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Value-Impact Assessment for a Candidate Operating Procedure Upgrade Program

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Prepared for
U.S. Nuclear Regulatory Commission

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RG

VALUE-IMPACT ASSESSMENT FOR A CANDIDATE OPERATING PROCEDURE UPGRADE PROGRAM

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ABSTRACT

This report documents a value-impact assessment that was undertaken to assist the U.S. Nuclear Regulatory Commission (NRC) in determining whether it should implement regulatory action that would specify requirements for the preparation of acceptable normal and abnormal operating procedures by the NRC's licensee nuclear power plants. The following steps were used in this assessment: 1) the NRC regulatory action was defined as the NRC requiring each U.S. nuclear power plant to undertake a candidate program to upgrade its normal and abnormal operating procedures, 2) the attributes effected by this action were identified, 3) the potential effects on the attributes were estimated, and 4) sensitivity analyses were performed to show how changes in important data would affect the expected changes in the attributes. These individual evaluations were then summarized and the value-impact results displayed.

EXECUTIVE SUMMARY

This report documents a value-impact assessment that was prepared for the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research. The objective of the assessment was to aid the NRC in determining whether it should implement regulatory action that would specify requirements for the preparation of acceptable normal and abnormal operating procedures by the NRC's licensee nuclear power plants.

STATEMENT OF THE PROBLEM

As part of the response to Three Mile Island Action Plan Item I.C.9, the NRC sponsored several projects aimed at assessing the current practices and problems associated with normal and abnormal operating procedures in U. S. nuclear power plants. The results of two of these projects were published in NUREG/CR-3968, Study of Operating Procedures in Nuclear Power Plants: Practices and Problems (Morgenstern et al. 1987) and in NUREG/CR-4613, Evaluation of Nuclear Power Plant Operating Procedures Classification and Interfaces: Problems and Techniques for Improvement (Barnes and Radford 1987). These studies indicated that operating procedures as well as the practices employed to guide the development, use, and administrative control of operating procedures in most, if not all, U.S. nuclear power plants are of unacceptably poor quality. The major substantive deficiencies in current operating procedures were found to include:

- Useability of operating procedures, considered as a group, falls within the minimally acceptable range.
- Operating procedures are often written in vague terms and lack specificity. They often fail to describe specific operator actions to be taken by operators in a step-by-step manner.
- Many operating procedures fail to provide clear indicators of when a particular objective has been achieved and when actions governed by the procedure have been completed and the procedure should be exited.
- Operating procedures have been found to be technically inaccurate.
- The large number and complexity of operating procedures at some plants create problems for procedure approval, use, and revision.
- Procedure classification schemes are deficient in human factors characteristics at many plants making it difficult for operators to make transitions between different classes of procedures.

CANDIDATE REGULATORY ACTION

To aid the NRC in determining whether it should implement regulatory action aimed at improving the current quality of operating procedures and to assess the costs and benefits of such regulatory action, an approach to upgrading procedures was developed. This approach consists of a five-part candidate Procedure Upgrade Program. In the program's first part, the NRC would develop a "good practices" document to assist licensees in the implementation of the Procedure Upgrade Program. In the second part, the NRC would convene an NRC-industry working group that would define the criteria by which licensees could identify which operations tasks should be proceduralized. Each individual plant would apply those criteria by identifying those operations tasks that are candidates for proceduralization in the third part of the program. The plants would also develop a plant-specific writers' guide and train its procedures writers in this part. In the next part, trained licensee personnel would upgrade operating procedures in accordance with the guidance developed in prior tasks. In the program's fifth part, the NRC would undertake a one-time inspection of all licensees' upgraded operating procedures. For purposes of determining the potential costs and benefits that would result from such a program, it was assumed the candidate program would apply to all licensee plants.

The objective of this candidate regulatory action would be to reduce the risk to public health and safety associated with nuclear power plant operations by improving the quality of operating procedures. Such improvement could be expected to result in reductions in operator error and in more efficient and effective performance of operator actions during normal and abnormal plant operations.

ALTERNATIVES

Only one alternative to the candidate Procedure Upgrade Program was considered. This alternative is to maintain the status quo as to current operating procedure and procedure development program quality in licensee nuclear power plants. In this value-impact analysis this alternative served as the baseline against which the candidate regulatory action was measured. Therefore, all values and impacts presented in this analysis were calculated relative to the status quo.

METHODOLOGY

The methodology used to develop estimates of the values and impacts associated with the candidate Procedure Upgrade Program was taken from the Handbook for Value-Impact Assessment (Heaberlin et al. 1983). Benefits were estimated based on the potential reduction to core-melt frequency and public health risks that would result from improving normal and abnormal operating procedures. Such improvements to operator performance were assumed to result from implementation of the program and subsequent use of higher quality procedures. Such improvements in operator performance were assumed to result in a reduction in the procedure-related contribution to specific transient initiating-events as well as to the procedure-related contribution to specific operator actions.

An 11-step approach was developed in order to quantitatively evaluate the effect of the candidate Procedure Upgrade Program on the annual core-melt frequency at nuclear power plants. The approach, which consisted of a sensitivity analysis technique, was used to: 1) estimate the current contribution to reactor core-melt frequency of procedure-related operational errors during normal and abnormal operating conditions, 2) estimate the same contribution assuming that the candidate program was implemented and operating procedures had been improved by specific measures, and 3) compare the "before" and "after" estimates of overall core-melt frequency to determine the potential reduction in core-melt frequency associated with the proposed improvements to current operating procedures. Each of these elements of the approach contained several unique steps.

CONSEQUENCES

Table S.1 shows the quantified results for each of the various values and impacts that would be affected by the candidate regulatory action. This table then summarizes these results by dividing the total estimated impact of the candidate program, measured in dollars, by the total estimated value of improving operating procedures, measured in person-rems. Because the NRC compares this resulting cost per person-rem avoided with a standard measure of $1,000 per person-rem, and because this standard figure is assumed to include the estimated potential impact of the candidate regulatory action on risk to offsite property, the estimated potential impact on Public Property Damage Avoided has not been included in this table nor in these summary calculations although it was quantified as part of the assessment. These summary results indicate that the best estimate of the cost of improving operating procedures would be approximately $750 per person-rem avoided.

Potential impacts on industry operation are not quantified in Table S.1. A relatively limited analysis performed for this attribute indicated, however, that improved operating procedures could also lead to additional benefits beyond those quantified in this assessment. High quality normal and abnormal operating procedures could be expected to cause a reduction in reactor trip frequency at U.S. nuclear power plants due to a decrease in procedure-related operational errors. A reduction in trips would have both beneficial safety and economic effects. Fewer trips would mean fewer challenges to plant safety systems as well as a reduction in the frequency of technical specification violations. As a beneficial economic consequence of a reduction in trips, the industry could decrease its expenses for replacement power purchased during outages following such trips. The industry could also reduce its expenses for revising and administering operating procedures if procedures are upgraded using the two-tier approach described in this assessment. Because procedures would be of higher quality and because the

TABLE S.1. Value-Impact Summary

Attribute	Units	Best Estimate	Best Case	Worst Case
Public Risk Reduction	(person-rem)	4.3E4	6.0E4	2.0E4
Avoided Occupational Exposure	(person-rem)	1.0E3	2.3E3	2.2E2
Total Value	(person-rem)	4.4E4	6.2E4	2.0E4
Avoided Onsite Property Damage	(dollars)	-1.1E7[(a)]	-2.4E7[(a)]	-2.6E6[(a)]
Industry Implementation Costs	(dollars)	4.2E7	2.1E7	6.2E7
Industry Operation Costs	(dollars)	NQ	NQ	NQ
NRC Development Costs	(dollars)	2.5E4	1.3E4	3.8E4
NRC Implementation Costs	(dollars)	2.0E6	1.0E6	3.0E6
NRC Operation Costs	(dollars)	NA	NA	NA
Total Impact[(b)]	(dollars)	3.3E7	-2.0E6[(a)]	6.2E7
Total Impact/Total Value	(dollars/ person-rem)	7.5E2	-3.2E1[(c)]	3.1E3

NQ = Not Quantified. Industry Operation Costs were analyzed on a qualitative, rather than a quantitative, level.
NA = Not Affected.
(a) Favorable impacts have a negative sign.
(b) Because the NRC is inclined to evaluate the results of value-impact assessments against a standard of $1,000 per person-rem, and because this standard figure is assumed to include the estimated potential impact of the candidate regulatory action on risk to offsite property, the estimated potential impact on Public Property Damage Avoided has not been included in this table.
(c) -3.2E1 signifies that, for the best case estimate, the regulatory action could reduce exposure to humans by 62,000 person-rems accompanied by a $2 million net benefit.

two-tier approach would result in plants having fewer normal operating procedures that are subject to rigorous development and control, the industry could expect to spend less effort over time for revision and administration of operating procedures.

ACKNOWLEDGMENTS

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1.0 INTRODUCTION

This report documents a value-impact assessment that was prepared by the Pacific Northwest Laboratory (PNL) for the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research. The purpose of the assessment was to aid the NRC in determining whether it should implement regulatory action that would specify requirements for the preparation of acceptable normal and abnormal operating procedures by the NRC's licensee nuclear power plants. The assessment estimated the impact of such NRC regulatory action on the 11 attributes set forth in NUREG/CR-3568, A Handbook for Value-Impact Assessment (Heaberlin et al. 1983).

A multistep approach was used in the preparation of this assessment. First, the candidate NRC regulatory action was defined as the NRC requiring each U.S. nuclear power plant to undertake a program to upgrade its normal and abnormal operating procedures. Next, the specific value-impact attributes affected by this action were identified. Because it is an integral part of the determination of the change of five of these attributes, changes in core-melt frequency that could result from NRC regulatory action were estimated. The potential effect on each of the 11 attributes was then evaluated. Sensitivity analyses were performed to show how changes in important assumptions and data would affect the expected changes in the attributes. These individual evaluations were then summarized and the value-impact results displayed.

1.1 BACKGROUND

The Three Mile Island Task Action Plan Item I.C.9 required the NRC to develop a long-term plan for upgrading plant procedures (NUREG-0660, NRC 1980). Human Factors Generic Issue 4.4, "Guidelines for Upgrading Other Procedures," which originates from this plan, requires: 1) recommendation of improvements in nuclear power plant normal and abnormal operating procedures, and 2) implementation of appropriate regulatory action. The NRC responded to the Action Plan by sponsoring several projects aimed at assessing the current practices and problems associated with normal and abnormal operating procedures in U.S. nuclear power plants. The results of two of these studies were published in NUREG/CR-3968, Study of Operating Procedures in Nuclear Power Plants: Practices and Problems (Morgenstern et al. 1987), and NUREG/CR-4613, Evaluation of Nuclear Power Plant Operating Procedures Classifications and Interfaces: Problems and Techniques for Improvement (Barnes and Radford 1987). These studies describe the problem that Human Factors Generic Issue 4.4 is intended to resolve.

For purposes of this value-impact assessment, "operating procedures" refers to hard-copy, written instructions that are provided to plant personnel to assist their performance of operations tasks under nonemergency plant conditions. Thus, operating procedures include those procedures that are typically identified throughout the industry as normal operating procedures and abnormal operating and alarm response procedures.

1.2 CURRENT OPERATING PROCEDURE DEFICIENCIES

The major findings of the two studies cited above indicate that operating procedures in most, if not all, U.S. nuclear plants are of unacceptably poor quality. The useability of operating procedures is one indicator of this poor quality. Useability is a measure of the procedures' ability to provide guidance to the operator that is accurate, complete, readable, and convenient to use. A clear table of contents, consistency of format throughout the procedure, and steps to express limits quantitatively rather than qualitatively are examples of measures of useability. In practice, operating procedures are often written in vague terms and lack specificity. They often fail to describe the specific actions to be taken by operators in a step-by-step manner. They fail to provide clear indicators of when a particular objective has been achieved and when actions governed by the procedure have been completed and the procedure should be exited. This lack of specificity decreases procedure useability particularly for relatively inexperienced operators. Operating procedures also typically do not conform to accepted human factors principles. For example, they often lack useful checklists, placekeeping aids, and other tools to enhance the ease and accuracy of procedure use.

The number of procedure classifications (e.g., normal, abnormal, emergency) and the content of those classes vary widely across the country. This can have implications for NRC review of procedures as well as the movement of information and transfer of trained personnel among plants. The human factors characteristics of procedure classification schemes are also deficient in many plants. Procedures are not organized in a meaningful way and operators are often not trained in how to efficiently locate procedures. Current classification schemes can make it difficult for operators to make transitions between different classes of procedures.

The number and complexity of operating procedures can also create problems. Some plants have such a large number of procedures as to be unmanageable. Many procedures contain more detail than is necessary. Plant review of new and revised procedures is also a problem. The large number of procedures to be reviewed and the systems some plants have created for procedure review can result in months of delay in formal procedure changes on one extreme or cursory approval without needed technical review on the other.

Two other important deficiencies were also identified. Coordination of the training function with procedure preparation and use is often very limited. Thus, procedure writers seldom benefit from the actual experience of training personnel in dealing with procedure problems. Additionally, operating procedures are not verified or validated before use at some plants creating a situation that contributes to technical inaccuracies.

These deficiencies in operating procedure useability, technical accuracy, classification, number, and complexity can lead to operator error and, thereby, be a threat to public health and safety. This is born out by the results of this value impact assessment described in detail below and by various studies and analyses that, for example, cite procedure deficiencies as root causes of significant operating events in licensee event reports (Trager 1988).

2.0 OBJECTIVES AND CANDIDATE REGULATORY ACTION

This value-impact assessment documents the expected costs and benefits of implementing a candidate regulatory action to correct current operating procedure deficiencies. This section of the report describes the basic objective of that action, the action itself, the intended results of that action, and the alternative to that action.

2.1 OBJECTIVE OF THE CANDIDATE REGULATORY ACTION

The objective of the candidate regulatory action, mandating a Procedure Upgrade Program at all licensee plants, is to reduce risk to public health and safety associated with nuclear power plant operation. That risk is to be reduced by improving the quality of operating procedures that, in turn, can be expected to result in reductions in operator error and in more efficient and effective performance of operator actions during normal and abnormal plant operations.

2.2 STATEMENT OF THE CANDIDATE REGULATORY ACTION

The regulatory action focuses on a candidate NRC program involving licensees and contractors directed at upgrading normal operating procedures and abnormal operating procedures. The need for such procedure improvement was identified in NRC-sponsored projects entitled "Program Plan for Assessing and Upgrading Operating Procedures for Nuclear Power Plants" (documented in Morgenstern et al. 1987) and "Study of Operating Procedures Classifications and Interfaces" (documented in Barnes and Radford 1987). That research found several deficiencies in normal operating procedures and abnormal operating procedures and in the practices employed by licensees to guide the development, use, and administrative control of those procedures. The content of this candidate Procedure Upgrade Program, which is described in detail in Appendix A, was first suggested in summary form in NUREG/ CR-3968, Study of Operating Procedures in Nuclear Power Plants: Practices and Problems (Morgenstern et al. 1987). For purposes of determining the potential costs and benefits that would flow from such a program, it has been assumed that the candidate NRC regulatory action would occur, and that it would apply to all licensee plants.

The specific activities to be undertaken by the industry and the NRC in developing and implementing the Procedure Upgrade Program under consideration can be briefly summarized here. The program would have five parts. In Task 1, the NRC would develop a "good practices" document to assist licensees in the implementation of the Procedure Upgrade Program. In Task 2, the NRC would convene an NRC-industry working group that would define the criteria by which licensees could identify which operations tasks should be proceduralized and evaluate and suggest improvements in the good practices document. In Task 3, each individual plant would apply these criteria by identifying those operations tasks that are candidates for proceduralization. Each plant would also develop a plant-specific writers' guide and train its procedures writers. In Task 4, the trained licensee personnel would upgrade operating procedures in accordance with the guidance developed in prior tasks. In Task 5 the NRC would undertake a one-time inspection of all licensees' upgraded operating procedures. In all this upgrade program would take approximately 4.5 years to complete.

2.3 INTENDED RESULT OF THE CANDIDATE REGULATORY ACTION

The candidate Procedure Upgrade Program would be expected to result in several improvements to procedures. The large number and complexity of procedures currently being used by most licensees have been found to be a major contributor to a multiplicity of problems in procedure use and administration. As a key element of the candidate Procedure Upgrade Program, each plant would analyze its operations tasks to determine those operations that require high-quality, detailed, step-by-step procedures. These tasks would normally be those that are complex, infrequently performed, and important to public health and safety. Procedures governing such operations tasks would be designated as Tier One procedures. Other operations tasks that are not safety-related would require some form of job performance aid, such as a checklist or calculation form, but would not require procedures of the high quality demanded by safety-related operations tasks. Such procedures would be designated as Tier Two procedures.

By differentiating between operations tasks on criteria such as these and revising current or creating new procedures to achieve the rigor and quality required by the nature of those tasks, substantial resources now being used for developing, reviewing, and administratively controlling <u>all</u> procedures could be redirected and concentrated on more rigorous development and maintenance of Tier One, safety-related procedures. Thus, the overall burden on plants should be reduced due to fewer resources being devoted to procedures that govern nonsafety-related operations. This should allow plants to concentrate more concern and resources on developing and maintaining more usable and technically accurate procedures to govern safety-related operations tasks.

Improvements in procedure useability would be another result of the candidate Procedure Upgrade Program. Operating procedures would be developed with the participation of people trained in sound human factors practices. This would produce procedures written in more definite and specific terms and in short, identifiable step-by-step means of accomplishing the operations task. When possible a single quantitative indicator telling the operator when the objective of each step has been accomplished would be specified. Procedures would be written in a consistent format and a usable table of contents would be provided. Clearly prepared graphs, figures, and flow charts would be provided to increase ease of use when necessary.

As procedures are written or revised, human-system interface analyses would be performed using the upgraded procedure at the location in the plant where the procedure would be performed or, if necessary, using simulators, mockups, or models of the displays and controls involved.

Technical accuracy of procedures would be improved through use of this program. For Tier One procedures, personnel from all relevant plant functional areas and, when necessary, equipment vendors would use relevant plant technical information and specifications that have been validated for accuracy to create technically accurate procedures. NRC generic communications, vendor bulletins, licensee event reports, and operating experience from other plants could also be used when necessary.

To improve procedure administrative controls, the upgrade program would require a number of procedure administrative control mechanisms. A process to ensure that procedures affected by design modifications or changes to other procedures be flagged for review and revision would be instituted. A means of ensuring that recommendations of users and training personnel based on their experience with the upgraded procedures would be fed back to utility staff responsible for the procedures would be initiated.

All procedures would continue to be subject to biennial reviews by plant personnel, but the content of these reviews would vary for Tier One and Tier Two procedures. Review of Tier One procedures should include a check of the procedures' use and revision histories, an assessment of any comments and recommendations that had been made on the procedures as well as any resulting action taken, a check of the relationship of the procedure to other procedures, and a walk-through of the procedure in the plant to ensure that the human-system interface information continues to reflect actual conditions of use. For Tier Two procedures, biennial reviews would be substantially less rigorous. The procedure-use histories would be reviewed and any inadequacies that are detected would be acted upon. This could include either purging procedures that are not needed or upgrading procedures to Tier One status.

Appendix A provides a detailed discussion of the candidate Procedure Upgrade Program and the types of changes in procedures and procedure administration that could be expected to result from that program.

2.4 DISCUSSION OF ALTERNATIVES

The only alternative to the proposed action that has been considered for this assessment is the status quo. Under the status quo, operating procedures used in most nuclear plants across the country would remain of poor quality. Furthermore, the programs by which licensees create, revise, and administer operating procedures would continue to have those deficiencies identified in previous research.

3.0 METHODOLOGY FOR ESTIMATING THE REDUCTION IN REACTOR CORE-MELT FREQUENCY

This chapter provides an overview of the methodology used in evaluating the reduction in reactor core-melt frequency associated with implementation of the candidate Procedure Upgrade Program described briefly in Section 2.2 and more extensively in Appendix A. The methodology was intended to provide an approach for estimating the potential reduction in reactor core-melt frequency associated with improved operating procedures. The methodology was adapted from the standard value-impact assessment methodology established in A Handbook for Value/Impact Assessment (Heaberlin et al. 1983), and was intended to be used as an extension to an existing nuclear power plant Probabilistic Risk Assessment (PRA).

The methodology relies on an existing PRA to provide a quantitative framework for modeling the safety significance of normal operating procedure and abnormal operating procedures in the operation of nuclear power plants. The reference PRA also provides the original risk equations for the plant, including the dominant accident sequences and their associated cut sets, along with the basic events that form the cut sets. Therefore, the reference PRA serves as a starting point for the methodology. The results obtained through the evaluation of the reference PRA, which is assumed to represent a generic plant, are then extended to reflect the results for the industry as a whole.

The approach outlined in this chapter consisted of a sensitivity analysis technique that was used to: 1) estimate the current contribution of procedure-related operational errors to reactor core-melt frequency during normal and abnormal operating conditions, 2) estimate the same contribution assuming that the quality of operating procedures is improved by specific measures, and 3) compare the "before" and "after" estimates of overall core-melt frequency to determine the potential reduction in core-melt frequency associated with the proposed improvements to current operating procedures.

Since the objective of this assessment was to evaluate the relative reduction in reactor core-melt frequency due to improved operating procedures, no attempt was made to reevaluate the baseline risk of the reference PRA. The methodology developed as part of this assessment can be characterized by an 11-step process, containing each of the three elements discussed above. This 11-step process is outlined in the following sections. A more detailed discussion of the safety assessment is contained in Appendix B.

3.1 THE 11-STEP PROCESS

The 11-step methodology is illustrated in Figure 3.1.

3.1.1 Estimate The Current Contribution of Procedure-Related Errors

The first element of the general approach consisted of a 6-step process. The objective of this process was to evaluate the current contribution

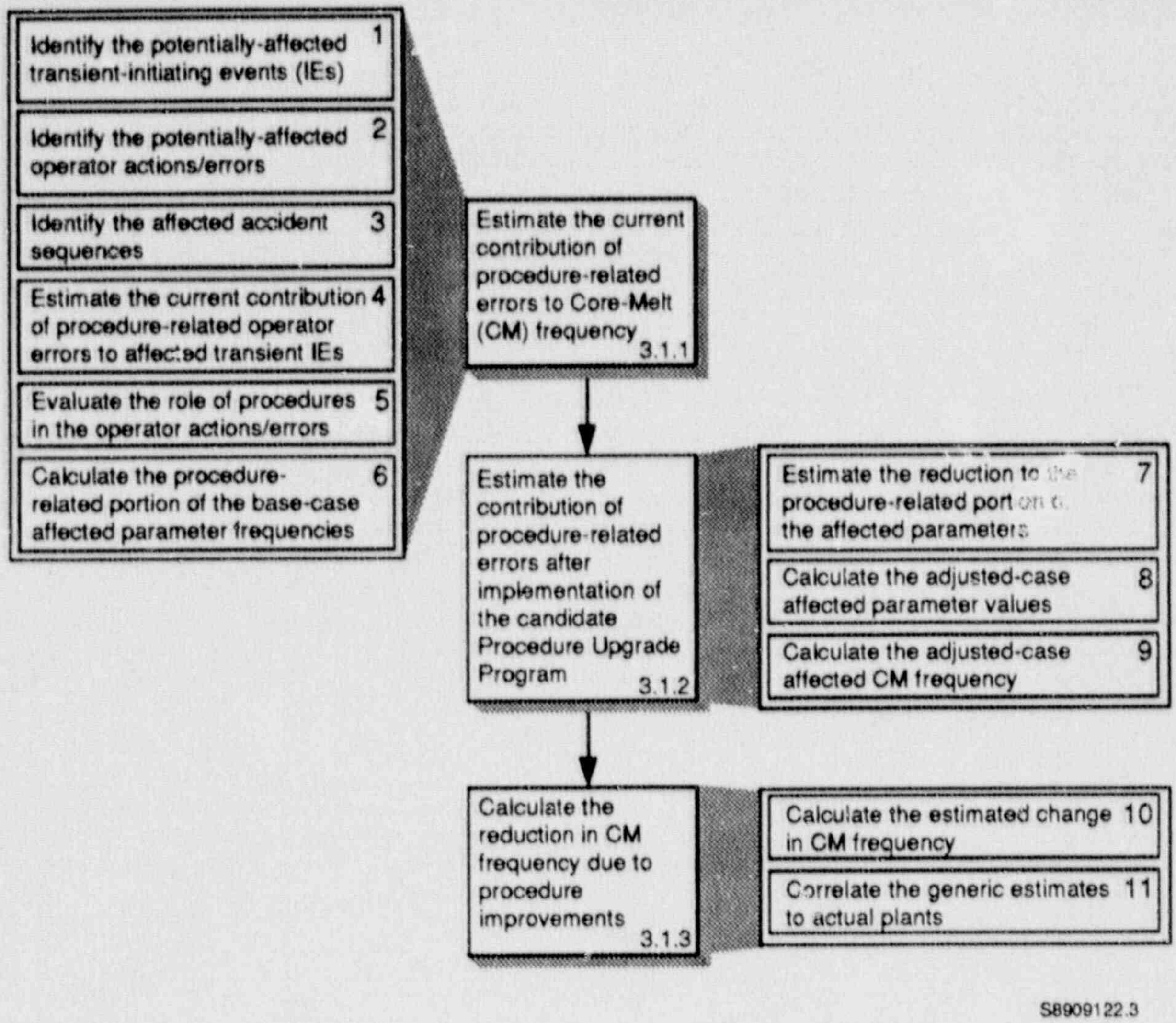

FIGURE 3.1. Process for Evaluating Core-Melt Frequency Reduction

of poor or inadequate procedures to the occurrence of procedure-related operational errors during normal and abnormal operating conditions. Each of the 6 steps involved in this process are briefly discussed below.

Step 1: Identify the Potentially Affected Transient Initiating-Events

The first step in developing estimates of the change in core-melt frequency was to identify transient initiating-events that could potentially be affected by the candidate Procedure Upgrade Program. It was assumed that in order for an operational error during normal operating conditions to have safety significance, it must result in a transient initiating-event.

This step was accomplished by reviewing the transient initiating-events modeled in the reference PRA in order to identify those that might be affected by improved normal and abnormal operating procedures. This review relied on the expert judgment of individuals with expertise in nuclear power plant operations. This step produced a list of potentially affected transient initiating-events that are modeled in the reference PRA and that contribute to the original risk equations of the plant. Each of these potentially affected transient initiating-events was, thus, determined to be an "affected parameter" that could potentially be altered by the candidate NRC regulatory action.

<u>Step 2</u>: Identify the Potentially Affected Operator Actions

In addition to the transient initiating-events identified in Step 1, there may also be specific operator actions that occur during normal and abnormal operating conditions that may be affected by the candidate Procedure Upgrade Program. This second step of the approach involved reviewing the descriptions of potential human errors and operator actions that are modeled in the reference PRA. The definition of each of these events was reviewed to determine if it might be affected (i.e., reduced in frequency) by improved operating procedures. In general, those operator actions modeled in the reference PRA that are in response to a specific failure (e.g. "recovery actions") are not included in this review since these actions are typically governed by emergency operating procedures.

In order to determine whether or not a particular action may be affected, it was necessary to consult with individuals who are knowledgeable in the operation of nuclear power plants and their written procedures. This step resulted in a list of potentially affected human errors and operator actions that are modeled in the reference PRA and that contribute to the original risk equations of the plant. Thus, each of these potentially affected operator actions also was determined to be an affected parameter.

<u>Step 3</u>: Identify the Affected Accident Sequences

In this step, all accident sequences and cut sets contained in the reference PRA that involve any of the potentially affected transient initiating-events identified in Step 1 or one or more of the potentially affected operator actions identified in Step 2 were identified. The result of this step was a listing of all core-melt accident sequences, along with their associated cut sets, that may be potentially affected by improved operating procedures.

Step 4: Estimate the Current Contribution of Procedure-Related Operator Errors to the Affected Transient Initiating-Events

A review of Licensee Event Reports (LERs) using the Sequence Coding and Search System (SCSS) database (NUREG/CR-3905, Green et al. 1985) was performed in this step. The review was conducted to estimate the current "base-case" contribution of procedure-related errors to the frequency of the transient initiating-events identified in Step 1. The SCSS database contains all current LERs submitted by nuclear power plant utilities after January 1, 198[illegible]. It has the capability of searching coded LERs to obtain detailed information regarding the causes of specific events. Of particular importance were those LERs that are assigned a cause/effect code of "SB" (task description inadequacy). A cause code of "SB" is assigned only to those LERs in which a task description inadequacy contributed to the occurrence of the event. Since these LERs are representative of incidents in which human errors occurred due to instructional inadequacies, improved procedures could potentially reduce the likelihood of these incidents.

