$\overline{P}_1$·Q·D·X	3.2E-7
					T_2·M·$\overline{P}_1$·Q·E·W	3.2E-7

TABLE A.3. (contd)

Accident Sequence	Sequence Frequency (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
S_3H	1.DE-5	γ	.5	3	$S_3 \cdot$HPRSCM	3.9E-6
(contribution from non-dominant minimal cut sets = 1E-6)		β	.0073	5	$S_3 \cdot$LPISCM	3.9E-6
		ε	.5	7	$S_3 \cdot D \cdot E$	6.4E-7
					$S_3 \cdot E \cdot W$	2.7E-7
					$S_3 \cdot D \cdot X$	2.7E-7
S_1D	6.7E-6	α	.01	1	$S_1 \cdot D$	2.3E-6
(contribution from non-dominant minimal cut sets = 1E-7)		γ	.2	3	$S_1 \cdot E$	2.3E-6
		β	.0073	5	$S_1 \cdot$CH3	5.0E-7
		ε	.8	7	$S_1 \cdot$CH4	5.0E-7
					$S_1 \cdot C$	3.3E-7
					$S_1 \cdot B$	3.3E-7
					$S_1 \cdot$LPISCM	3.0E-7

TABLE A.3. (contd)

Accident Sequence	Sequence Frequency (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_2MQFH	5.0E-6	γ	.5	2	$T_2 \cdot M \cdot \overline{P}_1 \cdot Q \cdot WXCM$	4.5E-6
(contribution from non-dominant minimal		β	.0073	4	$T_2 \cdot M \cdot P_1 \cdot Q \cdot W \cdot X$	1.3E-7
cut sets = 3E-7)		ε	.5	6	$T_2 \cdot M \cdot \overline{P}_1 \cdot Q \cdot 8 \cdot W$	4.1E-8
					$T_2 \cdot M \cdot \overline{P}_1 \cdot Q \cdot C \cdot X$	4.1E-8
S_3FH	4.2E-6	γ	.5	2	$S_3 \cdot WXCM$	3.9E-6
(contribution from non-dominant minimal		β	.0073	4	$S_3 \cdot W \cdot X$	1.1E-7
cut sets = 1E-7)		ε	.5	6	$S_3 \cdot B \cdot W$	3.5E-8
					$S_3 \cdot C \cdot X$	3.5E-8
S_2FH	1.3E-6	α	.01	1	$S_2 \cdot WXCM$	1.2E-6
		β	.0073	4	$S_2 \cdot X \cdot W$	3.5E-8
		ε	.8	6	$S_2 \cdot B \cdot W$	1.1E-8
					$S_2 \cdot C \cdot X$	1.1E-8
T_2MLUO	8.1E-6	γ	.5	3	$T_2 \cdot M \cdot PCSNR \cdot F1 \cdot G1$	5.9E-6
(contribution from non-dominant minimal		β	.0073	5	$T_2 \cdot M \cdot PCSNR \cdot F1 \cdot CH4$	2.1E-6
cut sets = 1E-7)		ε	.5	7		

TABLE A.3. (contd)

Accident Sequence	Sequence Frequency (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_2KMU	7.8E-6	γ	.5	3	$T_2 \cdot K \cdot M \cdot$HPMAN1	7.8E-6
		β	.0073	5		
		ε	.5	7		
S_2D	2.0E-6	α	.01	1	$S_2 \cdot$LPISCM	1.2E-6
(contribution from non-dominant minimal cut sets = 1E-7)		γ	.2	3	$S_2 \cdot E \cdot D$	2.0E-7
		β	.0073	5	$S_2 \cdot$C1$\cdot$B1	1.4E-7
		ε	.8	7	$S_2 \cdot$A1$\cdot$B1	1.4E-7
					$S_2 \cdot$CH1$\cdot$B1	7.0E-8
					$S_2 \cdot E \cdot$CH3	4.6E-8
					$S_2 \cdot$CH4$\cdot D$	4.6E-8
					$S_2 \cdot B \cdot D$	2.5E-8
					$S_2 \cdot E \cdot C$	2.5E-8
S_3D	1.4E-6	γ	.5	3	$S_3 \cdot$C1$\cdot$B1	4.5E-7
(contribution from non-dominant minimal cut sets = 1E-7)		β	.0073	5	$S_3 \cdot$A1$\cdot$B1	4.5E-7
		ε	.5	7	$S_3 \cdot$CH1$\cdot$B1	2.3E-7
					$S_3 \cdot$C1$\cdot$CH2	6.4E-8
					$S_3 \cdot$A1$\cdot$CH2	6.4E-8

TABLE A.3. (contd)

Accident Sequence	Sequence Frequency (ry^{-1})	Cont Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
S_3D (cont.)					$S_3 \cdot RCSRBCM$	4.2E-8
					$S_3 \cdot CH1 \cdot CH2$	3.3E-8
T_1MLUO	5.4E-6	γ	.5	3	$T_1 \cdot M \cdot F1 \cdot G1$	3.9E-6
(contribution from non-dominant minimal cut sets = 1E-7)		β	.0073	5	$T_1 \cdot M \cdot F1 \cdot CH4$	1.4E-6
		ε	.5	7		
T_3MLUO	1.1E-6	γ	.5	3	$T_3 \cdot M1 \cdot F1 \cdot G1$	7.8E-7
		β	.0073	5	$T_3 \cdot M1 \cdot F1 \cdot CH4$	2.8E-7
		ε	.5	7		
T_2MQD	1.5E-6	γ	.5	3	$T_2 \cdot M \cdot \overline{P}_1 \cdot Q \cdot C1 \cdot B1$	5.1E-7
		β	.0073	5	$T_2 \cdot M \cdot \overline{P}_1 \cdot Q \cdot A1 \cdot B1$	5.1E-7
		ε	.5	7	$T_2 \cdot M \cdot \overline{P}_1 \cdot Q \cdot CH1 \cdot B1$	2.6E-7
					$T_2 \cdot M \cdot \overline{P}_1 \cdot Q \cdot C1 \cdot CH2$	7.4E-8
					$T_2 \cdot M \cdot \overline{P}_1 \cdot Q \cdot A1 \cdot CH2$	7.4E-8
					$T_2 \cdot M \cdot \overline{P}_1 \cdot Q \cdot RCSRBCM$	4.8E-8
					$T_2 \cdot M \cdot \overline{P}_1 \cdot Q \cdot CH1 \cdot CH2$	3.8E-8

TABLE A.4. Elements of Dominant Minimal Cut Sets of Oconee Dominant Accident Sequences

Symbol	Description	Probability
T_1	Loss of offsite power.	0.2/ry
T_2	Loss-of-power-conversion-system (PCS) transient caused by other than a loss of offsite power.	3/ry
T_3	Transients requiring shutdown with the PCS initially available.	4/ry
S_1	Rupture of reactor coolant system (RCS) piping with diameter >10" but ≤13.5". The probability of event S_1 is taken to be that for a large LOCA (diameter >6") from WASH-1400 (1E-4/ry).	1E-4/ry
S_2	Rupture of RCS piping with diameter >4" but <10". The probability of event S_2 is taken to be the sum of those for a large LOCA and a small LOCA (diameter >2" but ≤6") from WASH-1400 (1E-4/ry + 3E-4/ry = 4E-4/ry).	4E-4/ry
S_3	Rupture of RCS piping with diameter ≤4". The probability of event S_3 is taken to be the sum of those for a small LOCA and a very small LOCA (diameter >1/2" but ≤2") from WASH-1400 (3E-4/ry + .001/ry = .0013/ry).	.0013/ry
B	Failure of a pump suction valve in train B of the low pressure/containment spray injection system (LP/CSIS). Event B occurs if either of the following fails: 1. A normally-open (NO) motor-operated valve (MOV). Its failure probability is the sum of the following contributory modes: operator error - .001 plugged - .0001 maintenance outage - .0021 .0032	.0033

TABLE A.4 (contd)

Symbol	Description	Probability
	2. A check valve (CV). Its failure probability is that for a hardware failure (.0001).	
	The failure probability of event B is the sum of the above.	
C	Failure of a pump suction valve in train A of the LP/CSIS	.0033
	The expansion of event C into its contributory failures is analogous to that for event B.	
D	Failure of a pump discharge valve in train A of the LP/CSIS.	.023
	Event D occurs if any of the following fails:	
	1. Either of two CVs. Each has a failure probability for hardware failure of .0001.	
	2. Either of two NO MOVs. Each has a failure probability of .0032 with the contributory modes as shown in event B.	
	3. A normally-closed (NC) MOV. Its failure probability is the sum of the following contributory modes: hardware - .001; plugged - .0001; control circuitry - .0064; maintenance outage - .0021; total .0096	
	4. An NO manual valve (ManV). Its failure probability is the sum of the following contributory modes: operator error - .0001; plugged - .0001; total .0002	

TABLE A.4 (contd)

Symbol	Description	Probability
	5. An NC ManV. Its failure probability is that for operator error (.001).	
	6. A pump. Its failure probability is the sum of the following contributory modes:	

hardware	-	.001
control circuitry	-	.0018
test outage	-	.0019
	-	.0047

	7. Valves in test line A inadvertently left open. The failure probability is that for human error (.001).	
	The failure probability of event D is the sum of the above: 2(.0001) + 2(.0032) + .0096 + .0002 + .001 + .0047 + .001 = .023. The factors of two account for the contributions from two CVs and two NO MOVs.	
E	Failure of a pump discharge valve in train B of the LP/CSIS	.023
	The expansion of event E into its contributory failures is analogous to that for event D.	
CH1	Failure of logic channel 1 of the engineered safeguards protective system (ESPS)	.0050
	Event CH1 occurs if there are single or double hardware failures in the logic channel (failure probability = .0029) or if there is a test or maintenance outage (failure probability = .0021). The failure probability of event CH1 is the sum of these.	
CH2	Failure of logic channel 2 of the ESPS	.0050
	The expansion of event CH2 into its contributory failures is analogous to that for event CH1.	

TABLE A.4 (contd)

Symbol	Description	Probability
CH3	Failure of logic channel 3 of the ESPS	.0050
	The expansion of event CH3 into its contributory failures is analogous to that for event CH1.	
CH4	Failure of logic channel 4 of the ESPS.	.0050
	The expansion of event CH4 into its contributory failures is analogous to that for event CH1.	
CONST1	Failure of the emergency feedwater system (EFWS) due to primarily hardware failure of the turbine pump train and both of the electric pump trains, or blockage of flow through both steam generator lines.	2.1E-4
	The expansion of event CONST1 into its contributory failures is somewhat complex. For more detail, see Addendum A.I.	
CONST2	Failure of the EFWS due to failure of both electric pump trains or blockage of flow through both steam generator lines.	6.3E-4
	The expansion of event CONST2 into its contributory failures is somewhat complex. For more detail, see Addendum A.I.	
PCSNR	Failure to restore the PCS within 30 min. following a T_2 transient.	0.1
M	Interruption of the PCS	1
M1	Interruption of the PCS (with T_3 initiator)	.01
LOPNRE	Failure to restore offsite or onsite AC power within approximately 40 min. This power is needed to operate the high pressure injection system (HPIS).	0.2
A1	Failure of a pump discharge valve in the discharge line common to both backup pumps (A & B) of the HPIS.	.0098
	Event A1 occurs if either of the following fails:	

TABLE A.4 (contd)

Symbol	Description	Probability
	1. An NC MOV. It has a failure probability of .0096 with the contributory modes as shown in event D. 2. An NO ManV. It has a failure probability of .0002 with the contributory modes as shown in event D. The failure probability of event A1 is the sum of the above.	
B1	Failure of a component in the main line (containing pump C) of the HPIS downstream from the borated water storage tank (BWST) isolation valve. Event B1 occurs if any of the following fails: 1. Either of two NC MOVs. Each has a failure probability of .0096 with the contributory modes as shown in event D. 2. One of three NO ManVs. Each has a failure probability of .0002 with the contributory modes as shown in event D. 3. Either of two CVs. Each has a failure probability for hardware failure of .0001. 4. HPIS pump C. Its failure probability is the sum of the following contributory modes: hardware - .001 control circuitry - .0011 lube oil becoming viscous - .01 service water not valved in - .001 maintenance outage - .0021 test outage - .0019 .017	.035

TABLE A.4 (contd)

Symbol	Description	Probability
	The failure probability of event B1 is the sum of the above with a multiple maintenance outage probability of .0021 removed: 2(.0096) + 3(.0002) + 2(.0001) + .017 - .0021 = .035.	
	The factors two and three account for the contributions from multiple valves of the same type.	
C1	Failure of a pump suction valve in the suction line (downstream from the BWST isolation valve) common to both backup pumps (A & B) of the HPIS.	.0098
	The expansion of event C1 into its contributory failures is analogous to that for event A1.	
(B_3)	Failure of both emergency AC hydroelectric generators	5E-4
	Event (B_3) occurs if any of the following occurs: 1. Both emergency hydroelectric generators fail on demand (each has a failure probability of .006). 2. Either hydroelectric generator fails on demand while the other is down for maintenance (with probability of .0058). 3. Both emergency DC batteries needed for generator startup fail (this probability is dominated by a common-cause miscalibration error and has a value of 4E-4).	
	The failure probability for event (B_3) is the sum of the above: $(.006)^2 + 2(.006)(.0058) + 4E-4 = 5E-4$. The factor of two accounts for the contribution from both possible pairings of a generator demand failure with the other generator's maintenance outage.	
K	Failure of the reactor protection system due to primarily test and maintenance faults (88% contribution).	2.6E-5
	The expansion of event K into its contributory failures is somewhat complex. For more detail, see Addendum A.II.	

TABLE A.4 (contd)

Symbol	Description	Probability
$\overline{P}_1$	Pressurizer safety/relief valves demanded open	.01
	Failure of any pressurizer safety/relief valve to reclose	.05
F1	Failure of a pump in train B of the low pressure service water system (LPSWS). Event F1 occurs if either of the following fails: 1. A normally-operating centrifugal pump. 2. A normally-operating vacuum pump. The failure probability of each is that for failure to run over a 24-hr period at a failure rate of 3E-5/hr. This gives a failure probability of 7.2E-4 for each. The failure probability of event F1 is the sum of these.	.0014
G1	Failure of a pump in train A of the LPSWS. Event G1 occurs if either of the following fails: 1. A normally-idle centrifugal pump. 2. A normally-idle vacuum pump. The failure of probability of each is the sum of the following contributory modes: hardware - .001 control circuitry - .0018 test outage - .0019 maintenance outage - .0021 ——— .0068 The failure probability of event G1 is the sum of these.	.014

TABLE A.4 (contd)

Symbol	Description	Probability
V	Undetected failure of both check valves combined with opening of the NC MOV for quarterly testing, all in either train of the LPIS discharging to the core flood nozzles. The expansion of event V into its contributory failures is somewhat complex. For more detail, see Addendum A.III. Note that a recent procedural modification at Oconee no longer allows the NC MOVs in the LPIS trains to be opened during power operation. Thus, the probability of event V has been decreased from its calculated value of 7.3E-5 to that for the WASH-1400 PWR, 4.0E-6. For calculational purposes, this is equivalent to dividing the equation for P(V) in Addendum A.III by an additional factor of 18.	4.0E-6
HHMAN	Operator fails to manually start the high head auxiliary service water system. This system is a backup to the EFWS.	.1
HPMAN	Operator fails to start the HPIS	.015
HPMAN1	Operator fails to start the HPIS during an ATWS sequence (extremely high stress)	.1
LPISCM	Common-cause failure to reclose valves in test train of the LP/CSIS	.003
HPRSCM	Common-cause failure of the operator to align suction of the high pressure recirculation system to the discharge of the low pressure recirculation system.	.003
RCSRBCM	Common-cause miscalibration of the sensor/bistables which actuate the HPIS. The sensor groups are the RCS low pressure and the reactor building high pressure sensors in logic channels 1 through 4 of the ESPS.	3.2E-5

TABLE A.4 (contd)

Symbol	Description	Probability
WXCM	Common-cause failure of the operator to open both containment sump suction valves in the low pressure/ containment spray recirculation system (LP/CSRS) at the start of recirculation.	.003
D'E	Failure of both trains A & B of the LP/CSIS due to a failure of a pump discharge valve in each train. The failure probability for event D'E is slightly lower than the product of the individual events because any double contribution from the same maintenance outage has been removed. Since both events D & E have a total maintenance contribution of .0063 from the three MOVs, $D \cdot E = (.023)^2 - (.0063)^2 = 4.9E-4$.	4.9E-4
W'X	Failure of both containment sump suction valves in the LP/CSRS. W corresponds to the valve in train A (NC MOV), X to the valve in train B (NC MOV). Each event W & X corresponds to failure of an NC MOV with the contributory modes as discussed in event D. Thus, each event has a failure probability of .0096. However, for event W'X the double maintenance contribution has been removed as above for event D'E: $W \cdot X = (.0096)^2 - (.0021)^2 = 8.8E-5$.	8.8E-5
B'W	Failure of both a pump suction valve in train B of the LP/CSIS and the containment sump suction valve in train A of the LP/CSRS. The double maintenance contribution has been removed as follows: $B \cdot W = (.0033)(.0096) - (.0021)^2 = 2.7E-5$.	2.7E-5

TABLE A.4 (contd)

Symbol	Description	Probability
C·X	Failure of both a pump suction valve in train A of the LP/CSIS and the containment sump suction valve in train B of the LP/CSRS. The double maintenance contribution has been removed as for event B·W.	2.7E-5
D·X	Failure of both a pump discharge valve in train A of the LP/CSIS and the containment sump suction valve in train B of the LP/CSRS. The double maintenance contribution has been removed as follows: D·X = (.023)(.0096) - (.0063)(.0021) = 2.1E-4.	2.1E-4
E·W	Failure of both a pump discharge valve in train B of the LP/CSIS and the containment sump suction valve in train A of the LP/CSRS. The double maintenance contribution has been removed as for event D·X.	2.1E-4
B·D	Failure of both a pump suction valve in train B of the LP/CSIS and a pump discharge valve in train A of the LP/CSIS. The double maintenance contribution has been removed as follows: B·D = (.0033)(.023) - (.0021)(.0063) = 6.3E-5.	6.3E-5
E·C	Failure of both a pump discharge valve in train B of the LP/CSIS and a pump suction valve in train A of the LP/CSIS. The double maintenance contribution has been removed as for event B·D.	6.3E-5

ADDENDUM A.I

TABLE A.I-1. Boolean Expansion of Terms CONST1 & CONST2

The Boolean expansion of CONST1 is the sum of the following terms:

Terms	Probabilities
A3˙B3	1.4×10^{-4}
E3˙F3˙G3	1.7×10^{-5}
A3˙G3˙F3	1.6×10^{-5}
E3˙F3˙B3	1.6×10^{-5}
E3˙P3˙G3	5.8×10^{-6}
E3˙P3˙B3	5.3×10^{-6}
A3˙G3˙P3	5.3×10^{-6}
$\sum$ = CONST1 =	2.1×10^{-4}

The Boolean expansion of CONST2 is the sum of the following terms:

Terms	Probabilities
E3˙G3	1.7×10^{-4}
E3˙B3	1.6×10^{-4}
A3˙G3	1.6×10^{-4}
A3˙B3	1.4×10^{-4}
$\sum$ = CONST2 =	6.3×10^{-4}

Note: Double and triple maintenance contributions have been removed from these terms. The calculational procedure for this removal is discussed in Section 6.1.2.1 as part of example issue I.A.2.2.

TABLE A.I-2. Boolean Terms Comprising CONST1 & CONST2

Boolean Term	Term Definition	Term Unavailability
A3	FDW-232 + FDW-317 + FDW-315	1.3×10^{-2}
B3	FDW-233 + FDW-318 + FDW-316	1.3×10^{-2}
E3	C575 + EFP-A + FDW-373 + FDW-370 + FDW-372	1.7×10^{-2}
G3	C576 + EFP-B + FDW-383 + FDW-380 + FDW-382	1.7×10^{-2}
F3	EFP-TD + FDW-88 + C-157 + C-156 + LPSW-137	1.1×10^{-1}*
P3	MS-90 + MS-91 + MS-93 + MS-94 + MS-95 + MS-87	3.6×10^{-2}

*A multiple maintenance outage unavailability of .0058 is removed from the Boolean sum.