By first identifying all LERs contained in the database that result in one of the affected transient initiating-events, and then searching further to determine the number of these LERs that have a cause code of SB, an estimate of the relative contribution of procedure-related errors to that transient initiating-event was obtained. This resulted in an estimate of the procedure-related portion for each of the specific transient initiating-events. This procedure-related portion of the base-case parameter frequencies is the only portion of the frequencies that is subject to reduction due to improved operating procedures.

Step 5: Evaluate the Role of Procedures in the Operator Actions/Errors

Just as in the case of transient initiating-events, operator actions and errors may also be caused or otherwise influenced by factors other than inadequate procedures. Therefore, improvements to procedures will not be entirely effective at eliminating all operator errors. In this step, expert judgment was used to evaluate the role of written procedures in the specific operator actions identified in Step 2. This allowed elimination of any portion of operator action probabilities that is not likely to be affected by improving operating procedures. This portion would, for example, account for events in which the operator is not likely to refer to operating procedures. This ensured that the operator actions that are being credited with reduced frequencies could indeed be affected by improved operating procedures. The result of this step was an estimate of the procedure-related portion of the base-case probabilities for each of the operator actions identified in Step 2.

Step 6: Calculate the Procedure-Related Portion of the Base-Case Affected Parameter Frequencies

Based on the results of Steps 4 and 5, the procedure-related portion of each of the base-case affected parameter values was calculated. These values represent the portion of the original values which is contributed by procedure-related errors. They also represent the only portion of the original parameter frequencies that can be reduced by improved operating procedures. For each of the transient initiating-events identified in Step 1, the procedure-related portion of the base-case parameter frequency was found to be a relatively small fraction of the total base-case parameter frequency.

3.1.2 Estimate The Contribution of Procedure-Related Errors After Implementation of the Candidate Procedure Upgrade Program

The second element of the methodology consisted of a 3-step process. The objective of this process was to evaluate the residual contribution of inadequate procedures to the occurrence of procedure-related operational errors after the candidate Procedure Upgrade Program has been implemented. Since it cannot be anticipated that improved procedures will completely eliminate procedure-related operational errors, it was necessary to estimate to what extent improved procedures will be effective. Each of the 3 steps involved in this process is discussed below.

Step 7: Estimate the Reduction to the Procedure-Related Portion of the Affected Parameters

Due to the lack of quantitative data regarding the effect of procedure quality on specific operator actions, it was necessary to exercise expert judgment in this step. A survey of expert opinion is considered to be a reasonable approach to estimate the expected improvement in operator performance (Comer et al. 1984). A survey, which is described in detail in Appendix C, solicited from experts the estimated reduction in each of the potentially affected parameters that they believed would result from implementation of the candidate Procedure Upgrade Program. The improvement in operator performance was estimated as a percentage reduction to the initial affected frequency or probability. This provided an estimate of a percentage reduction in the procedure-related portion of each of the affected parameter values due to improved procedures.

Step 8: Calculate the Adjusted-Case Affected Parameter Values

In this step, the adjusted-case (after procedure improvements) affected parameter values, based on the estimates provided by the experts in Step 7, were calculated by reducing the procedure-related portion of the affected parameter frequencies by the mean values of the expert judgments.

Step 9: Calculate the Adjusted-Case Affected Core-Melt Frequency

In this step, the adjusted-case affected parameter frequencies were substituted in the cut sets in place of the original base-case affected parameter values. The adjusted-case affected core-melt frequency was then calculated by summing the cut set frequencies for each of the affected accident sequences. Since the adjusted-case affected parameter frequencies were slightly less than the base-case affected parameter frequencies, the resultant adjusted-case affected core-melt frequency was also reduced.

3.1.3 Calculate the Reduction in Core-Melt Frequency Due to Procedure Improvements

The third element of the methodology consisted of a 2-step process whose objective was to evaluate the results from the previous two elements.

Step 10: Calculate the Estimated Change in Core-Melt Frequency

In this step, the estimated change in plant core-melt frequency due to improved operating procedures was calculated by subtracting the adjusted-case affected core-melt frequency (calculated in step 9) from the base-case affected core-melt frequency. The calculation of the base-case affected core-melt frequency included only those original accident sequences that contain one or more of the affected parameters.

Step 11: Correlate the Generic Estimates to Actual Plants

In this step the results that had been obtained for the generic plant modeled by the reference PRA were extended to represent the reactor core-melt frequency reduction potential at actual plants. This was accomplished by, first, using prior research on procedure quality (NUREG/CR-3968; Morgenstern et al. 1987) to create three power plant categories based on current quality of operating procedures--those having relatively good procedures, those with procedures of intermediate quality, and those having relatively poor procedures. Then, expert judgment was used to adjust the results for the generic plant to represent the change in core-melt frequency for each of the three actual plant categories.

4.0 CONSEQUENCES

The candidate Procedure Upgrade Program discussed in Section 2.2 and Appendix A is evaluated in this section to quantify the costs and benefits of instituting that program relative to maintaining the present condition of unacceptably poor operating procedures at nuclear power plants. The benefits (or "values") of the program are measured in terms of the avoided public and occupational health risks associated with implementation and operation of the program as well as the associated avoidance of property damage. The costs (or "impacts") include the industry costs associated with the implementation of the program and operation thereafter, and NRC costs for development, implementation, and operation of the program. Other nonquantifiable impacts are also discussed as supplementary considerations.

The benefits and costs associated with the candidate Procedure Upgrade Program are assessed as differentials using the current condition of operating procedures as a baseline. The costs and benefits of the upgrade program are therefore compared with those associated with Alternative A, that is the status quo. The potential effects of the Procedure Upgrade Program are identified in Figure 4.1.

In the context of this analysis, public health and safety includes both routine and normal risks and accident risks. Routine risks arise from those activities associated with the operation of the plant under normal

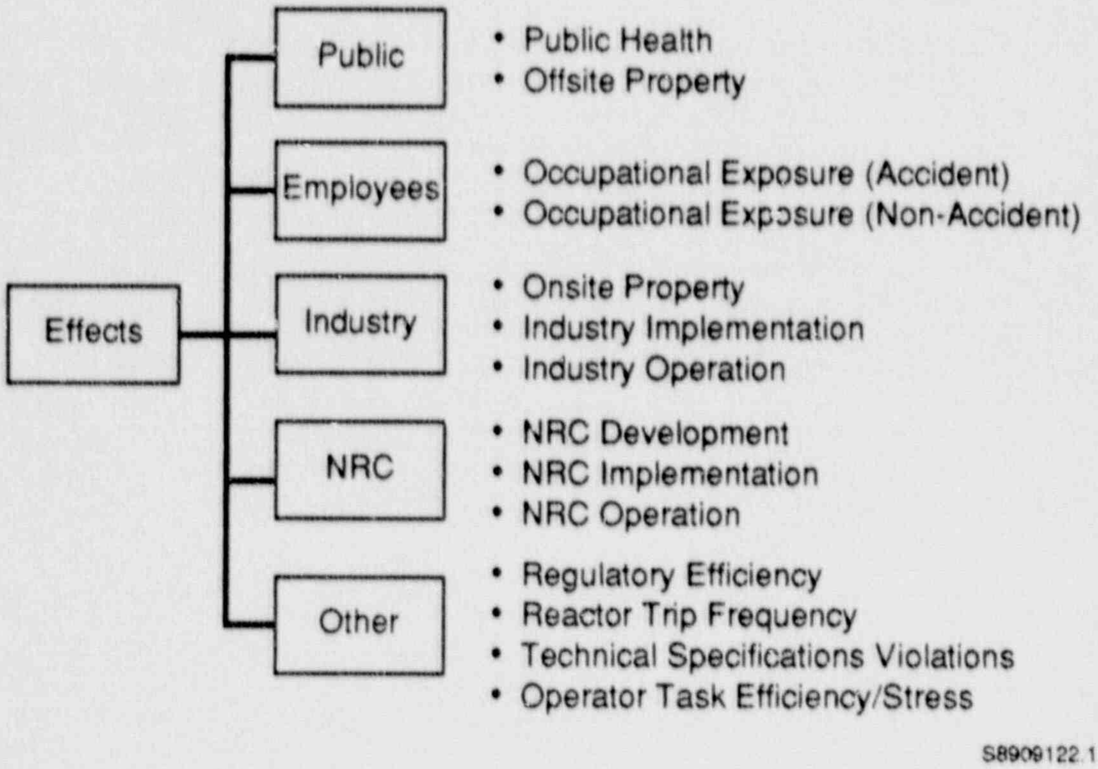

FIGURE 4.1. Identification and Characterization of Potential Effects

conditions. This routine risk is primarily the result of very low-level radiological releases to the atmosphere from routine operations of the plant. Since these annual radiological releases to the environment from routine operations are very small and are not expected to be significantly affected by the proposed action, emphasis is placed here on accident risks. Accident risks are those risks that arise from operating conditions other than those normally encountered. The principal contributor to accident risk at a nuclear power plant is a postulated core-melt accident. A hypothetical core-melt accident may involve damage to persons or property. The effects of these accident risks are discussed in Section 4.1.

Nuclear power plant workers are routinely exposed to low-level radiation while they work in radiation areas. This occupational exposure may be affected by changes in operating procedures that affect the frequency or duration that workers are exposed to radiation while working in radiation zones. For example, if new operating procedures required workers to spend additional time to perform a task in a radiation zone as compared with the time it had taken them before the procedures were changed, an increase in the routine exposure to occupational workers would be expected.

Workers may also incur occupational exposure due to accidents. Regulatory changes can have an impact on this component of occupational exposure if they affect either the frequency or the consequence of core-melt accidents. For example, if improved operating procedures reduce the expected core-melt frequency, a decrease in the accidental exposure to occupational workers would be expected.

Effects on industry costs are categorized as property damage costs if an accident occurs, costs to implement the candidate Procedure Upgrade Program, and operational costs attributable to the program. Costs incurred by the NRC are categorized as costs to develop the regulatory action, costs to implement the regulatory action, and operational costs. Each of these potential effects is discussed in Section 4.2.

Other potential effects of the candidate Procedure Upgrade Program include the reduction of unanticipated reactor trips and scrams. This could directly affect the capacity factors at plants where improved procedures help to prevent unanticipated trips and scrams. Although these effects primarily result in cost savings to the licensees, these qualified effects should also be considered in this case. Other possible effects would occur in the areas of technical specifications violations, operator task efficiency, and operator stress.

Table 4.1 provides a checklist of these potential effects and indicates the expected result of implementation of the candidate Procedure Upgrade Program.

TABLE 4.1. Checklist for Identification of Potential Effects

Potential Effect	Quantified Change	Qualitative Change	No Significant Change
Public Health and Safety	X		
Public Property	X		
Occupational Health and Safety	X		
Industry Implementation	X		
Industry Operation		X	
NRC Development	X		
NRC Implementation	X		
NRC Operation			X
Regulatory Efficiency		X	
Trip/Scram Frequency		X	
Violation of Technical Specs.		X	
Operator Efficiency/Stress		X	

4.1 ESTIMATION OF VALUES

This section presents estimates of the values associated with implementation of the candidate Procedure Upgrade Program. These values are calculated based on the change in core-melt frequency that could be expected from implementation of the program. Based on the estimated changes in core-melt frequency, the following values are calculated: public health risk (in person-rem), accidental occupational exposure (in person-rem), offsite property damage (in discounted dollars), and onsite property damage (in discounted dollars).

The estimated changes in core-melt frequency that form the basis for estimating the values of the candidate Procedure Upgrade Program are based on the risk equations contained in the NSAC/60 Probabilistic Risk Assessment (PRA) for Oconee Unit 3 (Sugnet et al. 1984). The Oconee-3 PRA was selected based on a review of several existing PRAs compared with several selection criteria. The specific criteria used in the selection process are discussed in Appendix B, Section B.2.1. Oconee Unit 3 is a nuclear power plant that utilizes a pressurized-water reactor (PWR) design. No distinction is made between reactor types in the risk-reduction calculations. Rather, the plant categorization is based on the quality of the current operating procedures in place at the plant. By taking this approach, the effects of implementing the program upon three distinct plant categories have been modeled.

As mentioned, these three plant categories are based on the current quality of operating procedures, and are categorized as follows: 1) those plants whose current procedures are considered to be of relatively good quality, 2) those plants whose current procedures are considered to be of intermediate quality, and 3) those plants whose current procedures are considered to be of relatively poor quality. This approach provides a meaningful estimate of the reduction in plant risk because: 1) it distinguishes between plants based on the current adequacy of operating procedures (rather than on reactor type), and 2) it allows the analysis to focus on one set of risk equations that serve to represent a "generic" plant, which can then be related to each of the three plant categories.

The basic approach used to develop estimates of the changes in core-melt frequency is described in detail in a report by Andrews et al. (1983). In that report, a general method for developing estimates of the changes in core-melt frequency that would arise from resolving generic safety issues is described. Briefly, the approach described by Andrews et al. (1983) consists of a sensitivity analysis technique. In the current report, estimates of the changes in core-melt frequency that would result from improvements to cut-set parameters are developed using the risk equations from the Oconee PRA. These improvements are postulated to result from implementation of the candidate Procedure Upgrade Program discussed in Section 2.2.

The estimated changes in core-melt frequency form the basis for the estimated changes in public health risk [person-rem/reactor-year (ry)], which are then integrated over the remaining lifetimes of the affected plants to obtain the total value for the entire industry. A similar approach is used to estimate the changes in accidental occupational exposure, offsite property damage, and onsite property damage. According to A Handbook for Value/Impact Assessment, use of this methodology can be described as an "intermediate effort" for performing value/impact analyses (Heaberlin et al. 1983). The reader is referred to Andrews et al. (1983) and Heaberlin et al. (1983) for additional details of the general methodology. An overview of the specific approach used in the core-melt frequency calculations is provided in Section 3.0 of this report and the entire safety assessment is described in detail in Appendix B.

The remainder of this section develops estimates of the values associated with implementation of the candidate Procedure Upgrade Program. The bases and rationale for determining the changes in cut-set parameters postulated to result from implementation of the Procedure Upgrade Program as well as the resulting changes in core-melt frequency (events/ry) are described in Section 4.1.1. Estimates of the avoided public health risk resulting from reductions in core-melt frequency are developed in Section 4.1.2. Section 4.1.3 develops estimates of the avoided accidental occupational exposure resulting from reductions in core-melt frequency. Offsite and onsite property damage avoidance costs are developed in Sections 4.1.4 and 4.1.5, respectively. A discussion of the potential values that have not been quantified is provided in Section 4.1.6.

4.1.1 Effects on Core-Melt Frequency

This section explains the bases and rationale for changing the values of the cut set parameters that are affected by the proposed action. First, the affected parameters from the reference PRA, their base-case values and adjusted-case values, and the quantitative effects on core-melt frequency and public health risks are estimated. Second, these estimates are modified by estimating new base-case values for each of the affected parameters corresponding to each of the three plant categories. Finally, these new base-case affected parameter values are used to recalculate the new base-case core-melt frequencies for the three plant categories.

The first step in developing estimates of the changes in core-melt frequency and public risk was to identify the cut set parameters contained in the reference PRA that would be affected by implementation of the Procedure Upgrade Program. This was accomplished by an in-depth review of the reference PRA. Based on engineering experience and experience with operating procedures use, several transient initiating-events were determined to be potentially affected by improved operating procedures. These potentially affected transient initiating-events are listed in Table 4.2. Each of the transient initiating-events identified in Table 4.2 was presumed to have a contribution from procedure-related operational errors. Several other transient initiating-events were also initially identified; however, upon further review, these were eliminated from consideration due to negligible contributions from procedure-related operational errors.

TABLE 4.2. Transient Initiating-Events Postulated to be Affected by the Candidate Procedure Upgrade Program

Event Identifier	Unadjusted Base-Case Frequency (events/year)	Event Description
T1	5.7	Reactor/Turbine Trip
T2	6.4E-1	Loss of Main Feedwater
T4	2.1E-1	Loss of Condenser Vacuum
T8	1.0E-2	Spurious ES Actuation Signal
T12	4.0E-3	Loss of Low Pressure Service Water

In addition to these transient initiating-events, several operator actions modeled in the reference PRA were judged to be potentially affected by improvements in procedures. These operator actions are listed in Table 4.3.

The core-melt accident sequences involving one or more of these potentially affected parameters are listed in Table 4.4.

Next, it was necessary to determine the effects that the candidate Procedure Upgrade Program might have on these affected parameters. There may be various contributors to the occurrence of each of these events, including electrical and hardware component failures, human errors, and equipment downtime due to testing, maintenance, and repair. Therefore, in order to evaluate the effect of improved procedures on these parameters, it was necessary to estimate the relative contribution of procedure-related operational errors to each of the affected parameter's base-case frequencies. Thus, it was necessary to determine what fraction of the original base-case frequency could be influenced by improvements to operating procedures. This fraction

TABLE 4.3. Operator Actions Postulated to be Affected by the Candidate Procedure Upgrade Program

Operator Action	Unadjusted Base-Case Probability	Description of Operator Action
RESSFSI	0.1	Operator fails to provide RCP seal injection from the SSF within 30 minutes of losing seal cooling via HPI
SW3BPPSH	0.002	Operator fails to start standby LPSW pump
HPRCPH	0.01	Operators fail to trip the RCPs following loss of seal cooling (within 15 minutes)
REIA2/6	0.055	Failure of the operating staff to recover IA prior to the depletion of the UST
RESW12	0.013	Failure of the operating staff to recover LPSW from another source before failure of all HPI pumps
RESW108	0.11	Failure of the operating staff to recover LPSW to HPI pumps given a failure of LPSW108

represents the portion of the original frequency that is subject to reduction and is referred to here as the procedure-related portion of the base case frequency.

For the transient initiating-events identified in Table 4.2, this determination was accomplished using the SCSS database, which contains coded information on all current Licensee Event Reports (LERs) submitted by nuclear power plant utilities after January 1, 1981. The SCSS database allowed the coded-LERs to be searched for the specific transient initiating-events identified in Table 4.2. Once the LERs that contain these events were identified, the SCSS database could then search the subset of LERs for those that have been assigned a cause/effect code representing "task description inadequacy". The task description inadequacy code identifies those events that include a deficiency in properly communicating all of the information necessary to perform a task. In determining what should be considered a task description inadequacy, the communication process is considered to be composed of five components: 1) perceiving an idea, 2) encoding the idea, 3) transmitting the idea, 4) decoding the transmission, and 5) understanding the idea. A breakdown in any of these components, whether by the information sender or the receiver, must be coded as a task description inadequacy (e.g., cause code "SB"). This code provides a reasonable estimate of the role that inadequate operating procedures have played in the various coded-1 LERs. The results of the LER search using this code, therefore, provided a useful

TABLE 4.4. Potentially Affected Core-Melt Accident Sequences[a]

Core-Melt Bin Type and Accident Sequence	Potentially Affected Parameters	Base-Case Frequency[b] (events/yr)
IB,TQUs	T_1,T_2,T_4,T_8,T_{12},HPRCPH,RESW12, SW3BPPSH	5.9E-7
ID,T6QU	HPRCPH	2.5E-7
IE,TQUs	T_{12}	1.0E-7
IIE,TQUYXs	T_1,T_2,T_4,T_8,T_{12}	5.8E-8
IIF,TQUYXs	T_8	2.2E-7
IIIA,T2BU	T_2	1.2E-6
IIIB,T4BU	T_4	4.1E-7
IIIC,TBU	T_1,T_2,T_4,T_8,T_{12}	4.2E-7
IIIF,TBU	T_1,T_2,T_4,T_8,T_{12},REIA2/6	4.7E-6
IIIG,TBU	T_1,T_2,T_4,T_8,T_{12},RESW12,RESSFSI RESW108,SW3BPPSH	1.5E-5
ATWS Sequences:		
I:6	T_1	1.7E-8
II:5	T_1	1.7E-8
III:15,14	T_1	3.4E-6
V:9,12,73,27	T_1,T_2	1.3E-6
VI:72,26,11,8	T_1,T_2	1.4E-6
	Total:	2.9E-5

(a) Based on the reference PRA (Sugnet et al. 1984).
(b) Reflects all cut sets contained in the accident sequences.

approximation of the contributions of procedure-related errors to the parameters' original frequencies. The results of the coded-LER review are given in Table 4.5.

The contribution of procedure-related operational errors to the original transient initiating-event frequencies was calculated by multiplying the

TABLE 4.5. Results of the Coded-LER Review

Event	Original Frequency (/year)	Event Description	Relative Contribution of Orig. Frequency from Procedure-related Errors
T1	5.7	Reactor/Turbine Trip	5%
T2	6.4E-1	Loss of Main Feedwater	5%
T4	2.1E-1	Loss of Condenser Vacuum	5%
T8	1.0E-2	Spurious ES Actuation Signal	2%
T12	4.0E-3	Loss of Low Pressure Service Water	2%

original parameter frequency by the fraction associated with procedure-related operational errors from the SCSS analysis.

It was assumed that the six operator actions identified in Table 4.3 could be improved upon by reducing the conditional probability of failure. Since each of these operator action failures may have contributions from factors other than procedures, it was necessary to determine the specific role that procedures play in each of these events. After all, if the operators do not typically refer to the written procedures for these actions, there can be no expected improvement in operator performance due to improvements made to procedures.

The six specific operator actions were reviewed in order to determine the potential affect of improved procedures on their occurrence. This evaluation was performed by a team of analysts at PNL who have expertise and experience in the area of nuclear power plant operating procedures. The experts were asked to estimate the importance that procedures play in each of the operator actions. The experts confirmed that procedures play a principle role in each of the operator actions identified. They judged that the affected portion of the base-case probabilities is equal to 100% of the base-case probability for each of the operator actions except for the RESW108 event. The affected probability for that event was judged to be 70% of the base-case probability (i.e., there may be some scenarios where even ideal procedures would not eliminate the chance of failure). This evaluation by the PNL experts provided estimates of the affected parameter probabilities for each of the operator actions, which can be included with the estimates of the affected frequency for each of the affected transient initiating-events. For additional information regarding the qualifications of the PNL experts, the reader is referred to Section B.2.3 of Appendix B.

The next step was the development of the adjusted-case values for the affected parameters that would result from implementation of the Procedure Upgrade Program. This was accomplished by developing and conducting a survey of expert opinion. The survey requested a group of experts to estimate the reduction affect of the candidate Procedure Upgrade Program on the specific events identified in Tables 4.2 and 4.3. The survey and the group of experts relied upon are described and discussed in detail in Appendix C. The results of the survey are illustrated in Table 4.6. The adjusted-case frequencies in that table were calculated by subtracting the base-case affected frequencies from the base-case frequencies and then adding the mean value of the adjusted-case affected frequencies from Table C.1.

TABLE 4.6. Affected Parameter Frequencies[(a)]

Event	Base-Case Frequency (/year)	Base-Case Affected Frequency (/year)	Adjusted-Case Frequency (/year)[(b)]
T1	5.7	2.85E-1	5.6057
T2	6.4E-1	3.20E-2	0.6342
T4	2.1E-1	1.05E-2	0.2070
T8	1.0E-2	2.00E-4	9.97E-3
T12	4.0E-3	8.0E-5	3.99E-3
RESSFSI	0.1	0.1	0.0527
SW3BPPSH	0.002	0.002	1.6E-3
HPRCPH	0.01	0.01	0.0078
REIA2/6	0.055	0.055	0.0262
RESW12	0.013	0.013	5.3E-3
RESW108	0.11	0.077	0.0768

(a) For the bottom six events, which are operator actions, the values are conditional probabilities rather than frequencies.
(b) From the survey of expert judgment.

In order to correlate the reference plant to the entire industry, it was necessary to modify the base-case estimates associated with the hypothetical generic plant to represent the variance in actual procedure quality that exists in the operating nuclear power plants across the country. This was accomplished by, first, using the results of prior research reported in NUREG/CR-3968, Study of Operating Procedures in Nuclear Power Plants: Practices and Problems (Morgenstern et al. 1987) to define three categories of plants based on current quality of operating procedures. That research, which evaluated and scored the quality of operating procedures being used in a large sample of operating nuclear power plants, concluded that operating procedures in many U.S. nuclear plants are of unacceptably poor quality. Though most plants have unacceptably poor quality procedures in an absolute sense, the procedure evaluation scores from that prior research allowed the definition of three categories of plants based on their relative current quality of operating procedures: 42% (or 50) of the plants have relatively poor procedures; 39% (or 46) of the plants have procedures of intermediate quality; and 19% (or 22) of the plants have relatively good procedures.

Then, expert judgment was used to modify the base-case affected parameter frequencies associated with the hypothetical generic plant to calculate new, modified base-case affected parameter frequency estimates for the three actual plant categories. PNL experts, relying on their diverse experience with operating procedures at various nuclear power plants, judged that for those plants whose current procedures are of relatively good quality, the procedure-related portion of the base-case frequency would be 70% less than for the generic model. For those plants whose current procedures are of intermediate quality, the PNL experts judged that the procedure-related portion of the base-case frequency would be 20% less than for the generic model. Finally, for those plants whose current procedures are considered to be of relatively poor quality, the PNL experts estimated that the procedure-related portion of the base-case parameter frequencies would be twice as large as for the generic model. These estimates were then used to calculate the modified base-case parameter frequencies for the three categories of actual plants. These modified base-case parameter frequencies, along with their corresponding adjusted-case frequencies, are given in Table 4.7.

The modified affected parameter values were incorporated into the original Oconee-3 risk equations for the core-melt accident sequences identified in Table 4.4. The results of these calculations provided an estimate of the potential core-melt frequency reduction for each category of plant. The results of these calculations are shown in Table 4.8.

TABLE 4.7. Modified Parameter Frequencies For Actual Plants

Event Identifier	Modified Base-Case Frequency (/year)			Adjusted-Case Frequency (/year)
	Current Quality of Operating Procedures			
	Relatively Poor	Intermediate	Relatively Good	
T1	5.794	5.681	5.634	5.606
T2	0.646	0.639	0.636	0.634
T4	0.213	0.209	0.208	0.207
T8	1.00E-2	9.99E-3	9.98E-3	9.97E-3
T12	4.01E-3	4.00E-3	3.99E-3	3.99E-3
RESSFSI[a]	0.147	0.091	0.067	0.053
SW3BPPSH	2.4E-3	1.9E-3	1.7E-3	1.6E-3
HPRCPH	1.2E-2	9.6E-3	8.5E-3	7.8E-3
REIA2/6	8.4E-2	4.9E-2	3.5E-2	2.6E-2
RESW12	2.1E-2	1.1E-2	7.6E-3	5.3E-3
RESW108	0.143	0.103	0.087	0.077

(a) The last six events are operator actions. Therefore, the numbers represent conditional probabilities rather than frequencies.

TABLE 4.8. Affected Accident Sequence Frequencies for Actual Plants

Core-Melt Bin Type and Accident Sequence	Modified Base-Case Frequency (events/yr) Relatively Poor	Intermediate	Relatively Good	Adjusted-Case Frequency (events/yr) All Plants
IB,TQUs	1.1E-6	5.0E-7	3.2E-7	2.3E-7
ID,T6QU	3.0E-7	2.4E-7	2.1E-7	1.9E-7
IE,TQUs	4.1E-8	4.1E-8	4.1E-8	4.1E-8
IIE,TQUYXs	4.9E-8	4.8E-8	4.8E-8	4.8E-8
IIF,TQUYXs	1.0E-8	9.9E-9	9.9E-9	9.9E-9
IIIA,T2BU	9.4E-7	9.3E-7	9.2E-7	9.2E-7
IIIB,T4BU	2.9E-7	2.9E-7	2.9E-7	2.9E-7
IIIC,TBU	2.7E-7	2.7E-7	2.6E-7	2.6E-7
IIIF,TBU	3.3E-6	2.0E-6	1.4E-6	1.0E-6
IIIG,TBU	2.9E-5	1.2E-5	6.9E-6	4.6E-6
ATWS I	1.74E-8	1.70E-8	1.69E-8	1.68E-8
ATWS II	1.74E-8	1.70E-8	1.69E-8	1.68E-8
ATWS III	3.48E-6	3.41E-6	3.38E-6	3.36E-6
ATWS V	1.31E-6	1.28E-6	1.27E-6	1.27E-6
ATWS VI	1.41E-6	1.38E-6	1.37E-6	1.37E-6
Total	4.15E-5	2.22E-5	1.65E-5	1.37E-5

The estimated reduction in core-melt frequency based on the results of these calculations are provided in Table 4.9. The upper and lower bounds have been calculated by adjusting the mean value of each of the adjusted-case parameter frequencies up and down by a factor of two times the standard deviation of the experts' judgments obtained by survey. In several cases, adjusting the mean value up by two times the standard deviation brought the value up to (or above) the original base-case value. Since it is not logical to predict an <u>increase</u> in event frequencies due to improved procedures, the original base-case values were used in these instances. This explains why the lower bound for the "Intermediate" and "Relatively Good" plants indicates a reduction in core-melt frequency of zero.

TABLE 4.9. Estimated Reduction in Core-Melt Frequency

	Reduction in Core-Melt Frequency ($\Delta\bar{F}$) (events/reactor-year) — Current Quality of Operating Procedures: Relatively Poor	Intermediate	Relatively Good
Best Estimate	2.8E-05	8.5E-06	2.8E-06
Upper Bound	3.4E-05	1.5E-05	9.1E-06
Lower Bound	1.7E-05	0	0

4.1.2 Avoided Public Health Risk

Once the estimated reduction in core-melt frequency was evaluated, the avoided public health risk was calculated using dose conversion factors to estimate the reduction in public health risk of each accident sequence and then summing these to determine the total reduction. Application of the dose conversion factors on the affected accident sequences results in the reduction of public risk (ΔW) shown in Table 4.10. These values represent the reduction in public risk per plant-year. Upper and lower bounds are based on the upper and lower bounds on the estimated reduction in core-melt frequency shown in Table 4.9.

TABLE 4.10. Estimated Reduction in Public Health Risk (ΔW) (person-rem/ry)

	Current Quality of Operating Procedures		
	Relatively Poor	Intermediate	Relatively Good
Best Estimate	24.9	7.6	2.5
Upper Bound	30.2	13.3	8.1
Lower Bound	15.1	0.0	0.0

To estimate the total avoided public health risks associated with implementation of the candidate program, the per-plant-year risk reduction must be integrated over the average remaining lives of all affected plants. This integration procedure develops estimates of the total avoided health risks over the remaining life cycle of operating plants and plants under construction. The formula used to develop the total avoided public health risk estimates is:

$$V_{PH} = \Delta W N \bar{T}$$

where V_{PH} = value of public health risk avoided (person-rem)

ΔW = avoided public dose per reactor-year (person-rem/ry)

N = number of affected facilities

$\bar{T}$ = average time new procedures are used at facilities (years).

The values of ΔW are shown in Table 4.10. The values of N vary depending on the current quality of procedures. Of the 118 plants listed in Table D.1 of Appendix D, 50 plants currently have relatively poor operating procedures, 46 have intermediate quality procedures, and 22 have relatively good procedures. This conclusion was drawn from prior research (Morgenstern et al. 1987) and has been discussed previously.

To determine $\bar{T}$, the average time new procedures will be used at the facilities, Table D.1 of Appendix D is used. This table lists the average lifetime of plants as 29.5 years as of 1989. It is assumed that improved procedures would be used beginning in 1993. Therefore, the current year, mid-1989 (1989.5) is subtracted from 1993. The difference, 3.5, is subtracted from 29.5 yielding 26 years as the value for $\bar{T}$. Table 4.11 lists the values of ΔW, N, $\bar{T}$, and V_{PH}.

TABLE 4.11. Summary of Avoided Public Health Risk

Current Quality of Operating Procedures	Type of Estimate	ΔW Reduction in Public Risk (per-rem/ry)	N Number of Facilities	$\bar{T}$ Average Time (y)	V_{PH} Value of Public Health Risk Avoided (person-rem)
Relatively Poor	Best	24.9	50	26	3.24E4
	Upper	30.2	50	26	3.93E4
	Lower	15.1	50	26	1.96E4
Intermediate	Best	7.6	46	26	9.09E3
	Upper	13.3	46	26	1.59E4
	Lower	0	46	26	0
Relatively Good	Best	2.5	22	26	1.43E3
	Upper	8.1	22	26	4.63E3
	Lower	0	22	26	0

4.1.3 Avoided Occupational Exposure

The avoided occupational exposure from accidents can be estimated as the product of the change in total core-melt probability and the occupation exposure likely to occur in the event of a major accident. This value can be calculated as:

$$V_{OHA} = \Delta\bar{F}N\bar{T}\ (D_{IO} + D_{LTO})$$

where V_{OHA} = value of occupational health risk due to accidents avoided (person-rem)

$\Delta\bar{F}$ = change in core-melt probability (events/reactor-year)

N = number of affected facilities

$\bar{T}$ = average time procedures are used at facilities (years)

D_{IO} = "immediate" occupational dose (person-rem)

D_{LTO} = long-term occupational dose (person-rem).

The values of $\Delta\bar{F}$ are shown in Table 4.9. The values of N and $\bar{T}$ are the same as defined earlier.

The immediate occupational exposure, D_{IO}, occurs at the time of the accident and during the immediate management of the emergency. The experience at Three Mile Island (TMI) was used to arrive at the estimates of this value. The average occupational exposure related to the incident was approximately 1 rem. For the first 4 months, a collective dose of 1000 person-rem could be attributed to the accident. After this time period, occupational exposure returned to preaccident levels. An upper bound can be estimated by assuming the average individual receives a dose equal to that of the maximum individual at TMI. The ratio of maximum to average dose for TMI is 4.2 to 1 rem; therefore, the collective upper bound dose would be 4200 person-rem. A lower bound of zero would indicate a case with no increase over the normal dose. Thus the values for D_{IO} are 1000 person-rem for the best estimate, 4200 person-rem for the upper bound, and 0 person-rem for the lower bound.

The second occupational dose due to an accident is long-term exposure, D_{LTO}. After the immediate response to an accident, a long process of cleanup and recovery takes place. The value used as a best estimate for this exposure is based on a study (Murphy and Holter 1982) of decommissioning a reference LWR following a major loss of coolant accident (LOCA) in which the emergency core cooling system (ECCS) is delayed in starting. All fuel cladding is assumed to rupture and significant fuel melting and core damage to occur. The containment building is extensively damaged and contaminated, and the auxiliary building undergoes some contamination. The estimated occupational radiation dose from cleanup and recovery is 20,000 person-rem. An upper bound of 30,000 person-rem and a lower bound of 10,000 person-rem are estimated by the authors of the study.

Combining the two types of occupational exposure results in a value of 21,000 person-rems for the best estimate, 34,200 person-rems for the upper bound, and 10,000 person-rems for the lower bound. Table 4.12 contains the values of $\Delta\bar{F}$, N, $\bar{T}$, and the combined occupational dose due to an accident. These are used to arrive at the value of occupational health risk avoided V_{OHA} for each alternative. The total value for the best estimate is 1000 person-rems, 2300 person-rems for the upper bound, and 221 person-rems for the lower bound.

TABLE 4.12. Summary of Avoided Occupational Exposure

Current Quality of Operating Procedures	Type of Estimate	$\Delta\bar{F}$ Change in Core Melt Probability (events/yr)	Number and Lifetime of Facilities N	($\bar{T}$)	$D_{IO} + D_{LTO}$ Additional Occupational Dose (person-rem/event)	VOHA Value of Avoided Occupational Exposure (person-rem)
Relatively Poor	Best	2.8E-5	50	(26)	2.1E4	7.644E2
	Upper	3.4E-5	50	(26)	3.4E4	1.512E3
	Lower	1.7E-5	50	(26)	1.0E4	2.210E2
Intermediate	Best	8.5E-6	46	(26)	2.1E4	2.135E2
	Upper	1.5E-5	46	(26)	3.4E4	6.135E2
	Lower	0	46	(26)	1.0E4	0
Relatively Good	Best	2.8E-6	22	(26)	2.1E4	3.363E1
	Upper	9.1E-6	22	(26)	3.4E4	1.780E2
	Lower	0	22	(26)	1.0E4	0

4.1.4 Public Property Damage Avoided

Offsite (public) property loss is one of the major impact categories for safety-related issues. In severe accidents, property damage offsite can exceed onsite damage. The impact of public property damage avoided can be calculated as follows:

$$V_{FP} = \Delta\bar{F} N D_G$$

where V_{FP} = impact of avoided offsite property damage ($)

$\Delta\bar{F}$ = change in core-melt probability (events/reactor-year)

N = number of affected facilities

D_G = generic present value of property damage conditional on release ($/event).

The values for $\Delta\bar{F}$ and N are the same as defined earlier.

Estimates of the generic present value of offsite property damage, D_G, are obtained by using base or scaled results from NUREG/CR-2723 (Strip 1982). This study reported offsite property costs for accidents at 91 U.S. sites with licensed reactors or construction permits. These costs are based directly on CRAC2 computer code results. This program is used to estimate accident consequences by the NRC (Ritchie 1982). The scaled values correspond to damage due to major accidents of release category SST 1. This assumption may tend to overestimate the reduction in offsite property damage costs.

These scaled values are discounted at a 10% rate to 1989. The present value of the property damage, D_G, can be calculated as:

$$D_G = C \times B$$

where $C = \dfrac{e^{(-.10\, t_i)} - e^{(-.10\, t_f)}}{.10}$

t_i = years before procedures take effect = 1993 - 1989 = 4 years

t_f = years from 1993 until end of procedure use = 26 years

B = scaled result for property damage.

For each alternative, C = 6.00. Table 4.13 contains the values for $\Delta\bar{F}$, N, B, and D_G. The best estimate for D_G was determined by calculating the mean value for the 154 reactors examined in NUREG/CR-2723 (Strip 1982). The upper and lower bound estimates of the scaled damage costs are values for Indian Point No. 2 and Maine Yankee, respectively (Strip 1982). These values are updated to 1989 dollars using the GNP deflator. The total impact for the

TABLE 4.13. Summary of Avoided Public Property Damage

Current Quality of Operating Procedures	Type of Estimate	$\Delta\bar{F}$ Change in Core Melt Probability (events/ry)	N Number of Plants	B Scaled Public Property Damage ($/event)	D_G Discounted Public Property Damage ($/event)	V_{FP} Impact of Avoided Public Property Damage($)
Relatively Poor	Best	2.8E-5	50	2.09E9	1.26E10	1.76E7
	Upper	3.4E-5	50	1.15E10	6.91E10	1.18E8
	Lower	1.7E-5	50	2.72E8	1.63E9	1.39E6
Intermediate	Best	8.5E-6	46	2.09E9	1.26E10	4.91E6
	Upper	1.5E-5	46	1.15E10	6.91E10	4.77E7
	Lower	0	46	2.72E8	1.63E9	0
Relatively Good	Best	2.8E-6	22	2.09E9	1.26E10	7.73E5
	Upper	9.1E-6	22	1.15E10	6.91E10	1.38E7
	Lower	0	22	2.72E8	1.63E9	0

best estimates of the public property damage avoided is $2.33E7; for the upper bound the impact is $1.80E8, and for the lower bound the impact is $1.39E6.

4.1.5 Onsite Property Damage Avoided

Onsite property costs from an accident are the economic costs to plant, equipment, land, and materials within the boundaries of the utility site. The onsite costs can be broken into three categories: the cost of interdicting or decontaminating onsite property; the cost of replacement power; and the capital cost of damaged plant equipment. The impact of onsite property damage avoided can be calculated as:

$$V_{OP} = \Delta\bar{F}NU$$

where V_{OP} = impact of avoided onsite property damage ($)

$\Delta\bar{F}$ = change in core-melt probability (events/reactor-year)

N = number of affected facilities

U = present value of property damage conditional on release ($/event).

The values for $\Delta\bar{F}$ and N are the same as defined earlier.

Generic estimates of the onsite property damage were obtained from a study that estimated the costs of clean-up, repair and refurbishment, and replacement power costs (Andrews et al. 1983). The values used from this study are those of a major LOCA in which ECCS is delayed. The cost of cleanup of a reactor is estimated to be \$373 million. The cost of repair and refurbishment is \$106 million. The cost of replacement power is \$1072 million spread equally over 10 years. Thus, the total generic cost of onsite property damage is estimated at \$1650 million over a 10-year period. The upper and lower bounds reflect a ±50 % spread, which is estimated to be indicative of the uncertainly level.

These generic values are discounted at a 10% rate to 1989. The present value of the onsite property damage, U, can be calculated as:

$$U = \left[\frac{\left(C_c + C_r + C_{rp}\right)}{m}\right] \frac{\left(e^{-rt_i}\right)}{r^2} \left[1 - e^{-r\left(t_f - t_i\right)}\right] \left(1 - e^{-rm}\right)$$

where U = present value of onsite property damage conditional upon release (\$/event)

C_c = cleanup cost (\$)

C_r = repair/refurbishment cost (\$)

C_{rp} = replacement power cost (\$)

t_i = years before procedures take effect, = 1993-1989 = 4 years

t_f = years from 1993 until end of procedure use = 26 years

r = discount rate (for 10 percent, r = .10)

m = years required to return utility to preaccident state = 10 years.

Table 4.14 contains the values of $\Delta\bar{F}$, N, and U as well as the generic costs. The total value for the best estimate is \$1.15E7, \$2.41E7 for the upper bound and \$2.64E6 for the lower bound.

4.1.6 Regulatory Efficiency

The proposed regulatory action would be expected to result in considerable benefits to regulatory efficiency. Many of these benefits would be derived from increased consistency in format, development, and administration of operating procedures among utilities that could be expected to result from the Procedure Upgrade Program. Such consistency should reduce the time NRC staff and contractors need during site visits to recognize and adapt to differences in structure, format, and overall philosophy of operating procedure hierarchy and responsibility. Consistency in operating procedures developed to a minimum requirement will facilitate NRC staff in making comparative value judgments regarding operations within plants and between utilities. NRC staff and its contractors performing activities such as operator licensing examinations, licensee event report investigation, and accident

TABLE 4.14. Summary of Avoided Onsite Property Damage

Current Quality of Operating Procedures	Type of Estimate	$\Delta\bar{F}$ Change in Core Melt Probability (events/ry)	N Number of Facilities	U Discounted Onsite Property Damage ($/event)	V_{op} Onsite Property Damage Avoided ($)
Relatively Poor	Best	2.8E-5	50	6.20E9	8.68E6
	Upper	3.4E-5	50	9.30E9	1.58E7
	Lower	1.7E-5	50	3.10E9	2.64E6
Intermediate	Best	8.5E-6	46	6.20E9	2.43E6
	Upper	1.5E-5	46	9.30E9	6.42E6
	Lower	0	46	3.10E9	0
Relatively Good	Best	2.8E-6	22	6.20E9	3.82E5
	Upper	9.1E-6	22	9.30E9	1.86E6
	Lower	0	22	3.10E9	0

analyses will also benefit. Greater consistency of procedures among utilities will facilitate their providing more accurate and better documented results when the NRC performs these activities.

Consistency between procedures developed to standard reference will enable the utilities to develop more meaningful plant performance indicators. NRC staff, especially Resident Inspectors, will benefit as they, too, can use the improved performance indicators to evaluate utilities.

Consistency should also lead to more credible NRC review of procedures. Without standard criteria, two NRC staff members reviewing the same procedure independently will use their own personal judgment to determine procedure accuracy. This can result in lack of consistency and possibly even lack of concurrence between the reviewers which, in turn, may cause a lack of credibility in the reviewers. When two reviewers use the same standard, increased consistency in their judgments should result.

It should also be easier for NRC staff and utility personnel to arrive at and agree upon a common understanding of what are and what are not adequate operating procedures. Operating procedures would be upgraded according to an agreed upon and relatively objective standard of quality as defined during the Procedure Upgrade Program. Thereafter, discussion between the NRC and utilities regarding the adequacy of operating procedures can be more fruitfully devoted to addressing specific procedure-related problems rather than in disagreements over what does and what does not constitute an adequate procedure.

Improved procedures will also reduce the likelihood that ineffective operating procedures will mask other operations problems. Currently, difficulties that are encountered when performing a poor quality procedure are often assumed to be the fault of the procedure when, in fact, they could be caused by a component or system problem. Identification of the root cause of the problem is delayed while procedure modifications or temporary changes are investigated. The root cause of the problem, even a reportable one, may, in fact, never be discovered because a procedure modification may well be found that substitutes for discovery of the true cause. High quality procedures would make it less likely that assumed procedure problems would mask actual root cause problems. Also in this regard, performance data analyses and root cause investigations are often hampered when insufficient data are recorded. Operating procedures developed to minimum standards will ensure that appropriate and sufficient data are taken and recorded.

Finally, upgrading plant operating procedures should decrease NRC procedure review time. Whether a procedure review is a formal review or just a spot check, NRC staff will require less time verifying that utilities are maintaining adequate procedures.

4.1.7 Qualitative Values

Several important qualitative safety considerations associated with the candidate Procedure Upgrade Program should be noted. This section provides a brief discussion of several factors that are difficult to quantify but are likely to be significantly affected by implementation of the Procedure Upgrade Program.

4.1.7.1 Reactor Trip Frequency

Reactor trip frequency has been considered quantitatively in this assessment as one of the affected initiating-events that could lead to a core-melt accident. The reduction in the core-melt frequency as a result of a reduction in reactor/turbine trip events is a significant value identified in this assessment. However, there are additional values associated with reducing the frequency of reactor trips that have not been quantified.

These additional values relate to the fact that reducing the frequency of unplanned reactor scrams is an important objective in itself because it results in a reduction in the frequency of challenges to plant safety systems. An important point to note is that the variation in fuel temperature that is initiated by a reactor scram causes a variation in the pressure inside the fuel rods. These variations in the temperature and pressure of the fuel can lead to additional stresses in the fuel cladding. A high frequency of reactor scrams, therefore, can weaken the strength of the first barrier of protection against release of radioactive material to the environment.

In addition, unplanned reactor scrams can also provide an environment in which an operator error or equipment malfunction can turn a relatively minor event into a serious accident. Therefore, reducing the frequency of reactor scrams and challenges to plant safety systems has been established as important safety goals.

4.1.7.2 Inadvertent Violations of Technical Specifications

The purpose of Technical Specifications is to protect public health and safety by imposing limits, opera.ing conditions, and other similar requirements on the operation of nuclear units. Violations of these Technical Specifications, therefore, can adversely affect the level of safety provided to the public. The candidate Procedure Upgrade Program may reduce the frequency of inadvertent violations of these Technical Specifications by providing better guidance to operators while performing complex procedures.

The safety significance associated with a reduction in Technical Specification violations is difficult to quantify, since it is dependent upon the specific violations that are expected to be reduced. Nonetheless, by reducing the frequency of inadvertent Technical Specifications violations, protection of the public health and safety would be facilitated, and economic benefits to licensees may result, although their magnitude is not readily quantifiable.

4.1.7.3 Procedure Quality as Contributor to Procedure Violations

Research recently conducted for the NRC suggests that poor procedure quality contributes to procedure violations which, in turn, can influence the likelihood of safety-threatening situations at nuclear power plants. This research, which is described in a report currently in preparation, found that procedures that are technically inaccurate, of low useability, or at an inappropriate level of detail for the operational action being controlled can contribute to procedure violations.

A technically-inaccurate procedure can tend to discourage workers from using the procedure when performing tasks because they know that following the procedure will not enable them to accomplish the task in an accurate and timely manner. On the other hand, inaccurate procedures are likely to lead to incidents or to the potential for incidents.

Procedures of low useability may also increase the likelihood that they will be violated. Procedures that are presented in an inappropriate medium for the conditions under which they will be used are likely either to encourage intentional procedure violations or to promote forced errors. Differing levels of both technical accuracy and useability of operating procedures were found to affect workers' motivations and, therefore, predispose them to ignore procedures in performing tasks.

The level of detail at which procedures are written may also affect the probability of violations. Procedures that are too highly detailed may allow insufficient flexibility in task performance and may, therefore, be detrimental to timely and efficient execution of operational tasks. Procedures may also be insufficiently detailed for the task at hand. In such situations, operators may tend to ignore the procedure in performing the operational task.

Although the data sources used did not permit this research to conclusively determine the significance of procedure violations in U.S. nuclear power plants, it did suggest that current levels of such violations may be a cause for concern. Procedure violations currently play a measurable role in operational problems. Extrapolation from the data collected for this research project suggest, for instance, that procedure violations currently contribute to about 7 radiation releases to the public, 57 excessive exposures of plant personnel, 1,135 technical specification violations, and 116 Engineered Safety Feature actuations each year in the United States. It should be noted that the poor quality of operating procedures may be only one of several causes of the incidents examined. Flawed procedures were, for instance, found to have had a role in causing up to 33% of the offsite radiation releases but only 5% of the Engineered Safety Feature actuations. The results of this research do tend to confirm, however, that safety benefits could accrue should the candidate regulatory action aimed at operating procedure improvement be adopted.

4.2 ESTIMATION OF COSTS

Unlike the value estimates, the cost estimates need not be calculated using the three categories of plants based on current quality of procedures. Instead, the costs associated with the new procedures are analyzed by the tasks required to implement them. As described in Appendix A, the process of implementation would occur in five tasks: 1) developing a Good Practices Document, 2) convening an NRC and industry working group, 3) defining licensee upgrade plans, 4) upgrading licensee operating procedures, and 5) conducting an NRC inspection of the new procedures. Within these tasks, both industry and NRC costs are incurred. Total costs for both sectors are summarized in Section 4.3.

In estimating the costs, two types of costs are used: labor costs and travel expenses. In order to determine wage rates, the data from NUREG/CR-4627 (Claiborne et. al, 1989, Table 4.2) are used. Both industry and NRC rates are defined. As suggested in this source, all industry rates from this table are doubled to reflect benefits. Travel expenses consist of airfare and per diem rates. Transportation costs are obtained through consultation with travel agencies. Per diem rates are based on the Department of Energy's maximum rates (DOE N 1500.33).

For each cost estimate, a range is determined to reflect uncertainty in the labor rates and variations in the travel costs. The upper and lower bounds for each cost estimate are generated by adding 50% to the best estimate. Similarly, 50% of the best estimate is subtracted to determine the lower bound.

Estimates of the industry costs for all tasks are discussed in Sections 4.2.1 and 4.2.2. The NRC costs are discussed in Sections 4.2.3, 4.2.4, and 4.2.5.

4.2.1 Industry Implementation Costs

To accomplish the Procedure Upgrade Program goals described in Appendix A, the industry would incur costs in three of the five tasks. These costs are described in detail by task in this section.

Task 1

Since Task 1 involves only NRC expenses, no industry costs are incurred.

Task 2

Table 4.15 lists the personnel requirements and labor costs for convening a working group.

TABLE 4.15. Task 2 Personnel Requirements and Wage Rates

Personnel Type	Wage Rate ($/hr)	Hours	Labor Costs ($)
Owners' Group Representative	52	384	19,968
Vendor Representative	52	192	9,984
Experts	52	192	9,984
Industry Group Representative	52	192	9,984
Licensee Representative	52	192	9,984

The wage rate used for the various industry personnel types is the rate for a utility engineering manager listed in NUREG/CR-4627 (Claiborne et al. 1989). According to this source, an engineering manager's wage rate is $26 per hour plus a two-fold increase for benefits. Thus, the full rate is $52 per hour. The hours required for completing Task 2 include the time for four owner's group representatives, two vendor representatives, two experts, two industry group representatives, and two licensee representatives. The labor costs are calculated by multiplying the wage rate by the number of required hours.

According to Figure A.1 in Appendix A, the costs will occur over the years 1990 and 1991. The year 1991, therefore, is used as the year from which the costs are discounted. At a 10% discount rate for 2 years, the factor 0.826 is used to convert the costs into 1989 dollars. Since the NRC pays for travel costs associated with this task, total industry costs are determined by multiplying the sum of the labor costs by the discount factor. The total industry cost associated with Task 2 is $59,904 x 0.826 = $49,481. To reflect uncertainty in the wage rates, $\pm$50% of the total is used to determine a range. The upper bound is approximately $74,221 and the lower bound is $24,740.

Task 3

Since this task requires the plants to thoroughly examine their operating procedures, all costs apply to industry. Table 4.16 highlights the personnel requirements and labor costs.

The wage rates for the plant operations manager, training specialist, and human factors specialist are assumed to be the same as that listed in NUREG/CR-4627 for a utility engineering manager. The wage rates for the plant engineer, senior reactor operator, and the writers are assumed to be the same as that listed for a plant engineer. The hours listed in Appendix A apply to one plant; therefore, the sum of the labor costs from Table 4.16 must be multiplied by 118 plants. Since the costs primarily occur in mid-1991, the discount factor, 0.790, reflects the average of 2 and 3 years at 10% rate. Discounting to 1989, the total industry cost is about $6,383,710. Adding 50% to this estimate, the upper bound is $9,575,560. The lower bound is $3,191,850.

Task 4

The requirements and costs for upgrading the procedures are specified in Table 4.17. The wage rate for a plant engineer as listed in NUREG/CR-4627 is applied to the human factors specialist and the training specialist. The clerical wage rate is assumed to be $22 per hour. The remaining staff are assumed to have a rate equivalent to a plant engineer. Since the costs for this task will be incurred between 1991 and 1993, 1992 is used as the year from which the costs are discounted. For 3 years at 10%, the discount factor is 0.751. Again, the labor costs in Table 4.17 apply to one plant. Multiplying the sum of the labor costs by 118 plants and by the discount factor yields the total industry costs, $35,128,170 with an upper bound of $52,692,260 and a lower bound of $17,564,090.

Task 5

All costs for this task are incurred by the NRC; therefore, no industry costs are accumulated.

TABLE 4.16. Task 3 Personnel Requirements and Wage Rates

Personnel Type	Wage Rate ($/hr)	Hours	Labor Costs ($)
Plant Operations Manager	52	80	4,160
Plant Engineer	38	480	18,240
Senior Reactor Operator	38	480	18,240
Training Specialist	52	160	8,320
Human Factors Specialist	52	120	6,240
Technical Writer	38	160	6,080
Procedure Writers	38	120	4,560
Clerical Support	22	120	2,640

TABLE 4.17. Task 4 Personnel Requirements and Wage Rates

Personnel Type	Wage Rate ($/hr)	Hours	Labor Costs ($)
Licensed reactor operator	38	3,840	145,920
Technical Writers	38	1,920	72,960
Human Factors Specialist	52	480	24,960
Plant Engineer	38	480	18,240
Training Specialist	52	480	24,960
Plant Safety Review Committee Members	38	600	22,800
Procedures Users (validation)	38	600	22,800
Procedures Users (training)	38	1,400	53,200
Clerical Support	22	480	10,560

Table 4.18 summarizes the industry implementation costs by task. As expected, the most costly task, Task 4, is the actual implementation of the upgrade program at all 118 reactors.

TABLE 4.18. Summary of Industry Implementation Costs

	Industry Implementation Costs ($)		
Task	Best Estimate	Upper Estimate	Lower Estimate
1	0	0	0
2	49,500	74,200	24,700
3	6,383,700	9,575,600	3,191,900
4	35,128,200	52,692,300	17,564,100
5	0	0	0
	41,561,400	62,342,100	20,780,700

4.2.2 Industry Operation Costs

After the implementation of the candidate Procedure Upgrade Program at a reactor, it is estimated that there would be cost savings due to two improvements. The first improvement would be due to a reduction in the amount of time and money spent in writing and updating procedures. The second improvement would be an increase in the plant capacity factor for the reactor. Both of these improvements are discussed in qualitative detail in the following subsections.

4.2.2.1 Operating Procedure Revisions and Administration

Each utility annually spends a substantial amount of time and money updating and revising normal operating procedures and abnormal operating procedures. The intent of the candidate Procedure Upgrade Program is to perform a one-time streamlining review of all current operating procedures and systematically implement new procedures. The candidate program assumes that licensees would adopt a two-tiered approach to the development, use, and administrative control of operating procedures. (See section A.3 of Appendix A.) Tier 1 procedures would be created for operations tasks that require high-quality, detailed, step-by-step procedures to be performed in a timely and error-free manner. These operations tasks would be likely to include those performed using safety-significant systems or components, complex tasks, tasks that are infrequently performed, tasks that involve the coordination of several workers' activities in one location or in dispersed locations, tasks that tend to be error-prone, and so on. Tier 2 procedures, on the other hand, would be created for tasks that require some form of job performance aid (e.g., checklists, tables, calculation forms), but for which a high-quality detailed procedure is unnecessary or undesirable. Such tasks would be those that are frequently performed and which any trained operator could perform with only a checklist reminder of step sequencing, tasks performed on simple or nonsafety-related systems or equipment, and so on.

Adoption of this two-tier approach could tend to result in a reduction in licensee costs for revision, review, and administration of operating procedures. Procedures created pursuant to the candidate program would be of higher quality and better fit to the operations tasks they control. These characteristics would tend to decrease the need for future procedure revisions. Also, there would be fewer procedures for licensees to deal with because job-performance aids, rather than procedures, could be used to guide Tier 2 tasks. Currently, the same level of resources is expended on the review, revision, and administrative control of all types of procedures, regardless of the safety-significance of the tasks they control. Because of the candidate program, procedure revision and administrative costs should be reduced because of a decrease in the number of procedures for which full-blown revision, reviews, and revision approvals may be necessary.