TABLE A.I-3. Component Failures Corresponding to Boolean Terms in CONST1 & CONST2

Component Description	Fault Identifiers	Failure Contributors	Q/Component
	FDW-232		
	FDW-317		
	FDW-233		
	FDW-318		
Check Valve	FDW-373		
	FDW-370		
	FDW-383	Hardware	1×10^{-4}
	FDW-380	Q Total	1×10^{-4}
	MS-91		
Electric Pump	EFP-A	Hardware	1×10^{-3}
	EFP-B	Control Circuitry	1.8×10^{-3}
		Maintenance	5.8×10^{-3}
		Fails to Run 24 hrs (3×10^{-5}/hr)	7.2×10^{-4}
		Q Total	9.3×10^{-3}
Air Operated Valve	FDW-315	Hardware	3×10^{-4}
(Normally Closed)	FDW-316	Control Circuitry	6.3×10^{-3}
	MS-93	Maintenance	5.8×10^{-3}
		Plugged	1×10^{-4}
		Q Total	1.3×10^{-2}

TABLE A.1-3. (contd)

Component Description	Fault Identifiers	Failure Contributors	Q/Component
Air Operated Valve (Normally Open)	MS-87	Operator Error	1×10^{-3}
		Plugged	1×10^{-4}
		Maintenance	5.8×10^{-3}
		Q Total	6.9×10^{-3}
Turbine governor Valve	MS-95	Plugged	1×10^{-4}
		DC oil pump fails	1×10^{-3}
		DC oil pump circuit	2×10^{-3}
		Maintenance	5.8×10^{-3}
		Q Total	8.9×10^{-3}
Turbine overspeed stop valve	MS-94	Plugged	1×10^{-4}
		Operator Error	1×10^{-3}
		Maintenance	5.8×10^{-3}
		Q Total	6.9×10^{-3}
Manual Valve	MS-90	Operator Error	1×10^{-4}
	C-575	Plugged	1×10^{-4}
	C-576		
	C-157		
		Q Total	2×10^{-4}
Manual Test Valve	FDW-88	Operator Error (leaves open after test)	1×10^{-3}
		Q Total	1×10^{-3}

TABLE A.I-3. (contd)

Component Description	Fault Identifier	Failure Contributors	Q/Component
Motor Operated	FDW-372	Plugged	1×10^{-4}
Valve	FDW-382	Operator Error	1×10^{-3}
(Normally Open)	C-156	Maintenance	5.8×10^{-3}
		Q Total	6.9×10^{-3}
Motor Operated	LPSW-137	Hardware	1×10^{-3}
Valve		Plugged	1×10^{-4}
(Normally Closed)		Control Circuitry	6.4×10^{-3}
		Maintenance	5.8×10^{-3}
		Q Total	1.3×10^{-2}
Turbine Pump	EFP-TD	Hardware[1]	9.1×10^{-2}
		Maintenance	5.8×10^{-3}
		Fails to Run 24 hrs (3×10^{-5}/hr)	7.2×10^{-4}
		Q Total	9.8×10^{-2}

[1]This unavailability is derived from plant test data for this pump taken from an April 25, 1979 letter from William O. Parker, Jr. (Duke Power) to Harold Denton (NRC).

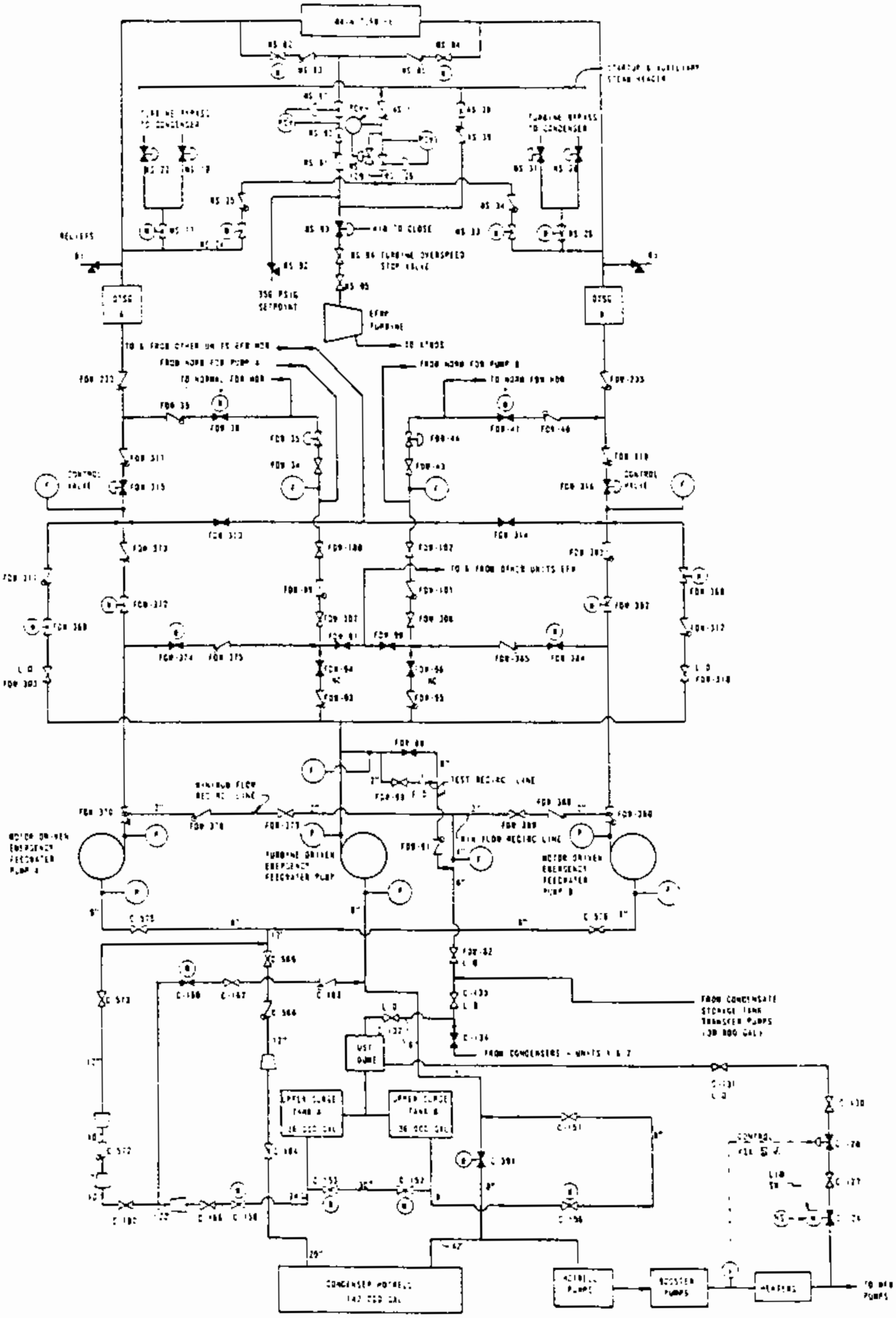

Figure A.I-1. Oconee Emergency Feedwater System

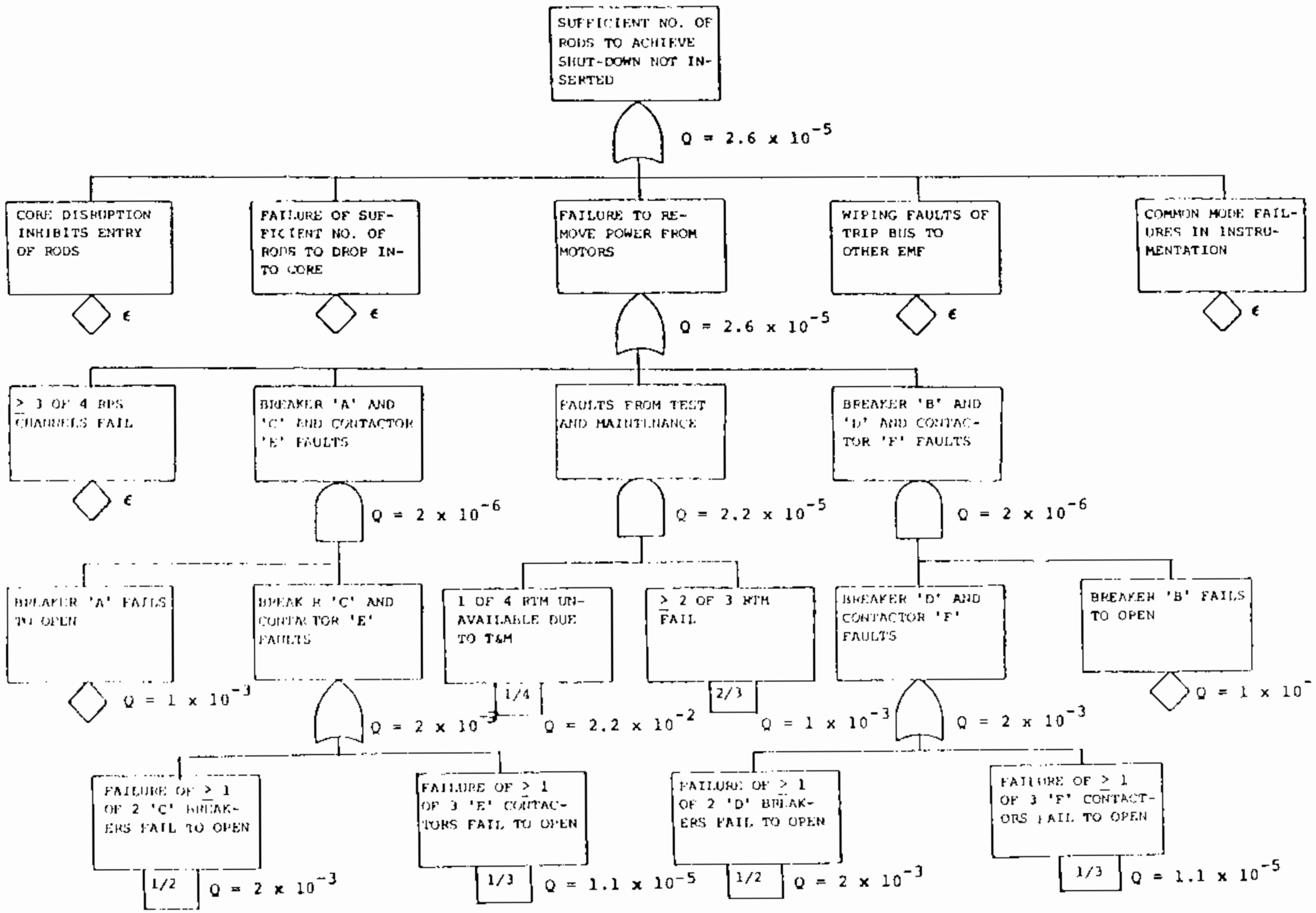

FIGURE A.II-1. Simplified Fault Tree for Oconee Reactor Protection System

ADDENDUM A.III

Determination of Oconee Interfacing Systems LOCA Failure Probability

Three failure modes have been identified for Oconee which result in the sequence V (valve failure), extra-containment LOCA:

A. Failure of two check valves and the isolation valve in either one of the two independent low pressure injection lines.

B. Failure of the one check valve, the manual valve and the isolation valve in the low pressure auxiliary spray cooling line.

C. Failure of the three isolation valves in the RCS hot leg low pressure suction line.

Failure modes A and C above will result in a large extra-containment LOCA because of the large pipe sizes. Failure modes A and C are also important because they preclude successful LPIS operation. Failure mode B will be constrained to a small extra-containment LOCA (S_1) by the 1-1/2 " diameter auxiliary spray cooling line.

The dominant failure combinations for the low pressure injection lines of the Oconee LPIS are described here. There are three valves which isolate the LPIS from the high RCS pressure. These include two check valves and a motor operated valve (normally closed). The three valves are arranged in series as shown in Figure III-1. The dominant failure mode for these three valves would be undetected failure of both check valves either by leakage or rupture, combined with opening of the motor operated valve for quarterly testing.

There are four possible failure mode combinations which dominate event V. For one train they are:

1) CF-14 CV Leaks; LP-48 CV Leaks; LP-17 MOV opened for Quarterly Test
2) CF-14 CV Leaks; LP-48 CV Ruptures; LP-17 MOV opened for Quarterly Test
3) CF-14 CV Ruptures; LP-48 CV Leaks; LP-17 MOV opened for Quarterly Test
4) CF-14 CV Ruptures; LP-48 CV Ruptures; LP-17 MOV opened for Quarterly Test

The analysis was based on the following assumptions:

1) The two check valves in each train (i.e., CF-14, LP-48) fail independently in time rather than sequentially in time as was done in the RSS. The reasoning behind this is that each check valve is pressurized by separate sources (i.e., CF-14 by the RCS, LP-48 by the core flooding tank).
2) Leak failures of concern are those caused by the failure of the check valves to reseat after a semi-annual flow test of the LPIS. These leaks are assumed to be large enough to fail the low pressure piping of the LPIS due to a subsequent water hammer if both check valves are subject to this failure and the MOV is opened. Other

smaller leaks, are not deemed to fail the LPIS since the associated flow rates and water hammer would not be severe enough to rupture the LPIS piping. The time of check valve reseat leak failure is therefore the LPIS flow test.

3) The following are the failure rates used in the analysis:

$$P\ (\text{Leak}) = \lambda_L = 3 \times 10^{-7}/\text{hr.}$$

$$P\ (\text{Rupture}) = \lambda_R = 1 \times 10^{-8}/\text{hr.}$$

The assumption is that these failure rates apply equally to the inboard and outboard check valves even though they are subject to a different pressure differential.

4) The check valve leak demand failure probability can be approximated by[1]:

$$Pd_L \approx P(\text{ Leak}) \times (\text{YBS})$$

where YBS is the time (4380 hours) between LPIS flow tests (or between shutdowns since this is when the LPIS is flow tested). The reason for this approach is that data does not exist for the reseat failure probability of a check valve.

5) The probability of sequence V per year can be estimated by calculating the probability per year of sequence V based on a 5 year average (this approach was also taken in the RSS). The reason for using this approach is that

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[1]See "PWR sensitivity to Alterations in the Interfacing System LOCA," EPRI NP-262, September 1976, pg. 6.

there appears to be no procedure for testing the integrity of the check valves.

The failure probability estimate for each of the four possible failure modes will be discussed separately. These estimates will then be combined to yield the final assessed probability of the Oconee interfacing system LOCA.

1) CF-14 CV Leaks; LP-48 CV Leaks; LP-17 MOV Opened for Quarterly Test

An estimate of the 5 year failure probability for this failure mode can be given as:

$$P(\text{Leak-Leak}) = [10(Pd_{L_{CF-14}})]^* \; [10(Pd_{L_{LP-48}})]$$

$$= 1.7 \times 10^{-4}$$

The factors of 10 originate from the fact that there are 10 LPIS flow tests in a 5 year period and therefore 10 opportunities for each check valve to fail to reseat.

It should be noted that in the RSS V assessment for the Surry plant that leak-leak failures were not considered. This is because early detection of this failure mode was possible during RCS heat up due to the fact that the MOV was in the normally open position and this failure would have been sensed by instruments in the control room.

2) CF-14 CV Leaks; LP-48 CV Ruptures; LP-17 MOV Opened for Quarterly Test

An estimate of the 5 year failure probability for this failure mode can be given as

$$P(\text{Leak} - \text{Rupture}) \approx [10(Pd_{L_{CF-14}})]*[\lambda_{R_{LP-48}} \tau_5]$$

$$= 5.8 \times 10^{-6}$$

where

τ_5 = Time of 5 years or 43800 hours.

3) <u>CF-14 CV Ruptures; LP-48 CV Leaks; LP-17 MOV Opened for Quarterly Test</u>

An estimate of the 5 year failure probability is the same as for the leak-rupture. Therefore:

$P(\text{Rupture-Leak}) = 5.8 \times 10^{-6}$.

4) <u>CF-14 CV Ruptures; LP-48 CV Ruptures; LP-17 MOV Opened for Quarterly Test</u>

An estimate of the 5 year failure probability is:

$$P(\text{Rupture} - \text{Rupture}) \approx \left[\lambda_{R_{CF-14}} \tau_5\right] \bullet \left[\lambda_{R_{LP-47}} \tau_5\right]$$

$$= 1.9 \times 10^{-7}$$

The final assessment of the probability of event V is found by summing the above failure mode probability estimates, multiplying the sum by 2 because there are two MOV-check valve trains, and dividing the sum by 5 to yield a per year estimate. This can be stated in equation form as:

$$P(V) \approx \frac{2}{5}\left[P(L - L) + P(L - R) + P(R - L) + P(R - R)\right]$$

$$= 7.3 \times 10^{-5}/\text{reactor year}^{*}.$$

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* To account for a procedural modification which no longer allows the MOV to be opened during power operation, this probability should be divided by a factor of 18 based on P(V) for the WASH-1400 PWR.

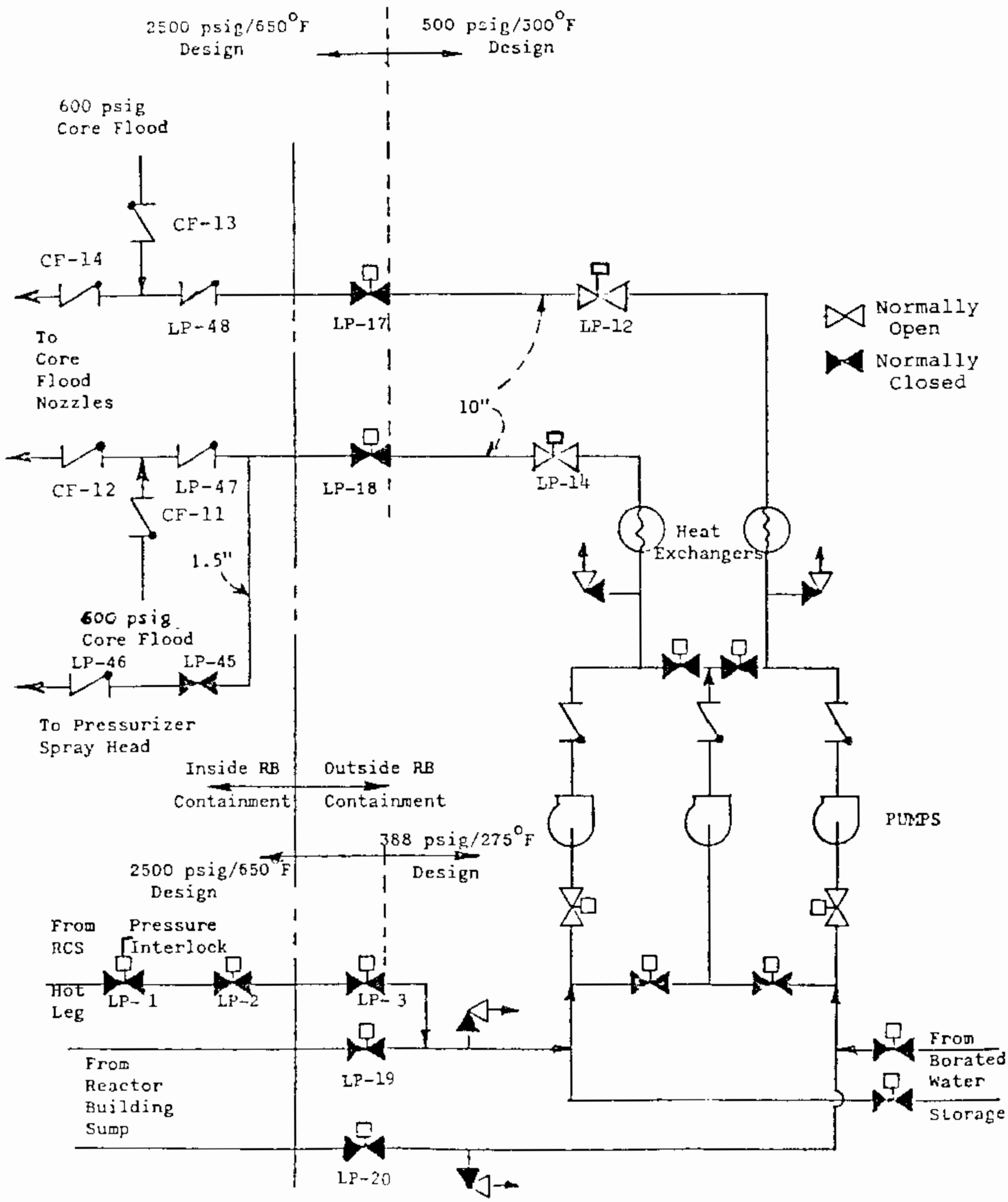

FIGURE A.III-1. Oconee #3 Low Pressure Injection System

REFERENCE

Kolb, G. et al. 1981. Reactor Safety Study Methodology Applications Program: Oconee #3 PWR Power Plant, NUREG/CR-1659/2 (Revision 1). Sandia National Laboratories, Albuquerque, New Mexico.