A limited analysis was performed to attempt to arrive at an approximate reduction in procedure revision costs that could result from the candidate program. Data from one plant suggest that the total annual costs of revising normal operating procedures and abnormal operating procedures is approximately $310,000 at that plant. Plant personnel and PNL researchers agreed that the two-tier approach could reduce such costs by as much as 50%, or about $155,000 per year. Extending that per-plant savings to the industry as a whole would indicate that the cost of revising operating procedures could be reduced by an estimated $18,000,000 annually by using the two-tier approach.

4.2.2.2 Plant Capacity Factors

As pointed out above, the implementation of the candidate Procedure Upgrade Program could reduce the frequency of unplanned reactor trips. While reducing reactor trip frequency is an important safety goal in itself, a reduction in unplanned trips would also result in an improvement in overall plant capacity factors. This increase in plant capacity factors could result from a decrease in the frequency of procedure-related operational errors that can lead to reactor trips. Fewer unplanned reactor trips would, in turn, lead to a reduction in required replacement power costs. Therefore, the costs incurred by licensees as a result of implementing the candidate program could be at least partially offset by a decrease in plant costs associated with increasing plant capacity factors.

The five transient initiating-events identified in Table 4.6 were used as part of the basis of a qualitative analysis of the economic benefits to licensee plants that could potentially result from this reduction of unplanned reactor trips. The sum of the reduction in frequency of these events due to improved procedures indicates an overall reduction in transient initiating-event frequency of approximately 0.1 per reactor-year. This frequency reduction was calculated by simply subtracting the adjusted- case frequencies from the base-case frequencies for each of the five transient initiating-events, and then summing those results. This provided an estimate of the overall reduction in trip initiating-event frequency using only those data used in this project's primary analysis of core-melt reduction leading to avoided public health risk.

Improved operating procedures may result in a reduction in other types of trip initiating-events in addition to those identified solely for purposes of estimating potential core-melt reduction. For instance, trips occurring at very low reactor power levels would not likely result in safety-significant consequences. Because these events would not be expected to lead to safety-significant consequences, they were not included in the analysis of potential core-melt reduction. Reduction of the frequency of these events could, however, reduce plant operating costs associated with unplanned outages.

The NRC's Performance Indicator program classifies and tracks those LERs that involve automatic scrams, safety significant failures, safety system actuations, and significant events. The annual industry average for automatic trips while the reactor was critical was 2.26 in 1988 (NRC 1989a). The values for the years 1987, 1986, and 1985 were 3.64, 4.41, and 5.22, respectively.

In addition, personnel related problems, including procedure deficiencies and human error, have been attributed to 26 and 30% of reactor scrams above 15% power for the years 1984 and 1985, respectively (Hebdon and Dennig 1987). Procedure deficiencies alone were identified as the cause of 4% of the total number of scrams in 1984 and 5% of the total number of scrams in 1985.

These sets of empirical data indicate that, although the total number of scrams is being reduced on the average, the contribution of procedure deficiencies to reactor trips is not. In fact, the 1984 and 1985 data indicate that the <u>relative</u> contribution of procedure deficiencies to scrams has increased slightly. Improved operating procedures could clearly help to reduce this contribution to unanticipated reactor scrams. In addition to reducing the frequency of scrams caused by procedure deficiencies, the candidate Procedure Upgrade Program could be expected to reduce the frequency of human errors leading to unanticipated reactor scrams.

Therefore, it appears reasonable to assume that implementation of the candidate Procedure Upgrade Program could, on the average, result in a reduction in unanticipated reactor trips of from 0.1 to 0.2 per reactor-year. These values correspond to approximately 5 and 10% of the 1988 estimated average annual reactor scrams frequency of 2.3. It should be noted that many plants could achieve a greater reduction, while some plants whose trip frequency is already low may not realize as large a reduction.

Based on the results of a previous study on transient initiating-event frequencies (Mackowiak et al. 1985), a reasonable estimate of the average outage time associated with transient initiating-events caused by procedure-related errors is 48 hours. This estimate assumes that there are no significant equipment failures associated with the transient and that there is no significant damage to hardware components resulting from the reactor trip. Assuming that the frequency of reactor trips could be reduced between 0.1 and 0.2 per reactor-year, that an unplanned reactor trip results in 2 days of downtime, and that replacement power costs are approximately $372,000 per day (Daling et al. 1989), the candidate program could be expected to produce a cost savings of from approximately $74,400 to $148,000 per reactor-year. The total estimated economic benefits of the candidate program associated with increased plant capacity factors for 118 operating plants could be from approximately $8,800,000 to approximately $17,600,000 per year.

4.2.3 NRC Development Costs

This section estimates the NRC development costs (i.e., the costs of preparations prior to implementation). The following steps are involved: 1) issue a completed procedure upgrade program to selected plants for comments and revise the program based on the comments, and 2) prepare and issue a Generic Letter.

The first step would involve 2 weeks of a consultant's time including a 4-day meeting in Washington, D.C., and 6 weeks of NRC staff time. The consultant's labor rate is assumed to be equivalent to that of a utility engineering manager (i.e., $52 per hour) for a total labor cost of $4,160. Assuming roundtrip airfare costs of $918 and a per diem rate in Washington, D.C. of $121, a 4-day meeting for the consultant would cost about $1,400. At a labor rate of $41 per hour, the NRC staff labor cost would be $9,840. Adding the labor costs to the travel expenses yields a $15,400 total cost for this step.

According to NUREG/CR-4627, the cost of preparing and issuing a complicated Generic Letter is $98 per power reactor (based on 100 reactors). Thus, the total cost would be $9,800. Although the procedures must be changed at 118 reactors, the total cost will not change. Only the cost per reactor would be modified to $83.

Adding the costs for both steps determines a total cost for NRC development of about $25,200. As with industry implementation costs, the upper bound is +50% of the best estimate, or $37,800. The lower bound is -50%, or $12,600.

4.2.4 NRC Implementation Costs

To implement the Procedure Upgrade Program outlined in Appendix A, the NRC would incur costs in three of the five tasks. These costs are described in detail by task in this section. The NRC incurs all travel costs for the implementation of the new procedures. Using NUREG/CR-4627, the average NRC wage rate of $41 per hour was used for NRC personnel in the labor cost calculations.

Task 1

Developing the Good Practices Document requires one human factors research scientist, two technical writers, one technical monitor, and clerical support. Table 4.19 outlines the wage rate, hours, and labor costs of these workers.

TABLE 4.19. Task 1 Personnel Requirements and Wage Rates

Personnel Type	Wage Rate ($/hr)	Hours	Labor Costs ($)
Human Factors Research Scientist	52	480	24,960
Technical Writers	41	640	26,240
Technical Monitor	41	160	6,560
Clerical	41	80	3,280

The human factors specialist is considered to be a consultant to the NRC with a wage rate equivalent to a utility engineering manager. The remaining personnel are considered to be NRC personnel with an average NRC wage rate. The four people involved with this task will also attend two 2-day meetings. Travel costs, however, will only be incurred by the consultant since the other workers will be based in Washington, D.C. Table 4.20 outlines the data used to determine the travel costs. The formula for determining travel costs is:

TABLE 4.20. Task 1 Travel Costs

Number of attendees	1
Airfare	$918
Per diem in Washington, DC	$121
Length of meetings (in days)	2
Number of meetings	2

Number of Attendees x Number of Meetings x [Airfare + (Per Diem x Days)]

Applying the data from Table 4.20 to the formula, the travel cost for this task is $2,320. Adding the labor and travel costs, and using the discount factor, 0.826, the total NRC cost for writing the document is $52,340. The upper bound is $78,500 and the lower bound is $26,170.

Task 2

As listed in Table 4.21, two NRC representatives and clerical support are the only NRC personnel requirements for this task.

TABLE 4.21. Task 2 Personnel Requirements and Wage Rates

Personnel Type	Wage Rate ($/hr)	Hours	Labor Costs ($)
NRC Representative	41	192	7,872
Clerical	41	80	3,280

Data used to determine travel costs associated with Task 2 are shown in Table 4.22. All industry personnel listed in Table 4.22 will be attending the six 2-day meetings in Washington, D.C. To make the costs conservative, the cost of a round trip airline ticket from Seattle to Washington, D.C. is used to determine the airfare value. The total travel costs are calculated using the formula described in Task 1.

TABLE 4.22. Task 2 Travel Costs

Number of attendees	12
Airfare	$918
Per diem in Washington, DC	$121
Length of meetings (in days)	2
Number of meetings	6

Since the costs will occur over the years 1990 and 1991, 1991 is used to discount the costs to 1989 dollars. At a 10% discount rate for 2 years, the factor 0.826 is used for the cost conversion. Adding labor costs and travel expenses, the total NRC cost for this task is $78,200 with an upper bound of $117,300 and a lower bound of $39,100.

Task 3

No NRC costs are associated with this task

Task 4

Only industry employees are needed for this task; therefore, no NRC costs are incurred for Task 4.

Task 5

All costs for this task are incurred by the NRC. As listed in Appendix A, either a NRC or contractor staff person is required for this task. Because the contractor's wage rate is the higher of the two, this rate is used to reflect conservative costs. The contractor and training specialist are assumed to have wage rates equivalent to a utility engineering manager. Applied to the NRC inspectors is the average NRC wage rate listed in NUREG/CR-4627. The costs and staffing conditions are outlined in Table 4.23.

TABLE 4.23. Task 5 Personnel Requirements and Wage Rates

Personnel Type	Wage Rate ($/hr)	Hours	Labor Costs ($)
Contractor Staff Person	52	160	8,320
Training Specialist	52	160	8,320
NRC Inspectors (in training)	41	240	9,840
NRC Inspection Team	41	60,800	2,492,800
Clerical Support	22	160	3,520

Travel costs for this task include two sets of costs. The first set of travel costs will be incurred by the 10 NRC inspectors traveling to Washington, D.C. for training. Since the inspectors will travel from all regions, Chicago is used as a typical origin. The per diem rate for Washington, D.C. is used. The second set of costs will be incurred by the inspection team. Travelling to plants within their region, the inspectors are assumed to travel a route similar in distance to the one from Chicago to Atlanta. The average per diem rate for the United States is used. Table 4.24 highlights the data used to determine the costs associated with both sets of travel expenses.

TABLE 4.24. Task 5 Travel Costs

	Set 1	Set 2
Number of attendees	10	3
Airfare	$500	$675
Per diem	$121	$ 66
Length of meetings (in days)	3	5
Number of meetings	1	118

Using the formula described in Task 1, the total travel cost for Set 1 is $8,630 and $355,770 for Set 2. Adding these costs to the labor costs and using a 10%, 4.5 year discount rate (0.652), the total NRC costs for Task 5 are $1,882,450. The upper bound is $2,823,680 and the lower bound is $941,230.

Table 4.25 summarizes the NRC implementation costs by task. As expected, the most costly task is the actual inspection of all 118 reactors to ensure compliance with the program.

TABLE 4.25. Summary of NRC Implementation Costs

	NRC Implementation Costs ($)		
Task	Best Estimate	Upper Estimate	Lower Estimate
1	52,300	78,500	26,200
2	78,200	117,300	39,100
3	0	0	0
4	0	0	0
5	1,882,500	2,823,700	941,200
	2,013,000	3,019,500	1,006,500

4.2.5 NRC Operation Costs

It is estimated that there would be no change in NRC operation costs due to the candidate Procedure Upgrade Program. NRC schedules and procedures for normal reactor inspections would not change, so no additional inspection burdens will be placed on the regional inspectors.

..3 VALUE-IMPACT ASSESSMENT SUMMARY

Tables 4.26, 4.27, and 4.28 summarize both the values and the impacts. The values for public property damage avoided and on-site property damage avoided are moved to the impact category to keep the units consistent. That is, all units expressed in terms of dollars are grouped together under the impact category. Similarly, units expressed in terms of person-rems are grouped under the value category. In moving the property figures to the impact group, the figures for the two attributes are changed to negative numbers, indicating a cost savings or benefit. Table 4.26 shows the best estimates for both the values and impacts. Table 4.27 shows the best case scenario: the highest public risk reduction and avoided occupational exposure, largest avoided property damage, and lowest industry and NRC costs. Similarly, Table 4.28 presents the worst case: the lowest public risk reduction and avoided occupational exposure, the lowest avoided property damage, and the highest industry and NRC costs. The latter two tables show the possible extreme scenarios that could occur.

As previously mentioned, the costs are determined for all plants regardless of the current quality of operating procedures. Therefore, to

TABLE 4.26. Summary of Best-Estimate Values and Impacts Integrated Over the Average Remaining Lifetimes of Affected Plants

Values (person-rem)	Current Quality of Procedures: Relatively Poor	Intermediate	Relatively Good	Related Page or Table
Public Risk Reduction	3.2E4	9.1E3	1.4E3	Table 4.11
Avoided Occupational Exposure	7.6E2	2.1E2	3.4E1	Table 4.12
Total Quantified Value	3.3E4	9.3E3	1.4E3	--
Impacts (dollars)				
Public Property	-1.8E7[a]	-4.9E6	-7.7E5	Table 4.13
Onsite Property	-8.7E6	-2.4E6	-3.8E5	Table 4.14
Industry Costs	1.7E7	1.6E7	7.9E6	--
NRC Costs	8.6E5	7.9E5	3.9E5	--
Total Quantified Impact	-8.8E6	9.5E6	7.1E6	--

(a) Negative impacts indicate cost savings.

TABLE 4.27. Summary of Best-Case Values and Impacts

Values (person-rem)	Current Quality of Procedures: Relatively Poor	Intermediate	Relatively Good	Related Page or Table
Public Risk Reduction	3.9E4	1.6E4	4.6E3	Table 4.11
Avoided Occupational Exposure	1.5E3	6.1E1	1.8E2	Table 4.12
Total Quantified Value	4.1E4	1.6E4	4.8E3	--
Impacts (dollars)				
Public Property	-1.2E8[a]	-4.8E7	-1.4E7	Table 4.13
Onsite Property	-1.6E7	-6.4E6	-1.9E6	Table 4.14
Industry Costs	8.7E6	8.1E6	3.9E6	--
NRC Costs	4.3E5	4.0E5	1.9E5	--
Total Quantified Impact	-1.3E8	-4.6E7	-1.2E7	--

(a) Negative impacts indicate cost savings.

TABLE 4.28. Summary of Worst Case Values and Impacts

Values (person-rem)	Current Quality of Procedures: Relatively Poor	Intermediate	Relatively Good	Related Page or Table
Public Risk Reduction	2.0E4	0	0	Table 4.11
Avoided Occupational Exposure	2.2E2	0	0	Table 4.12
Total Quantified Value	2.0E4	0	0	--
Impacts (dollars)				
Public Property	-1.4E6[a]	0	0	Table 4.13
Onsite Property	-2.6E6	0	0	Table 4.14
Industry Costs	2.6E7	2.4E7	2.2E7	--
NRC Costs	1.3E6	1.2E6	5.8E5	--
Total Quantified Impact	2.3E7	2.5E7	1.3E7	--

(a) Negative impacts indicate cost savings.

apply costs to the different categories of procedures, ratios are determined based on the number of plants in each category. For plants with relatively poor operating procedures, the total costs are multiplied by the ratio of poor plants to all plants. The ratio for plants in the relatively poor category is 50 plants divided by 118 total plants yielding 42%. Using the same formula for the 46 plants in the intermediate category and [illegible] the good category, the ratios are 39% and 19%, respectively. These percentages are multiplied by the total industry costs, including best estimates and upper and lower bounds. The three estimates for NRC costs are also multiplied by the percentages. Figure 4.2 is a graphical representation of the estimates of the values and costs. The best estimates and the upper and lower bounds are displayed.

Although plants in the relatively poor category have the largest range of values and costs, they would receive the most benefit in terms of public risk reduction and avoided occupational exposure. Also, according to the best estimate, these plants would have a cost savings of $8.8 million. Having a much smaller range, plants in the intermediate and relatively good categories show positive values would be gained but to a lesser extent than for those plants in the relatively poor category. Plants in these two categories would incur costs between ≈$7 million and $9 million.

An alternative way of considering the quantified results of the value-impact assessment is presented by Table 4.29. While the previous three tables show the estimated values and impacts for each of the three categories of plants based on current operating procedure quality, Table 4.29 summarizes these results by displaying the total effect on each value-impact attribute across all plants. This table also presents a summary calculation of these results by dividing the total estimated impact of the candidate program,

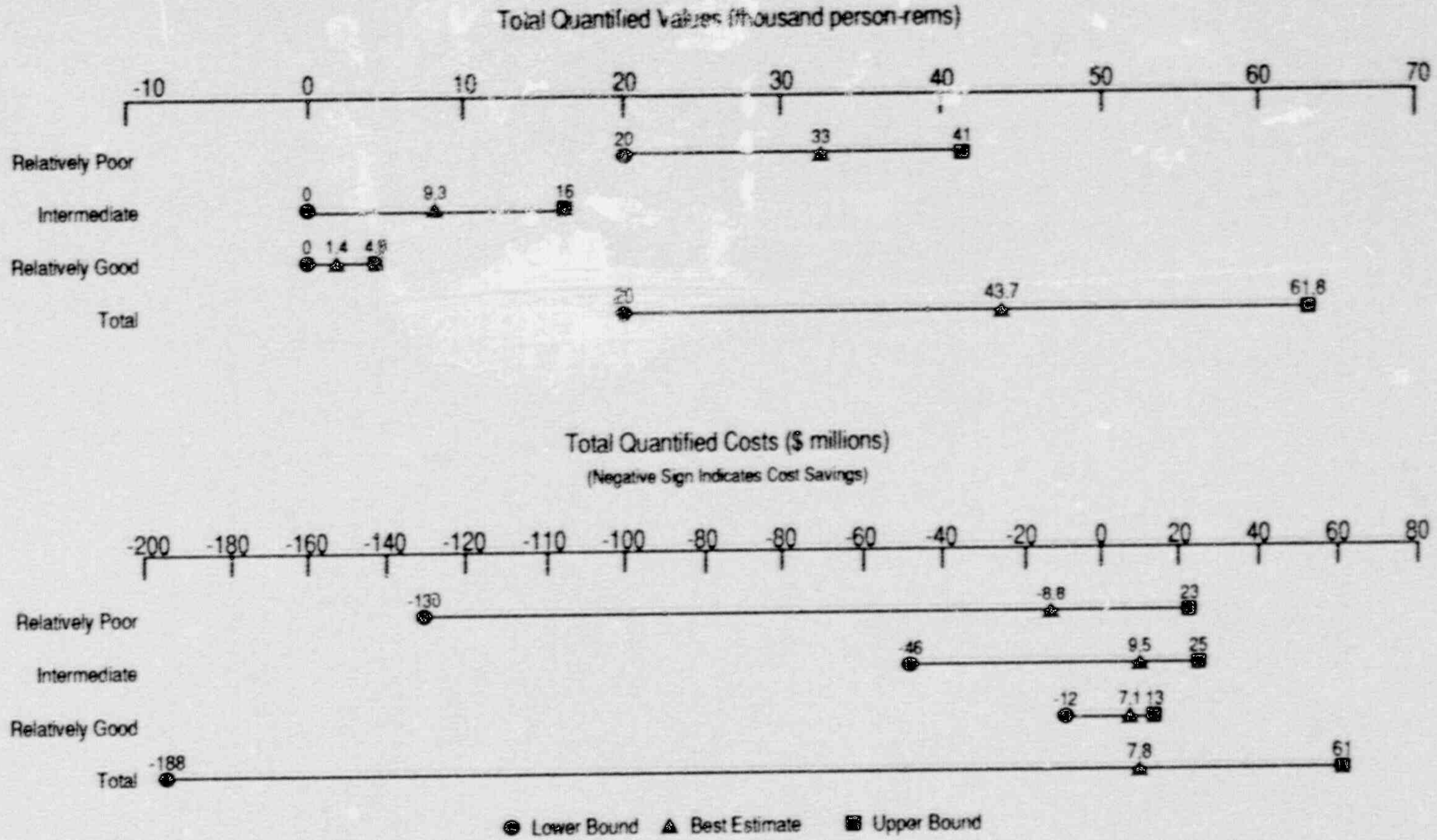

FIGURE 4.2. Range of Total Quantified Values and Costs for Each Category of Plants

TABLE 4.29. Value-Impact Summary

Attribute	Units	Best Estimate	Best Case	Worst Case
Public Risk Reduction	(person-rem)	4.3E4	6.0E4	2.0E4
Avoided Occupational Exposure	(person-rem)	1.0E3	2.3E3	2.2E2
Total Value	(person-rem)	4.4E4	6.2E4	2.0E4
Avoided Onsite Property Damage	(dollars)	-1.1E7[(a)]	-2.4E7[(a)]	-2.6E6[(a)]
Industry Implementation Costs	(dollars)	4.2E7	2.1E7	6.2E7
Industry Operation Costs	(dollars)	NQ	NQ	NQ
NRC Development Costs	(dollars)	2.5E4	1.3E4	3.8E4
NRC Implementation Costs	(dollars)	2.0E6	1.0E6	3.0E6
NRC Operation Costs	(dollars)	NA	NA	NA
Total Impact[(b)]	(dollars)	3.3E7	-2.0E6[(a)]	6.2E7
Total Impact/Total Value	(dollars/ person-rem)	7.5E2	-3.2E1[(c)]	3.1E3

NQ = Not Quantified. Industry Operation Costs were analyzed on a qualitative, rather than a quantitative, level.
NA = Not Affected.
(a) Favorable impacts have a negative sign.
(b) Because the NRC is inclined to evaluate the results of value-impact assessments against a standard of $1,000 per person-rem, and because this standard figure is assumed to include the estimated potential impact of the regulatory action on risk to public property, the estimated potential impact on Public Property Damage Avoided has not been included in this table.
(c) -3.2E1 signifies that, for the best case estimate, the regulatory action could reduce exposure to humans by 62,000 person-rems accompanied by a $2 million net benefit.

measured in dollars, by the total estimated value of improving operating procedures, measured in person-rems. Because the NRC compares this cost per person-rem avoided with a standard measure of $1,000 per person-rem, and because this standard figure is assumed to include the estimated potential impact of the candidate regulatory action on risk to offsite property, the estimated potential impact on Public Property Damage Avoided has not been included in this table nor in these summary calculations.

These summary results indicate that the best estimate of the cost of improving operating procedures would be approximately $750 per person-rem avoided. Under the best-case scenario, the candidate regulatory action would result in 62,000 person-rems avoided while at the same time generating a combined net benefit of $2 million for the industry and the NRC. Under the worst case scenario, improved procedures could be expected to result in a total cost of approximately $3,100 per person-rem avoided. Again, it should be noted that these summary results do not include the considerable estimated benefits of potential damage to public property avoided due to improved operating procedures.

Improved operating procedures could lead to additional benefits beyond those quantified in this assessment and summarized in these four tables. High quality normal and abnormal operating procedures could be expected to cause a reduction in reactor trip frequency at U.S. power plants due to a decrease in procedure-related operational errors. A reduction in trips would have both beneficial safety and economic effects. Fewer trips would mean fewer challenges to plant safety systems as well as a reduction in the frequency of technical specification violations. As a beneficial economic consequence of a reduction in trips, the industry could decrease its expenses for replacement power purchased during outages following trips. The industry could also reduce its expenses for revising and administering operating procedures if procedures are upgraded using the two-tier approach described in this assessment. Because procedures would be of higher quality and because the two-tier approach would lead to plants having fewer normal operating procedures that are subject to rigorous development and control, the industry could expect to spend less effort over time for revision and administration of operating procedures.

5.0 SENSITIVITY ANALYSIS

Although a variety of uncertainties are associated with both the physical phenomena and the calculational tools described in this report, a rigorous quantitative uncertainty analysis is beyond the scope of this effort. The description of uncertainties in this report is primarily qualitative with some limited quantification. In this section, it is the intention to ameliorate this shortcoming by performing a limited sensitivity analysis on several of the key elements of the analysis.

Sensitivity analyses that focus on public health risk factors permit analysis of the importance of the various factors that contribute to the public health risk. The analysis results can be used to: 1) identify and quantify the effects of major contributors to risk, 2) identify ways to decrease the uncertainty in the evaluation, and 3) study the effects of possible design or regulatory changes on the risk. Similarly, sensitivity analyses that focus on cost factors allow evaluations to be made regarding the various factors that contribute to economic costs. Most sensitivity studies are performed by repeating the calculations with a changed value for the parameter of interest. In general, the dependence of the final result on a particular parameter is complex; although, in some cases, the parameter enters simply into the calculations and the sensitivity can be determined directly.

For this value-impact assessment, the areas presenting the greatest uncertainty in the risk calculations are the dose conversion factors, the results of the SCSS analysis, and the evaluation of the relationship between the generic plant model and the three categories of actual plants. The key assumption in the cost calculations is that of the discount rate applied to future costs. To test the effects of these uncertainties and assumptions on the results of the value-impact assessment provided in Section 4.3, each of these areas of interest is addressed in the following subsections.

5.1 DISCOUNT RATE (SENSITIVITY ANALYSIS #1)

The best estimates of the following impacts are calculated here using a discount rate of 5% rather than the 10% rate applied in the main analysis.

Public Property

The value of public property damage avoided can be calculated as follows:

$$V_{FP} = \Delta\bar{F}ND_G$$

where V_{FP} = impact of avoided offsite property damage ($)

$\Delta\bar{F}$ = change in core-melt probability (events/reactor-year)

N = number of affected facilities

D_G = generic present value of property damage conditional on release ($/event).

The values for $\Delta\bar{F}$ and N are the same as defined in Chapter 4. The present value of the property damage (D_G) can be calculated as:

$$D_G = C \times B$$

where $C = \dfrac{e^{(-0.05\ t_i)} \quad {(-0.05\ t_f)}}{0.05}$

t_i = years before procedures take effect = 1993 - 1989 = 4 years

t_f = years from 1993 until end of procedure use = 26 years

B = scaled result for property damage = same values as defined in Chapter 4.

The value of C is:

$$C = \frac{e^{[-.05(4)]} - e^{[-.05(26)]}}{0.05} = 10.95$$

Table 5.1 lists the value of public property damage avoided when a 5% discount rate is applied.

Onsite Property

The formula for the value of onsite property damage avoided is:

$$V_{OP} = \Delta\bar{F}NU$$

where V_{OP} = impact of avoided onsite property damage ($)

$\Delta\bar{F}$ = change in core-melt probability (events/reactor-year)

N = the number of facilities affected

U = present value of property damage conditional on release ($/event).

TABLE 5.1. Sensitivity Analysis #1: Avoided Public Property Damage

Current Quality of Operating Procedures	Type of Estimate	$\Delta\bar{F}$ Change in Core Melt Probability (events/ry)	N Number of Plants	B Scaled Public Property Damage ($/event)	D_G Discounted Public Property Damage ($/event)	V_{FP} Impact of Avoided Public Property Damage ($)
Relatively Poor	Best	2.8E-5	50	2.09E9	2.29E10	3.21E7
	Upper	3.4E-5	50	1.15E10	1.26E11	2.14E8
	Lower	1.7E-5	50	2.72E8	2.98E9	2.53E6
Intermediate	Best	8.5E-6	46	2.09E9	2.29E10	8.95E6
	Upper	1.5E-5	46	1.15E10	1.26E11	8.67E7
	Lower	0	46	2.72E8	2.98E9	0
Relatively Good	Best	2.8E-6	22	2.09E9	2.29E10	1.41E6
	Upper	9.1E-6	22	1.15E10	1.26E11	2.52E7
	Lower	0	22	2.72E8	2.98E9	0

The values for $\Delta\bar{F}$ and N are the same as defined earlier. The present value of the onsite property damage (U) can be calculated as:

$$U = \left[\frac{\left(C_c + C_r + C_{rp}\right)}{m}\right] \frac{\left(e^{-rt_i}\right)}{r^2} \left[1 - e^{-r\left(t_f - t_i\right)}\right] \left(1 - e^{-rm}\right)$$

where U = present value of onsite property damage conditional upon release ($/event)

C_c = cleanup cost ($)

C_r = repair/refurbishment cost ($)

C_{rp} = replacement power cost ($)

t_i = years before procedures take effect, = 1993-1989 = 4 years

t_f = years from 1993 until end of procedure use, = 26 years

r = discount rate (for 5%, r = 0.05)

m = years required to return utility to preaccident state, 10 years.

The values of C_c, C_r, and C_{rp} are the same as defined in Chapter 4. Applying the numbers to the formula, the value for U is:

$$U = \frac{(1650M)}{10} \times \frac{e^{-[(0.05)(4)]}}{(0.05)^2} \times [1 - e^{-.05(26-4)}] \times [1 - e^{-(0.5)(10)}]$$

$$= 155M \times 327.49 \times 0.67 \times 0.394$$

$$= 1.426E10$$

The values for onsite property damage avoided using a 5% discount rate are summarized in Table 5.2.

TABLE 5.2. Sensitivity Analysis #1: Avoided Onsite Property Damage

Current Quality of Operating Procedures	Type of Estimate	$\Delta\bar{F}$ Change of Core-Melt Probability (events/ry)	N Number of Facilities	U Discounted Onsite Property Damage ($/event)	V_{op} Onsite Property Damage Avoided ($)
Relatively Poor	Best	2.8E-5	50	1.43E10	2.00E7
	Upper	3.4E-5	50	2.14E10	3.64E7
	Lower	1.7E-5	50	7.13E9	6.06E6
Intermediate	Best	8.5E-6	46	1.43E10	5.58E6
	Upper	1.5E-5	46	2.14E10	1.48E7
	Lower	0	46	7.13E9	0
Relatively Good	Best	2.8E-6	22	1.43E10	8.78E5
	Upper	9.1E-6	22	2.14E10	4.28E6
	Lower	0	22	7.13E9	0

Industry and NRC Costs

Since the implementation costs are analyzed by tasks that span different years, the costs and the 5% discount rate factors for the five tasks are shown in Table 5.3. The costs for Tasks 1 and 2 are discounted from 1991. The costs for Task 3 primarily occur in mid-199[illegible]; therefore, the costs are discounted over an average of 2 and 3 years. The costs for Task 4 are discounted from 3 years in the future. Finally, the discount factor for 4 years is used for Task 5. Using the labor and travel costs described in Chapter 4, Table 5.3 lists the total implementation costs for each sector for each task. Since the NRC development costs are not discounted, the change in the discount rate will not affect the estimates.

Table 5.4 illustrates the differences in costs between the two discount rates. By using the 5% rate, the total impact is negative, indicating cost savings. The choice of discount rates, therefore, is an important decision when estimating the costs of implementing new procedures.

TABLE 5.3. Sensitivity Analysis #1: Implementation Costs

Task	Discount Factors	Industry Implementation Costs: Best Estimate	Industry Implementation Costs: Upper Estimate	Industry Implementation Costs: Lower Estimate	NRC Implementation Costs: Best Estimate	NRC Implementation Costs: Upper Estimate	NRC Implementation Costs: Lower Estimate
1	0.907	0	0	0	57,500	86,200	28,700
2	0.907	54,300	81,500	27,200	85,900	128,800	42,900
3	0.886	7,159,400	10,739,200	3,579,700	0	0	0
4	0.864	40,413,800	60,620,700	20,206,900	0	0	0
5	0.823	0	0	0	2,376,200	3,564,300	1,188,100
		47,627,500	71,441,400	23,813,800	2,519,600	3,779,300	1,259,700

TABLE 5.4. Sensitivity Analysis #1: Best Estimates of Impacts

Impacts ($)	10% Rate	5% Rate
Public Property	-2.4E7	-4.2E7
Onsite Property	-1.1E7	-2.6E7
Industry Costs	2.0E6	2.5E6
NRC Costs	2.0E6	2.5E6
Total Impact	9.0E6	-1.8E7

5.2 DOSE CONVERSION FACTORS (SENSITIVITY ANALYSIS #2)

The dose conversion factors used in the calculations throughout this value-impact assessment correspond directly with those used in the reference PRA (Sugnet et at. 1984). These dose conversion factors, which are provided in Table 5.5, are reported as a range of possible consequences of a core-melt accident for each release category. In order to ensure conservatism in the final estimate of public risk, the upper limit on the estimated range is used as the "best estimate" for all calculations of public risk in this value-impact assessment, as well as in the reference PRA. In this sensitivity analysis however, the lower limit of the range is used to calculate a lower bound to the public health risk. Since the best estimate and the upper bound are (in this case) equivalent, this sensitivity analysis provides only an estimate of how much lower the public health risk results would go if the lower bounds of the population doses were used as dose conversion factors.

TABLE 5.5. Sensitivity Analysis #2: Consequence Ranges Affecting Public Risk

Release Category	Population Dose (person-rem)
1A	1.0E8 to 3.0E8
1B	1.0E6 to 4.0E7
2	1.0E6 to 1.0E8
3	No effect
4	0 to 1.0E6
5	No effect

Adapted from the reference PRA (Sugnet et al. 1984).

Since this sensitivity analysis does not affect the change in core-melt frequency, only the public risk estimates are affected. Therefore, the only value-impact assessment attribute that is affected by a change in core-melt frequency is public health. All other attribute values will remain constant. The results of this sensitivity analysis are shown in Tables 5.6 and 5.7.

TABLE 5.6. Sensitivity Analysis #2: Core-Melt Frequencies

Plant Category Based on Current Quality of Operating Procedures	Reduction in the Core-Melt frequency (core-melt events/ry)	ΔW Reduction in the Public Health Risk (person-rem/ry)	
		Best/Upper Estimate	Lower Estimate
Relatively Poor			
Best Estimate	2.8E-5	24.9	0.47
Upper Estimate	3.4E-5	30.2	0.57
Lower Estimate	1.7E-5	15.1	0.29
Intermediate			
Best Estimate	8.5E-6	7.6	0.[illegible]4
Upper Estimate	1.5E-5	13.3	0.25
Lower Estimate	0	0	0
Relatively Good			
Best Estimate	2.8E-6	2.5	0.05
Upper Estimate	9.1E-6	8.1	0.15
Lower Estimate	0	0	0

TABLE 5.7. Sensitivity Analysis #2: Public Health Risk

Current Quality of Operating Procedures	Type of Estimate	ΔW Reduction in Public Health Risk (person-rem/ry)	N Number of Facilities	$\bar{T}$ Ave. Time (y)	V_{PH} Value of Public Health Risk Avoided (person-rem)
Relatively Poor	Best	0.47	50	26	6.11E2
	Upper	0.57	50	26	7.41E2
	Lower	0.29	50	26	3.77E2
Intermediate	Best	0.14	46	26	1.67E2
	Upper	0.25	46	26	2.99E2
	Lower	0	46	26	0
Relatively Good	Best	0.05	22	26	2.86E1
	Upper	0.15	22	26	8.58E1
	Lower	0	22	26	0

The values in Table 5.7 can be compared to the corresponding values associated with the upper limit dose conversation factors appearing in Table 4.11.

The formula used to develop the total avoided public health risk estimates is:

$$V_{PH} = \Delta W N \bar{T}$$

where V_{PH} = value of public health risk avoided (person-rem)

ΔW = avoided public dose per reactor-year (person-rem/ry)

N = number of affected facilities

$\bar{T}$ = average time new procedures are used at facilities (years).

The values of N and $\bar{T}$ are the same as defined in Chapter 4.

5.3 SCSS DATABASE RESULTS

The results of the SCSS analysis established a baseline from which to evaluate the current contribution of procedure-related operational errors to the transient initiating-events identified as being potentially affected by the Procedure Upgrade Program. The results of the SCSS analysis indicated that only a relatively small fraction of each of the transient initiating-events could be affected by improving operating procedures. The estimated contribution of procedure-related errors to the five affected transient event frequencies ranges from 1% to 5%, as given in Table 4.5.

In this sensitivity analysis, each of the 11 affected parameters have had their base-case estimated contribution from procedure-related errors adjusted up and down by a factor of 2. For example, the SCSS database predicted that approximately 5% of all reactor/turbine trips can be attributed to procedure-related operational errors. In this sensitivity analysis, values of 10% and 2.5% were used as upper and lower bounds on this contribution. Once the upper- and lower-bound base-case affected frequencies were calculated, the best estimate of the relative reduction in the base-case affected frequency was applied. The results of this sensitivity analysis are shown in Table 5.8.

5.4 CURRENT QUALITY OF OPERATING PROCEDURES (SENSITIVITY ANALYSIS #4)

The base-case position of the three categories of plants relative to the generic (or hybrid) plant indicates their current relative quality of operating procedures. This relationship is discussed in Section 4.1.1. These relative positions have been quantitatively evaluated by a panel of PNL experts. These estimates, listed in Table 4.7, are used as the "best estimate" values for all the calculations used in the value-impact assessment, including this sensitivity analysis. In order to generate the upper and lower bounds, each of these estimates has been adjusted up and down by a factor of 2. The results are shown in Table 5.9.

TABLE 5.8. Sensitivity Analysis #3: Core-Melt and Public Health Results[a]

Plant Category which is Based on Quality of Current Operating Procedures	Reduction in the Core-Melt Frequency (core-melt events/ry)	Reduction in the Public Health Risk (person-rem/ry)
Relatively Poor		
Best Estimate	2.8E-5	24.9
Upper Estimate	2.83E-5	25.2
Lower Estimate	2.79E-5	24.8
Intermediate		
Best Estimate	8.5E-6	7.6
Upper Estimate	8.61E-6	7.7
Lower Estimate	8.45E-6	7.5
Relatively Good		
Best Estimate	2.8E-6	2.5
Upper Estimate	2.83E-6	2.5
Lower Estimate	2.79E-6	2.5

(a) Since the change in the core-melt frequency associated with this sensitivity calculation is so small, it is evident that the overall results of the value-impact assessment are not particularly sensitive to this calculation. Therefore, the values for core-melt frequency reduction provided here are extended to estimates of the public risk reduction but to no other values.

TABLE 5.9. Sensitivity Analysis #4: Core-Melt and Public Health

Plant Category which is Based on Quality of Current Operating Procedures	$\Delta\bar{F}$ Reduction in the Core-Melt frequency (core-melt events/ry)	ΔW Reduction in the Public Health Risk (person-rem/ry)
Relatively Poor		
Best Estimate	2.8E-5	24.9
Upper Estimate	7.7E-5	68.5
Lower Estimate	1.1E-5	9.8
Intermediate		
Best Estimate	8.5E-6	7.6
Upper Estimate	2.1E-5	18.7
Lower Estimate	3.8E-6	3.4
Relatively Good		
Best Estimate	2.8E-6	2.5
Upper Estimate	5.7E-6	5.1
Lower Estimate	1.2E-6	1.1

Public Risk

The changes in the reduction in the public health risk are shown in Table 5.9. The new risk estimates are applied to the formula described in Chapter 4. Table 5.10 shows the value of public health risk avoided.

TABLE 5.10. Sensitivity Analysis #4: Avoided Public Health Risk

Current Quality of Operating Procedures	Type of Estimate	ΔW Reduction in Public Risk (person-rem/ry)	N Number of Facilities	$\bar{T}$ Ave. Time (y)	VPH Value of Public Health Risk Avoided (person-rem)
Relatively Poor	Best	24.9	50	26	3.24E4
	Upper	68.5	50	26	8.91E4
	Lower	9.8	50	26	1.27E4
Intermediate	Best	7.6	46	26	9.09E3
	Upper	18.7	46	26	2.24E4
	Lower	3.4	46	26	4.07E3
Relatively Good	Best	2.5	22	26	1.43E3
	Upper	5.1	22	26	2.92E3
	Lower	1.1	22	26	6.29E2

Table 5.9 also lists the estimates of the modified core-melt frequencies for each category of plants. The modified values of $\Delta\bar{F}$ are used to determine new values of the avoided occupational exposure, public property damage avoided, and onsite property damage avoided as discussed in the following subsections.

Occupational Exposure

The value of avoided occupational exposure can be calculated as follows:

$$V_{OHA} = \Delta\bar{F}N\bar{T}\ (D_{IO} + D_{LTO})$$

where V_{OHA} = value of occupational health risk due to accidents avoided (person-rems)

$\Delta\bar{F}$ = change in core-melt probability (events/reactor-year)

N = number of affected facilities

$\bar{T}$ = average time procedures are used at facilities (years)

D_{IO} = "immediate" occupational dose (person-rem)

D_{LTO} = long-term occupational dose (person-rem).

The values for N, $\bar{T}$, D_{IO}, and D_{LTO} are the same as defined in Chapter 4. The new values for occupational exposure avoided are shown in Table 5.11.

TABLE 5.11. Sensitivity Analysis #4: Avoided Occupational Exposure

Current Quality of Operating Procedures	Type of Estimate	$\Delta\bar{F}$ Change in Core-Melt Probability (events/ry)	Number and Lifetime of Facilities N ($\bar{T}$)	$D_{IO} + D_{LTO}$ Additional Occupational Dose (person-rem/ event)	V_{OHA} Value of Avoided Risk (person-rem)
Relatively Poor	Best	2.8E-5	50 (26)	2.1E4	7.644E2
	Upper	7.7E-5	50 (26)	3.4E4	3.403E3
	Lower	1.1E-5	50 (26)	1.0E4	1.430E2
Intermediate	Best	8.5E-6	46 (26)	2.1E4	2.135E2
	Upper	2.1E-5	46 (26)	3.4E4	8.539E2
	Lower	3.8E-6	46 (26)	1.0E4	4.545E1
Relatively Good	Best	2.8E-6	22 (26)	2.1E4	3.363E1
	Upper	5.7E-6	22 (26)	3.4E4	1.109E2
	Lower	1.2E-6	22 (26)	1.0E4	6.864E0

Public Property

The impact of public property damage avoided can be calculated as follows:

$$V_{FP} = \Delta\bar{F}ND_G$$

where V_{FP} = impact of avoided offsite property damage ($)

$\Delta\bar{F}$ = change in core-melt probability (events/reactor-year)

N = number of affected facilities

D_G = generic present value of property damage conditional on release ($/event).

The values for N, B, and D_G are the same as defined in Chapter 4. Table 5.12 shows the results.

TABLE 5.12. Sensitivity Analysis #4: Avoided Public Property Damage

Current Quality of Operating Procedures	Type of Estimate	$\Delta\bar{F}$ Change in Core-Melt Probability (events/ry)	N Number of Plants	B Scaled Public Property Damage ($/event)	D_G Discounted Public Property Damage ($/event)	V_{FP} Impact of Avoided Pub. Property Damage ($)
Relatively Poor	Best	2.8E-5	50	2.09E9	1.26E10	1.76E7
	Upper	7.7E-5	50	1.15E10	6.91E10	2.66E8
	Lower	1.1E-5	50	2.72E8	1.63E9	8.97E5
Intermediate	Best	8.5E-6	46	2.09E9	1.26E10	4.91E6
	Upper	2.1E-5	46	1.15E10	6.91E10	6.68E7
	Lower	3.8E-6	46	2.72E8	1.63E9	2.85E5
Relatively Good	Best	2.8E-6	22	2.09E9	1.26E9	7.73E5
	Upper	5.7E-6	22	1.15E10	6.91E10	8.67E6
	Lower	1.2E-6	22	2.72E8	1.63E9	4.30E4

Onsite Property

The impact of onsite property damage avoided can be calculated as:

$$V_{OP} = \Delta\bar{F}NU$$

where V_{OP} = impact of avoided onsite property damage ($)

$\Delta\bar{F}$ = change in core-melt probability (events/reactor-year)

N = the number of facilities affected

U = present value of property damage conditional on release ($/event).

The values for N and U are the same as defined in Chapter 4. Table 5.13. highlights the results.

TABLE 5.13. Sensitivity Analysis #4: Avoided Onsite Property Damage

Current Quality of Operating Procedures	Type of Estimate	$\Delta\bar{F}$ Change in Core-Melt Probability (events/ry)	N Number of Facilities	U Discounted Onsite Property Damage ($/event)	V_{op} Onsite Property Damage Avoided ($)
Relatively	Best	2.8E-5	50	6.20E9	8.68E6
Poor	Upper	7.7E-5	50	9.30E9	3.58E7
	Lower	1.1E-5	50	3.10E9	1.71E6
Intermediate	Best	8.5E-6	46	6.20E9	2.43E6
	Upper	2.1E-5	46	9.30E9	8.99E6
	Lower	3.8E-6	46	3.10E9	5.42E5
Relatively	Best	2.8E-6	22	6.20E9	3.82E5
Good	Upper	5.7E-6	22	9.30E9	1.17E6
	Lower	1.2E-6	22	3.10E9	8.19E4

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APPENDIX A

DESCRIPTION OF CANDIDATE
OPERATING PROCEDURE UPGRADE PROGRAM

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DESCRIPTION OF CANDIDATE

OPERATING PROCEDURE UPGRADE PROGRAM

The approach to upgrading operating procedures presented in this appendix was developed solely for the purpose of performing the value-impact assessment. It is considered by the authors to be a reasonable candidate approach based upon prior experience with the emergency operating procedure upgrade effort and other prior research.

A.1 INTRODUCTION

The Three Mile Island Action Plan Item I.C.9, "Long-Term Plan for Upgrading of Procedures," required the NRC to undertake a course of action to improve the quality of nuclear power plant operating procedures (NUREG-0660, 1980). In response to this Action Plan Item, the NRC has sponsored several projects to assess practices and problems associated with both normal operating procedures and abnormal operating procedures in U.S. nuclear power plants. The findings of these projects indicate that the improvements in the technical accuracy and useability of current operating procedures are warranted. See, for example, NUREG/CR-3968, Study of Operating Procedures in Nuclear Power Plants: Practices and Problems and (Morgenstern et al. 1987) and Evaluation of Nuclear Power Plant Operating Procedures Classifications and Interfaces: Problems and Techniques for Improvement, (Barnes and Radford 1987). The present study has been undertaken to assess the costs and benefits of a program to bring about such improvements. This appendix describes the upgrade program that was used as the candidate in the value-impact assessment.

For the purposes of this report, the term "operating procedures" is defined as the hard-copy, written instructions that are provided to plant personnel to assist their performance of operations tasks under nonemergency plant conditions. The class of operating procedures includes those procedures that are typically identified throughout the industry as normal operating procedures and abnormal operating procedures. The term "procedures programs" refers to those practices employed by licensees to guide the development, use, and administrative control of operating procedures.

The need to improve operating procedures and the procedures programs that govern them is based upon the assumption that deficiencies in the technical accuracy and usability of operating procedures, which result from inadequacies in licensee procedures programs, can lead to operator error and, thereby, potentially jeopardize public health and safety. For example, procedural deficiencies are often cited as root causes of significant operating events in licensee event reports from U.S. nuclear power plants and from the plants of other nations (INPO 1985; Trager 1988). The potentially serious consequences of inadequate procedures programs (e.g., pro forma safety reviews and lax enforcement of policies regarding operating procedures compliance) were amply demonstrated by the Chernobyl accident (NUREG-1250, Rev. 1, 1987, NRC 1987).

The findings reported in NUREG/CR-4613, Evaluation of Nuclear Power Plant Operating Procedures Classifications and Interfaces: Problems and Techniques for Improvement (Barnes and Radford 1987) indicated, however, that not all normal operating procedures and abnormal operating procedures are important to safety and that the procedures that are safety-significant differ between plants. The report concluded, therefore, that some, but not all, operating procedures should receive as much licensee and NRC scrutiny for technical accuracy and useability as do emergency operating procedures.

The candidate Procedure Upgrade Program described herein and used as the basis of the value-impact assessment is intended to require licensees to focus their upgrade efforts on those operating procedures that can prevent operator errors that could affect public health and safety. It includes a description of the procedures program components that lead to high-quality operating procedures. To provide the rationale for the candidate upgrade program, this appendix presents a brief review of the problems associated with operating procedures and licensee procedures programs that were identified in prior projects. The scope and components of the candidate Procedure Upgrade Program are then outlined, and the report concludes with a detailed implementation plan.

A.2 REVIEW OF OPERATING PROCEDURES PROBLEMS

As noted above, significant deficiencies were identified in the quality of operating procedures and in the licensee procedures programs that govern their development, use, and administrative control in the course of conducting NRC-sponsored projects entitled "Program Plan for Assessing and Upgrading Operating Procedures for Nuclear Power Plants" (the Operating Procedures Project) and "Study of Operating Procedure Classifications and Interfaces" (the Classifications and Interfaces Project). These deficiencies are summarized below.

A.2.1 Findings of the Operating Procedures Project

Procedure useability refers to the extent to which task instructions are consistently presented, complete, readable, and easy to follow. As part of the Operating Procedures Project, an extensive sample of procedures from 46 plants were rated with respect to format, style of presentation, content, and overall useability. The rating was done by analysts with prior experience in drafting procedure preparation guidelines and reviewing nuclear plant maintenance procedures, emergency operating procedures, and normal operating procedures. This evaluation indicated that the average rating of procedure useability fell into the minimally acceptable range. Only 4% of the procedures rated were assessed to be of high useability, while 20% of the sample received the lowest useability rating of "very poor."

The evaluation also indicated that current operating procedures lack specificity. A substantial percentage of the procedures reviewed failed to specify follow-on actions to the procedures and did not include reference to either single or multiple indicators that would allow the operators to ensure that the objective of procedure steps had been achieved. In general, the procedures were written in vague terms (e.g., "start pumps as needed") and lacked explicit guidance for taking action, such as a description of the appropriate entry and exit conditions for the procedure.

Other good practices in presenting procedures that help to eliminate common errors in procedure use also were violated in many of the procedures evaluated. For example, long procedures frequently did not include tables of contents, tabs, or indexes. Placekeeping aids were often missing, and 69% of the procedures did not identify or highlight the most recent revisions to the procedure.

In addition to the evaluation of the sample of operating procedures, procedures programs were studied during site visits and during meetings of a peer review group. This peer review group consisted of six members selected on the basis of their experience and background regarding procedure development and operational use and the numerous areas of plant operations that require procedures. The most significant finding of these activities was that operating procedures are increasing in number and complexity as licensees respond with procedural "fixes" to NRC pressure and operational experience. The consequences of this increasing proceduralization are a multiplicity of problems in procedure use and administration. As the procedures become more detailed, their applicability decreases. The narrow applicability then requires that either new procedures be written to cover several normal variations on performing a particular task or that temporary changes be made to the procedures. The new or revised procedures are then often required by current licensee procedures programs to be reviewed by plant safety review committees, which delays the implementation of the revisions. These delays further encourage use of the temporary change process, which, in some plants, involves minimal safety review. Further, because plant safety review committees are overburdened by the number of procedures that they must review, the reviews they perform are often hurried and may be ineffective.

Two other important procedures program deficiencies were identified during the Operating Procedures Project. Coordination of the training function with procedure preparation and use was found to be very limited. One consequence of this lack of coordination was that processes either did not exist or were ineffective in providing feedback from training personnel to procedure writers about problems in the procedures that should lead to revisions. In addition, operating procedures were not required to be verified or validated before use at some plants and full-scale simulator validation of normal operating procedures was found to be particularly rare. Thus, methods to ensure that operating procedures are technically accurate do not appear to be in widespread use.

A.2.2 Findings of the Classifications and Interfaces Project

The Classifications and Interfaces Project focused, in part, on the manner in which transitions (i.e., interfaces) among procedures are managed and presented to operators, since procedure interfaces have been shown to promote operator errors in simulator research (Cauley and Schroeder 1985). The findings of this project suggested that the sources of operator interface errors can be linked to deficiencies in how interfaces are signalled in operating procedures and to deficiencies in operator training with regard to using procedures. Findings of this project further indicated that licensees have typically neither developed nor implemented effective systems for tracking procedure interfaces. Consequently, when equipment is modified or procedures are revised, the other procedures that are affected by these changes may not be flagged for revision. References to procedures that no longer exist or to steps that have been renumbered, for example, can obviously promote operator error.

In summary, operating procedures and licensee operating procedures programs continue to demonstrate many of the weaknesses that were found in emergency operating procedure before the licensees and the NRC undertook the emergency operating procedure upgrade program (NUREG/CR-1977, NRC 1981). Although some licensees have voluntarily begun to improve their operating procedures and to apply to normal operating procedures, and abnormal operating procedures the guidance required for the development, use, and control of emergency operating procedure, the findings of recent emergency operating procedure inspections indicate that voluntary upgrades are not widespread (NUREG-1358, NRC 1989). Further, much has been learned about procedures in nuclear power plants since the emergency operating procedure upgrade program was initiated that can also be applied to improving the effectiveness of other types of operating procedures. The candidate Procedure Upgrade Program presented in the sections to follow is one possible approach to upgrading operating procedures. It is designed to overcome the weaknesses identified in current operating procedures and procedures programs, and to incorporate the lessons learned from the emergency operating procedure upgrade program.

A.3 SCOPE OF THE CANDIDATE UPGRADE PROGRAM

Current guidance to the industry from the NRC that pertains to the development, use, and administrative control of normal operating procedures and abnormal operating procedures (ANSI/ANS-3.2, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, 1981; U.S. Nuclear Regulatory Commission Regulatory Guide 1.33; and Appendix B to 10 CFR 50, Parts V and VI) focuses primarily on the activities for which procedures should be provided and does not specify methods for ensuring that useable, technically accurate procedures are generated. Lessons learned from the emergency operating procedure upgrade program indicate that the industry currently appears to have insufficient knowledge of the appropriate methods and, as a consequence, licensees are dedicating resources to their operating procedures programs for relatively little return in terms of reducing human error in plant operations (NUREG-1358, NRC 1989b). Therefore, it is important that the upgrade program include guidance for the industry in how to develop high-quality operating procedures.

Not all plant operations have implications for public health and safety, however, and so fall outside the NRC's scope of concern. In addition, many operations tasks may not require error-free performance, or may be performed so frequently that high-quality, written procedures are unnecessary for the trained operator. Further, it is impossible and undesirable to develop procedures for every operations task that must be performed in a nuclear power plant. Dedicating significant resources to the development and administrative control of procedures for many tasks is not only unnecessary, but contributes to the overburdening of plant procedures review systems that has been observed to result from the trend toward increasing proceduralization in many plants. Therefore, the candidate upgrade program would guide licensees in focusing their efforts on developing high-quality procedures for tasks that are significant to safety, are intolerant of human error, and for which other techniques of reducing human error are inappropriate (e.g., man-machine interface redesign, improved labeling).

To achieve this goal, the candidate upgrade program assumes that licensees would adopt a two-tiered approach to the development, use, and administrative control of operating procedures. This approach would involve the design of criteria for discriminating between operations tasks that require high-quality, detailed, step-by-step procedures to be performed in a timely and error-free manner (Tier One procedures) and tasks that require some form of job performance aid (e.g., checklists, tables, calculation forms), but for which a high-quality, detailed procedure is unnecessary or undesirable (Tier Two procedures). Tasks in the Tier One group would be identified through engineering and human factors analyses, and would be likely to include tasks performed using safety-significant systems or components, complex tasks, tasks that are infrequently performed, tasks that involve the coordination of several workers' activities in one location or in dispersed locations, tasks that operating experience indicates are error-prone, and so on. Tasks in the Tier Two group would include frequently performed tasks that any trained operator can perform with only a checklist reminder of step sequencing, tasks performed on simple or nonsafety-related systems or equipment, and so on.

Because the operation of many components and systems is unique to individual plants, and because the training levels and expertise of operations personnel differ between plants, it is unlikely that the NRC could provide generic guidance to the industry about which specific procedures should fall into the Tier One and Tier Two groups. However, the NRC and the industry can work together to identify the criteria and methods that should be used to identify tasks that require Tier One versus Tier Two procedures. To ensure the effectiveness of the upgrade program in limiting the number of Tier One procedures, the NRC and industry group should be as specific as possible in the criteria that are developed.

By rechanneling some of the resources now dedicated to the development, review, and administrative control of all operating procedures to the more rigorous development of Tier One procedures, the burden on plant operating procedures programs would be reduced, and more useable and technically accurate procedures would be available to guide the performance of important operations tasks. In addition, the application of technical writing and human factors principles to the development of both Tier One and Tier Two procedures would be expected to reduce human error in operations, resulting in fewer equipment failures and less frequent challenges to plant safety systems. As a consequence, public health and safety would benefit.

A.4 COMPONENTS OF THE CANDIDATE PROGRAM

The components of the candidate upgrade program would differ for Tier One and Tier Two procedures. Because they are important to safety, Tier One operating procedures would be subjected to the more rigorous development, use, and administrative control processes, similar to those now required for emergency operating procedures (NUREG-0899, NRC 1982). Tier Two procedures would also be improved because of the potential they represent for indirectly impacting plant safety, but would require a smaller investment of utility resources than the Tier One procedures. For all operating procedures, the upgrade program would address: 1) the availability of procedure writing expertise; 2) the availability of sound technical bases for the procedures; 3) man-machine interface requirements; 4) procedure verification; 5) procedure validation; 6) documentation of the procedure development process; 7) user training; 8) procedure use policies; and 9) administrative controls. These program components are briefly discussed in the following sections.