APPENDIX B

RISK PARAMETERS FOR GRAND GULF 1 BWR

The risk equation for Grand Gulf 1 (General Electric BWR/6 with Mark III containment) has been summarized for the dominant accident sequences contributing to the four BWR core-melt release categories as defined in WASH-1400. The Grand Gulf results have been extracted from its RSSMAP study, NUREG/CR-1659/4 (Hatch 1981) and are provided here in Tables B.1 through B.4, with Addendum B.1. The information is presented so as to be compatible with the technique described in Section 3.D for estimating the risk reduction.

Tables B.1 through B.4 are analogous with their counterparts in Appendix A of this report. The introductory comments given in Appendix A of this report are applicable here. For additional detail and information, the analyst is referred to the Grand Gulf RSSMAP report (Hatch 1981).

TABLE B.1. Grand Gulf Dominant Accident Sequences and Frequencies (Reactor-Year^{-1})

Accident Sequence	BWR Release Category (based on WASH-1400) 1	2	3	4
T_1PQI	α 1.6E-8	δ 1.6E-6		
$T_{23}PQI$	α 3.7E-8	δ 3.7E-6		
T_1PQE			γ 1.2E-7	δ 1.2E-7
$T_{23}PQE$			γ 2.7E-7	δ 2.7E-7
SI	α 4.6E-8	δ 4.6E-6		
T_1QW		δ 6.2E-6		
$T_{23}QW$		δ 1.2E-5		
$T_{23}C$		δ 5.4E-6		
T_1QUV			γ 9.5E-7	δ 9.5E-7
Non-Dominant	1E-8		1E-7	3E-7
Total	1.1E-7	3.4E-5	1.4E-6	1.6E-6

TABLE B.2. Symbols Used in Table B.1

Initiating Events

T_1 - A loss of offsite power transient.

T_{23} - Any other transient which requires an emergency reactor shutdown.

S - A small LOCA (the break area is less than one square foot).

System, Component, or Functional Failures

C - Failure to render the reactor subcritical.

E - Failure of the Emergency Core Cooling System.

I - Failure of residual heat removal systems after a LOCA (including transient induced LOCAs).

P - Failure of a safety/relief valve to reseat.

Q - Failure of the Power Conversion System.

U - Failure of the High Pressure Core Spray and Reactor Core Isolation Cooling System.

V - Failure of the low pressure ECCS systems to provide core flow.

W - Failure of the residual heat removal systems after a transient.

Containment Failure Modes[a]

α - Containment failure due to a steam explosion.

γ - Containment failure due to an overpressure caused by rapid hydrogen burning.

δ - Containment failure due to an overpressure caused by gas generation.

(a) The symbols used for the Grand Gulf containment failure modes are somewhat different from those used in the RSS.

TABLE B.3. Dominant Minimal Cut Sets of Grand Gulf Dominant Accident Sequences

Accident Sequence	Seq. Freq. (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_1PQI	1.6E-6	α	.01	1	T_1*P*LOPNRE*LOPNRL*DIESEL1*DIESEL2*RECOVERY	1.2×10^{-7}
					T_1*P*LOPNRE*LOPNRL*VGA2*DIESEL2*RECOVERY	7.9×10^{-8}
(contribution from non-dominant minimal cut sets = 1E-7)		δ	1.0	2	T_1*P*LOPNRE*LOPNRL*VGB2*DIESEL1*RECOVERY	7.9×10^{-8}
					T_1*P*LOPNRE*LOPNRL*DIESEL1*SSB*RECOVERY	7.0×10^{-8}
					T_1*P*LOPNRE*LOPNRL*DIESEL2*SSA*RECOVERY	7.0×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGA2*VGB2*RECOVERY	5.3×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGA1*DIESEL2*RECOVERY	5.0×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGB1*DIESEL1*RECOVERY	5.0×10^{-8}
					T_1*P*LOPNRE*LOPNRL*SB*DIESEL1*RECOVERY	4.6×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGA2*SSB*RECOVERY	4.6×10^{-8}
					T_1*P*LOPNRE*LOPNRL*LA2*DIESEL2*RECOVERY	4.6×10^{-8}
					T_1*P*LOPNRE*LOPNRL*SA*DIESEL2*RECOVERY	4.6×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGB2*SSA*RECOVERY	4.6×10^{-8}
					T_1*P*LOPNRE*LOPNRL*LB2*DIESEL1*RECOVERY	4.6×10^{-8}
					T_1*P*LOPNRE*LOPNRL*SSA*SSB*RECOVERY	4.1×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGA1*VGB2*RECOVERY	3.3×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGA2*VGB1*RECOVERY	3.3×10^{-8}
					T_1*P*LOPNRE*LOPNRL*LA2*VGB2*RECOVERY	3.1×10^{-8}
					T_1*P*LOPNRE*LOPNRL*LB2*VGA2*RECOVERY	3.1×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGA1*SSB*RECOVERY	2.9×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGB1*SSA*RECOVERY	2.9×10^{-8}
					T_1*P*LOPNRE*LOPNRL*LA2*SSB*RECOVERY	2.7×10^{-8}
					T_1*P*LOPNRE*LOPNRL*SA*SSB*RECOVERY	2.7×10^{-8}
					T_1*P*LOPNRE*LOPNRL*SB*SSA*RECOVERY	2.7×10^{-8}
					T_1*P*LOPNRE*LOPNRL*LB2*SSA*RECOVERY	2.7×10^{-8}
					T_1*P*LOPNRE*LOPNRL*DIESEL1*V2*RECOVERY	2.6×10^{-8}
					T_1*P*LOPNRE*LOPNRL*DIESEL2*V1*RECOVERY	2.6×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGA1*VGB1*RECOVERY	2.1×10^{-8}
					T_1*P*LOPNRE*LOPNRL*LA2*VGB1*RECOVERY	1.9×10^{-8}
					T_1*P*LOPNRE*LOPNRL*LB2*VGA1*RECOVERY	1.9×10^{-8}
					T_1*P*LOPNRE*LOPNRL*SA*SB*RECOVERY	1.8×10^{-8}
					T_1*P*LOPNRE*LOPNRL*LA2*LB2*RECOVERY	1.8×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGA2*V2*RECOVERY	1.8×10^{-8}

TABLE B.3. (contd)

Accident Sequence	Seq. Freq. (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_1PQI (Cont.)					T_1*P*LOPNRE*LOPNRL*VGB2*V1*RECOVERY	1.8×10^{-8}
					T_1*P*LOPNRE*LOPNRL*SSA*V2*RECOVERY	1.5×10^{-8}
					T_1*P*LOPNRE*LOPNRL*SSB*V1*RECOVERY	1.5×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGA1*V2*RECOVERY	1.1×10^{-8}
					T_1*P*LOPNRE*LOPNRL*VGB1*V1*RECOVERY	1.1×10^{-8}
					T_1*P*LOPNRE*LOPNRL*LA2*V2*RECOVERY	1.0×10^{-8}
					T_1*P*LOPNRE*LOPNRL*SA*V2*RECOVERY	1.0×10^{-8}
					T_1*P*LOPNRE*LOPNRL*SB*V1*RECOVERY	1.0×10^{-8}
					T_1*P*LOPNRE*LOPNRL*LB2*V1*RECOVERY	1.0×10^{-8}
					T_1*P*LOPNRE*LOPNRL*V1*V2*RECOVERY	5.9×10^{-9}
T_{23}PQI*	3.7E-6	α	.01	1	T_{23}*P*Q1*VGA2*VGB2*RECOVERY	5.0×10^{-7}
		δ	1.0	2	T_{23}*P*Q1*VGB2*SSA*RECOVERY	3.4×10^{-7}
					T_{23}*P*Q1*VGA2*SSB*RECOVERY	3.4×10^{-7}
					T_{23}*P*Q1*VGA2*VGB1*RECOVERY	2.5×10^{-7}
					T_{23}*P*Q1*VGA1*VGB2*RECOVERY	2.5×10^{-7}
					T_{23}*P*Q1*VGA2*LB2*RECOVERY	2.3×10^{-7}
					T_{23}*P*Q1*VGB2*LA2*RECOVERY	2.3×10^{-7}
					T_{23}*P*Q1*SSA*SSB*RECOVERY	1.6×10^{-7}
					T_{23}*P*Q1*VGB1*SSA*RECOVERY	1.3×10^{-7}
					T_{23}*P*Q1*VGA1*SSB*RECOVERY	1.3×10^{-7}
					T_{23}*P*Q1*LB2*SSA*RECOVERY	1.0×10^{-7}
					T_{23}*P*Q1*LA2*SSB*RECOVERY	1.0×10^{-7}
					T_{23}*P*Q1*VGA1*VGB1*RECOVERY	1.0×10^{-7}
					T_{23}*P*Q1*VGA1*LB2*RECOVERY	8.5×10^{-8}
					T_{23}*P*Q1*VGB1*LA2*RECOVERY	8.5×10^{-8}
					T_{23}*P*Q1*SA*SB*RECOVERY	6.9×10^{-8}
					T_{23}*P*Q1*LA2*LB2*RECOVERY	6.9×10^{-8}
					T_{23}*P*Q1*VGA2*SBC*RECOVERY	3.2×10^{-8}
					T_{23}*P*Q1*VGA2*BCACT*RECOVERY	3.2×10^{-8}
					T_{23}*P*Q1*VGB2*SAC*RECOVERY	3.2×10^{-8}
					T_{23}*P*Q1*VGB2*LRACT*RECOVERY	3.2×10^{-8}

*See Addendum B.I

TABLE B.3. (contd)

Accident Sequence	Seq. Freq (ry^{-1})	Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_{23}PQI					T_{23}*P*Q1 *SAC*SSB*RECOVERY	2.8×10^{-8}
(Cont.)					T_{23}*P*Q1 *SSB*LRACT*RECOVERY	2.8×10^{-8}
					T_{23}*P*Q1 *SBC*SSA*RECOVERY	2.8×10^{-8}
					T_{23}*P*Q1 *SSA*BCACT*RECOVERY	2.8×10^{-8}
					T_{23}*P*Q1 *PA27*VGB2*RECOVERY	2.2×10^{-8}
					T_{23}*P*Q1 *PB27*VGA2*RECOVERY	2.2×10^{-8}
					T_{23}*P*Q1 *VGA1*BCACT*RECOVERY	2.0×10^{-8}
					T_{23}*P*Q1 *VGA1*SBC*RECOVERY	2.0×10^{-8}
					T_{23}*P*Q1 *LRACT*VGB1*RECOVERY	2.0×10^{-8}
					T_{23}*P*Q1 *VGB1*SAC*RECOVERY	2.0×10^{-8}
					T_{23}*P*Q1 *PA27*SSB*RECOVERY	1.9×10^{-8}
					T_{23}*P*Q1 *PB27*SSA*RECOVERY	1.9×10^{-8}
					T_{23}*P*Q1 *SAACC*SB*RECOVERY	1.9×10^{-8}
					T_{23}*P*Q1 *LRACT*LB2*RECOVERY	1.9×10^{-8}
					T_{23}*P*Q1 *LA2*SBC*RECOVERY	1.9×10^{-8}
					T_{23}*P*Q1 *LB2*SAC*RECOVERY	1.9×10^{-8}
					T_{23}*P*Q1 *LA2*BCACT*RECOVERY	1.9×10^{-8}
					T_{23}*P*Q1 *SA*SBACC*RECOVERY	1.9×10^{-8}
T_1PQE*	2.3E-7	γ	.5	3	T_1*P*Q*OP*H*R	1.1×10^{-8}
(contribution from non-dominant minimal cut sets = 4E-8)		δ	.5	4	T_1*P*Q*OP*LOPNRE*DIESEL3*R	1.1×10^{-8}
					T_1*P*Q*LOPNRE*DIESEL1*DIESEL2*DIESEL3*R	9.5×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*DIESEL3*R*LC	5.8×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL2*DIESEL3*L*R	5.6×10^{-9}
					T_1*P*Q*LOPNRE*SSA*DIESEL2*DIESEL3*R	5.6×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*DIESEL2*H*R	5.6×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*SSB*DIESEL3*R	5.6×10^{-9}
					T_1*P*Q*LOPNRE*BATA*DIESEL2*DIESEL3	5.2×10^{-9}
					T_1*P*Q*OP*LOPNRE*SSC*R	4.3×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*DIESEL2*SSC*R	3.7×10^{-9}
					T_1*P*Q*LOPNRE*LB2*DIESEL1*DIESEL3*R	3.7×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*DIESEL3*R*LB1	3.4×10^{-9}
					T_1*P*Q*LOPNRE*SSA*DIESEL3*R*LC	3.4×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*H*R*LC	3.4×10^{-9}

*See Addendum B.I

TABLE B.3. (contd)

Accident Sequence	Seq. Freq. (ry^{-1})	Cont. Fail Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_1PQE					T_1*P*Q*LOPNRE*SSB*DIESEL3*L*R	3.2×10^{-9}
(Cont.)					T_1*P*Q*LOPNRE*L*H*R*DIESEL2	3.2×10^{-9}
					T_1*P*Q*LOPNRE*SSA*DIESEL2*H*R	3.2×10^{-9}
					T_1*P*Q*LOPNRE*SSA*SSB*DIESEL3*R	3.2×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*SSB*H*R	3.2×10^{-9}
					T_1*P*Q*BATA*LOPNRE*DIESEL3*LC	3.2×10^{-9}
					T_1*P*Q*BATA*LOPNRE*DIESEL2*H	3.0×10^{-9}
					T_1*P*Q*BATA*LOPNRE*SSB*DIESEL3	3.0×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*SSC*R*LC	2.3×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL2*SSC*L*R	2.2×10^{-9}
					T_1*P*Q*LOPNRE*SSA*DIESEL2*SSC*R	2.2×10^{-9}
					T_1*P*Q*LOPNRE*LB2*SSA*DIESEL3*R	2.2×10^{-9}
					T_1*P*Q*LOPNRE*LB2*DIESEL1*H*R	2.2×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*SSB*SSC*R	2.2×10^{-9}
					T_1*P*Q*LOPNRE*V1*DIESEL2*DIESEL3*R	2.1×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*V2*DIESEL3*R	2.1×10^{-9}
					T_1*P*Q*BATA*LOPNRE*DIESEL2*SSC	2.0×10^{-9}
					T_1*P*Q*BATA*LOPNRE*LB2*DIESEL3	2.0×10^{-9}
					T_1*P*Q*LOPNRE*SSA*DIESEL3*R*LB1	2.0×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*H*R*LB1	2.0×10^{-9}
					T_1*P*Q*LOPNRE*SSA*H*R*LC	2.0×10^{-9}
					T_1*P*Q*LOPNRE*SSB*L*H*R	1.9×10^{-9}
					T_1*P*Q*LOPNRE*SSA*SSB*H*R	1.9×10^{-9}
					T_1*P*Q*BATA*LOPNRE*DIESEL3*LB1	1.9×10^{-9}
					T_1*P*Q*OP*HACT*R	1.8×10^{-9}
					T_1*P*Q*BATA*LOPNRE*H*LC	1.8×10^{-9}
					T_1*P*Q*BATA*LOPNRE*SSB*H	1.8×10^{-9}
					T_1*P*Q*LOPNRE*LB2*DIESEL1*SSC*R	1.4×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*SSC*R*LB1	1.3×10^{-9}
					T_1*P*Q*LOPNRE*SSA*SSC*R*LC	1.3×10^{-9}
					T_1*P*Q*LOPNRE*V1*DIESEL3*R*LC	1.3×10^{-9}
					T_1*P*Q*LOPNRE*SSB*SSC*L*R	1.3×10^{-9}
					T_1*P*Q*LOPNRE*LB2*SSA*H*R	1.3×10^{-9}
					T_1*P*Q*LOPNRE*SSA*SSB*SSC*R	1.3×10^{-9}
					T_1*P*Q*LOPNRE*V2*DIESEL3*L*R	1.2×10^{-9}
					T_1*P*Q*LOPNRE*SSA*V2*DIESEL3*R	1.2×10^{-9}

TABLE B.3. (contd)

Accident Sequence	Seq. Freq. (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_1PQE					T_1*P*Q*LOPNRE*V1*DIESEL2*H*R	1.2×10^{-9}
					T_1*P*Q*LOPNRE*V1*SSB*DIESEL3*R	1.2×10^{-9}
(cont.)					T_1*P*Q*LOPNRE*DIESEL1*V2*H*R	1.2×10^{-9}
					T_1*P*Q*BATA*LOPNRE*SSC*LC	1.2×10^{-9}
					T_1*P*Q*BATA*LOPNRE*SSB*SSC	1.2×10^{-9}
					T_1*P*Q*BATA*LOPNRE*LB2*H	1.2×10^{-9}
					T_1*P*Q*LOPNRE*SSA*H*R*LB1	1.2×10^{-9}
					T_1*P*Q*BATA*LOPNRE*V2*DIESEL3	1.2×10^{-9}
					T_1*P*Q*BATA*LOPNRE*H*LB1	1.1×10^{-9}
					T_1*P*Q*LOPNRE*DIESEL1*DIESEL2*V3*R	8.7×10^{-10}
					T_1*P*Q*LOPNRE*LB2*SSA*SSC*R	8.4×10^{-10}
					T_1*P*Q*LOPNRE*V1*DIESEL2*SSC*R	8.2×10^{-10}
					T_1*P*Q*LOPNRE*LB2*V1*DIESEL3*R	8.2×10^{-10}
					T_1*P*Q*LOPNRE*DIESEL1*V2*SSC*R	8.2×10^{-10}
					T_1*P*Q*BATA*LOPNRE*LB2*SSC	7.8×10^{-10}
					T_1*P*Q*LOPNRE*SSA*SSC*R*LB1	7.8×10^{-10}
					T_1*P*Q*LOPNRE*V1*DIESEL3*R*LB1	7.6×10^{-10}
					T_1*P*Q*LOPNRE*V1*H*R*LC	7.5×10^{-10}
					T_1*P*Q*BATA*LOPNRE*SSC*LB1	7.3×10^{-10}
					T_1*P*Q*LOPNRE*V2*L*H*R	7.2×10^{-10}
					T_1*P*Q*LOPNRE*SSA*V2*H*R	7.2×10^{-10}
					T_1*P*Q*LOPNRE*V1*SSB*H*R	7.2×10^{-10}
					T_1*P*Q*BATA*LOPNRE*V2*H	6.7×10^{-10}
					T_1*P*Q*LRACT*H*R*LC	5.7×10^{-10}
					T_1*P*Q*BCACT*L*H*R	5.4×10^{-10}
					T_1*P*Q*LOPNRE*DIESEL1*V3*R*LC	5.3×10^{-10}
					T_1*P*Q*LOPNRE*DIESEL2*V3*L*R	5.1×10^{-10}
					T_1*P*Q*LOPNRE*SSA*DIESEL2*V3*R	5.1×10^{-10}
					T_1*P*Q*LOPNRE*DIESEL1*SSB*V3*R	5.1×10^{-10}
					T_1*P*Q*LOPNRE*V1*SSC*R*LC	5.0×10^{-10}
					T_1*P*Q*LOPNRE*V2*SSC*L*R	4.8×10^{-10}
					T_1*P*Q*LOPNRE*SSA*V2*SSC*R	4.8×10^{-10}
					T_1*P*Q*LOPNRE*LB2*V1*H*R	4.8×10^{-10}
					T_1*P*Q*LOPNRE*V1*SSB*SSC*R	4.8×10^{-10}
					T_1*P*Q*BATA*LOPNRE*DIESEL2*V3	4.8×10^{-10}

TABLE B.3. (contd)