A.4.1 Procedure Writing Expertise

A first step in upgrading operating procedures is ensuring that utility personnel possess the capabilities to produce useable procedures. For both Tier One and Two procedures, it is important that the utility develop a detailed procedure writers' guide that would assist it in applying technical writing and human factors principles to the preparation of operating procedures. NUREG-0899, Guidelines for the Preparation of Emergency Operating Procedure (NRC 1982), describes some of the useability objectives that should be addressed in any procedure writers' guide, but, with the exception of Appendix B regarding the presentation of logic statements, does not provide detailed examples of acceptable methods for presenting task instructions. Because it does not appear that this information is widely known throughout the industry (as indicated by the lessons learned from the emergency operating procedure upgrade program as well as the findings of the Operating Procedures and Classification and Interfaces Projects), a detailed good practices document that would serve as the technical basis for utility writers' guides could be developed.

Results of the emergency operating procedure upgrade program also indicate, however, that a detailed procedure writers' guide is insufficient to ensure that the procedures developed will be useable. Procedure writers may not use the writers' guide or may not fully understand how to apply the information it contains. Similarly, those individuals responsible for reviewing procedures to ensure that they are consistent with the writers' guide are often not fully familiar with the guidance it presents. Consequently, the candidate program would include requirements for procedure writers and quality assurance auditors to be trained in applying the writers' guide.

A.4.2 Technical Information

In addition to ensuring that plant personnel possess the technical writing and human factors knowledge required to produce useable procedures, it is necessary that all of the relevant information be available to produce technically accurate operating procedures. For many Tier Two procedures, the last version of the procedure, a vendor technical manual, and an experienced, licensed operator may, in combination, represent sufficient technical information to prepare a technically accurate procedure. For Tier One procedures, however, a team approach would be necessary to ensure that complete information is brought to bear on procedure development.

Licensees would identify at least two teams of plant personnel to participate in the upgrade process. The use of two teams would speed the procedure upgrades and would allow for independent verification and validation of the new or revised procedures by an equally qualified group. Each team would be comprised of licensed operators, training personnel, engineering personnel, vendor representatives (when necessary technical information about plant equipment and systems is proprietary), and representatives of other plant functional areas (e.g., maintenance, chemistry, radiation protection) when the task that is being proceduralized involves more than operations personnel. The roles and responsibilities of each team member would be specified, and one individual would be assigned primary responsibility for the development of each procedure and for maintaining a record of the procedure development process.

In the course of upgrading existing procedures or developing new operating procedures, the Tier One teams may determine that the technical information available to them is neither accurate nor complete. For example, when procuring equipment, many licensees do not require their vendors to validate the technical manuals they supply, or vendors may withhold some technical information as proprietary. In these cases, equipment operating histories, engineering analyses, or the personal experience of operations personnel must serve as the primary sources of technical information. A process for ensuring that relevant plant technical specifications, NRC generic communications, vendor bulletins, licensee event reports, and operating experiences from other plants are translated into the procedures correctly is also necessary.

A.4.3 Man-Machine Interface Requirements

In many plants, the primary method of writing or revising a procedure is for a single operator to sit down at a desk and draft it based upon his or her memory of the control room configuration and the requirements of the task. Not surprisingly, this method often fails to yield a useable procedure that is appropriate for the conditions under which it will be performed.

Consequently, the upgrade program would specify that the development and revision of Tier One operating procedures involve man-machine interface analyses performed at the location in the plant at which the procedure will be performed or, when the location is inaccessible, that mock-ups or models of the displays and controls involved be provided.

These man-machine interface analyses would provide several types of information to the procedure writer. Information would be collected regarding labelling and the units of measure on controls and displays, equipment location, the most efficient ordering of procedure steps, the likely effects of the ambient illumination, noise levels and other performance shaping factors on task performance during normal and degraded conditions, the requirements for additional tools and equipment, the number of individuals required to accomplish the task and how their activities can best be coordinated, as well as the overall practicality of requiring an operator to use a hard-copy, written procedure in that task environment.

Licensees may choose to conduct similar analyses in the development of Tier Two procedures. However, much of this information could be gathered concurrently with the verification of Tier Two procedures as described below.

A.4.4 Procedure Verification

Procedure verification is the process used to ensure that a procedure is technically correct, that there is a correspondence between the content of the procedure and the control room/plant hardware, that the language and nomenclature used in the procedure are consistent with the terms familiar to the procedure users, and that the procedure is written in accordance with the requirements of the operating procedures writers' guide. For Tier One procedures, the verification process would be formally conducted and documented by the Tier One team that did not write the procedure. Verification would include desk-top reviews of the technical information used to develop the procedure and of the consistency of the procedure with the writers' guide requirements. In addition, the adequacy of the procedure in terms of the man-machine interface requirements would be evaluated during walk-throughs of the control room and other locations at which the procedure must be performed.

Verification of Tier Two procedures would also be conducted by an individual who did not write the procedure and would be documented. Both desk-top reviews and walk-throughs would be employed. Verification and collection of the man-machine interface information discussed above may be combined for Tier Two procedures, however.

A.4.5 Procedure Validation

Procedure validation is the process used to ensure that a newly written or revised procedure accomplishes its purpose and that it is useable. At the licensee's discretion, Tier Two procedures could be validated during the first time that they are used to operate equipment rather than prior to use.

The results of the first-use validation would be documented, however, and the procedure revised to reflect the information gained.

Tier One procedures would be subjected to a more thorough validation than Tier Two procedures and would be validated before the procedure is used. The validation would be conducted by the Tier One team that did not write the procedure and, when appropriate, would involve use of the procedure by operators in the control room simulator. As with emergency operating procedure validation exercises, scenarios would be developed that involve use of the procedure under normal conditions as well as under conditions of likely equipment and system unavailability or malfunctioning (e.g., start-up with some equipment tagged out for maintenance). Both experienced and newly licensed operators would be involved in the validation exercises to ensure that the procedure is clear to both types of operators. Procedures that will be performed by non-licensed operators or by personnel outside of the control room would also be validated with the involvement of the intended users of the procedures. Findings of the validation exercises would be documented and lead to procedure revisions. If the revisions are substantial, the validation would be repeated on the new version of the procedure.

A.4.6 Procedure Safety Reviews

The type of safety review required for Tier One and Tier Two procedures would differ to ensure that plant safety review committees do not continue to be overburdened by unproductive and unnecessary procedure review requirements. For Tier Two procedures, safety reviews would be conducted only by a responsible, qualified individual within the operations department. For example, before it is used, a Tier Two valve line-up checklist could be reviewed for safety implications by the control room shift supervisor on-duty or by the operations manager, at the licensee's discretion.

Tier One procedures, however, would be completely reviewed by the individuals who sit on the plant safety review committee. For each procedure, the focus and scope of the review to be conducted by the individuals on the plant safety review committee would be specified before the procedure is reviewed. Thus, for example, the regulatory compliance representative would review the procedure for compliance with technical specifications and other regulatory requirements, while the radiation protection manager reviews the procedure in terms of its implications for activities in his or her functional area. The focus and scope of the reviews would be tailored to the characteristics of the procedure, rather than subjecting all Tier One procedures to a standard review process.

A.4.7 Procedure Documentation

Documentation of the procedure development and review process is necessary for both Tier One and Tier Two procedures. The documentation would include the rationale for defining the procedure as either Tier One or Tier Two, the technical justification for the steps included, the results of man-machine interface analyses, the results of the verification and validation activities, and the conclusions of the safety review process. In addition, a list of other plant procedures that are referenced by the procedure and that refer to the procedure would be developed and maintained.

This documentation would serve three primary purposes. First, the development documentation would assist in the verification and validation activities, so that those individuals responsible for verifying and validating the procedure will understand the reasons for how the procedure has been developed and will be able to evaluate those reasons. The documentation would also provide a procedure history for assessing possible future revisions to the procedures to ensure that the procedure developers' intentions are not inappropriately violated when a revision is considered, or that a previously rejected revision is not unintentionally repeated. Third, the referencing information would assist in ensuring that other procedures that are affected by a change to one procedure can also be updated, if needed.

A.4.8 Training of Users

Two types of procedure-related training would be provided to the users of the procedures. The first type would be offered to all operations personnel and should focus on interpretation and use policies for the procedures. This training would include discussion of the conventions used to present information in the procedures (e.g., the two-column format, how cautions and notes and other types of supplementary information are presented, the meaning of flowchart symbols) as well as the appropriate methods of using the procedures (e.g., emphasis on reading cautions, how to perform concurrent and other types of nonsequential steps, verbatim compliance with Tier One procedures). The second type of training would involve practice at using Tier One procedures in the simulator or in a walk-through, as appropriate. For Tier Two procedures, training would involve a walk- or talk-through of the procedure with a senior operations staff person before beginning task performance. In addition, each licensee would establish criteria for determining the type of revisions that require retraining on the procedure and for determining the type of retraining that is required (e.g, desk-top reviews, walk- or talk-throughs, simulators).

A.4.9 Use Policies

Experience in the military, commercial aviation, and the aerospace industries has indicated that the requirement for verbatim compliance with procedures (i.e., requiring that procedures be followed in a step-by-step manner) reduces human error in operations tasks. Prerequisites necessary to support a verbatim compliance policy, however, are that the procedures are technically accurate, useable, and have been designed to address the most common off-normal conditions. In the nuclear power industry, therefore, verbatim compliance should apply only to Tier One procedures.

Clearly, however, procedures cannot be written to apply to every situation that is likely to be encountered in a nuclear power plant. Procedure use policies must, therefore, allow for operator judgment and temporary changes to address unusual circumstances. Because of the importance to safety of Tier One procedures, however, policies for allowing operator discretion in circumstances where a decision to violate the procedure must be made under time pressure in order to prevent significant damage to plant systems or to protect public health and safety would be similar to policies currently in place for emergency operating procedures. Temporary change policies for Tier One procedures would be similar to those for emergency operating procedures as well.

Use policies for Tier Two procedures would allow greater user discretion than Tier One procedures, since these procedures would not have been developed as thoroughly and would not be as detailed. It is important, however, that departures from the Tier Two procedures be documented and that the documentation be forwarded to the person responsible for the procedure to ensure that the Tier Two procedures are revised when the departure was necessitated by design modifications, useability limitations, or other factors that are not temporary in nature.

A.4.10 Administrative Controls

This candidate Procedure Upgrade Program would necessitate a number of administrative control mechanisms to support both Tier One and Tier Two procedures. Of particular importance is a process to ensure that procedures affected by design modifications or changes to other procedures be flagged for review and revision. Identification of the procedures that would be affected by any modifications to existing procedures, systems, or equipment would be necessary to ensure that Tier One procedures remain technically accurate, given the verbatim compliance policy.

A second important administrative control mechanism that would be developed by licensees is a method to ensure that feedback from users and training personnel about the procedures reaches the individuals responsible for the procedure. Procedures are living documents and may require revision as the composition, experience, and qualifications of the operations staff changes over time. To ensure that trainers and users are reinforced for providing feedback, the person responsible for each procedure would inform the source of the feedback whether or not a revision will be made to the procedure and why or why not. However, because feedback from users often reflects only individual preferences that do not affect either the technical accuracy or useability of the procedure for others, the person responsible for the procedure would be required to review the procedure documentation package and revision history and to carefully evaluate the need for a procedure change. Clearly, revisions to Tier One procedures that are initiated out of personal preference are costly and contribute to the overburdening of procedure programs that has been observed.

Both Tier One and Tier Two procedures would continue to be subject to biennial review. For Tier One procedures, these reviews would include an overview of the procedure's use and revision histories, an assessment of feedback documents and decisions made about the feedback, a check of the relationship of the procedure to other procedures, and a walk-through of the procedure in the plant to ensure that the man-machine interface information continues to reflect the actual conditions of use. Revisions made to Tier

One procedures during the period between the biennial reviews would not substitute for the complete review described above.

Biennial reviews of Tier Two procedures would be less thorough. These reviews would provide the opportunity to evaluate the procedure's use history and to purge any procedures that have not been and may not be of value. In addition, use and feedback histories may indicate that the procedure should be upgraded to a Tier One procedure. Interim revisions to Tier Two procedures could satisfy the biennial review requirement.

A.5 PROGRAM IMPLEMENTATION

In this section, the specific activities to be conducted by the industry and the NRC in developing and implementing the candidate Procedure Upgrade Program are discussed. Figure A.1 illustrates the estimated period of performance that would be required to implement the tasks associated with the upgrade program. In addition, the personnel involved in performing each task, the estimated task schedules, and the products of the tasks are described, based upon the information needed in the value-impact assessment.

A.5.1 Task 1: Development of Good Practices Document

The initial task of the candidate upgrade program would consist of a review and synthesis of information relevant to developing, using, and controlling operating procedures in nuclear power plants. The goal of this information survey would be to produce a good practices document to assist licensees in the implementation of upgraded operating procedures.

The proposed document would be comprehensive. One section would consist of recommendations regarding criteria for identifying tasks which should be conducted with the assistance of procedures and for identifying the tasks from among these which require either Tier One or Tier Two treatment. The document would also more completely describe the program components presented in Section A.4 above, and provide specific examples of acceptable methods for developing, using, and controlling operating procedures. A major appendix of the document would present a sample operating procedures writers' guide and example normal operating procedure and abnormal operating procedure, and alarm response procedures written in accordance with the writers' guide. Another appendix would present an example curriculum for a procedure writers' training course. This document could reduce implementation costs of the upgrade program for licensees as well as increase the likelihood that the licensee programs would be effective.

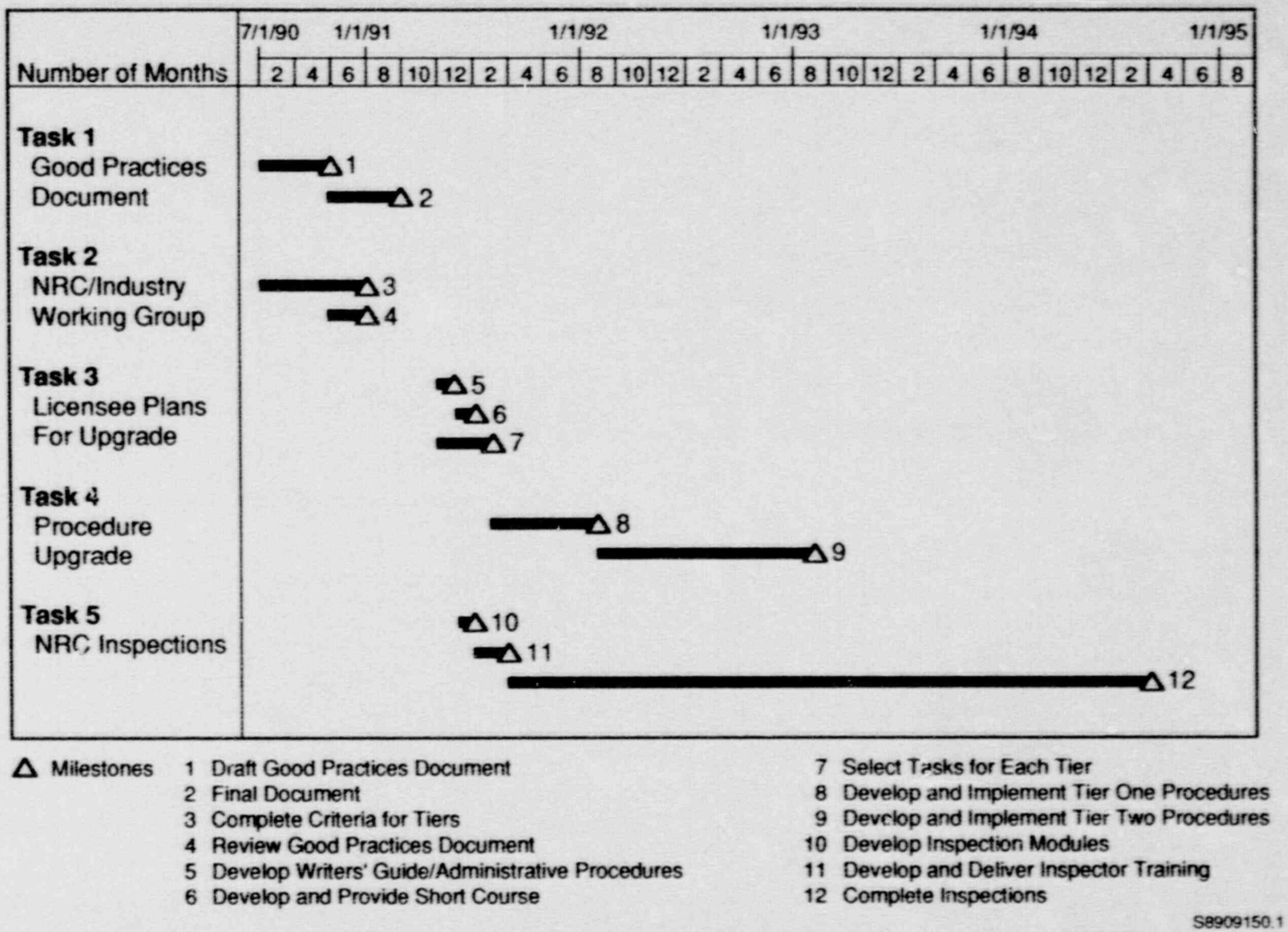

FIGURE A.1. Estimates Period of Performance for the Candidate Operating Procedure Upgrade Program

Personnel:

1 human factors research scientist 3 person-months	480 hours
2 technical writers 2 person-months each	640 hours
Clerical support 2 person-weeks	80 hours

Schedule:

4 months to complete first draft
4 months to receive comments and complete final document

A.5.2 Task 2: Convene a NRC/Industry Working Group

While the good practices document is being developed, the NRC would convene a working group composed of knowledgeable NRC and industry personnel to discuss the upgrade of operating procedures. This group might include representatives of INPO, NUMARC, the Owners' Groups, other major vendors, experts in procedures from the aerospace and other regulated, proceduralized industries, and several representatives of individual licensees who have already initiated upgrades of their operating procedures. Experts in procedures from other nations could also be considered for participation (e.g., France).

The working group would serve two purposes. The primary goal would be to define the criteria that licensees would use to identify which tasks should be proceduralized, and to assign procedures to Tier One or Tier Two. A secondary goal would be to evaluate the good practices document described above and provide comments on it to Task 1 personnel.

Personnel:

2 NRC representatives 2 days/month/person	192 hours
4 Owners' Group representatives 2 days/month/person	384 hours
2 vendor representatives 2 days/month/person	192 hours
2 experts 2 days/month/person	192 hours
2 industry group representatives 2 days/month/person	192 hours

2 licensee representatives 2 days/month/person	192 hours
Clerical support 2 person weeks	80 hours

Schedule:

6 months to complete criteria
2 months to review good practices document (performed concurrently with completion of criteria)

A.5.3 Task 3: Licensee Upgrade Plans

Following publication of the upgrade program's good practices document, several major activities would be required of licensees before they begin revising their operating procedures. The first activity would involve defining the operations tasks for each plant that are candidates for proceduralization to identify areas where procedures are required but do not currently exist and to select those which require Tier One versus Tier Two treatment. This activity is likely to require involvement of the plant operations manager, plant engineers, senior reactor operators, training personnel, and human factors specialists. The second major activity would involve the preparation of the plant-specific operating procedures writers' guide and the training of procedure writers. This group would also revise plant administrative procedures to guide the development, use, and control of the upgraded operating procedures.

Personnel:

1 plant operations manager 2 person-weeks	80 hours
1 plant engineer 3 person-months	480 hours
1 senior reactor operator 3 person-months	480 hours
1 training specialist 1 person-month	160 hours
1 human factors specialist 3 person-weeks	120 hours
1 technical writer 1 person-month	160 hours

5 procedure writers or auditors 3 days/person in training	120 hours
Clerical support 3 person-weeks	120 hours

Schedule:

1 month to complete writers' guide and revise administrative procedures
1 month to develop and conduct procedure writers' short course
3 months to select tasks for Tier One and Two procedures

A.5.4 Task 4: Licensee Upgrades of Operating Procedures

At the conclusion of the planning phase, licensees would begin upgrading their operating procedures in accordance with the process described in the good practices document and in Section A.4 above. Tier One procedures would receive first attention, followed by the revision of Tier Two procedures.

Personnel:

2 licensed reactor operators 12 months/person	3,840 hours
2 technical writers 6 months/person	1,920 hours
2 human factors specialist 1.5 months/person	480 hours
2 plant engineers 1.5 months/person	480 hours
2 training specialists 1.5 months/person	480 hours
5 plant safety review committee members 3 weeks/person	600 hours
5 procedure users 3 weeks/Tier One validation	600 hours
60 procedure users 3 days/person in training	1,400 hours
Clerical support 3 person-months	480 hours

Schedule:

6 months to complete Tier One procedure development process
12 months to complete Tier Two procedure development process

A.5.5 Task 5: NRC Inspections of Upgraded Operating Procedures

About one year after the NRC has published the good practices document, the NRC would begin a one-time inspection of all licensees' upgraded operating procedures. These inspections would be intended to assess the technical accuracy and useability of the new normal operating procedure and abnormal operating procedure, and, secondarily, the viability of the licensees' upgrade programs. In addition, an inspection module to be conducted by resident inspectors would be developed for use when, in the resident inspector's opinion, the quality of the licensee's operating procedures or the operating procedures program required evaluation.

Prior to the initial inspections, it would be necessary to develop a temporary inspection instruction and to train inspectors in conducting the inspections. This training, particularly emphasizing the human factors aspects of the procedures, would reduce the burden on headquarters personnel for supporting the inspections and developing the capabilities of regional inspectors for evaluating all types of licensee procedures.

Personnel:

1 NRC or contractor staff person 1 person-month	160 hours
1 training specialist 2 person-months	160 hours
10 NRC inspectors in training 3 days/person	240 hours
5 NRC inspection team personnel 118 plants - 13 days/plant	60,800 hours
Clerical support 2 person-months	160 hours

Schedule:

1 month to develop temporary and on-going inspection modules
2 months to develop and deliver 3 days of training to inspectors
3 years to complete initial inspections

APPENDIX B

SAFETY ASSESSMENT DETAILS

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SAFETY ASSESSMENT DETAILS

B.1 INTRODUCTION

The objective of this safety assessment was the evaluation of the potential for reduction in public risk presented by the candidate Procedure Upgrade Program described in Appendix A. This candidate program would be directed at the upgrading of normal operating procedures and abnormal operating procedures by NRC licensee power plants. Since this assessment was concerned with the potential risk <u>reduction</u>, no attempt was made to reevaluate the baseline risk of reactor operations. Additionally, an evaluation of the safety significance of emergency operating procedures was beyond the scope of this assessment.

The general approach taken in this safety assessment consisted of a sensitivity analysis technique which was used to: 1) estimate the current contribution to reactor core-melt frequency of procedure-related operational errors during normal and abnormal operating conditions, 2) estimate the same contribution assuming that the quality of operating procedures is improved by specific measures, and 3) compare the "before" and "after" estimates of overall core-melt frequency to determine the potential reduction in core-melt frequency associated with the potential improvements to current operating procedures. After arriving at an estimate of the reduction in core-melt frequency, standard dose conversion factors were applied to estimate the resulting reduction to public health risk.

The methodology described here was intended to be used as an extension to an existing nuclear PRA. The methodology relied on an existing PRA to provide a quantitative framework for modeling the role of normal operating procedures and abnormal operating procedures in the operation of a nuclear power plant. The reference PRA also provided the original risk equations for the plant being evaluated, including the dominant accident sequences and their associated cut sets, along with the basic events that form the cut sets. These accident sequences were used to represent a hypothetical generic reactor that served as a model for much of the safety assessment. Therefore, the reference PRA served as a starting point for the safety assessment.

It was assumed that, in order for procedure-related operational errors to have safety significance, normal plant operations must be disturbed. Following the occurrence of an abnormal event or disturbance, the resulting instability of the plant could result in a plant accident if certain failures were to occur consecutively. Only certain specified abnormal events are generally considered in a PRA. These events are referred to as initiating-events (i.e., events that require the plant to trip). Initiating-events are generally classified as either external events, such as fire, flood, or earthquake, or internal events. Internal events are divided into loss-of-coolant accidents (LOCAs) and transients. A transient is a condition which causes a requirement for reactor shutdown not caused by a LOCA. (For PRA purposes, a loss of secondary coolant is typically classified as a transient, not a LOCA.) There are other internal abnormal events that may occur in addition to LOCAs and transients, such as a leak in the spent fuel pool. However, since these other types of events contribute only negligibly to the overall risk of the facility, they are typically not evaluated in PRAs.

Most PRAs also consider the consequences of specific accident sequences, in addition to identifying the initiating-events. Consequence analysis refers to the analysis of the health and financial effects that result from a release of radioactive material that, in turn, resulted from some initiating-event or events. Associated with consequence analysis is an analysis of emergency response (e.g., evacuation of personnel who could be affected by the release of the radioactive material). Improved normal operating procedures and abnormal operating procedures were expected to reduce the frequency of transient initiating-events (and the potential for accidental release of radioactive material), but were not expected to significantly affect the consequence of a release should one occur. Therefore, we did not address consequence analysis or emergency response since actions performed under these conditions would be directed primarily by emergency operating procedures.

Human reliability analyses performed as input to PRAs have typically been applied only to the activities related to internal initiating-events (LOCAs and transients). In this safety assessment, the emphasis was placed on transient initiating-events since it is unlikely that improvements to operating procedures could significantly affect the probability of a LOCA. Similarly, if a LOCA were to occur, it is likely that emergency operating procedure would be implemented and, therefore, improvements to normal operating procedures and abnormal operating procedures would not be effective. Likewise, operator responses to external initiating-events were not addressed in this analysis unless the required operator actions are commonly provided for in normal operating procedures or abnormal operating procedures.

B.2 EVALUATION METHODOLOGY

It was assumed that, in order for there to exist a significant risk to public health and safety from the operation of a nuclear power plant, normal plant operations must be disturbed. As mentioned previously, a plant disturbance (initiating-event) may be in the form of an external event or an internal event. Since operator errors are not likely to contribute to the occurrence of external events, only internal initiating-events were considered in this analysis. Similarly, since the operator response to LOCAs are typically given in the emergency operating procedures, only transient initiating-events were considered.

As a result, a key assumption of the methodology was that, in order for operational errors during normal operating conditions to have safety significance, they must manifest themselves as transient initiating-events leading to a plant trip. The general approach used here followed the guidelines established in A Handbook for Value-Impact Assessment (Heaberlin et al. 1983) and relied on the combined use of the reference PRA (Sugnet et. al, 1984) along with the Sequence Coding and Search Systems (SCSS) database (NUREG/CR-3905, Greene et al. 1985).

B.2.1 Selection of Representative Plant

One of the first and most important decisions made in this safety assessment was that of selecting the representative plant model to be used for the core-melt frequency calculations. This decision involved the selection of the specific PRA or PRAs to be used as the generic model(s). Since the calculation of the reduction in public health risk was to be evaluated based on the risk equations contained in the reference PRA(s), the overall results of the safety assessment depended to a large extent on this choice. Ideally, several PRAs can be used as models of different classes of plants. However, the lack of certain necessary technical attributes in available PRAs as well as limitations in resources may not allow an in-depth analysis of multiple PRA models.

Value-impact assessments frequently use two reference PRAs, one for each reactor type, to model technical issues that affect both PWRs and BWRs. For many technical safety issues, the fundamental differences in the design and operation of the two reactor types may make the proposed regulatory action significantly more or less attractive at one or the other of the two types of reactors. The use of a separate PRA for each reactor type allows comparisons to be made regarding the relative utility of implementing the proposed regulatory action at one or the other type of plant, and accordingly facilitates the making of independent regulatory decisions, if necessary.

After careful analysis of how normal operating procedures and abnormal operating procedures are actually used in plants of both PWR and BWR design, it was concluded that there are no fundamental differences between these two reactor types that would result in a significant difference in the potential affect of improved procedures. It was determined that improved operating procedures would most likely result in similar benefits at both PWRs and BWRs. Instead, it was determined that current quality of operating procedures being used at individual plants, whether they be of PWR or BWR design, is a more important distinction than reactor type. In other words, it was determined that the variation in the current quality of operating procedures at individual plants has a much greater potential effect on safety than do the design differences between the two reactor types. For example, the safety significance of operating procedures at two plants that have nearly identical reactor designs may be substantially different due to differences in the current quality of operating procedures at the plants. Similarly, there may be two reactors whose designs are very different, but due to similar procedures development programs currently in place at the plants (and, therefore, similar quality of current operating procedures), the overall safety significance of procedures at the two plants may be very nearly the same.