Accident Sequence	Seq. Freq. (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_{23}PQE*	5.4E-7	γ	.5	3	T_{23}*P*Q*OP*R*H	3.8×10^{-7}
(contribution from		δ	.5	4	T_{23}*P*Q*OP*R*HACT	6.4×10^{-8}
non-dominant					T_{23}*P*Q*OP*RACT*H	2.6×10^{-8}
minimal cut					T_{23}*P*Q*R*LRACT*H*LC	2.0×10^{-8}
sets = 1E-8)					T_{23}*P*Q*R*BCACT*L*H	1.9×10^{-8}
					T_{23}*P*Q*R*LRACT*LB2*H	1.3×10^{-8}
					T_{23}*P*Q*R*LRACT*H*LB1	1.2×10^{-8}
SI*	4.6E-6	α	.01	1	S*VGA2*VGB2	6.2×10^{-7}
		δ	1.0	2	S*VGB2*SSA	4.2×10^{-7}
					S*VGA2*SSB	4.2×10^{-7}
					S*VGA2*VGB1	3.2×10^{-7}
					S*VGA1*VGB2	3.2×10^{-7}
					S*VGA2*LB2	2.8×10^{-7}
					S*LA2*VGB2	2.8×10^{-7}
					S*SSA*SSB	1.9×10^{-7}
					S*VGB1*SSA	1.6×10^{-7}
					S*VGA1*SSB	1.6×10^{-7}
					S*LB2*SSA	1.3×10^{-7}
					S*LA2*SSB	1.3×10^{-7}
					S*VGA1*VGB1	1.3×10^{-7}
					S*VGA1*LB2	1.1×10^{-7}
					S*LA2*VGB1	1.1×10^{-7}
					S*SA*SB	8.6×10^{-8}
					S*LA2*LB2	8.6×10^{-8}
					S*VGA2*SBC	4.0×10^{-8}
					S*VGA2*BCACT	4.0×10^{-8}
					S*VGB2*SAC	4.0×10^{-8}
					S*VGB2*LRACT	4.0×10^{-8}
					S*SAC*SSB	3.5×10^{-8}
					S*LRACT*SSB	3.5×10^{-8}
					S*SSA*SBC	3.5×10^{-8}
					S*BCACT*SSA	3.5×10^{-8}
					S*PA27*VGB2	2.7×10^{-8}
					S*VGA2*PB27	2.7×10^{-8}
					S*VGA1*BCACT	2.5×10^{-8}
					S*VGA1*SBC	2.5×10^{-8}
					S*LRACT*VGB1	2.5×10^{-8}
					S*VGB1*SAC	2.5×10^{-8}

*See Addendum B.I

TABLE B.3. (contd)

Accident Sequence	Seq. Freq. (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
SI (Cont.)					S*PA27*SSB	2.4×10^{-8}
					S*PB27*SSA	2.4×10^{-8}
					S*SAACC*SB	2.4×10^{-8}
					S*LRACT*LB2	2.4×10^{-8}
					S*LA2*SBC	2.4×10^{-8}
					S*LB2*SAC	2.4×10^{-8}
					S*LA2*BCACT	2.4×10^{-8}
					S*SA*SBACC	2.4×10^{-8}
T_1QW	6.2E-6	6	1.0	2	T_1*LOPNRE*LOPNRL*DIESEL1*DIESEL2*RECOVERY1	1.1×10^{-6}
(contribution from non-dominant minimal cut sets = 5E-7)					T_1*LOPNRE*LOPNRL*SSA*DIESEL2*RECOVERY1	6.4×10^{-7}
					T_1*LOPNRE*LOPNRL*DIESEL1*SSB*RECOVERY1	6.4×10^{-7}
					T_1*LOPNRE*LOPNRL*VGB1*DIESEL1*RECOVERY1	4.5×10^{-7}
					T_1*LOPNRE*LOPNRL*VGA1*DIESEL2*RECOVERY1	4.5×10^{-7}
					T_1*LOPNRE*LOPNRL*SSA*SSB*RECOVERY1	3.7×10^{-7}
					T_1*LOPNRE*LOPNRL*VGB1*SSA*RECOVERY1	2.6×10^{-7}
					T_1*LOPNRE*LOPNRL*VGA1*SSB*RECOVERY1	2.6×10^{-7}
					T_1*LOPNRE*LOPNRL*V1*DIESEL2*RECOVERY1	2.4×10^{-7}
					T_1*LOPNRE*LOPNRL*DIESEL1*V2*RECOVERY1	2.4×10^{-7}
					T_1*LOPNRE*LOPNRL*VGA1*VGB1*RECOVERY1	1.9×10^{-7}
					T_1*LOPNRE*LOPNRL*SSA*V2*RECOVERY1	1.4×10^{-7}
					T_1*LOPNRE*LOPNRL*V1*SSB*RECOVERY1	1.4×10^{-7}
					T_1*LOPNRE*LOPNRL*VGB1*V1*RECOVERY1	1.0×10^{-7}
					T_1*LOPNRE*LOPNRL*VGA1*V2*RECOVERY1	1.0×10^{-7}
					T_1*LOPNRE*LOPNRL*V1*V2*RECOVERY1	5.4×10^{-8}
					T_1*LOPNRE*LOPNRL*VGA2*DIESEL2*R*RECOVERY1	3.7×10^{-8}
					T_1*LOPNRE*LOPNRL*VGB2*DIESEL1*R*RECOVERY1	3.7×10^{-8}
					T_1*LOPNRE*LOPNRL*SAC*DIESEL2*RECOVERY1	3.6×10^{-8}
					T_1*LOPNRE*LOPNRL*DIESEL1*SBC*RECOVERY1	3.6×10^{-8}
					T_1*LOPNRE*LOPNRL*BATB*DIESEL1*RECOVERY1	3.0×10^{-8}
					T_1*LOPNRE*LOPNRL*BATA*DIESEL2*RECOVERY1	3.0×10^{-8}
					T_1*LOPNRE*LOPNRL*VGA2*VGB2*R*RECOVERY1	2.5×10^{-8}
					T_1*LOPNRE*LOPNRL*VGB2*SCVB*DIESEL1*RECOVERY1	2.3×10^{-8}
					T_1*LOPNRE*LOPNRL*VGA2*SCVA*DIESEL2*RECOVERY1	2.3×10^{-8}

TABLE B.3. (contd)

Accident Sequence	Seq. Freq. (ry^{-1})	Cont. Fail Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
$T_{23}QW$* (contribution from non-dominant minimal cut sets = 1E-6)	1.2E-5	δ	1.0	2	T_{23}*Q1*SSA*SSB*RECOVERY1	3.2×10^{-6}
					T_{23}*Q1*VGB1*SSA*RECOVERY1	1.9×10^{-6}
					T_{23}*Q1*VGA1*SSB*RECOVERY1	1.9×10^{-6}
					T_{23}*Q1*VGA1*VGB1*RECOVERY1	9.4×10^{-7}
					T_{23}*Q1*VGA2*VGB2*R*RECOVERY1	3.0×10^{-7}
					T_{23}*Q1*VGA2*SSB*R*RECOVERY1	2.6×10^{-7}
					T_{23}*Q1*VGB2*SSA*R*RECOVERY1	2.6×10^{-7}
					T_{23}*Q1*SSA*SBC*RECOVERY1	2.6×10^{-7}
					T_{23}*Q1*SAC*SSB*RECOVERY1	2.6×10^{-7}
					T_{23}*Q1*VGA2*VGB1*R*RECOVERY1	1.9×10^{-7}
					T_{23}*Q1*VGA1*VGB2*R*RECOVERY1	1.9×10^{-7}
					T_{23}*Q1*VGB1*SAC*RECOVERY1	1.9×10^{-7}
					T_{23}*Q1*VGA1*SBC*RECOVERY1	1.9×10^{-7}
					T_{23}*Q1*LA2*VGB2*R*RECOVERY1	1.8×10^{-7}
					T_{23}*Q1*VGA2*LB2*R*RECOVERY1	1.8×10^{-7}
					T_{23}*Q1*VGB2*SCVB*SSA*RECOVERY1	1.7×10^{-7}
					T_{23}*Q1*VGA2*SCVA*SSB*RECOVERY1	1.7×10^{-7}
					T_{23}*Q1*LA2*SSB*R*RECOVERY1	1.5×10^{-7}
					T_{23}*Q1*LB2*SSA*R*RECOVERY1	1.5×10^{-7}
$T_{23}C$	5.4E-6	δ	1.0	2	T_{23}*C	5.4×10^{-6}
T_1QUV	1.9E-6	γ	.5	3	T_1*LOPNRE*OP*R*DIESEL3	1.1×10^{-7}
					T_1*LOPNRE*R*DIESEL1*DIESEL2*DIESEL3	9.5×10^{-8}
		δ	.5	4	T_1*LOPNRE*OP*R*H	6.4×10^{-8}
					T_1*LOPNRE*R*DIESEL1*DIESEL3*LC	5.8×10^{-8}
					T_1*LOPNRE*R*SSA*DIESEL2*DIESEL3	5.6×10^{-8}
					T_1*LOPNRE*R*DIESEL1*SSB*DIESEL3	5.6×10^{-8}
					T_1*LOPNRE*R*DIESEL1*DIESEL2*H	5.6×10^{-8}
					T_1*LOPNRE*R*DIESEL2*DIESEL3*L	5.6×10^{-8}
					T_1*LOPNRE*BATA*DIESEL2*DIESEL3	5.2×10^{-8}
					T_1*LOPNRE*OP*R*SSC	4.3×10^{-8}
					T_1*LOPNRE*R*DIESEL1*DIESEL2*SSC	3.7×10^{-8}
					T_1*LOPNRE*R*LB2*DIESEL1*DIESEL3	3.7×10^{-8}
					T_1*LOPNRE*R*LB1*DIESEL1*DIESEL3	3.4×10^{-8}
					T_1*LOPNRE*R*SSA*DIESEL3*LC	3.4×10^{-8}
					T_1*LOPNRE*R*DIESEL1*H*LC	3.4×10^{-8}
					T_1*LOPNRE*R*SSA*SSB*DIESEL3	3.2×10^{-8}

*See Addendum B.I

TABLE B.3. (contd)

Accident Sequence	Seq. Freq. (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_1QUV					T_1*LOPNRE*R*SSA*DIESEL2*H	3.2×10^{-8}
(Cont.)					T_1*LOPNRE*R*SSB*DIESEL1*H	3.2×10^{-8}
					T_1*LOPNRE*R*SSB*DIESEL3*L	3.2×10^{-8}
					T_1*LOPNRE*R*DIESEL2*L*H	3.2×10^{-8}
					T_1*LOPNRE*BATA*DIESEL3*LC	3.2×10^{-8}
					T_1*LOPNRE*BATA*SSB*DIESEL3	3.0×10^{-8}
					T_1*LOPNRE*BATA*DIESEL2*H	3.0×10^{-8}
					T_1*LOPNRE*R*DIESEL1*SSC*LC	2.3×10^{-8}
					T_1*LOPNRE*R*SSA*DIESEL2*SSC	2.2×10^{-8}
					T_1*LOPNRE*R*LB2*SSA*DIESEL3	2.2×10^{-8}
					T_1*LOPNRE*R*DIESEL1*SSB*SSC	2.2×10^{-8}
					T_1*LOPNRE*R*LB2*DIESEL1*H	2.2×10^{-8}
					T_1*LOPNRE*R*DIESEL2*SSC*L	2.2×10^{-8}
					T_1*LOPNRE*R*V1*DIESEL2*DIESEL3	2.1×10^{-8}
					T_1*LOPNRE*R*DIESEL1*V2*DIESEL3	2.1×10^{-8}
					T_1*LOPNRE*BATA*DIESEL2*SSC	2.0×10^{-8}
					T_1*LOPNRE*BATA*LB2*DIESEL3	2.0×10^{-8}
					T_1*LOPNRE*R*SSA*DIESEL3*LB1	2.0×10^{-8}
					T_1*LOPNRE*R*DIESEL1*H*LB1	2.0×10^{-8}
					T_1*LOPNRE*R*SSA*H*LC	2.0×10^{-8}
					T_1*LOPNRE*R*SSA*SSB*H	1.9×10^{-8}
					T_1*LOPNRE*R*SSB*L*H	1.9×10^{-8}
					T_1*LOPNRE*BATA*DIESEL3*LB1	1.9×10^{-8}
					T_1*LOPNRE*BATA*H*LC	1.8×10^{-8}
					T_1*LOPNRE*BATA*SSB*H	1.8×10^{-8}
					T_1*LOPNRE*R*LB2*DIESEL1*SSC	1.4×10^{-8}
					T_1*LOPNRE*R*DIESEL1*SSC*LB1	1.3×10^{-8}
					T_1*LOPNRE*R*SSA*SSC*LC	1.3×10^{-8}
					T_1*LOPNRE*R*V1*DIESEL3*LC	1.3×10^{-8}
					T_1*LOPNRE*R*SSA*SSB*SSC	1.3×10^{-8}
					T_1*LOPNRE*R*LB2*SSA*H	1.3×10^{-8}
					T_1*LOPNRE*R*SSB*SSC*L	1.3×10^{-8}
					T_1*LOPNRE*R*SSA*V2*DIESEL3	1.2×10^{-8}
					T_1*LOPNRE*R*V1*SSB*DIESEL3	1.2×10^{-8}
					T_1*LOPNRE*R*V1*DIESEL2*H	1.2×10^{-8}
					T_1*LOPNRE*R*DIESEL1*V2*H	1.2×10^{-8}
					T_1*LOPNRE*R*V2*DIESEL3*L	1.2×10^{-8}
					T_1*LOPNRE*BATA*SSC*LC	1.2×10^{-8}

TABLE B.3. (contd)

Accident Sequence	Seq. Freq. (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_1QUV					T_1*LOPNRE*BATA*LB2*H	1.2×10^{-8}
(Cont.)					T_1*LOPNRE*BATA*SSB*SSC	1.2×10^{-8}
					T_1*LOPNRE*R*SSA*H*LB1	1.2×10^{-8}
					T_1*LOPNRE*BATA*V2*DIESEL3	1.2×10^{-8}
					T_1*LOPNRE*BATA*H*LB1	1.1×10^{-8}
					T_1*LOPNRE*OP*R*V3	1.0×10^{-8}
					T_1*LOPNRE*R*DIESEL1*DIESEL2*V3	8.7×10^{-9}
					T_1*LOPNRE*R*LB2*SSA*SSC	8.4×10^{-9}
					T_1*LOPNRE*R*V1*DIESEL2*SSC	8.2×10^{-9}
					T_1*LOPNRE*R*LB2*V1*DIESEL3	8.2×10^{-9}
					T_1*LOPNRE*R*DIESEL1*V2*SSC	8.2×10^{-9}
					T_1*LOPNRE*BATA*LB2*SSC	7.8×10^{-9}
					T_1*LOPNRE*R*SSA*SSC*LB1	7.8×10^{-9}
					T_1*LOPNRE*R*V1*DIESEL3*LB1	7.6×10^{-9}
					T_1*LOPNRE*R*V1*H*LC	7.5×10^{-9}
					T_1*LOPNRE*BATA*SSC*LB1	7.3×10^{-9}
					T_1*LOPNRE*R*SSA*V2*H	7.2×10^{-9}
					T_1*LOPNRE*R*V1*SSB*H	7.2×10^{-9}
					T_1*LOPNRE*R*V2*L*H	7.2×10^{-9}
					T_1*LOPNRE*BATA*V2*H	6.7×10^{-9}
					T_1*LOPNRE*R*DIESEL1*V3*LC	5.3×10^{-9}
					T_1*LOPNRE*R*SSA*DIESEL2*V3	5.1×10^{-9}
					T_1*LOPNRE*R*SSB*DIESEL1*V3	5.1×10^{-9}
					T_1*LOPNRE*R*DIESEL2*V3*L	5.1×10^{-9}
					T_1*LOPNRE*R*V1*SSC*LC	5.0×10^{-9}
					T_1*LOPNRE*R*SSA*V2*SSC	4.8×10^{-9}
					T_1*LOPNRE*R*V1*SSB*SSC	4.8×10^{-9}
					T_1*LOPNRE*R*LB2*V1*H	4.8×10^{-9}
					T_1*LOPNRE*R*V2*SSC*L	4.8×10^{-9}
					T_1*LOPNRE*BATA*DIESEL2*V3	4.8×10^{-9}
					T_1*LOPNRE*R*V1*V2*DIESEL3	4.7×10^{-9}
					T_1*LOPNRE*BATA*V2*SSC	4.5×10^{-9}
					T_1*LOPNRE*R*V1*H*LB1	4.5×10^{-9}
					T_1*LOPNRE*OP*R*SCC	3.7×10^{-9}
					T_1*LOPNRE*OP*R*HACT	3.7×10^{-9}
					T_1*LOPNRE*R*LB2*DIESEL1*V3	3.4×10^{-9}

TABLE B.4. Elements of Dominant Minimal Cut Sets of Grand Gulf Dominant Accident Sequences

Symbol	Description	Probability
T_1	Transient initiated by loss of offsite power	0.2/ry
T_{23}	Transient other than loss of offsite power which requires a reactor shutdown	7/ry
S	Small LOCA (rupture area <1 ft^2)	.0014/ry
C	Failure to achieve reactor subcriticality	7.7E-7
	Event C occurs if both of the following occur: 1. The reactor protection system (RPS) fails. The RPS fails if either the reactor protection logic system (RPLS) fails or three or more adjacent control rods fail to insert. The failure probabilities for these are taken from WASH-1400 to be: RPLS - 1.9E-6 control rods - 5.8E-6 The first is dominated by common-cause human errors in test and calibration of sensor switches. The overall failure probability for the RPS is the sum of the above (7.7E-6). 2. The recirculation pumps fail to trip or the operator fails to take appropriate action to shutdown the reactor, given RPS failure. These are dominated by the operator failing to manually initiate the standby liquid control system or to manually initiate control rod insertion. The estimated failure probability is 0.1 from WASH-1400. The failure probability of event C is the product of the above: (7.7E-6)(0.1) = 7.7E-7.	
DIESEL1	Failure of diesel generator #1 to provide emergency power	.036
	The failure probability of event DIESEL1 is the sum of the following contributory modes:	

TABLE B.4. (contd)

Symbol	Description	Probability
	Failure to start - .030 maintenance outage - .0064 .036	
DIESEL2	Failure of diesel generator #2 to provide emergency power	.036
	The expansion of event OIESEL2 into its contributory failures in analogous to event DIESEL1	
DIESEL3	Failure to diesel generator #3 to provide emergency power	.036
	The expansion of event DIESEL3 into its contributory modes is analogous to event DIESEL1.	
BATA	Failure of emergency DC battery A	.001
BATB	Failure of emergency DC battery B	.001
H	Loss of flow path from condensate storage tank (CST) to core spray nozzles in the high pressure core spray system (HPCSS)	.021
	Event H occurs if any of the following fail:	
	1. One of three check valves (CVs). The failure probability of each is that for hardware (.0001).	
	2. A normally-open (NO) motor-operated valve (MOV). Its failure probability is the sum of the following contributory modes: plugged - .0001 maintenance outage - .0058 .0059	

TABLE B.4. (contd)

Symbol	Description	Probability
	3. A normally-closed (NC) MOV. Its failure probability is the sum of the following contributory modes: hardware - .001 plugged - .0001 control circuitry - .0003 maintenance outage - .0058 .0072	
	4. A pump. Its failure probability is the sum of the following contributory modes: hardware - .001 control circuitry - .001 maintenance outage - .0058 .0078	
	The failure probability of event H is the sum of the above: 3(.0001) + .0059 + .0072 + .0078 = .021. The factor of three accounts for the contribution from three CVs.	
LOPNRE	Failure to recover offsite power within 30 min.	0.2
LOPNRL	Failure to recover offsite power within ~30 hrs, given LOPNRE	0.1
HACT	Failure of actuating circuit of HPCSS	.0012
	The failure probability of event HACT is the sum of the following contributory modes: functional - .001 test outage - .00023 .0012	
P	Failure of a safety/relief valve to reseat	0.1

TABLE B.4. (contd)

Symbol	Description	Probability
R	Loss of either of the following flow paths in the reactor core isolation cooling system (RCICS):	.051

1. CST to core spray nozzle.
2. Main steam line to turbine pump to suppression pool.

Event R occurs if any of the following fail:

1. One of five CVs. Each has a hardware failure probability of .0001.
2. Either of two NO (locked) manual valves (ManVs). The failure probability of each is the sum of of the following contributory modes:

   plugged - .0001

   operator error - .0001

   .0002

3. One of four NO MOVs. Each has a failure probability of .0059 as shown in event H.
4. Either of two NC MOVs. Each has a failure probability of .0072 as shown in event H.
5. A trip throttling valve for the RCIC turbine pump. Its failure probability is .0013.
6. A turbine governing valve for the RCIC turbine pump. Its failure probability is .0022.
7. A turbine pump. Its failure probability is .001.
8. An electric pump. It has a failure probability of .0078 as shown in event H.