Therefore, it was concluded that comparisons made between multiple baseline PRAs would not be useful in these circumstances due to the substantial differences in the current quality of operating procedures between individual plants. Little or no additional insight would have been gained by performing the risk calculations on multiple PRA models. This followed from the determination that the importance of the differences in the current quality of operating procedures among individual plants exceeds that of any fundamental differences in the importance of procedures among plants of different reactor type. Because there appeared to be little benefit to performing an evaluation of a second model, it was decided that the resources that would have been required to evaluate a second PRA would be more effectively used by focussing additional attention on a single PRA.

The actual selection of the reference PRA was based on the following criteria: 1) does the PRA contain sufficient detail in the modeling of transient initiating-events necessary to support evaluations of the impact of procedure-related operational errors?, 2) to what extent does the PRA model human performance?, 3) is the plant configured in a manner which supports its use as a representative LWR (i.e., are there any unique characteristics that could be significant)?, and 4) is the PRA relatively current?

Based on a review of the existing PRA studies, the NSAC/60 Oconee-3 PRA (Sugnet et al. 1984) was judged to be the best choice because it was the PRA then available that best satisfied each of these four criteria. Other PRA models considered were found to be inadequate in meeting one or more of the criteria. This was especially the case with the modeling of human performance; the Oconee PRA models human error better than the other PRAs reviewed. Although this choice of a reference PRA does affect the overall results of the analysis (as would the selection of any other single PRA), it was judged that differences in the quality of current operating procedures between plants results in larger uncertainties than are introduced by relying on the use of one PRA to serve as a generic model. The interested reader is referred to Section 5 of this report for a discussion of other sources of uncertainty in this analysis and the results of a sensitivity analysis performed on several of the key elements of the analysis.

B.2.2 Limitations and Assumptions of the Study

The response of a particular plant to a specific transient initiating-event will, of course, depend upon the particular design of the plant among other variables. Although there are several common transients that occur at plants with different designs, there are also many transients that are unique to, or particularly important to, each specific plant. Therefore, each particular transient could produce different consequences in plants of different reactor type or vendor design. However, due to limited resources, it was determined that a detailed evaluation of the specific responses of different plants to each particular transient initiating-event was not possible.

As a result, this analysis did not attempt to evaluate the various different responses to plant transients that would occur in plants of dissimilar designs. Since the objective of this analysis was to evaluate the potential safety benefit that might be realized due to improved operating procedures across the entire industry, this analysis (although based on the risk equations of one reference plant) was intended to model a hypothetical "generic" nuclear power plant.

Reactor Power Level Considerations

Except for a transient initiated by a reactivity insertion such as a control rod withdrawal, transients starting at zero power should not have any significant safety consequences. This implies that, below some non-zero power level, transients not initiated by a reactivity insertion should also not have significant safety consequences. For the purposes of this analysis, it was assumed that transients occurring at a reactor power level below 20% of the full power level do not contribute significantly to public health risks. Therefore, these transients were not included in the evaluation of transient frequencies.

Learning Curve Considerations

It is common for nuclear power plants to experience a greater frequency of transient initiating-events in their early years of operation. Therefore, it is not surprising that certain transients show a smaller frequency in years subsequent to the first or second year of operation because of the complexity of learning to operate a nuclear power plant. In this analysis, it was assumed that the average plant is beyond this "learning curve". It should be noted that, although most of the commercial nuclear power plants are beyond this point, for those plants that are not, the improvements to plant operating procedures may have additional safety value. Therefore, supplementary consideration should be given to the application of the results of this assessment to those plants that are in the first few years of operation.

B.2.3 Use of PNL Expert Panel

As with most studies that deal with issues involving human performance and human modeling, expert judgment was relied upon in this study at various points to provide estimates of values for which there are no solid data available. At these points in the analysis, members of a panel of PNL experts in the area of nuclear power plant operations were convened in order to develop estimates of values which were needed to proceed with the analysis. This group of PNL experts consisted of six individuals with diverse experience with operating procedures at various nuclear power plants.

Most of these PNL experts are certified NRC reactor operator examiners; many have several years experience with nuclear power plant operations. One of the PNL experts is a certified Senior Reactor Operator while another holds a Ph.D in Physics. The PNL expert panel also contained several individuals who were involved in the development and review of Procedure Development Packages for emergency operating procedure. In addition, several panelists have experience in evaluating human factors and operator performance issues in the context of nuclear power plant operations.

As mentioned above, this group of PNL experts was relied upon at several points of the analysis. These points included the identification of the affected transient initiating-events and operator actions, the evaluation of the role of operating procedures in the operator actions identified as being potentially affected, and the estimation of the relationship between the three categories of actual plants and the hypothetical generic plant.

Not all of the experts contributed to each of the judgments made in this analysis. Rather, the experts were primarily relied upon for judgments which pertained to issues dealing directly with their own particular areas of expertise.

B.2.4 Evaluation Methods and Results

The approach outlined in Figure B.1 was applied to the reference PRA, which had been selected to represent a hypothetical generic plant. The results of the analysis are provided in the following subsections.

B.2.4.1 Identification of Affected Parameters

After reviewing the transient initiating-events contained in the reference PRA, the following parameters were identified as being potentially affected by improved operating procedures. The affected parameters were then separated based on whether they were believed to be potentially affected by improvements to normal operating procedures or by improvements to abnormal operating procedures. This distinction was not critical to this analysis, however, because the candidate Procedure Upgrade Program would apply to both normal operating procedures and abnormal operating procedures.

Normal Operating Procedures

Five transient initiating-events were identified as being most likely affected by improvements to normal operating procedures. Table B.1 lists the affected parameters along with each parameter's original, base-case frequency.

Each of the transient initiating-events identified in Table B.1 was assumed to have a contribution (albeit small) from procedure-related operational errors. The improvement that was expected with these initiating-events is a reduction in the contribution to each of the events from operator errors that result from inadequate or poor procedures.[a] In order to

(a) Hardware failures and other types of equipment failures that may not be affected by operating procedures are the principal contributors to these transients. However, it was assumed that there is a small contribution to each of these transient initiating-event frequencies from operator errors that are made as a result of inadequate or poor operating procedures.

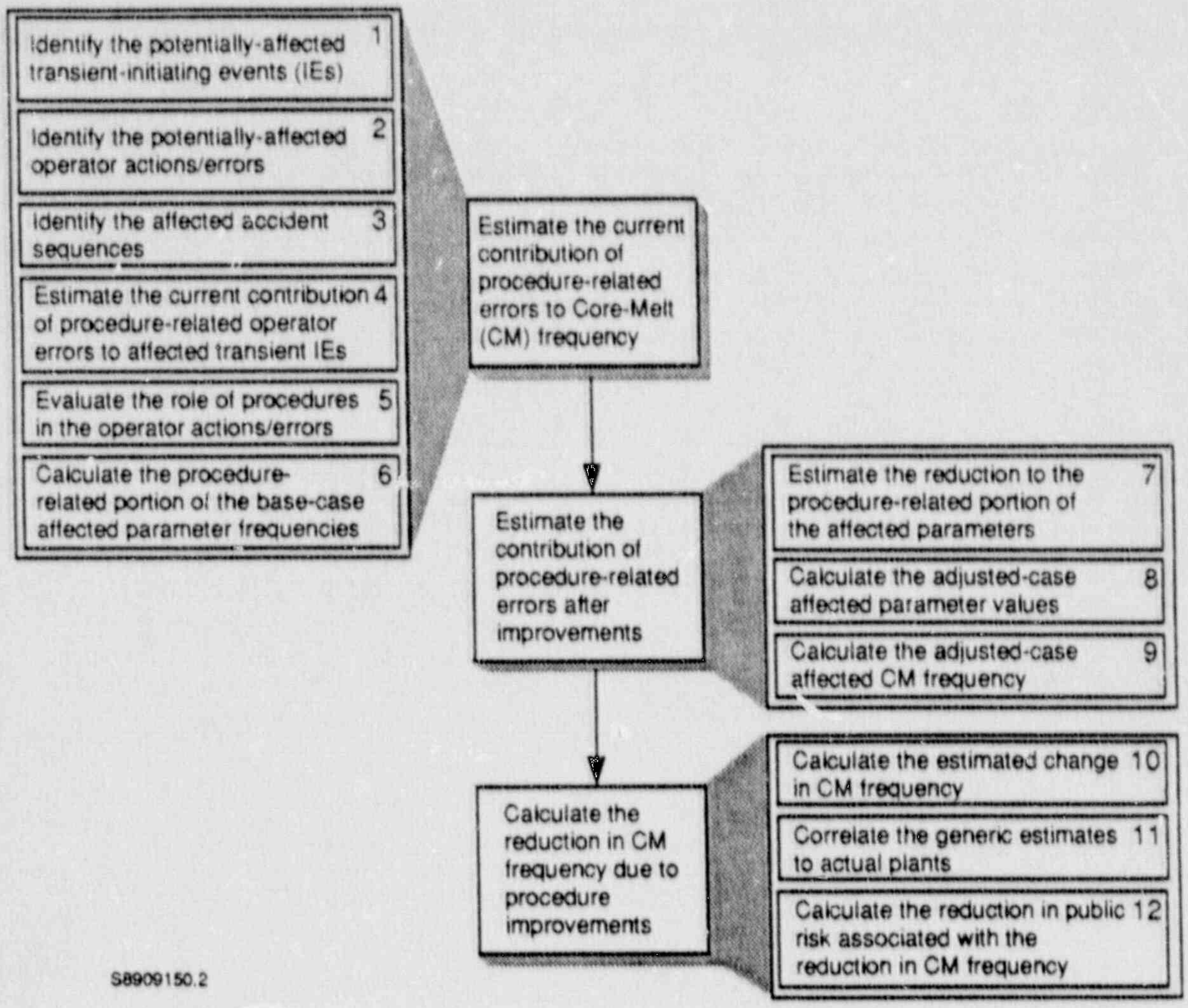

FIGURE B.1. Process for Evaluating Core-Melt Frequency Reduction

TABLE B.1. Potentially Affected Transient Initiating-Events

Event Symbol	Original Frequency (events/year)	Event Description
T1	5.7	Reactor/Turbine Trip
T2	6.4E-1	Loss of Main Feedwater
T4	2.1E-1	Loss of Condenser Vacuum
T8	1.0E-2	Spurious ES Actuation Signal
T12	4.0E-3	Loss of Low Pressure Service Water

determine the potential improvement (reduction) to these transient initiating-events, it was necessary to estimate the relative contribution that procedure-related operator errors play in each of the transient initiating-event frequencies.

Abnormal Operating Procedures

The six events shown in Table B.2 represent the "operator actions," which are modeled in the reference PRA, that were identified as being most likely affected by improvements to abnormal operating procedures. These operator actions may be improved by reducing their conditional probability. Since each of these operator actions may have contributions from factors other than procedures, it was necessary to determine the specific role that procedures play in each of these events (after all, if operators do not typically refer to the procedures for these actions, there can be no expected improvement in operator performance regardless of improvements made to procedures).

B.2.4.2 Identification of Affected Accident Sequences

The minimal cut sets that comprise the PRA's core-melt sequences for internal initiating-events were reviewed in order to identify those that contain one or more of the affected parameters listed in Tables B.1 and B.2. The core-melt accident sequences that involve one or more of the potentially affected parameters are listed in Table B.3. Since there are hundreds of cut-sets contained in the affected accident sequences, only the accident sequences themselves are listed here. The reader interested in the detailed cut sets that comprise the affected accident sequences is referred to the reference PRA (Sugnet et al. 1984).

TABLE B.2. Potentially Affected Operator Actions

Event Symbol	Original Conditional Probability	Description of Operator Action
RESSFSI	0.1	Operator fails to provide RCP seal injection from the SSF within 30 min. of losing seal cooling via HPI
SW3BPPSH	0.002	Operator fails to start standby LPSW pump
HPRCPH	0.01	Operators fail to trip the RCPs following loss of seal cooling (within 15 minutes)
REIA2/6	0.055	Failure of the operating staff to recover IA prior to the depletion of the UST
RESW12	0.013	Failure of the operating staff to recover LPSW from another source before failure of all HPI pumps
RESW108	0.11	Failure of the operating staff to recover LPSW to HPI pumps given a failure of LPSW108

TABLE B.3. Potentially Affected Core-Melt Accident Sequences[a]

Core-Melt Bin Type and Accident Sequence	Potentially Affected Parameters	Base-Case Frequency (events/yr)
IB,TQUs	T_1,T_2,T_4,T_8,T_{12},HPRCPH,RESW12, SW3BPPSH	5.9E-7
ID,T6QU	HPRCPH	2.5E-7
IE,TQUs	T_{12}	1.0E-7
IIE,TQUYXs	T_1,T_2,T_4,T_8,T_{12}	5.8E-8
IIF,TQUYXs	T_8	2.2E-7
IIIA,T2BU	T_2	1.2E-6
IIIB,T4BU	T_4	4.1E-7
IIIC,TBU	T_1,T_2,T_4,T_8,T_{12}	4.2E-7
IIIF,TBU	T_1,T_2,T_4,T_8,T_{12},REIA2/6	4.7E-6
IIIG,TBU	T_1,T_2,T_4,T_8,T_{12},RESW12,RESSFSI, RESW108,SW3BPPSH	1.5E-5
ATWS Sequences:		
I:6	T_1	1.7E-8
II:5	T_1	1.7E-8
III:15,14	T_1	3.4E-6
V:9,12,73,27	T_1,T_2	1.3E-6
VI:72,26,11,8	T_1,T_2	1.4E-6
	Total:	2.9E-5

(a) Based on the reference PRA (Sugnet et al. 1984).

B.2.4.3 Establish Base-Case Contribution of Procedure-Related Errors

The next step was to determine the portion of these affected parameters that might be affected by the candidate Procedure Upgrade Program. It was recognized that there may be various contributors to each of these transient initiating-events including electrical and hardware component failures, human errors, and equipment downtime due to testing, maintenance, and repair. In order to evaluate the effect of the Procedure Upgrade Program on these parameters, it was necessary to estimate the relative contribution of procedure-related operational errors to each affected parameter's base-case frequency. In other words, it was necessary to determine what fraction of the original base-case frequency could be influenced by improvements to operating procedures. This fraction represents the portion of the frequency which can be reduced and is referred to here as the procedure-related portion of the base-case affected frequency.

In the case of the transient initiating-events, this evaluation was accomplished using the SCSS database which contains coded information on all current LERs submitted by nuclear power plant utilities after January 1, 1981. The SCSS database allowed the coded LERs to be searched for the specific transient initiating-events identified in Table B.1. Once these LERs were identified, the SCSS database then searched the subset of LERs for those that have been assigned a cause/effect code representing "task description inadequacy". LERs which were identified with this code provided a reasonable estimate of the role that inadequate operating procedures have played in the recorded incidents. After reviewing the coded LERs and comparing the frequency of those with the SB code to those without it, an estimate of the relative contribution of procedure-related errors to the transient initiating-events was calculated. The results of the coded-LER review are given in Table B.4.

TABLE B.4. Results of the Coded-LER Review Analysis

Event	Original Base-Case Frequency (/year)	Event Description	Relative Contribution of Orig. Frequency from Procedure-related Errors
T1	5.7	Reactor/Turbine Trip	5%
T2	6.4E-1	Loss of Main Feedwater	5%
T4	2.1E-1	Loss of Condenser Vacuum	5%
T8	1.0E-2	Spurious ES actuation Signal	2%
T12	4.0E-3	Loss of Low Pressure Service Water	2%

The base-case procedure-related frequency is the portion of the original base-case frequency that is potentially affected by improved procedures. This procedure-related frequency was calculated by multiplying the original base-case frequency by the estimated contribution from procedure-related errors.

In other words, the contribution of procedure-related operational errors to the original transient initiating-event frequencies was calculated using the following formula:

$$P_i^{aff} = P_i * \alpha_i$$

where P_i^{aff} = the portion of the original base-case parameter frequency affected by procedure-related errors

P_i = the original base-case parameter frequency

α_i = the fraction of the original base-case affected parameter frequency that can be associated with procedure-related operational errors (derived from the SCSS analysis).

The results of these calculations are shown in the third column of Table B.5.

TABLE B.5. Affected Transient Initiating-Event Frequencies

Event	Base-Case Frequency (/year)	Procedure-Related Portion of Base-Case Frequency (/year)	Adjusted-Case Frequency (/year)[a]
T1	5.7	2.85E-1	5.6057
T2	6.4E-1	3.20E-2	0.6342
T4	2.1E-1	1.05E-2	0.2070
T8	1.0E-2	2.00E-4	9.97E-3
T12	4.0E-3	8.00E-5	3.99E-3

(a) From the survey of expert judgment discussed in Section B.2.4.4.

Similarly to the transient initiating-events, each operator action identified in Table B.2 was reviewed to determine that portion of the base-case frequency that could potentially be affected by improved procedures. This evaluation was performed by a team of PNL experts who have expertise in the areas of human performance and nuclear plant operations, and specifically with the affects on improved procedures of operator error probabilities. These experts were asked to estimate the importance that procedures play in each of the operator actions. The qualifications of the PNL experts are discussed in Section B.2.3.

The experts concluded that procedures could play a dominant role in each of the operator actions identified. They judged that the procedure-related portion of the base-case probabilities is equal to 100% of the base-case probability for each of the operator actions except for the RESW108 event, whose procedure-related probability was judged to be 70% of the base-case probability (e.g. there may be some scenarios for this event where even ideal procedures would not reduce the chance of failure). This provided estimates of the procedure-related parameter probabilities for each of the operator actions, which can be combined with the previous estimates of the procedure-related frequency for each of the affected transient initiating events. The results of these judgments are shown in the third column of Table B.6.

TABLE B.6. Affected Operator Action Probabilities

Event	Base-Case Probability	Procedure-Related Portion of Base-Case Probability	Adjusted-Case Probability[a]
RESSFSI	0.1	0.1	0.0527
SW3BPPSH	0.002	0.002	1.6E-3
HPRCPH	0.01	0.01	7.8E-3
REIA2/6	0.055	0.055	0.0262
RESW12	0.013	0.013	5.3E-3
RESW108	0.11	0.077	0.0768

(a) From the survey of expert judgment discussed in Section B.2.4.4.

B.2.4.4 Change in the Affected Parameter Values

Next, the adjusted-case values for all 11 of the affected parameters that would result from implementation of the candidate Procedure Upgrade Program were estimated. This was accomplished by developing and conducting a survey of expert opinion. The survey, which requested experts to estimate the potential reduction to the frequency or probability of each of these events that could be expected due to improving operating procedures, is discussed in detail in Appendix C. The results of the survey, in the form of adjusted-case frequencies and probabilities, are presented in Tables B.5 and B.6, respectively.

B.2.4.5 Estimating the Resulting Reduction in Public Risk

Next, the base-case procedure-related frequencies and probabilities associated with the generic plant model were modified to account for the variation in procedure quality that actually exists in the operating nuclear plants in the United States. This was accomplished by, first, using the results of prior research reported in NUREG/CR-3968, Study of Operating Procedures in Nuclear Power Plants: Practices and Problems (Morgenstern et al. 1987) to define three categories of plants based on current quality of operating procedures. That research, which evaluated and scored the quality of operating procedures being used in a large sample of operating nuclear power plants, concluded that operating procedures in many U.S. nuclear plants are of unacceptably poor quality. Though most plants have unacceptably poor quality procedures in an absolute sense, the procedure evaluation scores from that prior research allowed the definition of three categories of plants based on their relative current quality of operating procedures: 42% (or 50) of the plants have relatively poor procedures; 39% (or 46) of the plants have procedures of intermediate quality; and 19% (or 22) of the plants have relatively good procedures. Then, using the engineering judgment of the team of PNL experts, the original base-case values of the affected parameters were reestimated for each of the three plant categories. That is, new, "modified" (before procedure improvement) base-case frequencies for the 11 events were estimated for each of the three plant categories based on the original base-case probabilities associated with the generic plant.

PNL experts, relying on their diverse experience with operating procedures at various nuclear power plants, judged that for those plants whose current procedures are of relatively good quality, the procedure-related portion of the base-case frequency would be 70% less than for the generic model. For those plants whose current procedures are of intermediate quality, the PNL experts judged that the procedure-related portion of the base-case frequency would be 20% less than for the generic model. Finally, for those plants whose current procedures are considered to be of relatively poor quality, the PNL experts estimated that the procedure-related portion of the base-case parameter frequencies would be twice as large as for the generic model. These estimates where then used to calculate the modified base-case parameter frequencies for the three categories of actual plants.

This relationship is portrayed in Figure B.2. The original base-case values of the affected parameters are portrayed as the heavy dotted line labeled "Generic Plant (Base-Case)." The new, modified base-case values for the three plant categories that recognize the actual variance in current procedure quality are the three thin dotted lines labeled accordingly. The heavy solid line labeled "Adjusted-Case (All Plants)" portrays the probabilities of the affected parameters that were estimated to obtain at all plants

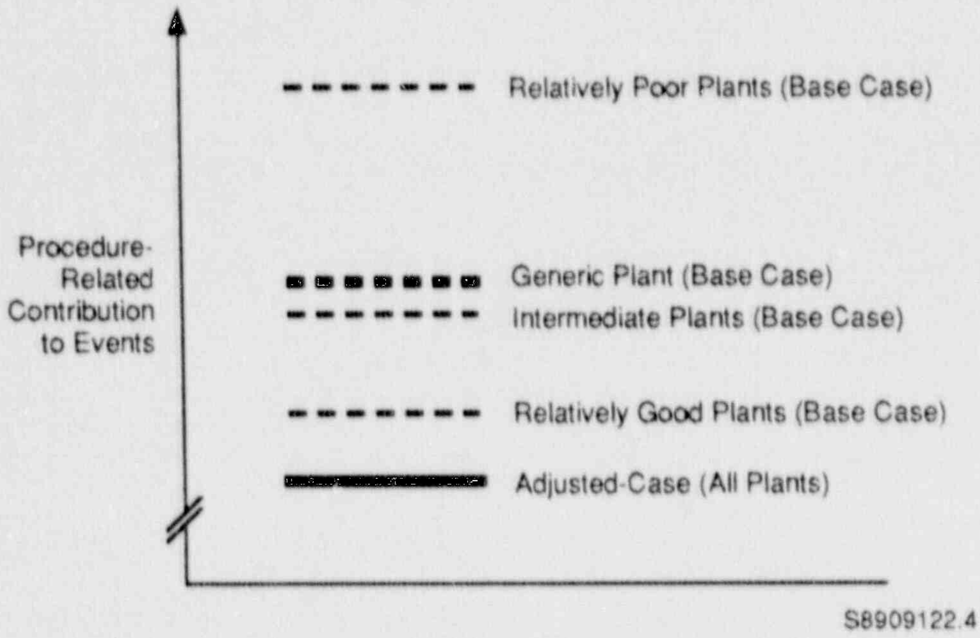

FIGURE B.2. Relationship Between Actual Plant Categories and the Generic Model after implementation of the Procedure Upgrade Program due to the creation and use of new, high-quality operating procedures. The actual values of modified base-case parameters, along with their corresponding adjusted-case parameters, are given in Table B.7.

TABLE B.7. Modified Parameter Frequencies For Actual Plants

Event Identifier	Modified Base-Case Frequency (/year) — Current Quality of Operating Procedures: Relatively Poor	Intermediate	Relatively Good	Adjusted-Case Frequency (/year)
T1	5.794	5.681	5.634	5.606
T2	0.646	0.639	0.636	0.634
T4	0.213	0.209	0.208	0.207
T8	1.00E-2	9.99E-3	9.98E-3	9.97E-3
T12	4.01E-3	4.00E-3	3.99E-3	3.99E-3
RESSFSI[(a)]	0.147	0.091	0.067	0.053
SW3BPPSH	2.4E-3	1.9E-3	1.7E-3	1.6E-3
HPRCPH	1.2E-2	9.6E-3	8.5E-3	7.8E-3
REIA2/6	8.4E-2	4.9E-2	3.5E-2	2.6E-2
RESW12	2.1E-2	1.1E-2	7.6E-3	5.3E-3
RESW108	0.143	0.103	0.087	0.077

(a) The last six events are operator actions. Therefore, the numbers represent conditional probabilities rather than frequencies.

The modified affected parameter values were then incorporated into the original reference PRA risk equations for the core-melt accident sequences identified in Table B.3. The results of these calculations provided an estimate of the potential core-melt frequency reduction for each category of plants.

The results of these calculations are provided in Tables B.8 and B.9. The upper and lower bounds have been calculated by adjusting the mean value of each of the adjusted-case parameter frequencies up and down by a factor of two times the standard deviation of the experts' judgments.

In order to extend these estimates of core-melt frequency reduction to estimates of the reduction in public health risk, dose conversion factors were applied to these estimates. Since each of the affected accident sequences identified in Table B.3 may result in an accident with a different release category, these dose conversion factors were applied to each accident sequence individually, then summed to represent the total reduction in public risk. The dose conversion factors summarized in Table B.10 were used to evaluate this expected public risk reduction.

TABLE B.8. Affected Accident Sequence Frequencies for Actual Plants

Core-Melt Bin Type and Accident Sequence	Modified Base-Case Frequency (events/yr) Relatively Poor	Intermediate	Relatively Good	Adjusted-Case Frequency (events/yr) All Plants
IB,TQUs	1.1E-6	5.0E-7	3.2E-7	2.3E-7
ID,T6QU	3.0E-7	2.4E-7	2.1E-7	1.9E-7
IE,TQUs	4.1E-8	4.1E-8	4.1E-8	4.1E-8
IIE,TQUYXs	4.9E-8	4.8E-8	4.8E-8	4.8E-8
IIF,TQUYXs	1.0E-8	9.9E-9	9.9E-9	9.9E-9
IIIA,T2BU	9.4E-7	9.3E-7	9.2E-7	9.2E-7
IIIB,TRBU	2.9E-7	2.9E-7	2.9E-7	2.9E-7
IIIC,TBU	2.7E-7	2.7E-7	2.6E-7	2.6E-7
IIIF,TBU	3.3E-6	2.0E-6	1.4E-6	1.0E-6
IIIG,TBU	2.9E-5	1.2E-5	6.9E-6	4.6E-6
ATWS I	1.743-8	1.70E-8	1.69E-8	1.68E-8
ATWS II	1.74E-8	1.70E-8	1.69E-8	1.68E-8
ATWS III	3.48E-6	3.41E-6	3.38E-6	3.36E-6
ATWS V	1.31E-6	1.28E-6	1.27E-6	1.27E-6
ATWS VI	1.48E-6	1.38E-6	1.37E-6	1.37E-6
Total	4.15E-5	2.22E-5	1.65E-5	1.37E-5

TABLE B.9. Estimated Reduction in Core-Melt Frequencies

	Reduction in Core-Melt Frequency (ΔF) (events/reactor-year) Current Quality of Operating Procedures Relatively Poor	Intermediate	Relatively Good
Best Estimate	2.8E-5	8.5E-6	2.8E-6
Upper Bound	3.4E-5	1.5E-5	9.1E-6
Lower Bound[a]	1.7E-5	0	0

(a) In several cases, adjusting the mean value up by two times the standard deviation brought the value up to (or above) the original base-case value. This explains why the lower bound for the "Intermediate" and "Relatively Good" plants indicates a reduction in core-melt frequency of zero.

The details of these calculations followed those provided in the reference PRA. Since those calculations are rather complex, the interested reader is referred to the reference PRA (Suget et al. 1984) for detailed information on core-melt bins, containment safeguard states, and containment response. Suffice it to say here that, for those accident sequences identified in Table B.3, the average value of the dose conversion factor is approximately 8.90E+5 person-rem/CM accident. This approximate value, which is

TABLE B.10. Summary of Consequence Ranges for Release Categories[a]

Release Category	Population Dose[b] (person-rem)
1A	1.0E+8 to 3.0E+8
1B	1.0E+6 to 4.0E+7
2	1.0E+6 to 1.0E+8
3	No effect
4	0 to 1.0E+6
5	No effect

(a) Taken from the reference PRA (Sugnet et al. 1984)
(b) To ensure conservatism, the upper limit on the range is used as the best estimate of the population dose.

indicative of a core-melt accident of moderate severity, can be used to calculate the public health risk reduction values based on the core-melt frequency reductions given in Table B.8.

Application of those dose conversion factors to each of the affected accident sequences identified in Table B.3 results in public risk reduction as given in Table B.11. Upper and lower bounds are based on the upper and lower bounds on the estimated reduction in core-melt frequency as given in Table B.9.