The failure probability of event R is the sum of the above: 5(.0001) + 2(.0002) + 4(.0059) + 2(.0072) + .0013 + .0022 + .001 + .0078 = .051. The multiplicative factors account for contributions from multiple valves of the same type

TABLE B.4. (contd)

Symbol	Description	Probability
RACT	Failure of the actuating circuit of RCICS	.0012
	The expansion of event RACT into its contributory failures is analogous to event HACT	
L	Loss of flow path from suppression pool to core spray nozzles in the low pressure core spray system (LPCSS)	.021
	Event L occurs if any of the following fail: 1. Either of two CVs. Each has a hardware failure probability of .0001. 2. An NO (key-locked) MOV. It has a failure probability of .0059 as shown in event H. 3. An NC (key-locked) MOV. It has a failure probability of .0072 as shown in event H. 4. An NO (locked) MOV. It has a failure probability of .0002 as shown in event R. 5. A pump. It has a failure probability of .0078 as shown in event H.	
	The failure probability of event L is the sum of the above: 2(.0001) + .0059 + .0072 + .0002 + .0078 = .021. The factor of two account for the contribution from two CVs.	
LRACT	Failure of the actuating circuit for LPCSS and for train A of the residual heat removal system (RHRS)	.0012
	The expansion of event LRACT into its contributory failures is analogous to that for event HACT	
OP	Failure of the operator to manually initiate the automatic depressurization system	.0015
LA2	Loss of flow path from the suppression pool through the pump in train A of the low pressure coolant injection system (LPCIS)	.014
	Event LA2 occurs if any of the following fail: 1. A CV. It has a hardware failure probability of .0001.	

TABLE B.4. (contd)

Symbol	Description	Probability
	2. An NO MOV. It has a failure probability of .0059 as shown in event H.	
	3. An NO (locked) ManV. It has a failure probability of .0002 as shown in event R.	
	4. A pump. It has a failure probability of .0078 as shown in event H.	
	The failure probability of event LA2 is the sum of the above.	
LB1	Failure of a valve in the piping from the pump to the reactor vessel in train B of the LPCIS	.013
	Event LB1 occurs if any of the following fail:	
	1. A CV. It has a hardware failure probability of .0001.	
	2. An NO MOV. It has a failure probability of .0059 as shown in event H.	
	3. An NC MOV. It has a failure probability of .0072 as shown in event H.	
	4. An NO (locked) ManV. It has a failure probability of .0002 as shown in event R.	
	The failure probability of event LB1 is the sum of the above.	
LB2	Loss of flow path from the suppression pool through the pump in train B of the LPCIS	.014
	The expansion of event LB2 into its contributory failures is analogous to event LA2.	
LC	Loss of flow path from the suppression pool to the reactor vessel in train C of the LPCIS	.022
	Event LC occurs if any of the following fail:	
	1. Either of two CVs. Each has a hardware failure probability of .0001.	

TABLE B.4. (contd)

Symbol	Description	Probability
	2. An NO MOV. It has a failure probability of .0059 as shown in event H.	
	3. An NC MOV. It has a failure probability of .0072 as shown in event H.	
	4. Either of two NO (locked) ManVs. Each has a failure probability of .0002 as shown in event R.	
	5. A pump. It has a failure probability of .0078 as shown in event H.	
	The failure probability of event LC is the sum of the above: 2(.0001) + .0059 + .0072 + 2(.0002) + .0078 = .022. The factors of two account for the contributions from two CVs and two NO ManVs.	
BCACT	Failure of the actuating circuit for trains B and C of the RHRS	.0012
	The failure probability of event BCACT is the sum of the following contributory modes:	
	functional - .001	
	test outage - .00024	
	.0012	
PA27	Failure of RHRS pump A to continue running for ~30 hrs	8.1E-4
PB27	Failure of RHRS pump B to continue running for ~30 hrs	8.1E-4
VGA1	Failure of a value in the inlet/outlet piping of the RHRS or the standby service water system (SSWS) for RHRS heat exchanger A	.015
	Event VGA1 occurs if any of the following fail:	
	1. Either of two NC MOVs. Each has a failure probability of .0072 as shown in event H.	
	2. Either of two NO (locked) ManVs. Each has a failure probability of .0002 as shown in event R.	

TABLE B.4. (contd)

Symbol	Description	Probability
	The failure probability of event VGA1 is the sum of the above: 2(.0072) + 2(.0002) = .015. The factors of two account for contributions from two NC MOVs and two NO ManVs.	
VGA2	Failure of a valve in any of the following for train A of the RHRS:	.024

1. bypass line for RHRS heat exchanger A
2. suppression pool return line
3. inlet/outlet piping of the SSWS for the RHRS A room cooler
4. inlet/outlet piping of the SSWS for the RHRS pump A seal cooler.

Event VGA2 occurs if any of the following fail:

1. Either of two NO MOVs. Each has a failure probability of .0059 as shown in event H.
2. An NO, must-close MOV. Its failure probability is the sum of the following contributory modes:

   hardware – .001

   control circuitry – .0003

   .0013

3. An NC MOV. It has a failure probability of .0072 as shown in event H.
4. One of four NO ManVs. The failure probability of each is the sum of the following contributory modes:

   human error – .00073

   plugged – .0001

   .00083

TABLE B.4. (contd)

Symbol	Description	Probability
	The failure probability of event VGA2 is the sum of the above: 2(.0059) + .0013 + .0072 + 4(.00083) = .024. The factors of two and four account for the contributions from two NO MOVs and four NO ManVs.	
VGB1	Failure of a valve in the inlet/outlet piping of the RHRS or the SSWS for RHRS heat exchanger B.	.015
	The expansion of event VGB1 into its contributory failures is analogous to event VGA1.	
VGB2	Failure of a valve in any of the following for train B of the RHRS:	.024
	1. bypass line for RHRS heat exchanger B	
	2. suppression pool return line	
	3. inlet/outlet piping of the SSWS for the RHRS B room cooler	
	4. inlet/outlet piping of the SSWS for the RHRS pump B seal cooler.	
	The expansion of event VGB2 into its contributory failures is analogous to event VGA2.	
SA	Failure of a valve in train A of the suppression pool makeup system (SPMS).	.014
	Event SA occurs if either of two NC MOVs fails. Each has a failure probability of .0072 as shown in event H.	
	The failure probability of event SA is the sum of that for each NC MOV.	
SB	Failure of a valve in train B of the SPMS.	.014
	The expansion of event SB into its contributory failures is analogous to event SA.	
SAACC	Failure of actuation and control circuitry for train A of the SPMS.	.0012
	The expansion of event SAACC into its contributory failures is analogous to event HACT.	

TABLE B.4. (contd)

Symbol	Description	Probability
SBACC	Failure of actuation and control circuitry for train B of the SPMS. The expansion of event SBACC into its contributory failures is analogous to event HACT.	.0012
SSA	Loss of flow path into and through pump A of the SSWS, including the pump A oil cooler. Event SSA occurs if any of the following fail: 1. A pump. It has a failure probability of .D078 as shown in event H. 2. A CV. It has a hardware failure probability of .0001. 3. Either of two NO MOVs. Each has a failure probability of .0059 as shown in event H. 4. An NO ManV. It has a failure probability of .00083 as shown in event VGA2. The failure probability for event SSA is the sum of the above: .0078 + .0001 + 2(.0059) + .00083 = .021. The factor of two accounts for the contribution from two NO MOVs.	.021
SSB	Loss of flow path into and through pump B of the SSWS, including the pump B oil cooler. The expansion of event SSB into its contributory failures is analogous to event SSA.	.D21
SSC	Loss of flow path into and through pump C of the SSWS Event SSC occurs if any of the following fail: 1. A pump. It has a failure probability of .0078 as shown in event H. 2. A CV. It has a hardware failure probability of .0001. 3. An NO MOV. It has a failure probability of .0059 as shown in event H.	.014

TABLE B.4. (contd)

Symbol	Description	Probability
	4. An NO (locked) ManV. It has a failure probability of .0002 as shown in event R.	
	The failure probability of event SSC is the sum of the above.	
SAC	Failure of the actuation and control circuitry for train A of the SSWS.	.0012
	The expansion of event SAC into its contributory failures is analogous to event HACT.	
SBC	Failure of the actuation and control circuitry for train B of the SSWS.	.0012
	The expansion of event SBC into its contributory failures is analogous to event HACT.	
SCC	Failure of the actuation and control circuitry for train C of the SSWS.	.0012
	The expansion of event SCC into its contributory failures is analogous to event HACT.	
V1	Failure of the inlet/outlet valve of the SSWS for the jacket cooler of diesel generator #1.	.0080
	Event V1 occurs if either of the following fail:	
	1. An NC MOV. It has a failure probability of .0072 as shown in event H.	
	2. An NO ManV. It has a failure probability of .00083 as shown in event VGA2.	
	The failure probability of event V1 is the sum of the above.	
V2	Failure of the inlet/outlet valve of the SSWS for the jacket cooler of diesel generator #2.	.0080
	The expansion of event V2 into its contributory failures is analogous to event V1.	

TABLE B.4. (contd)

Symbol	Description	Probability
V3	Failure of an inlet/outlet valve in either of two SSWS flow paths for the two jacket coolers of diesel generator #3.	.0033
	Event V3 occurs if one of four NO ManVs fails. Each has a failure probability of .00083 as shown in event VGA2.	
	The failure probability of event V3 is the sum of that for each NO ManV.	
Q	Failure of the power conversion system (PCS) to provide makeup water.	1
Q1	Failure of the PCS to remove decay heat in ~30 hrs.	.0070
	The failure probability of event Q1 is taken from the analogous event for the WASH-1400 BWR, .0070.	
RECOVERY	Failure to restore maintenance/test faults or to take other corrective action within 28 hrs.	0.23
RECOVERY1	Failure to restore maintenance/test faults or to take other corrective action within 30 hrs.	0.21
SCVA	Loss of flow path into and through heat exchanger A of the RHRS.	.032
	Event SCVA occurs if any of the following fail: 1. A CV. It has a hardware failure probability of .0001. 2. Either of two NO, must-close MOVs. Each has a failure probability of .0013 as shown in event VGA2. 3. One of four NC MOVs. Each has a failure probability of .0072 as shown in event H	
	The failure probability of event SCVA is the sum of the above: .0001 + 2(.0013) + 4(.0072) = .032. The factors of two and four account for the contributions from two NO, must-close MOVs and four NC MOVs.	

TABLE B.4. (contd)

Symbol	Description	Probability
SCVB	Loss of flow path into and through heat exchanger B of the RHRS.	.032
	The expansion of event SCVB into its contributory failures is analogous to event SCVA.	

ADDENDUM B.I

When calculating the frequencies of the minimal cut sets for the accident sequences $T_{23}PQI$, T_1PQE, $T_{23}PQE$, SI, and $T_{23}QW$ (see Table B.3), double contributions from the same maintenance outages for the following pairs of terms must be removed from their products as follows:

1. For sequences $T_{23}PQI$ and SI:

$$\text{VGA2} \cdot \text{VGB2} = (.024)^2 - [2(.0058)]^2 = 4.4\text{E}{-4}$$

$$\left.\begin{matrix} \text{VGB2} \cdot \text{SSA} \\ \text{VGA2} \cdot \text{SSB} \end{matrix}\right\} = (.024)(.021) - 2(3)(.0058)^2 = 3.0\text{E}{-4}$$

$$\left.\begin{matrix} \text{VGA2} \cdot \text{VGB1} \\ \text{VGA1} \cdot \text{VGB2} \end{matrix}\right\} = (.024)(.015) - [2(.0058)]^2 = 2.3\text{E}{-4}$$

$$\left.\begin{matrix} \text{VGA2} \cdot \text{LB2} \\ \text{VGB2} \cdot \text{LA2} \end{matrix}\right\} = (.024)(.014) - [2(.0058)]^2 = 2.0\text{E}{-4}$$

$$\text{SSA} \cdot \text{SSB} = (.021)^2 - [3(.0058)]^2 = 1.4\text{E}{-4}$$

$$\left.\begin{matrix} \text{VGB1} \cdot \text{SSA} \\ \text{VGA1} \cdot \text{SSB} \end{matrix}\right\} = (.015)(.021) - 2(3)(.0058)^2 = 1.1\text{E}{-4}$$

$$\left.\begin{matrix} \text{LB2} \cdot \text{SSA} \\ \text{LA2} \cdot \text{SSB} \end{matrix}\right\} = (.014)(.021) - 2(3)(.0058)^2 = 9.2\text{E}{-5}$$

$$\text{VGA1} \cdot \text{VGB1} = (.015)^2 - [2(.0058)]^2 = 9.0\text{E}{-5}$$

$$\left.\begin{matrix} \text{VGA1} \cdot \text{LB2} \\ \text{VGB1} \cdot \text{LA2} \end{matrix}\right\} = (.015)(.014) - [2(.0058)]^2 = 7.5\text{E}{-5}$$

$$\left.\begin{matrix} \text{SA} \cdot \text{SB} \\ \text{LA2} \cdot \text{LB2} \end{matrix}\right\} = (.014)^2 - [2(.0058)]^2 = 6.1\text{E}{-5}$$

2. For sequences T_1PQE and $T_{23}PQE$ (in the first, i.e., most dominant, minimal cut set ONLY):

$$\text{H} \cdot \text{R} = (.021)(.051) - 3(7)(.0058)^2 = 3.6\text{E}{-4}$$

3. For sequence $T_{23}QW$:

$$SSA \cdot SSB = (.021)^2 - [2(.0058)]^2 = 3.1E-4$$

$$\left.\begin{array}{l} VGB1 \cdot SSA \\ VGA1 \cdot SSB \end{array}\right\} = (.015)(.021) - [2(.0058)]^2 = 1.8E-4$$

$$VGA1 \cdot VGB1 = (.015)^2 - [2(.0058)]^2 = 9.0E-5$$

REFERENCE

Hatch, S. et al. 1981. Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant, NUREG/CR-1659/4. Sandia National Laboratories, Albuquerque, New Mexico.

APPENDIX C

NUCLEAR POWER PLANT CHARACTERIZATION

This appendix provides information on nuclear power plant age, principal vendors, and size useful in determining where safety issue resolutions are applicable. These characteristics are also used in calculating the average plant life ($\overline{T}$) for groups of plants.

The calculation of the average remaining life of reactors affected by the resolution of a safety issue ($\overline{T}$) can be completed in four steps:

1) Determine plants affected and divide into backfit and forward-fit categories.
2) Multiply forward-fit plants by their total expected life. Thirty-years was assumed for this category.
3) Sum remaining lives in existing plants by assuming a 35-year life and subtracting past service years.
4) Sum backfit and forward-fit life and divide by the total number of plants.

An estimate of the number of plants (N) and average remaining life ($\overline{T}$) in each of the four reactor categories (backfit, forward-fit, BWR, PWR) was completed. Results are shown in Table C.1.

If specific plants or vendors are involved, a specific calculation must be performed using the method discussed above. Additional sources of data (for example Nuclear Power Experience) may need to be consulted if further differentiation between plants by subsystem or performance is required.

TABLE C.1 Type and Life of Nuclear Power Plants*

Reactor Supplier	Type	No. of Units (N)		Average Remaining Life ($\overline{T}$)(years)	
		Completed	Planned	Completed	Planned
1. Westinghouse	PWR	31	30	27.5	30
2. General Electric	BWR	24	20	25.2	30
3. Babcock & Wilcox	PWR	8	5	27.8	30
4. Combustion Engineering	PWR	8	8	27.9	30

	N	$\overline{T}$(years)
All PWR	90	28.8
Backfit	47	27.7
Forward-fit	43	30
All BWR	44	27.4
Backfit	24	25.2
Forward-fit	20	30
All Plants	134	28.3
Backfit	71	26.9
Forward-fit	63	30

* Excluding Humboldt Bay, TMI-2, Shippingport, La Crosse BWR, Fort St. Vrain, and Hanford-N

TABLE C.2 Plant-Specific Characteristics

Combustion Engineering

Completed

Name	(Net MWe)	Start Date	Backfit (yrs)
Calvert Cliffs 1	850	5/75	28
Calvert Cliffs 2	850	4/77	30
Maine Yankee	825	12/72	25
Millstone 2	870	12/75	28
Palisades	740	12/71	24
Fort Calhoun 1	478	9/73	26
Arkansas Nuclear I-2	858	3/80	33
St Lucie 1	777	12/76	29

Under Construction

Name	(Net MWe)	Start Date
St Lucie 2	777	5/83
Waterford 3	1104	7/83
Palo Verde 1	1270	5/83
Palo Verde 2	1270	5/84
Palo Verde 3	1270	5/86
WNP-3	1240	12/86
San Onofre 2	1100	82
San Onofre 3	1100	9/83

TABLE C.2 (contd)

Babcock & Wilcox

Completed

Name	(Net MWe)	Start Date	Backfit (yrs)
Three-Mile Island 1	792	9/74	27
*Three-Mile Island 2	880	12/78	31
Davis-Besse 1	906	11/77	30
Arkansas Nuclear I-1	836	12/74	27
Oconee 1	860	7/73	26
Oconee 2	860	9/74	27
Oconee 3	860	12/74	27
Crystal River 3	825	3/77	30
Rancho Seco	913	4/75	28

Under Construction

Name	(Net MWe)	Start Date
Midland 1	805	7/84
Midland 2	805	12/83
Bellefonte 1	1213	11/86
Bellefonte 2	1213	11/89
North Anna 3	907	89

* Shut-down indefinitely.

TABLE C.2 (contd)

Westinghouse

Completed

Name	(Net MWe)	Start Date	Backfit (yrs)
Haddam Neck	582	1/68	21
Indian Point 2	873	7/74	27
Beaver Valley 1	833	4/77	30
Indian Point 3	965	8/76	29
Salem 1	1090	6/77	30
Salem 2	1115	10/81	34
Robert E. Ginna	490	3/70	23
Yankee	175	6/61	14
Zion 1	1040	10/73	26
Zion 2	1040	9/74	27
Donald C. Cook 1	1054	8/75	28
Donald C. Cook 2	1094	7/78	31
Prairie Island 1	520	12/73	26
Prairie Island 2	520	12/74	27
Point Beach 1	497	12/70	23
Point Beach 2	497	10/72	25
Kewaunee	535	6/74	27
Joseph M. Farley 1	829	12/77	30
Joseph M. Farley 2	829	7/81	34
Robinson 2	665	3/71	24
McGuire 1	1180	12/81	34
Turkey Point 3	666	12/72	25
Turkey Point 4	666	9/73	26
Sequoyah 1	1148	7/81	34
Sequoyah 2	1148	6/82	35
Surry 1	775	12/72	25
Surry 2	775	5/73	26
North Anna 1	865	6/78	31
North Anna 2	890	12/80	33
Trojan	1130	5/76	29
San Onofre 1	436	1/68	21

TABLE C.2 (contd)

Westinghouse

Under Construction

Name	(Net MWe)	Start Date
Beaver Valley 2	833	5/86
Millstone 3	1150	5/86
Seabrook 1	1150	2/84
Seabrook 2	1150	5/86
Byron 1	1120	2/84
Byron 2	1120	2/85
Braidwood 1	1120	10/85
Braidwood 2	1120	10/86
Carroll County 1	1120	99
Carroll County 2	1120	2000
Wolf Creek	1150	5/84
Marble Hill 1	1130	86
Marble Hill 2	1130	87
Callaway 1	1150	4/84
Shearon Harris 1	900	9/85
Shearon Harris 2	900	3/89
McGuire 2	1180	10/83
Catawba 1	1145	6/85
Catawba 2	1145	6/87
Vogtle 1	1100	3/87
Vogtle 2	1100	9/88
Virgil C. Summer 1	900	82
Watts Bar 1	1177	11/84
Watts Bar 2	1177	12/85
South Texas Project 1	1250	86
South Texas Project 2	1250	88
Comanche Peak 1	1150	84
Comanche Peak 2	1150	85
Diablo Canyon 1	1084	82
Diablo Canyon 2	1106	83

TABLE C.2 (contd)

General Electric

Completed

Name	(Net MWe)	Start Date	Backfit (yrs)
Pilgrim 1	670	12/72	25
Oyster Creek 1	620	12/69	22
Nine Mile Point 1	610	12/69	22
Millstone 1	660	12/70	23
Peach Bottom 2	1065	7/74	27
Peach Bottom 3	1065	12/74	27
James A. Fitzpatrick	821	6/77	28
Vermont Yankee	514	11/72	25
Dresden 1	207	8/60	13
Dresden 2	794	8/70	23
Dresden 3	794	10/71	24
Quad-Cities 1	789	8/72	25
Quad-Cities 2	789	10/72	25
Big Rock Point	63	12/62	15
Duane Arnold	545	5/74	27
Cooper	778	7/74	27
Monticello	536	7/71	24
Brunswick 1	790	3/77	30
Brunswick 2	790	11/75	28
Edwin I. Hatch 1	786	12/75	28
Edwin I. Hatch 2	786	8/79	32
Browns Ferry 1	1067	8/74	27
Browns Ferry 2	1067	3/75	28
Browns Ferry 3	1067	3/77	30
*Humboldt Bay 3	63	8/63	16

*Shut-down indefinitely.

TABLE C.2 (contd)

General Electric

Under Construction

Name	(Net MWe)	Start Date
Shoreham	820	3/83
Nine Mile Point 2	1080	10/86
Susquehanna 1	1050	5/83
Susquehanna 2	1050	84
Limerick 1	1055	4/85
Limerick 2	1055	10/87
Hope Creek 1	1070	12/86
Zimmer 1	810	83
Perry 1	1205	5/84
Perry 2	1205	5/88
Lasalle 1	1078	9/82
Lasalle 2	1078	10/83
Fermi 2	1100	11/83
Clinton 1	950	8/84
River Bend 1	940	12/85
Allens Creek 1	1200	91
Skagit-Hanford 1	1288	91
Skagit-Hanford 2	1288	93
WNP-2	1100	2/84
Grand Gulf 1	1250	2/83

TABLE C.2 (contd)

	Name	(Net MWe)	Start Date
LWBR	- Westinghouse		
	Shippingport	6D	12/57
BWR	- Allis-Chalmers		
	La Crosse BWR	50	11/69
HTGR	- General Atomics		
	Fort St. Vrain	330	1/79
LGR	- General Electric		
	Hanford-N	860	7/66

REFERENCES

American Nuclear Society. 1982. "World List of Nuclear Power Plants." Nuclear News 25/10:79.

Petroleum Information Corporation. Nuclear Power Experience, Denver, Colorado.

APPENDIX D

DOSE CALCULATION FACTORS

This appendix presents dose calculation factors to be used in conjuction with methods described in Section 3.0 of this report to calculate dose reductions due to resolution of safety issues.

TABLE D.1. Public Dose Consequence Factors

Category	Whole Body Dose Consequence Factor (man-rem)	
	Core Melt	Non Core-Melt
PWR 1A*	5.4E+6	
PWR 1B	4.4E+6	
PWR 2	4.8E+6	
PWR 3	5.4E+6	
PWR 4	2.7E+6	
PWR 5	1.0E+6	
PWR 6	1.5E+5	
PWR 7	2.3E+3	
PWR 8		7.5E+4
PWR 9		1.2E+2
BWR 1	5.4E+6	
BWR 2	7.1E+6	
BWR 3	5.1E+6	
BWR 4	6.1E+5	
BWR 5		2.0E+1

* Assumed to be PWR-1

The following are descriptions of the release categories used in this study. This information was extracted from Appendix VI, Section 2 of the Reactor Safety Study (WASH-1400):

GENERAL REMARKS

In order to define the various releases that might occur, a series of release categories were identified for the postulated types of containment failure in both BWRs and PWRs. The probability of each release category and the associated magnitude of radioactive releases (as fractions of the initial core radioactivity that might leak from the containment structure) are used as input data to the consequence model.

In addition to probability and release magnitude, the parameters that characterize the various hypothetical accident sequences are time of release, duration of release, height of release, and energy content of the released plume. The time of release refers to the time interval between the start of the hypothetical accident and the release of radioactive material from the containment building to the atmosphere; it is used to calculate the initial decay of radioactivity. The duration of release is the total time during which radioactive material is emitted into the atmosphere; it is used to account for continuous releases by adjusting for horizontal disperion due to wind meander. These parameters, time and duration of release, represent the temporal behavior of the release in the dispersion model. Finally, the height of release and the energy content of the released plume gas affect the manner in which the plume would be dispersed in the atmosphere.

ACCIDENT DESCRIPTIONS

PWR 1

This release category can be characterized by a core meltdown followed by a steam explosion on contact of molten fuel with the residual water in the reactor vessel. The containment spray and heat removal systems are also assumed to have failed and, therefore, the containment could be at a pressure above ambient at the time of the steam explosion. It is assumed that the steam explosion would rupture the upper portion of the reactor vessel and breach the containment barrier, with the result that a substantial amount of radioactivity might be released from the containment in a puff over a period of about 10 minutes. Due to the sweeping action of gases generated during containment-vessel meltthrough, the release of radioactive materials would continue at a relatively low rate thereafter. The total release would contain approximately 70% of the iodines and 40% of the alkali metals present in the core at the time of release. Because the containment would contain hot pressurized gases at the time of failure, a relatively high release rate of sensible energy from the containment could be associated with this category. This category also includes certain potential accident sequences that would involve the occurrence of core melting and a steam explosion after containment rupture due to overpressure. In these sequences, the rate of energy release would be lower, although still relatively high.

PWR 2

This category is associated with the failure of core-cooling systems and core melting concurrent with the failure of containment spray and heat-removal systems. Failure of the containment barrier would occur through overpressure, causing a substantial fraction of the containment atmosphere to be released in a puff over a period of about 30 minutes. Due to the sweeping action of gases generated during containment vessel meltthrough, the release of radioactive material would continue at a relatively low rate thereafter. The total release would contain approximately 70% of the iodines and 50% of the alkali metals present in the core at the time of release. As in PWR release category 1, the high temperature and pressure within containment at the time of containment failure would result in a relatively high release rate of sensible energy from the containment.

PWR 3

This category involves an overpressure failure of the containment due to failure of containment heat removal. Containment failure would occur prior to the commencement of core melting. Core melting then would cause radioactive materials to be released through a ruptured containment barrier. Approximately 20% of the iodines and 20% of the alkali metals present in the core at the time of release would be released to the atmosphere. Most of the release would occur over a period of about 1.5 hours. The release of radioactive material from containment would be caused by the sweeping action of gases generated by the reaction of the molten fuel with concrete. Since these gases would be initially heated by contact with the melt, the rate of sensible energy release to the atmosphere would be moderately high.

PWR 4

This category involves failure of the core-cooling system and the containment spray injection system after a loss-of-coolant accident, together with a concurrent failure of the containment system to properly isolate. This would result in the release of 9% of the iodines and 4% of the alkali metals present in the core at the time of release. Most of the release would occur continuously over a period of 2 to 3 hours. Because the containment recirculation spray and heat-removal systems would operate to remove heat from the containment atmosphere during core melting, a relatively low rate of release of sensible energy would be associated with this category.

PWR 5

This category involves failure of the core cooling systems and is similar to PWR release category 4, except that the containment spray injection system would operate to further reduce the quantity of airborne radioactive material and to initially suppress containment temperature and pressure. The containment barrier would have a large leakage rate due to a concurrent failure of the containment system to properly isolate, and most of the radioactive material would be released continuously over a period of several hours. Approximately 3% of the iodines and 0.9% of the alkali metals present in the core would be released. Because of the operation of the containment heat-removal systems, the energy release rate would be low.

PWR 6

This category involves a core meltdown due to failure in the core cooling systems. The containment sprays would not operate, but the containment barrier would retain its integrity until the molten core proceeded to melt through the concrete containment base mat. The radioactive materials would be released into the ground, with some leakage to the atmosphere occurring upward through the ground. Direct leakage to the atmosphere would also occur at a low rate prior to containment-vessel meltthrough. Most of the release would occur continuously over a period of about 10 hours. The release would include approximately 0.08% of the iodines and alkali metals present in the core at the time of release. Because leakage from containment to the atmosphere would be low and gases escaping through the ground would be cooled by contact with the soil, the energy release rate would be very low.

PWR 7

This category is similar to PWR release category 6, except that containment sprays would operate to reduce the containment temperature and pressure as well as the amount of airborne radioactivity. The release would involve 0.002% of the iodines and 0.001% of the alkali metals present in the core at the time of release. Most of the release would occur over a period of 10 hours. As in PWR release category 6, the energy release rate would be very low.

PWR 8

This category approximates a PWR design basis accident (large pipe break), except that the containment would fail to isolate properly on demand. The other engineered safeguards are assumed to function properly. The core would not melt. The release would involve approximately 0.01% of the iodines and 0.05% of the alkali metals. Most of the release would occur in the 0.5-hour period during which containment pressure would be above ambient. Because containment sprays would operate and core melting would not occur, the energy release rate would also be low.

PWR 9

This category approximates a PWR design basis accident (large pipe break), in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into the containment. The core would not melt. It is assumed that the minimum required engineered safeguards would function satisfactorily to remove heat from the core and containment. The release would occur over the 0.5-hour period during which the containment pressure would be above ambient. Approximately 0.00001% of the iodines and 0.00006% of the alkali metals would be released. As in PWR release category 8, the energy release rate would be very low.

BWR 1

This release category is representative of a core meltdown followed by a steam explosion in the reactor vessel. The latter would cause the release of a substantial quantity of radioactive material to the atmosphere. The total release would contain approximately 40% of the iodines and alkali metals present in the core at the time of containment failure. Most of the release would occur over a 1/2 hour period. Because of the energy generated in the steam explosion, this category would be characterized by a relatively high rate of energy release to the atmosphere. This category also includes certain sequences that involve overpressure failure of the containment prior to the occurrence of core melting and a steam explosion. In these sequences, the rate of energy release would be somewhat smaller than for those discussed above, although it would still be relatively high.

BWR 2

This release category is representative of a core meltdown resulting from a transient event in which decay-heat-removal systems are assumed to fail. Containment overpressure failure would result, and core melting would follow. Most of the release would occur over a period of about 3 hours. The containment failure would be such that radioactivity would be released directly to the atmosphere without significant retention of fission products. This category involves a relatively high rate of energy release due to the sweeping action of the gases generated by the molten mass. Approximately 90% of the iodines and 50% of the alkali metals present in the core would be released to the atmosphere.

BWR 3

This release category represents a core meltdown caused by a transient event accompanied by a failure to scram or failure to remove decay heat. Containment failure would occur either before core melt or as a result of gases generated during the interaction of the molten fuel with concrete after reactor-vessel meltthrough. Some fission-product retention would occur either in the suppression pool or the reactor building prior to release to the atmosphere. Most of the release would occur over a period of about 3 hours and would involve 10% of the iodines and 10% of the alkali metals. For those sequences in which the containment would fail due to overpressure after core melt, the rate of energy release to the atmosphere would be relatively high. For those sequences in which overpressure failure would occur before core melt, the energy release rate would be somewhat smaller, although still moderately high.

BWR 4

This release category is representative of a core meltdown with enough containment leakage to the reactor building to prevent containment failure by overpressure. The quantity of radioactivity released to the atmosphere would be significantly reduced by normal ventilation paths in the reactor building and potential mitigation by the secondary containment filter systems. Condensation in the containment and the action of the standby gas treatment system on the releases would also lead to a low rate of energy release. The radioactive material would be released from the reactor building or the stack at an elevated level. Most of the release would occur over a 2-hour period and would involve approximately 0.08% of the iodines and 0.5% of the alkali metals.

BWR 5

This category approximates a BWR design basis accident (large pipe break) in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into containment. The core would not melt, and containment leakage would be small. It is assumed that the minimum required engineered safeguards would function satisfactorily. The release would be filtered and pass through the elevated stack. It would occur over a period of about 5 hours while the containment is pressurized above ambient and would involve approximately 6×10^{-9}% of the iodines and 4×10^{-7}% of the alkali metals. Since core melt would not occur and containment heat-removal systems would operate, the release to the atmosphere would involve a negligibly small amount of thermal energy.

TABLE D.2. Estimated Occupational Radiation Dose from Cleanup, Repair and Refurbishment (man-rem) [Murphy 1982]

Activity	Accident* Scenario 1	Accident* Scenario 2	Accident* Scenario 3
Cleanup	670	4,580	12,100
Repair and Refurbishment[(a)]	1,210	3,060	7,760
Total	1,880	7,640	19,860

(a) Based on immediate dismantlement estimates.

* These scenarios are described in Section 3.3. In summary, they are as follows:

- Scenario 1 - a small LOCA in which ECCS functions as intended. Some fuel cladding ruptures, but no fuel melts. The containment building is moderately contaminated, but there is minimal physical damage.
- Scenario 2 - a small LOCA in which ECCS is delayed. Fifty percent of the fuel cladding ruptures, and some fuel melts. The containment building is extensively contaminated but there is minimal physical damage. (This scenario is presumed to simulate the TMI-2 accident.)
- Scenario 3 - a major LOCA in which ECCS is delayed. All fuel cladding ruptures, and there is significant fuel melting and core damage. The containment building is extensively contaminated and physically damaged. The auxiliary building undergoes some contamination.

TABLE D.3. General Occupational Dose Rates in Radiation Zones**

Area	Dose Rate
Inside Containment (Reactor Shutdown)	25 mR/hr
Outside Containment	2.5 mR/hr

** Use of dose rates from Chapter 12 of the plant FSAR is recommended over these values. (See Section 3.4.2.1.) These are very general values and provided only for convenience.

REFERENCES

Murphy, E. and G. Holter. 1982. Technology, Safety, and Costs of Decommissioning Reference Light Water Reactors Following Postulated Accidents, NUREG/CR-2601. Pacific Northwest Laboratory, Richland, Washington.

WASH-1400. 1975. Reactor Safety Study. U.S. Nuclear Regulatory Commission, Washington, D.C.

APPENDIX E

BASES FOR COST ESTIMATION

The cost information presented in this appendix is used in conjunction with cost analysis methods described in Section 4.0. Costs associated with a SIR are divided into industry and NRC categories. Industry costs are defined as all costs associated with the implementation, operation and maintenance of the SIR. The industry cost savings due to accident avoidance is also quantified for each issue. Future NRC costs are associated with the development of a SIR, the review of industry implementation actions associated with SIR compliance, and ongoing reviews of the licensee to assure proper operation/maintenance of the SIR. This appendix provides information for use in calculating industry and NRC labor costs, industry accident-avoidance cost, and industry replacement power costs. Other costs are estimated on a case-by-case basis.

INDUSTRY AND NRC STAFF LABOR COSTS

The development of average staff labor costs for both industry and NRC personnel is based on the following assumptions. For all professional staff manpower cost estimates, $100,000/man-year is used. Assuming 30 days of annual leave and 10 paid holidays, for a total of 220 work days or 44 work weeks per year, results in an average staff labor cost of $2,270 per person-week.

Regional labor costs for industry can deviate by as much as 17% from the national average (Manion 1980). Costs at individual nuclear plant locations might deviate even more. In addition, the owner/licensee labor cost will depend on the values used to estimate fringe benefits, taxes, insurance, and other owner/licensee overhead expenses. Nevertheless, this industry labor cost estimate is judged to be reasonable for purposes of these analyses.

INDUSTRY REPLACEMENT POWER COSTS

The value assumed for the purchase of replacement power during each outage day attributable to the implementation of the SIR is $300,000. The actual cost of replacement power for a specific plant will depend on may factors, including the capacity of the plant, the capacity of the utility's total system, the size of the system margin, and the cost of the replacement power at the time.

INDUSTRY ACCIDENT CONSEQUENCE COSTS

The costs of reactor cleanup, repair/refurbishment, and replacement power for use in calculating the industry cost savings due to accident avoidance are given here for the three accident scenarios described in Section 3.3. Scenario 3 refers to core-melt accidents, and its associated cost will be applicable in most safety issues. Scenarios 1 and 2 refer to non-core-melt accidents, and their associated costs may be applicable in certain specialized safety issues.

Scenario[a]	Costs[b]
1	= $72M Cleanup + $49M Repair/Refurbish + $600M Replacement Power = $720M over a 5 1/2-year period.
2	= $165M Cleanup + $48M Repair/Refurbish + $822M Replacement Power = $1035M over a 7 1/2-year period.
3	= $373M Cleanup + $106M Repair/Refurbish + $1172M Replacement Power = $1650M over a 10-year period.

(a) These scenarios are described in Section 3.3 and summarized in Table D.2.
(b) These costs were developed in NUREG/CR-2601 (Murphy 1982).

Costs in the above table are based on engineering estimates. Scenario 3 costs are similar to those used by the U.S. NRC (NUREG-0933) for prioritization puroses. NRC estimates are based on available information for cleanup and loss-of-use costs at the Three Mile Island 2 plant discounted to their present worth for future potential accidents.

REFERENCES

1. Manion, W. and T. La Guardia. 1980. Decommissioning Handbook, DOE/EV/10128-1. Nuclear Energy Services, Inc., Danbury, Connecticut.

2. Murphy, E. and G. Holter. 1982. Technology, Safety, and Costs of Decommissioning Reference Light Water Reactors Following Postulated Accidents, NUREG/CR-2601, Pacific Northwest Laboratory, Richland, Washington.

3. NUREG-0933. 1982. A Prioritizatin of Generic Safety Issues, U.S. Nuclear Regulatory Commission, Washington, D.C.

APPENDIX F

UNCERTAINTY ANALYSIS

To facilitate the evaluation of risk, dose and costs for the numerous safety issues, reasonably generic methods for uncertainty analyses are desirable. Thus, although issue-specific uncertainty techniques can be utilized where appropriate, it is anticipated that the generic approach developed here will suffice for most issues. This approach is extenoed to the specification of "standardized" values for the uncertainties on the various risk, dose and cost parameters.

F.1 UNCERTAINTIES ON PARAMETERS RELATED TO ACCIDENT-AVOIDANCE

The parameters related to accident-avoidance are the following:

- $(\Delta W)_{Total}$, the public risk reduction (for all affected plants)
- ΔU, the occupational dose reauction due to accident-avoidance
- ΔH, the industry cost savings due to accident-avoidance.

The following generic approach forms the basis for estimating uncertainties in these accident-avoidance terms.

The uncertainty analyses required to estimate upper and lower bounds on these terms are complicated because each is a random variable formed from the difference between two other random variables, e.g., $V = S - T$. The approach typically taken to estimate uncertainties in V involves random sampling from the probability distributions assumed to govern the parameters (random variables) S and T. These are combined via a Monte Carlo computer code to yield an approximate probability distribution on V. From this, upper and lower bounds can be found. The use of Monte Carlo techniques is too time-consuming for this project. Simpler, more approximate methods are developed here.

Consider the two random variables, S and T, with the following "best estimates" and error bounds:

Variable	Best Estimate	Upper Bound	Lower Bound
S	$\hat{S}$	S_u	S_ℓ
T	$\hat{T}$	T_u	T_ℓ

Define random variable V formed from the difference between S and T. It will have a best estimate $\hat{V}$ and error bounds V_u and V_ℓ for which the following relations hold:

$$\hat{V} = \hat{S} - \hat{T}$$

$$V_u \leq S_u - T_\ell$$

$$V_\ell \geq S_\ell - T_u$$

Conservative limits on V_u and V_ℓ can be set from the above as follows:

$$(V_u)_{max} = S_u - T_\ell$$

$$(V_\ell)_{min} = S_\ell - T_u$$

So long as S > T, $(V_u)_{max}$ will be >0. However, $(V_\ell)_{min}$ can be ≤ 0 if $T_u \geq S_\ell$. If V is so defined that it cannot assume negative values, then the following holds:

$$(V_\ell)_{min} = \max[0,\ (S_\ell - T_u)].$$

These are conservative approximations to the "true" error bounds on V = S - T. For convenience, the "max" and "min" subscripts are dropped and the error bounds on V are taken as:

$$V_u = S_u - T_\ell$$

$$V_\ell = \max[0,\ (S_\ell - T_u)]$$

F.1.1 Public Risk Reduction

The public risk reduction for one plant is:

$$\Delta W = W - W^*$$

where W is the base-case, affected public risk and W* is the adjusted-case, affected public risk. This equation is analogous to that for V = S - T with V, S, and T replaced by ΔW, W, and W* respectively. Thus, the error bounds on the public risk reduction will be:

$$(\Delta W)_u = W_u - W^*_\ell$$

$$(\Delta W)_\ell = \max[0, (W_\ell - W^*_u)]$$

The general formula for the affected public risk is:

$$W = \sum_i (R_i \sum_j F_{ij})$$

This is analogous to the public risk formula given in Section 3.1 except that here only the affected core-melt accident sequence frequencies and the affected core-melt release category frequencies and dose factors are involved. This formula can be rewritten as follows:

$$W = \bar{F} \cdot \bar{R}$$

where $\bar{F}$ = affected core-melt frequency

$\bar{R}$ = average dose factor for all affected core-melt release categories.

The latter has a value of:

$$\bar{R} = \frac{W}{\bar{F}} = \frac{\sum_i (R_i \sum_j F_{ij})}{\sum_i \sum_j F_{ij}}$$

Returning to the variables S and T defined earlier, the additional assumption is made that both are lognormal with "error factors" defined as follows:

$$f_S = S_u/\hat{S} = \hat{S}/S_\ell$$

$$f_T = T_u/\hat{T} = \hat{T}/T_\ell$$

Define a new random variable Z formed from the product of S and T. An approximation for the error factor of Z is the following (Pepping 1981):

$$f_Z = \exp\sqrt{\ln^2 f_S + \ln^2 f_T}$$

where $Z = S \cdot T$. The error bounds on Z will be:

$$Z_u = \hat{Z} f_Z$$

$$Z_\ell = \hat{Z}/f_Z$$

where $\hat{Z} = \hat{S} \cdot \hat{T}$. Z will be lognormal.

Replacing Z, S, and T by W, $\bar{F}$, and $\bar{R}$ respectively in the above equations yields the following:

$$f_W = \exp\sqrt{\ln^2 f_{\bar{F}} + \ln^2 f_{\bar{R}}}$$

$$W_u = \hat{W} f_W$$

$$W_\ell = \hat{W}/f_W$$

where $W = \bar{F} \cdot \bar{R}$

$\hat{W}$ = best estimate of W (the base-case value of the affected public risk).

Analogously,

$$W^* = \sum_i (R_i \sum_j F^*_{ij})$$
$$= \bar{F}^* \cdot \bar{R}^*$$

where $\bar{R}^* = \dfrac{W^*}{\bar{F}^*} = \dfrac{\sum_i (R_i \sum_j F^*_{ij})}{\sum_i \sum_j F^*_{ij}}$

The error bounds on W^* will be:

$$W^*_u = \hat{W}^* f_{W^*}$$
$$W^*_\ell = \hat{W}^* / f_{W^*}$$

where $\hat{W}^*$ = best estimate of W^* (the adjusted-case value of the affected public risk)

$$f_{W^*} = \exp\sqrt{\ln^2 \bar{f}_{F^*} + \ln^2 \bar{f}_{R^*}}$$

Therefore, for $\Delta W = W - W^*$, the error bounds will be:

$$(\Delta W)_u = W_u - W^*_\ell$$
$$= \hat{W} f_W - \hat{W}^* / f_{W^*}$$
$$(\Delta W)_\ell = \max[0, (W_\ell - W^*_u)]$$
$$= \max[0, (\hat{W}/f_{\hat{W}} - \hat{W}^* f_{W^*})]$$

The lower bound will be zero if:

$$\hat{W}/f_W - \hat{W}^* f_{W^*} \leq 0$$
$$\hat{W}/\hat{W}^* \leq f_W f_{W^*}$$

The only difference in the risk equation for the adjusted case as compared to the base case is the adjusted-case values of the affected parameters. Thus, any difference in uncertainty between the base and adjusted-case affected risk would arise only from a difference in the uncertainties of the affected parameters between the base and adjusted cases. The adjusted-case uncertainties of the affected parameters are assumed to be similar to those for the base case. Correspondingly, the uncertainty in the adjusted-case, affected public risk will be similar to that in the base case. This implies that $f_{W*} = f_W$. Thus, so long as the following holds, the lower bound on the affected public risk will be zero:

$$\hat{W}/\hat{W}* \leq f_W^2$$

For the total public risk reduction, the error bounds become:

$$[(\Delta W)_{Total}]_u = \sum_x N_x \bar{T}_x [(\Delta W)_x]_u$$

$$[(\Delta W)_{Total}]_\ell = \sum_x N_x \bar{T}_x [(\Delta W)_x]_\ell$$

where x = the index of the representative plant-type (BWR, PWR)

N_x = the number of affected plants to which plant-type x corresponds

$\bar{T}_x$ = the average remaining operating life of plant-type x

$\left.\begin{matrix}[(\Delta W)_x]_u \\ [(\Delta W)_x]_\ell\end{matrix}\right\}$ = the upper(u)/lower(ℓ) bound on the public risk reduction for plant-type x

Standardized Uncertainty Values

In WASH-1400, a detailed uncertainty analysis was performed for the core-melt frequency via Monte Carlo simulation. An error factor of 5 was estimated at a 90% confidence level. This is comparable to more recent estimates for the general uncertainty associated with the core-melt frequency in a risk study.

WASH-1400 also provided an estimate of the uncertainty on early fatalities in the consequence analysis. An error factor of 4 was estimated at a 90% confidence level. This is at the low end of the range of more recent estimates associated with the consequence analysis in a risk study. This range of error factors is approximately 5-20.

It will be assumed that the affected core-melt frequency for a representative plant as used in this study has an uncertainty comparable to that in WASH-1400. Thus, the error factor on the affected core-melt frequency at a 90% confidence level is taken to be 5 in the base case ($f_{\bar{F}} = 5$).

Presumably the uncertainty in the average dose factor as used in this study is comparable to that for the general consequence analysis in a risk study. Thus, the error factor on the average dose factor (base case) is taken to lie in the range from 5-20 ($f_{\bar{R}}$ = 5-20) at a 90% confidence level.

The corresponding uncertainty in the base-case, affected public risk can be found from the previously derived formula:

$$f_W = \exp\sqrt{\ln^2 f_{\bar{F}} + \ln^2 f_{\bar{R}}}$$

For $f_{\bar{F}} = 5$ and $f_{\bar{R}} = 5$,

$$f_W = \exp\sqrt{\ln^2 5 + \ln^2 5}$$

$$= 10$$

For $f_{\bar{F}} = 5$ and $f_{\bar{R}} = 20$,

$$f_W = \exp\sqrt{\ln^2 5 + \ln^2 20}$$

$$= 30$$

Conservatively, the larger error factor will be assumed, $f_W = 30$.

With this standardized value, the error bounds on the public risk reduction can be obtained (at a 90% confidence level):

$$(\Delta W)_u = \hat{W}f_W - \hat{W}^*/f_W^*$$

$$= 30\hat{W} \text{ since } \hat{W} \geq \hat{W}^* \text{ and } f_{W^*} \geq f_W \geq 30$$

$$(\Delta W)_\ell = \max\,[0,(\hat{W}/f_W - \hat{W}^* f_{W^*})]$$

$$= 0 \text{ if } \hat{W}/\hat{W}^* \leq f_W^2 \leq 30^2 \leq 900$$

It is extremely unlikely that any issue will generate a 900-fold reduction in the affected public risk from the base to the adjusted case. Even a 100-fold reduction (for the lower estimate of $f_W = 10$) is highly unlikely. Thus, the lower bound on the public risk reduction should be zero for all issues.

For the total public risk reduction, the standardized error bounds (at a 90% confidence level) become:

$$[(\Delta W)_{Total}]_u = 30\sum_x N_x \bar{T}_x \hat{W}_x$$

$$[(\Delta W)_{Total}]_\ell = 0$$

where x, N_x, and $\bar{T}_x$ are defined as before

$\hat{W}_x$ = best-estimate of the base-case, affected public risk for plant-type x

F.1.2 Occupational Dose Reduction Due to Accident-Avoidance

The occupational dose reduction due to accident-avoidance for one plant is defined as:

$$\Delta(\bar{F}D_R) = D_R\bar{F} - D_R\bar{F}^*$$

where D_R = occupational dose due to reactor cleanup and repair following an accident

$\bar{F}$, $\bar{F}^*$ = affected core-melt frequency (base and adjusted case)

This equation is analogous to that for $V = S - T$ with V, S, and T replaced by $\Delta(\bar{F}D_R)$, $D_R\bar{F}$, and $D_R\bar{F}^*$, respectively. Thus, the error bounds on $\Delta(\bar{F}D_R)$ will be:

$$\left[\Delta(\bar{F}D_R)\right]_u = (D_R\bar{F})_u - (D_R\bar{F}^*)_\ell$$

$$\left[\Delta(\bar{F}D_R)\right]_\ell = \max\left[0, \left\{ (D_R\bar{F})_\ell - (D_R\bar{F}^*)_u \right\}\right]$$

Since $\bar{F} \geq \bar{F}^*$, $D_R\bar{F}$ must be $\geq D_R\bar{F}^*$, restricting $\Delta(\bar{F}D_R)$ to non-negative values. Thus, $[\Delta(\bar{F}D_R)]_\ell$ can be no lower than zero.

$\bar{F}$ (and $\bar{F}^*$) and D_R are assumed to be lognormal with error factors $f_{\bar{F}}$ (and $f_{\bar{F}^*}$) and f_{D_R}, respectively. Thus, the equation for $D_R\bar{F}$ (and $D_R\bar{F}^*$) is analogous to that for $Z = S \cdot T$ when Z, S, and T are replaced by $D_R\bar{F}$ (or $D_R\bar{F}^*$), D_R, and $\bar{F}$ (or F^*) respectively. The following error bounds result:

$$(D_R\bar{F})_u = \hat{D}_R\hat{\bar{F}} f_{D_R\bar{F}}$$

$$(D_R\bar{F})_\ell = \hat{D}_R\hat{\bar{F}} / f_{D_R\bar{F}}$$

where $D_R\bar{F} = \exp \sqrt{\ln^2 f_{D_R} + \ln^2 f_{\bar{F}}}$

Similarly,

$$(D_R\bar{F}^*)_u = \hat{D}_R\hat{\bar{F}}^* f_{D_R\bar{F}^*}$$

$$(D_R\bar{F}^*)_\ell = \hat{D}_R\hat{\bar{F}}^*/f_{D_R\bar{F}^*}$$

where $D_RF^* = \exp \sqrt{\ln^2 f_{D_R} + \ln^2 f_{F^*}}$

Therefore, for $\Delta(\bar{F}D_R) = D_R\bar{F} - D_R\bar{F}^*$, the error bounds will be:

$$\left[\Delta(\bar{F}D_R)\right]_u = \hat{D}_R(\hat{\bar{F}} f_{D_R\bar{F}} - \hat{\bar{F}}^*/f_{D_R\bar{F}^*})$$

$$\left[\Delta(\bar{F}D_R)\right]_\ell = \max \left[0, \hat{D}_R(\hat{\bar{F}}/f_{D_R\bar{F}} - \hat{\bar{F}}^* f_{D_R\bar{F}^*})\right]$$

The lower bound will be zero if:

$$\hat{\bar{F}}/f_{D_R\bar{F}} - \hat{\bar{F}}^* f_{D_R\bar{F}^*} \leq 0$$

$$\hat{\bar{F}}/\hat{\bar{F}}^* \leq f_{D_R\bar{F}} f_{D_R\bar{F}^*}$$

In Section F.1.1 it was concluded that $f_{W^*} = f_W$. The argument presented there for the affected public risk is equally applicable to the affected core-melt frequency. Thus, it is assumed that $f_{\bar{F}^*} = f_{\bar{F}}$. This implies that $f_{D_R\bar{F}^*} = f_{D_R\bar{F}}$ from their formulae. Thus, so long as the following holds, the lower bound on the occupational risk reduction due to accident-avoidance will be zero:

$$\hat{\bar{F}}/\hat{F}^* \leq f^2_{D_R F}$$

For the total occupational dose reduction due to accident-avoidance (ΔU), the error bounds become:

$$(\Delta U)_u = \sum_x N_x T_x \left[\Delta(\bar{F}D_R)_x\right]_u$$

$$(\Delta U)_\ell = \sum_x N_x \bar{T}_x \left[\Delta(\bar{F}D_R)_x\right]_\ell$$

where x, N_x, and T_x are defined as in Section F.1.1.

$$\left.\begin{matrix}\left[\Delta(\bar{F}D_R)_x\right]_u \\ \left[\Delta(\bar{F}D_R)_x\right]_\ell\end{matrix}\right\}$$ = the upper(u)/lower(ℓ) bound on the occupational dose reduction due to accident-avoidance for plant-type x

Standardized Uncertainty Values

The error factor on the base-case, affected core-melt frequency ($f_{\bar{F}}$) has been assumed to have a standardized value of 5 (at a 90% confidence level). For the occupational dose due to cleanup and repair of the reactor (D_R), the error factor (f_{D_R}) is presumed to have a standardized value of 2 (at a 90% confidence level). The corresponding uncertainty in $D_R F$ becomes (at a 90% confidence level):

$$f_{D_R \bar{F}} = \exp \sqrt{\ln^2 f_{D_R} + \ln^2 f_{\bar{F}}}$$

$$= \exp \sqrt{\ln^2 2 + \ln^2 5}$$

$$= 6$$

With this standardized value, the error bounds on the occupational dose reduction due to accident-avoidance $\Delta(\bar{F}D_R)$ can be obtained (at a 90% confidence level):

$$\left[\Delta(\bar{F}D_R)\right]_u = \hat{D}_R(\hat{\bar{F}} f_{D_R\bar{F}} - \hat{\bar{F}}^*/f_{D_R\bar{F}^*})$$

$$= 6\hat{D}_R\hat{\bar{F}} \text{ since } \hat{\bar{F}} \geq \hat{\bar{F}}^* \text{ and } f_{D_R\bar{F}^*} \geq f_{D_R\bar{F}}$$

$$\geq 6$$

$$\left[\Delta(\bar{F}D_R)\right]_\ell = \max\left[0,\ \hat{D}_R(\hat{\bar{F}}/f_{D_R\bar{F}} - \hat{\bar{F}}^* f_{D_R\bar{F}^*})\right]$$

$$= 0 \text{ if } \hat{\bar{F}}/\hat{\bar{F}}^* \leq f^2_{D_R\bar{F}}$$

$$\leq 6^2$$

$$\leq 36$$

A 36-fold reduction in the affected core-melt frequency from the base to the adjusted case is unlikely for any issue. Thus, the lower bound on the reduction in occupational dose due to accident-avoidance should be zero for all issues.

For the total occupational dose reduction due to accident-avoidance, the standardized error bounds (at a 90% confidence level) become:

$$(\Delta U)_u = 6\hat{D}_R \sum_x N_x \bar{T}_x \hat{\bar{F}}_x$$

$$(\Delta U)_\ell = 0$$

where x, N_x, $\bar{T}_x$, and $\hat{D}_R$ (best estimate) are defined as before

$\hat{\bar{F}}_x$ = best estimate of the base-case, affected core-melt frequency for plant-type x

F.1.3 Industry Cost Savings Due to Accident-Avoidance

The industry cost savings due to accident-avoidance for one plant is defined as:

$$\Delta(\bar{F}A) = A\bar{F} - A\bar{F}^*$$

$\bar{F}$ and $\bar{F}^*$ are lognormal variables with error factors $f_{\bar{F}}$ and $f_{\bar{F}^*}$, respectively. A, the industry cost for reactor cleanup, repair, and refurbishment (plus replacement power) following a core-melt accident, is also taken to be lognormal with an error factor f_A. Thus, the equation for $A\bar{F}$ (and $A\bar{F}^*$) is analogous to that for $Z = S \cdot T$ where Z, S, and T are replaced by $A\bar{F}$ (or $A\bar{F}^*$), A, and $\bar{F}$ (or $\bar{F}^*$) respectively. The following error bounds result:

$$(A\bar{F})_u = \hat{A}\hat{\bar{F}} f_{A\bar{F}}$$

$$(\hat{A}\hat{\bar{F}})_\ell = \hat{A}\hat{\bar{F}} / f_{A\bar{F}}$$

where $f_{A\bar{F}} = \exp\sqrt{\ln^2 f_A + \ln^2 f_{\bar{F}}}$

Similarly,

$$(A\bar{F}^*)_u = \hat{A}\hat{\bar{F}}^* f_{A\bar{F}^*}$$

$$(A\bar{F}^*)_\ell = \hat{A}\hat{\bar{F}}^* / f_{A\bar{F}^*}$$

where $f_{A\bar{F}^*} = \exp \sqrt{\ln^2 f_A + \ln^2 f_{\bar{F}^*}}$

Therefore, for $\Delta(\bar{F}A) = A\bar{F} - A\bar{F}*$, the error bounds will be:

$$\lfloor \Delta(\bar{F}A) \rfloor_u = \hat{A}(\hat{\bar{F}} f_{A\bar{F}} - \hat{\bar{F}}*/f_{A\bar{F}*})$$

$$\lfloor \Delta(\bar{F}A) \rfloor_\ell = \max \lfloor 0, \hat{A}(\hat{\bar{F}}/f_{A\bar{F}} - \hat{\bar{F}}* f_{A\bar{F}*}) \rfloor$$

The lower bound will be zero if:

$$\hat{\bar{F}}/f_{A\bar{F}} - \hat{\bar{F}}* f_{A\bar{F}*} \leq 0$$

$$\hat{\bar{F}}/\hat{\bar{F}}* \leq f_{A\bar{F}} f_{A\bar{F}*}$$

Since it has been concluded that $f_{\bar{F}*} = f_{\bar{F}}$, it follows that $f_{A\bar{F}*} = f_{A\bar{F}}$. Thus, so long as the following holds, the lower bound on $\Delta(\bar{F}A)$ will be zero:

$$\hat{\bar{F}}/\hat{\bar{F}}* \leq f^2_{A\bar{F}}$$

For the total industry cost savings due to accident-avoidance (ΔH), the error bounds become:

$$(\Delta H)_u = \sum_x N_x \bar{T}_x \left[\Delta(\bar{F}A)_x\right]_u$$

$$(\Delta H)_\ell = \sum_x N_x \bar{T}_x \left[\Delta(\bar{F}A)_x\right]_\ell$$

where x, N_x, and $\bar{T}_x$ are defined as in Section F.1.1

$\left.\begin{matrix}\left[\Delta(\bar{F}A)_x\right]_u \\ \left[\Delta(\bar{F}A)_x\right]_\ell\end{matrix}\right\}$ = the upper(u)/lower(ℓ) bound on the industry cost savings due to accident-avoidance for plant-type x

Standardized Uncertainty Values

The standardized error factor on the affected core-melt frequency has been assumed to be 5 (at a 90% confidence level), i.e., $f_{\bar{F}} = 5$. It will be assumed that the industry cost for reactor cleanup, repair, and refurbishment, plus replacement power, following a core-melt accident has a standardized error factor of 2 (at a 90% confidence level), i.e., $f_A = 2$. The standardized error factor on $A\bar{F}$ becomes (at a 90% confidence level):

$$f_{A\bar{F}} = \exp \sqrt{\ln^2 f_A + \ln^2 f_{\bar{F}}}$$

$$= \exp \sqrt{\ln^2 2 + \ln^2 5}$$

$$= 6$$

The resulting error bounds on $\Delta(\bar{F}A)$ become:

$$\Delta(\bar{F}A)_u = \hat{A}(\hat{\bar{F}} f_{A\bar{F}} - \hat{\bar{F}}^\star / f_{A\bar{F}\star})$$

$$= 6\hat{A}\hat{\bar{F}} \text{ since } \hat{\bar{F}} \geq \hat{\bar{F}}^\star \text{ and } f_{A\bar{F}\star} \geq f_{A\bar{F}}$$

$$\geq 6$$

$$\Delta(\bar{F}A)_\ell = \max \left[0,\ \hat{A}(\hat{\bar{F}}/f_{A\bar{F}} - \hat{\bar{F}}^\star f_{A\bar{F}\star})\right]$$

$$= 0 \text{ if } \hat{\bar{F}}/\hat{\bar{F}}^\star \leq f_{A\bar{F}}^2$$

$$\leq 6^2$$

$$\leq 36$$

A 36-fold reduction in the affected core-melt frequency from the base to the adjusted case is unlikely for any issue. Thus, this lower bound should be zero for all issues.

For the total industry cost savings due to accident-avoidance, the standardized error bounds become:

$$(\Delta H)_u = 6\hat{A} \sum_x N_x \bar{T}_x \hat{F}_x$$

$$(\Delta H)_\ell = 0$$

where x, N_x, $\bar{T}_x$, and $\hat{A}$ (best estimate) are defined as before

$\hat{F}_x$ = best estimate of the base-case, affected core-melt frequency for plant-type x

F.2 UNCERTAINTIES IN REMAINING DOSE AND COST PARAMETERS

The remaining parameters related to dose and cost are the following:

- G, the occupational dose increase for implementation, operation, and maintenance of the SIR
- S_I, future cost to the industry for SIR implementation, operation, and maintenance
- S_N, future cost to the NRC for SIR development, support of SIR implementation, and review of SIR operation and maintenance.

The following generic approach forms the basis for estimating uncertainties in these terms.

In general, if a variable M is assumed to have an incremental uncertainty of d_M which is the same for its upper and lower bounds, these bounds will be (Green 1972):

$$M_u = \hat{M} + d_M$$

$$M_\ell = \hat{M} - d_M$$

where $\hat{M}$ is the best estimate.

Consider a second random variable Q with incremental uncertainty d_Q and error bounds as follows:

$$Q_u = \hat{Q} + d_Q$$

$$Q_\ell = \hat{Q} - d_Q$$

The incremental uncertainty on the sum of M and Q can be approximated as follows so long as $d_M < \hat{M}$ and $d_Q < \hat{Q}$:

$$d_{(M+Q)} = \sqrt{d_M^{\ 2} + d_Q^{\ 2}}$$

The error bounds on the sum will be:

$$(M+Q)_u = (\hat{M}+\hat{Q}) + d_{(M+Q)}$$

$$(M+Q)_\ell = (\hat{M}+\hat{Q}) - d_{(M+Q)}$$

The incremental uncertainty on the product of M and Q can be approximated as follows so long as $d_M < \hat{M}$ and $d_Q < \hat{Q}$ (Green 1972):

$$d_{MQ} = \hat{M}\hat{Q}\sqrt{\left(\frac{d_M}{\hat{M}}\right)^2 + \left(\frac{d_Q}{\hat{Q}}\right)^2}$$

The error bounds on the product will be:

$$(MQ)_u = \hat{M}\hat{Q} + d_{MQ}$$

$$(MQ)_\ell = \hat{M}\hat{Q} - d_{MQ}$$

These formulae will be used in deriving uncertainty estimates for the dose and cost terms.

F.2.1 Occupational Dose Increase for SIR Implementation, Operation, and Maintenance

The total occupational dose increase for SIR implementation, operation, and maintenance is defined as:

$$G = N(\bar{T}D_o + D)$$

where N = number of reactors affected by the SIR
$\bar{T}$ = average remaining operating life of reactors affected
D_o = annual incremental dose increase for SIR operation/maintenance
D = incremental dose increase for SIR implementation

D_o and D are further defined as follows:

$$D_o = L_o M_o$$

where L_o = annual utility staff labor for operation and maintenance of the SIR
M_o = occupational dose rate for the location where the operation and maintenance are performed.

$$D = LM$$

where L = utility staff labor needed to install and test the safety fix
M = occupational dose rate for the location where the installation and testing are performed.

L_o, L, M_o, and M are assumed to be lognormal with error factors f_{L_o}, f_L, f_{M_o}, and f_M, respectively. Thus, the equations for D_o and D are analogous to that for $Z = S \cdot T$ when Z, S, and T are replaced by D_o (or D), L_o (or L), and M_o (or M), respectively. The following error bounds result:

$$(D_o)_u = \hat{D}_o f_{D_o}$$

$$(D_o)_\ell = \hat{D}_o / f_{D_o}$$

where $f_{D_o} = \exp\sqrt{\ln^2 f_{L_o} + \ln^2 f_{M_o}}$

$$D_u = \hat{D} f_D$$

$$D_\ell = \hat{D} / f_D$$

where $f_D = \exp\sqrt{\ln^2 f_L + \ln^2 f_M}$

Rewriting $G = N(\bar{T}D_o + D)$ as the following:

$$G = N\bar{T}D_o + ND$$

one can find the error bounds on G as follows.

For the sum of two variables $Y = S + T$:

$$\hat{Y} = \hat{S} + \hat{T}$$

$$Y_u \leq S_u + T_u$$

$$Y_\ell \geq S_\ell + T_\ell$$

Conservative limits can be set as follows:

$$(Y_u)_{max} = S_u + T_u$$

$$(Y_\ell)_{min} = S_\ell + T_\ell$$

For convenience, the "max" and "min" subscripts are dropped and the error bounds on Y are taken as:

$$Y_u = S_u + T_u$$

$$Y_\ell = S_\ell + T_\ell$$

If S and T are lognormal,

$$Y_u = \hat{S}f_S + \hat{T}f_T$$

$$\leq (\hat{S} + \hat{T}) \max(f_S, f_T)$$

$$\leq \hat{Y} \max(f_S, f_T)$$

$$Y_\ell = \hat{S}/f_S + \hat{T}/f_T$$

$$\geq (\hat{S} + \hat{T})/\max(f_S, f_T)$$

$$\geq \hat{Y}/\max(f_S, f_T)$$

Since Y is also taken to be lognormal, one obtains:

$$Y_u = \hat{Y}f_Y \leq \hat{Y} \max(f_S, f_T)$$

$$f_Y \leq \max(f_S, f_T)$$

$$Y_\ell = \hat{Y}/f_Y \geq \hat{Y}/\max(f_S, f_T)$$

$$f_Y \leq \max(f_S, f_T)$$

Thus,

$$(f_Y)_{max} = \max(f_S, f_T)$$

Dropping the "max" subscript yields the following error factor for Y:

$$f_Y = \max(f_S, f_T)$$

Replacing Y, S, and T by G, $N\bar{T}D_o$ and ND respectively, the error bounds on G become the following:

$$G_u = \hat{G}f_G$$

$$G_\ell = \hat{G}/f_G$$

where $\hat{G} = N(\bar{T}\hat{D}_o + \hat{D})$

$f_G = \max(f_{N\bar{T}D_o}, f_{ND}) = \max(f_{D_o}, f_D)$, since N and $\bar{T}$ are constants

$(f_N = f_{\bar{T}} = 1)$

If $\hat{G}<0$, the error bounds are modified as follows:

$$G_u = \hat{G}/f_G$$

$$G_\ell = \hat{G}f_G$$

where $\hat{G}$ and f_G are defined as above.

Standardized Uncertainty Values

The error factors on the terms related to occupational dose rates and staff hours involving implementation of issue resolution are all presumed to be around 2 (at a 90% confidence level):

$$\left.\begin{matrix} f_{L_o} \\ f_L \\ f_{M_o} \\ f_M \end{matrix}\right\} = 2$$

Thus, the error bounds on D_o and D become (at a 90% confidence level):

$$f_{D_o} = \exp\sqrt{\ln^2 f_{L_o} + \ln^2 f_{M_o}}$$

$$= \exp\sqrt{\ln^2 2 + \ln^2 2}$$

$$= 3$$

$$(D_o)_u = \hat{D}_o f_{D_o}$$

$$= 3\hat{D}_o$$

$$(D_o)_\ell = \hat{D}_o / f_{D_o}$$

$$= \hat{D}_o / 3$$

$$f_D = \exp\sqrt{\ln^2 f_L + \ln^2 f_M}$$

$$= \exp\sqrt{\ln^2 2 + \ln^2 2}$$

$$= 3$$

$$D_u = \hat{D} f_D$$

$$= 3\hat{D}$$

$$D_\ell = \hat{D} / f_D$$

$$= \hat{D}/3$$

The standardized error bounds (at a 90% confidence level) on the total occupational dose increase G become:

$$G_u = 3\hat{G}$$

$$G_\ell = \hat{G}/3$$

where $\hat{G} = N(\bar{T}\hat{D}_o + \hat{D})$

If $\hat{G}<0$, the error bounds are modified as follows:

$$G_u = \hat{G}/3$$
$$G_\ell = 3\hat{G}$$

where $\hat{G}$ is defined as above.

F.2.2 Industry Cost for SIR Implementation, Operation, and Maintenance

The total industry cost for SIR implementation, operation, and maintenance is defined as:

$$S_I = N(\bar{T}I_o + I)$$

where N and $\bar{T}$ are defined as in Section F.2.1

I_o = annual incremental industry cost for SIR operation/maintenance

I = incremental industry cost for SIR implementation.

The cost terms I_o and I are assumed to have incremental uncertainties d_{I_o} and d_I respectively such that:

$$(I_o)_u = \hat{I}_o + d_{I_o}$$

$$(I_o)_\ell = \hat{I}_o - d_{I_o}$$

$$I_u = \hat{I} + d_I$$

$$I_\ell = \hat{I} - d_I$$

Rewriting $S_I = N(\bar{T}I_o + I)$ as the following:

$$S_I = N\bar{T}I_o + NI$$

one can find the error bounds on S_I by using the formulae derived in Section F.2 for uncertainties on the sum and product of variables with incremetal uncertainties.

$$(S_I)_u = \hat{S}_I + d_{S_I}$$

$$(S_I)_\ell = \hat{S}_I - d_{S_I}$$

where $\hat{S}_I = N(\bar{T}\hat{I}_o + \hat{I})$

$$d_{S_I} = \sqrt{d^2_{N\bar{T}I_o} + d^2_{NI}}$$

$$= \sqrt{(N\bar{T}d_{I_o})^2 + (Nd_I)^2}, \text{ since N and } \bar{T} \text{ are constants } (d_N = d_{\bar{T}} = 0)$$

<u>Standardized Uncertainty Values</u>

It will be assumed that the incremental uncertainties in I_o and I (at a 90% confidence level) are 50% of the best estimates. Thus,

$$d_{I_o} = \hat{I}_o/2$$

$$d_I = \hat{I}/2$$

The standardized error bounds (at a 90% confidence level) on the total industry cost S_I become:

$$(S_I)_u = \hat{S}_I + d_{S_I}$$

$$(S_I)_\ell = \hat{S}_I - d_{S_I}$$

where $\hat{S}_I = N(\bar{T}\hat{I}_o + \hat{I})$

$$d_{S_I} = \sqrt{(N\bar{T}\hat{I}_o/2)^2 + (N\hat{I}/2)^2}$$

$$= \frac{1}{2}\sqrt{(N\bar{T}\hat{I}_o)^2 + (N\hat{I})^2}$$

F.2.3 NRC Cost for SIR Development, Support of SIR Implementation, and SIR Operation/Maintenance Review

The total NRC cost related to SIR development, implementation, operation and maintenance is defined as:

$$S_N = C_D + N(\bar{T}C_o + C)$$

where N and $\bar{T}$ are defined as in Section F.2.1

C_D = future NRC cost for SIR development

C_o = annual incremental NRC cost for review of SIR operation/maintenance

C = incremental NRC cost for support of SIR implementation.

The cost terms C_D, C_o, and C are assumed to have incremental uncertainties d_{C_D}, d_{C_o}, and d_C respectively such that:

$$(C_D)_u = \hat{C}_D + d_{C_D}$$

$$(C_D)_\ell = \hat{C}_D - d_{C_D}$$

$$(C_o)_u = \hat{C}_o + d_{C_o}$$

$$(C_o)_\ell = \hat{C}_o - d_{C_o}$$

$$C_u = \hat{C} + d_C$$

$$C_\ell = \hat{C} - d_C$$

Rewriting $\hat{S}_N = \hat{C}_D + N(\bar{T}\hat{C}_o + \hat{C})$ as the following:

$$S_N = C_D + N\bar{T}C_o + NC$$

one can find the error bounds on S_N by using the formulae derived in Section F.2 for uncertainties on the sum and product of variables with incremental uncertainties.

$$(S_N)_u = \hat{S}_N + d_{S_N}$$

$$(S_N)_\ell = \hat{S}_N - d_{S_N}$$

where $\hat{S}_N = \hat{C}_D + N(\bar{T}\hat{C}_o + \hat{C})$

$$d_{S_N} = \sqrt{d^2_{C_D} + d^2_{N\bar{T}C_o} + d^2_{NC}}$$

$$= \sqrt{d_{C_D}^2 + (N\bar{T}d_{C_o})^2 + (ND_c)^2}, \text{ since N and } \bar{T} \text{ are constants } (d_N = d_{\bar{T}} = 0)$$

Standardized Uncertainty Values

It will be assumed that the incremental uncertainties in C_D, C_o, and C (at a 90% confidence level) are 50% of the best estimates. Thus,

$$d_{C_D} = \hat{C}_D/2$$

$$d_{C_o} = \hat{C}_o/2$$

$$d_C = \hat{C}/2$$

The standardized error bounds (at a 90% confidence level) on the total NRC cost S_N become:

$$(S_N)_u = \hat{S}_N + d_{S_N}$$

$$(S_N)_\ell = \hat{S}_N - d_{S_N}$$

where $\hat{S}_N = \hat{C}_D + N(\bar{T}\hat{C}_o + \hat{C})$

$$d_{S_N} = \sqrt{(\hat{C}_D/2)^2 + (N\bar{T}\hat{C}_o/2)^2 + (N\hat{C}/2)^2}$$

$$= \frac{1}{2}\sqrt{\hat{C}_D{}^2 + (N\bar{T}\hat{C}_o)^2 + (N\hat{C})^2}$$

F.3 UNCERTAINTIES ON COMBINATIONS OF SAFETY ISSUE RANKING PARAMETERS

The preceding sections developed generic methods for estimating uncertainties on the six parameters for use in determining safety issue priorities: $(\Delta W)_{Total}$, ΔU, ΔH, G, S_I, and S_N. Standardized values have been given to facilitate the uncertainty analyses.

Both best estimates and error bounds will be calculated for these parameters using the techniques developed in this document. Several options exist for using these best estimates and error bounds to establish safety issue priorities. One is the arithmetic combination of two or more of these parameters to define some ranking measure. The arithmetic combination of the best estimates is straightforward. However, techniques for arithmetically combining uncertainties are not. In lieu of rigorous Monte Carlo methods, some approximate procedures are given for arithmetically combining uncertainties. Three cases will be discussed:

1. combining uncertainties for random variables with unknown distributions (the "general" case)
2. combining uncertainties for random variables with distributions symmetric about the best estimates (the "symmetric" case)
3. combining uncertainties for random variables with lognormal distributions (the "lognormal" case)

Under the assumptions made in the preceeding sections, the parameters related to accident-avoidance (the delta parameters) have no known distributions; G is lognormally distributed in the standardized case; and both S_I and S_N have symmetric distributions. Of course, a change in assumptions could alter these results. Still, under current assumptions, any combination of parameters involving at least one of the delta variables will normally require use of the "general" case for uncertainty analysis. However, the potential exists for use of the "symmetric" and "lognormal" techniques for uncertainty analysis on combinations of parameters.

F.3.1 The General Case

Only conservative limits can be placed on the error bounds when variables with unknown distributions are combined. Consider two random variables X and Y, the best estimates $\hat{X}$ and $\hat{Y}$ and error bounds X_u, X_ℓ, Y_u and Y_ℓ. Nothing is known about their distributions. The following equations will hold for the sum, difference, product and quotient of these variables:

Sum	Difference
$Z = X + Y$	$Z = X - Y$
$\hat{Z} = \hat{X} + \hat{Y}$	$\hat{Z} = \hat{X} - \hat{Y}$
$Z_u \leq X_u + Y_u$	$Z_u \leq X_u - Y_\ell$
$Z_\ell \geq X_\ell + Y_\ell$	$Z_\ell \geq X_\ell - Y_u$

Product	Quotient
$Z = XY$	$Z = X/Y$
$\hat{Z} = \hat{X}\hat{Y}$	$\hat{Z} = \hat{X}/\hat{Y}$
$Z_u \leq X_u Y_u$	$Z_u \leq X_u/Y_\ell$
$Z_\ell \geq X_\ell Y_\ell$	$Z_\ell \geq X_\ell/Y_u$

Conservative limits on the error bounds result if the equalities are assumed in the equations for the error bounds. For the difference, the lower bound Z_ℓ cannot be negative if Z is so defined as to be a non-negative parameter (e.g., the public risk reduction).

F.3.2 The Symmetric Case

The random variables X and Y are now assumed to have symmetric distributions with incremental uncertainties d_x and d_y about their best estimates. The error bounds on X and Y are then:

$$X_u = \hat{X} + d_x \qquad Y_u = \hat{Y} + d_y$$
$$X_\ell = \hat{X} - d_x \qquad Y_\ell = \hat{Y} - d_y$$
$$(d_x < \hat{X}) \qquad (d_y < \hat{Y})$$

The arithmetic combinations of these variables will also have symmetric distributions. The error bounds on the sum, difference, product, and quotient of these variables will be (Green 1972):

Sum/Difference

$$Z = X \pm Y$$
$$Z_u = \hat{Z} + d_z$$
$$Z_\ell = \hat{Z} - d_z$$

$$\text{where } \hat{Z} = \hat{X} \pm \hat{Y}$$

$$d_z = \sqrt{d_x^2 + d_y^2}$$

Product/Quotient

$$Z = XY \text{ or } X/Y$$

$$Z_u = \hat{Z} + d_z$$

$$Z_\ell = \hat{Z} - d_z$$

$$\text{where } \hat{Z} = \hat{X}\hat{Y} \text{ or } \hat{X}/\hat{Y}$$

$$d_z = \hat{X}\hat{Y}\sqrt{\left(\frac{d_x}{\hat{X}}\right)^2 + \left(\frac{d_y}{\hat{Y}}\right)^2}$$

F.3.3 The Lognormal Case

The random variables X and Y are now assumed to have lognormal distributions with error factors f_x and f_y. The error bounds on X and Y are then:

$$X_u = \hat{X}f_x \qquad Y_u = \hat{Y}f_y$$

$$X_\ell = \hat{X}/f_x \qquad Y_\ell = \hat{Y}/f_y$$

No simple approximation exists for the error bounds on the sum or difference unless one variable clearly dominates the other. In that case, the sum/difference is approximately lognormal and the bounds become (Pepping 1981):

Sum/Difference (X >> Y)

$$Z = X \pm Y$$

$$Z_u = \hat{Z}f_z$$

$$Z_\ell = \hat{Z}/f_z$$

$$\text{where } \hat{Z} = \hat{X} \pm \hat{Y}$$

$$f_z = \exp\left[\frac{1}{2}\sqrt{(\hat{X}\ \ln f_x)^2 + (\hat{Y}\ \ln f_y)^2}\right]$$

No restriction of variable dominance applies for estimating the error bounds on the product or quotient, which is itself lognormal. These bounds are (Pepping 1981):

<u>Product/Quotient</u>

$$Z = XY \text{ or } X/Y$$

$$Z_u = \hat{Z} f_z$$

$$Z_\ell = \hat{Z}/f_z$$

$$\text{where } \hat{Z} = \hat{X}\hat{Y} \text{ or } \hat{X}/\hat{Y}$$

$$f_z = \exp\sqrt{\ln^2 f_x + \ln^2 f_y}$$

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16. ABSTRACT *(200 words or less)*

Pacific Northwest Laboratory has developed a methodology, with examples, to calculate the risk, dose and cost impacts of implementing resolutions to reactor safety issues. This report is an applications guide to issue-specific calculations. A description of the approach, mathematical models, work sheets and step-by-step examples are provided.

Analyses using this method are intended to provide comparable results for many issues at a cost of two staff-weeks per issue. Results will be used by the NRC to support decisions related to issue priorities in allocation of resources to complete safety issue resolutions.

17. KEY WORDS AND DOCUMENT ANALYSIS

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