TABLE B.11. Estimated Reduction in Public Health Risk

	Reduction in Public Health Risk (ΔW) (person-rem/reactor-year)		
	Current Quality of Operating Procedures		
	Relatively Poor	Intermediate	Relatively Good
Best Estimate	24.9	7.6	2.5
Upper Bound	30.2	13.3	8.1
Lower Bound	15.1	0	0

APPENDIX C

OPERATING PROCEDURES IMPROVEMENT SURVEY

APPENDIX C

OPERATING PROCEDURES IMPROVEMENT SURVEY

C.1 INTRODUCTION

This value-impact assessment uses a multistep methodology to evaluate the contribution to reduction in reactor core-melt frequency that could be expected from improving the quality of operating procedures in nuclear power plants. The general approach taken, as described in Appendix B, can be characterized as: 1) estimate the contribution to reactor core-melt frequency of procedure-related operational errors during normal and abnormal operating conditions assuming use of procedures of a (relatively poor)[a] quality currently found in most plants, 2) estimate the same contribution assuming that the quality of operating procedures is improved due to the candidate Procedure Upgrade Program instituted in response to NRC regulatory action, and 3) compare the "before-improvement" and "after-improvement" estimates of overall core-melt frequency to determine the potential reduction in core-melt frequency that could be expected from upgraded procedures.

The purpose of this appendix is to describe the survey of expert opinion that was used in the second phase mentioned above to estimate what change in procedure-related operational errors could be expected if operating procedure quality is improved by specific measures.

The survey was constructed using as guidance the principles for obtaining human reliability data and estimates set forth in NUREG/CR-3688, Generating Human Reliability Estimates Using Expert Judgment, Volumes 1 and 2, (Comer et al. 1984). That report describes a project that evaluated two techniques--paired comparisons and direct numerical estimation--for using expert judgment to generate human error probability estimates and associated uncertainty bounds. The authors concluded that each technique should have sufficient validity for use in developing human error probability estimates. They found that the convergent validity of each technique was more than sufficient for deriving human error probability estimates to support any type of probabilistic study in which human error is a consideration. The report presents detailed procedures for using each technique. With certain minor modifications, these procedures were followed in the creation and administration of this survey.

(a) Findings reported in Morgenstern 1987 and Barnes and Radford 1987.

C.2 PREPARATION OF THE SURVEY INSTRUMENT

This section describes the reasons for choice of survey technique and the contents of the survey instrument. A copy of the survey instrument is included as Attachment 1 to this appendix.

C.2.1 Choice of Technique to Collect Expert Judgment

Direct numerical estimation, as described in Comer et al. (1984), was the technique used in this survey to gather data regarding potential reductions in event probabilities due to improved operating procedures. In their report Comer et al. concluded that there are only minor differences in the validity of the results of the two survey techniques that they examined. They suggested, therefore, that selection of technique from among these two can be based on practical considerations rather than upon result reliability. Direct numerical estimation was chosen for this survey for several reasons. This technique requires a smaller number of experts to attain statistical reliability. Of the two techniques, direct estimation requires less of the experts' time, and data analysis is relatively simple. Also, a survey using this technique could be performed effectively and inexpensively by mail rather than gathering all the experts in one place for a data collection session.

C.2.2 Contents of the Survey Instrument

The survey instrument consisted of three parts. The first of these parts presented introductory material that explained the context and purpose of the survey. The experts were told that 11 separate events that can occur at nuclear power plants had been identified that could, at least partially, be influenced by the use of normal and/or abnormal operating procedures. It was explained that six of these events were operator recovery actions that were required to be taken following specific equipment failures. The remaining five events were anticipated transients that could occur as a result of equipment malfunction or as a result of operator errors. (Refer to Appendix B for a detailed description of the source and nature of these 11 events.) The experts were told that it was expected that operators would likely refer to written operating procedures before or while taking the six recovery actions and that it could also be expected that poor or inadequate operating procedures could contribute, at least to a small extent, to the occurrence of the five transient initiating-events. The remainder of the introductory material consisted of brief instructions for filling out the survey. The experts were asked to contact the data collection administrator if they had any questions regarding the survey.

The second part of the survey instrument consisted of a 1-page summary of the problems that have been found to exist with operating procedures currently in use in nuclear power plants and some of the ways that they could potentially be improved. For example, the survey stated that procedures sometimes fail to describe the specific actions to be taken by operators in a step-by-step manner. The experts were told that this shortcoming could be rectified by having procedures written in short, concise, identifiable steps with only one operator action per step. Nine such potential procedure improvements were briefly described. The purpose of this summary was to help the experts envision a situation in which procedures of higher quality than they are used to were being used in nuclear power plants. It was assumed, then, that the experts could combine their experience with procedures of current, rather poor, quality with these ideas of how procedures could be improved and, with both in mind, make informed judgments as to how improved procedures may affect the frequency of occurrence of the 11 events.

The rest of the body of the survey instrument dealt with these eleven events. A separate page was devoted to each of the six operator recovery actions and to each of the five transient initiating-events. First, the event was briefly described. For example, Operator Action #1 was described as follows:

> "Following a loss of normal Reactor Coolant Pump (RCP) seal injection from the High Pressure Injection (HPI) pumps, the operating staff fails to provide RCP seal injection from the Safe Shutdown Facility (SSF) within approximately 30 minutes. RCP seal-injection flow is normally provided by the HPI pump that is operating to supply the Reactor Coolant System makeup. In this situation, the RCP-seal makeup fails due to system faults and must be recovered from the Safe Shutdown Facility within 30 minutes of seal leakage to prevent leakage in excess of the SSF makeup capability."

These event descriptions were derived by translating from rather obscure descriptions of the events in the reference PRA to wording that would be more likely to be understood by personnel more used to nuclear power plant operations language. The clarity and level of detail of these event descriptions were checked in a pretest setting using PNL personnel familiar with nuclear plant control room operations and operating procedure usage.

Next, the current probability or frequency of the event was stated. This attribute of each of the 11 events had been derived during the first stages of this project's risk assessment calculations and was characterized there as the "base-case procedure-related parameter frequencies." That is, the probability or frequency of each of the 11 events as stated in the survey was that portion of the total probability of the event as given in the reference PRA that may potentially be affected by improvements in normal or abnormal operating procedures. It was assumed that these base-case probabilities and frequencies obtain under current operating conditions in which operating procedures of relatively poor quality are used.

In the presentation of the base-case frequency for each of the five transient initiating-events, the experts were explicitly told that the event frequency was due to procedure-related operator error. This was necessary to make sure that the experts realized that they were dealing with only the procedure-related portion of the events' overall frequencies. A similar explicit statement was not necessary in the case of the six operator actions because the procedure-related portion of the events' probabilities was at or close to 100% of their overall probabilities.

For each event a scale was placed on the page beside the written explanatory material. For the six operator action events, the scale was stated both in terms of "Probability" and in "Chance of Occurrence." These scales were calibrated in probabilities of from "1.0" to "0.0000001" and in corresponding chances of occurrence of from "1 chance in 1" to "1 chance in 10,000,000." Thus, this scale allowed the experts to think in terms of either probabilities or in "chances," e.g., 1 chance in 50.

The scale used for the five transient initiating-events was stated in terms of "Base Frequency (X times per year)" and in "Base Frequency (1 time per Y years)." These scales were calibrated in frequencies of from "1.0" time per year to "0.0000001" times per year and in corresponding frequencies of from "1 time in 1 year" to "1 time in 10,000,000 years." Again, this scale allowed the experts to think in terms of two alternative types of frequency.

The calibration marks on each of these scales were set at an equal distance from each other. In other words, the actual distance between the marks for "1 in 1 year"and "1 in 2 years" was, for example, equal to the distance between "1 in 10,000 years" and "1 in 20,000 years." The scales were constructed in this way with the intention of providing simple, straight-forward scales that would be easy to use, provide sufficient variation in frequencies to be able to accommodate the range of the 11 subject frequencies, and to produce meaningful survey results.

The base probability or frequency as stated in the description of each event was marked on the appropriate place on the scale. The experts were asked to put a slash across the frequency scale to indicate the frequency with which they thought the particular event would occur due to procedure-related operator errors if operating procedures were improved as had been described earlier in the survey instrument.

On each of these scales the smallest increment of change was a 50% decrease in event probability. Such large increments between calibration points could potentially have caused an experimental bias toward overestimation of decrease in event probability. The actual survey results showed, however, that this distance between calibration points apparently did not produce such bias. Fully one-half of the experts' marks on the survey scales fell between the calibration points.

C.3 CHOICE OF EXPERTS

Comer et al. (1984) state that familiarity with the tasks to be judged is the primary qualification of subject matter experts. The experts asked to make judgments regarding the impact of improved procedures were, therefore, chosen for their in-depth knowledge of plant systems, operations, and control room procedures. All are very familiar with the use of operating procedures in control room operations. Because of the varying backgrounds and current work experience of the various experts, it was assumed that the quality of the different operating procedures with which they have experience would tend to approximate the range of procedure quality found in nuclear plants across the country.

The subject matter expert qualifications suggested by Comer et al. (1984) do not include prior experience in making probabilistic judgments. These experts chosen for this survey were not necessarily experienced in making probabilistic judgments nor were they given instruction in making such judgments as part of this survey process. Experience with probabilistic judgment making was not considered a necessary qualification in this case because the experts were to be given the base-case probability or frequency for each event that could serve as an "anchor" with which to estimate new levels assuming improved procedures were being used. Having the benefit of these anchor probabilities was thought to obviate the need for any particular prior experience in probabilistic judgment making on the part of the experts.

Comer et al. (1984) suggest that as few as six experts would be sufficient for direct estimation though using more than six would tend to ensure the necessary statistical reliability of the survey results. Nine experts were originally contacted and asked to fill out this survey instrument. Of this number, eight experts were either Senior Reactor Operators or Licensed Operators at one of five nuclear power plants. All of these five plants are pressurized water reactor plants designed by Babcock & Wilcox (B&W). The scope of the survey was confined to B&W plants because the probabilistic risk assessment (PRA) that was used as a basis for estimating change in core-melt frequency for this project is for the Oconee plant (Sugnet et al. 1984), which was designed by B&W. The subject operations events, the hardware involved, and the operating procedures that would be used in conjunction with those events that were taken from the Oconee PRA were sufficiently B&W plant-specific as to require the reactor operators to be drawn only from B&W-designed plants. The ninth expert of the original group chosen is a member of the training staff at the NRC Training Center at Chattanooga, Tennessee. He is a former reactor operator at a B&W-designed plant with considerable experience in procedure use.

Because less than a 100% return of the survey from this original group of nine experts was expected, it was decided to include five PNL staff members in the group to be surveyed. Like the original group, each of these people was chosen for his or her substantial experience in nuclear power plant control room operations and the use of operating procedures in those operations. Only one of these PNL staff members has hands-on experience with the operations of and procedure use in B&W-designed plants, however. The others do have considerable experience with control room operations of various plants designed by Westinghouse. Because the controlling functions of Westinghouse plants are quire similar to those of B&W plants, it was decided that these PNL staff members could provide sufficiently well informed judgment to make a positive contribution to the survey.

C.4 DATA COLLECTION

Surveys instruments were mailed to the 14 experts. The survey instrument had been designed to be straightforward and easily understandable and to elicit expert judgment without further oral instruction. It was estimated that the survey could be completed within 30 to 45 minutes.

Ten completed surveys were returned over a period of from 3 to 8 weeks from the date in which they were mailed to the experts. Six of the nine surveys sent to power plant and training center personnel were completed. Of the other three, one completed survey was apparently lost in the mails; the other two experts did not respond due to major work commitments. Four of the five surveys sent to PNL staff members were filled out and returned. Based on the substantial consistency among the experts' judgments and on conversations with some of the 10 experts who filled out the survey, the survey was apparently clear and relatively easy to fill out. The survey appears to have been effective in eliciting expert opinion regarding changes in operator error that could be expected to result from improving operating procedures.

C.5 SURVEY RESULTS

The numerical results from this survey are set forth in Table C.1. This table presents the base probabilities for each of the 11 events that were presented to the survey participants as the probabilities associated with the use of procedures of current quality. The new probabilities estimated by each of the 10 experts for each of the 11 events assuming the use of high quality procedures are presented. The table then states the arithmetic mean of these new probabilities. These mean values represent the collective expert judgment of the survey participants as to what the new probabilities of each of the eleven events would be if procedures are upgraded pursuant to an NRC regulatory action. These mean values are expressed in one more significant digit than are the base probabilities and the estimates of the individual experts. This is deemed acceptable because the experts were given scales ranging from 1.0 to 0.0000001 and, therefore, their estimates could be extended to more than five significant digits.

Comer et al. (1984) recommend that results of expert surveys be used only if there is reasonable agreement among the experts regarding the estimates. They suggest using the Kendall coefficient of concordance as a measure of the extent of that agreement. This coefficient measures the average correlation among the various experts on a zero-to-one scale where 0 is no agreement and 1 is complete agreement.

Calculation of this coefficient for the survey results gives a result of 0.887. This result indicates considerable agreement among the experts as to the affect of procedure improvement and can be taken as an indication that the experts interpreted the events described in the survey instrument quite similarly. While the result of this test of consistency is quite high, it should be noted that part of the reason for this result may be that the

TABLE C.1. Operating Procedures Improvement Survey Results

	Operator Events						Transient Events				
Base	1	2	3	4	5	6	7	8	9	10	11
Probabilities	0.100	0.002	0.010	0.056	0.012	0.110	0.280	0.032	0.010	0.0002	0.00008
Survey Participant ID#s											
1	0.067	0.002	0.007	0.020	0.008	0.067	0.200	0.025	0.008	0.0002	0.00008
2	0.020	0.001	0.005	0.010	0.005	0.010	0.133	0.013	0.005	0.0001	0.00007
3	0.050	0.001	0.002	0.056	0.003	0.050	0.280	0.010	0.010	0.0002	0.00007
4	0.100	0.001	0.010	0.005	0.001	0.005	0.100	0.050	0.010	(a)	0.00008
5	0.067	0.002	0.010	0.020	0.010	0.020	0.222	0.032	0.010	0.0002	0.00008
6	0.050	0.002	0.010	0.025	0.006	0.050	0.200	0.022	0.003	0.0002	0.00007
7	0.005	0.002	0.002	0.056	0.001	0.020	0.280	0.032	0.009	0.0001	0.00005
8	0.100	0.001	0.012	0.020	0.004	0.100	0.167	0.028	0.002	0.0002	0.00007
9	0.040	0.002	0.010	0.040	0.005	0.006	0.075	0.020	0.008	0.0001	0.00005
10	0.028	0.002	0.010	0.010	0.010	0.110	0.250	0.030	0.010	0.0002	0.00007
Mean	**0.0527**	**0.0016**	**0.0078**	**0.0262**	**0.0053**	**0.0438**	**0.1907**	**0.0262**	**0.0075**	**0.00017**	**0.00007**
Std. Dev.	0.0316	0.0005	0.0036	0.0184	0.0033	0.0385	0.0716	0.0113	0.0031	0.0001	0.00001

(a) For this event the participant estimated a probability of 0.1000 because this event had occurred at least twice at his plant. This estimate was eliminated from the results because the participant had ignored the base probability and, therefore, there is no way to determine what affect he would ascribe to improved procedures.

experts were given an "anchor" -- the base probabilities -- that may have produced some inherent consistency by establishing a basic range for the experts' estimates.

This reasonably high across-expert consistency does not necessarily indicate that the survey results are a strictly accurate prediction of what the eleven events' probabilities or frequencies would be if operating procedures are improved. According to Comet et al. (1984), however, human error probabilities derived using the prescribed procedures and achieving this degree of consistency can reliably be used to support probabilistic risk assessment analyses such as this determination of change in core-melt frequency. To the extent, therefore, that the original base-case probabilities and frequencies derived from the reference PRA are accurate, these adjusted-case survey results can be used with confidence to represent the relative change in event occurrence due to improved procedures.

A t-test, which is a statistical method for measuring the difference between a known value and one derived from a sample survey, was also performed for the survey results. The t-test allows a determination as to whether the individual means are different from the base probabilities with a confidence level of 95%. For 8 of the 11 events the mean probability results are lower than the base probabilities with a confidence level of 95%. For events #3, #8, and #10 the confidence levels do not reach 95%; they are 91%, 86%, and 92%, respectively. In events #3 and #8 for which the mean is not significantly less, there were experts that estimated that the probability of the event would increase due to improved procedures.[a] When those responses are removed the difference becomes significant for event #8 but not for event #3. Further analysis of this latter event indicates that half of the respondents estimated no change in probability.

(a) In both instances, the expert's slash mark across the scale was very close to, though slightly above, the base-case probability mark. It appears likely that each expert may have intended to signify no change in probability rather than a (counterintuitive) increase in probability due to improved procedures.

Battelle Pacific Northwest Laboratories

OPERATING PROCEDURES IMPROVEMENT SURVEY

This survey is being conducted as part of a project to determine what would be the potential benefits of improving normal and abnormal operating procedures in commercial nuclear power plants around the United States. This project is being conducted for the Nuclear Regulatory Commission by the Battelle Pacific Northwest Laboratories.

The purpose of this survey is to obtain your opinion on the potential effects of improving normal and abnormal operating procedures at nuclear power plants. In order to make this evaluation, eleven separate events that occur at nuclear power plants have been identified. Each of these events is expected to be influenced (at least partially) by the use of normal and/or abnormal operating procedures. The first six events are operator recovery actions which are required to be taken following specific equipment failures. The remaining five events are anticipated transients that may occur as a result of equipment malfunction or as a result of operator errors due to poor or inadequate written procedures.

It is expected that operators would likely refer to written operating procedures before or while taking the six recovery actions described in this survey. Likewise, it is expected that poor or inadequate operating procedures could contribute (at least to a small extent) to occurrence of the five transient initiating events described in this survey. Of course, these eleven events are only a small sample of potential safety-significant events that are influenced by operating procedures. However, for this study, only these eleven events are being specifically evaluated.

For each of the eleven events, you will be given an estimated frequency of occurrence assuming that normal and abnormal operating procedures of current quality are being used. The frequencies of occurrence of the six operator recovery actions are expressed in a chance of occurrence, for example, one chance in ten. The frequencies of the transient initiating events are expressed in times per year, for example, .28 times per year. Then you will be asked for your opinion of how each frequency of occurrence would change if those operating procedures were improved in the ways described in the survey.

Please follow these instructions:

1) First, read "How Operating Procedures Could Be Improved" which generally describes some of the ways in which utilities can improve their current normal and abnormal operating procedures.

2) Then, for each of the eleven events, read the information provided and put a horizontal slash across the scale at the point that describes what you think the event's frequency would be <u>if improved operating procedures were being used</u>.

3) If you believe that the frequency would not change due to improved operating procedures, please place your horizontal slash at the place indicated by the arrow, i.e., at the present frequency.

4) If you would like to provide any information that would explain you opinion about a change in event frequency, please write that information at the bottom of the page.

5) Please send the completed survey to Thomas Grant, Battelle Human Affairs Research Centers, 4000 N.E. 41st Street, Seattle, Washington 98105. If you have any questions, please call Thomas Grant at (206) 525-3130.

Thanks very much for completing this survey.

How Operating Procedures Could Be Improved

The purpose of this survey is to determine the potential benefits that would result if nuclear power plants around the country improved their current normal (NOPs) and abnormal (AOPs) operating procedures. For you to be able to answer this questionnaire, you need to know how NOPs and AOPs could be improved. Therefore, the following is a short summary of some of the problems that have been found with NOPs and AOPs currently used in several power plants and some of the ways that they potentially could be improved.

- Procedures are sometimes written in vague terms and do not specifically describe what the operator must do to correctly complete the required operation. **Procedures could be improved by writing them in more definite and specific terms.**

- Procedures sometimes fail to describe the specific actions to be taken by operators in a step-by-step manner. **Procedures could be rewritten in short, concise, identifiable steps with only one action per step.**

- Current procedures sometimes fail to provide clear indicators of when a particular objective has been achieved or when the procedure has been completed and should be exited. **Procedures could be improved by specifying a single quantitative indicator to tell the operator when the objective of the step has been accomplished. Also, procedures could more clearly identify the controls, indicators, and equipment relevant to the procedure.**

- Some procedures have been found to be technically inaccurate. For example, a procedure may refer to panel F when panel E is intended. **Procedures could be improved by making sure that such technical inaccuracies are eliminated.**

- Different procedures used by the same operators may be written in different formats making it difficult for operators to quickly understand and use the procedures. **Procedures could be rewritten using a consistent format to increase the ease of use.**

- Procedures sometimes do not contain a table of contents. **Procedures could be improved by adding a table of contents that would make it easier for operators to quickly find the needed section of the procedure.**

- Procedures sometimes lack checklists, placekeeping aids, and other tools to enhance the ease and accuracy with which a procedure can be used. **Procedures could be improved by including these aids as well as clearly prepared graphs, figures, and flow charts to increase the ease of use.**

- Procedures may lack cautions, warnings, or explanations describing how fast in responding or how sensitive particular controls are. **Procedures could be improved by including this information.**

- Procedures are sometimes not matched to the level of experience of the operators using them. In some cases, **procedures could be improved by writing procedures at two levels, for experienced and for less experienced operating personnel.**

These are a few of the improvements that could be made to current normal and abnormal operating procedures. In summary, such improvements would be intended to increase operators' ability to assess needed information, to decrease response time in abnormal situations, and to decrease errors due to confusion.

ATTACHMENT

PLEASE CONSIDER THE FOLLOWING INFORMATION

Operator Action #1

Operator Action Description:
Following a loss of normal Reactor Coolant Pump (RCP) seal injection from the High Pressure Injection (HPI) pumps, the operating staff fails to provide RCP seal injection from the Safe Shutdown Facility (SSF) within approximately 30 minutes.

Situation:
RCP seal-injection flow is normally provided by the HPI pump that is operating to supply the Reactor Coolant System makeup. In this situation, the RCP-seal makeup fails due to system faults and must be recovered from the Safe Shutdown Facility within 30 minutes of seal leakage to prevent leakage in excess of the SSF makeup capability.

Current event probability:
Research shows that in this situation there is currently a probability of about 1 chance in 10 of the operating staff not being able to provide RCP seal injection within 30 minutes. That probability is marked on the scale to the right.

Probability	Chance of Occurrence
1.0	1 chance in 1
.5	1 chance in 2
.2	1 chance in 5
.1	1 chance in 10
.05	1 chance in 20
.02	1 chance in 50
.01	1 chance in 100
.005	1 chance in 200
.002	1 chance in 500
.001	1 chance in 1,000
.0005	1 chance in 2,000
.0002	1 chance in 5,000
.0001	1 chance in 10,000
.00005	1 chance in 20,000
.00002	1 chance in 50,000
.00001	1 chance in 100,000
.000005	1 chance in 200,000
.000002	1 chance in 500,000
.000001	1 chance in 1,000,000
.0000005	1 chance in 2,000,000
.0000002	1 chance in 5,000,000
.0000001	1 chance in 10,000,000

Please put a slash across the scale to indicate the probability with which you think this operator error would occur if NOPs and AOPs were improved as described in How Operating Procedures Could Be Improved.

PLEASE CONSIDER THE FOLLOWING INFORMATION

Operator Action #2

Operator Action Description:
Operator fails to start standby Low Pressure Service Water (LPSW) pump.

Situation:
The operating Low Pressure Service Water pump fails to continue running. The operator must start the standby pump to provide vital cooling to several systems and components, including the Reactor Building cooling units and room cooling for several areas. There will be numerous direct and indirect indications of the loss of LPSW. Diagnosis is not thought to be a problem. Time is not long before important effects result, especially HPI-pump burnup due to the loss of pump-motor cooling.

Current event frequency:
Research shows that in this situation there is currently a probability of about <u>1 chance in 500</u> of the operating staff failing to start the standby LPSW pump. That probability is marked on the scale to the right.

Probability	Chance of Occurrence
1.0	1 chance in 1
.5	1 chance in 2
.2	1 chance in 5
.1	1 chance in 10
.05	1 chance in 20
.02	1 chance in 50
.01	1 chance in 100
.005	1 chance in 200
.002	1 chance in 500
.001	1 chance in 1,000
.0005	1 chance in 2,000
.0002	1 chance in 5,000
.0001	1 chance in 10,000
.00005	1 chance in 20,000
.00002	1 chance in 50,000
.00001	1 chance in 100,000
.000005	1 chance in 200,000
.000002	1 chance in 500,000
.000001	1 chance in 1,000,000
.0000005	1 chance in 2,000,000
.0000002	1 chance in 5,000,000
.0000001	1 chance in 10,000,000

Please put a slash across the scale to indicate the probability with which you think this operator error would occur if NOPs and AOPs were improved as described in How Operating Procedures Could Be Improved.

PLEASE CONSIDER THE FOLLOWING INFORMATION

Operator Action #3

Operator Action Description:
Operators fail to trip the Reactor Coolant Pumps (RCPs) following loss of seal cooling within 15 minutes.

Current event frequency:
Research shows that in this situation there is currently a probability of about <u>1 chance in 100</u> of the operating staff failing to trip the Reactor Coolant Pumps (RCPs) following loss of seal cooling within 15 minutes. That probability is marked on the scale to the right.

Please put a slash across the scale to indicate the probability with which you think this operator error would occur if NOPs and AOPs were improved as described in How Operating Procedures Could Be Improved.

Probability	Chance of Occurrence
1.0	1 chance in 1
.5	1 chance in 2
.2	1 chance in 5
.1	1 chance in 10
.05	1 chance in 20
.02	1 chance in 50
.01	1 chance in 100
.005	1 chance in 200
.002	1 chance in 500
.001	1 chance in 1,000
.0005	1 chance in 2,000
.0002	1 chance in 5,000
.0001	1 chance in 10,000
.00005	1 chance in 20,000
.00002	1 chance in 50,000
.00001	1 chance in 100,000
.000005	1 chance in 200,000
.000002	1 chance in 500,000
.000001	1 chance in 1,000,000
.0000005	1 chance in 2,000,000
.0000002	1 chance in 5,000,000
.0000001	1 chance in 10,000,000

PLEASE CONSIDER THE FOLLOWING INFORMATION

Operator Action #4

Operator Action Description:
The operating staff fails to recover Instrument Air (IA) prior to the depletion of the Upper Surge Tank (UST) which is a source of water for the Auxiliary Feedwater Pumps. Two to six hours are available to recover the IA depending on the volume of the Upper Surge Tank.

Current event frequency:
Research shows that in this situation there is currently a probability of about 1 chance in 18 of the operating staff not being able to recover Instrument Air prior to the depletion of the Upper Surge Tank. That probability is marked on the scale to the right.

Please put a slash across the scale to indicate the probability with which you think this operator error would occur if NOPs and AOPs were improved as described in How Operating Procedures Could Be Improved.

Probability	Chance of Occurrence
1.0	1 chance in 1
.5	1 chance in 2
.2	1 chance in 5
.1	1 chance in 10
.05	1 chance in 20
.02	1 chance in 50
.01	1 chance in 100
.005	1 chance in 200
.002	1 chance in 500
.001	1 chance in 1,000
.0005	1 chance in 2,000
.0002	1 chance in 5,000
.0001	1 chance in 10,000
.00005	1 chance in 20,000
.00002	1 chance in 50,000
.00001	1 chance in 100,000
.000005	1 chance in 200,000
.000002	1 chance in 500,000
.000001	1 chance in 1,000,000
.0000005	1 chance in 2,000,000
.0000002	1 chance in 5,000,000
.0000001	1 chance in 10,000,000

PLEASE CONSIDER THE FOLLOWING INFORMATION

Operator Action #5

Operator Action Description:
Failure of the operating staff to recover Low Pressure Service Water from another source before failure of all HPI pumps.

Situation:
The Low Pressure Service Water (LPSW) systems fail and the operators fail to cycle the HPI pumps to prevent burnout or fail to get LPSW from another source. Loss of LPSW during normal operations affects the operating HP pumps. However, the operator can cycle the HPI pumps at least once to buy some extra time. Service water may also be obtained from another unit (if available) or from the High Pressure Service Water (HPSW) system, depending on the reason for the LPSW failure. The HPI pumps heat up to the high-temperature alarm in 10 to 15 minutes. The operator is assumed to set up for a manual reactor trip, anticipating RCP alarms in about 5 to 10 minutes. Since the loss of LPSW will be annunciated, diagnosis is not assumed to be a problem. The operator is expected to try to restart all LPSW pumps and if not successful will align HPSW or LPSW from other units, if available. The RCP trip and the actions associated with shutting down the reactor add confusion to LPSW recovery efforts. The operator has about 35 to 40 minutes to recover LPSW after the RCP trip.

Current event frequency:
Research shows that in this situation there is currently a probability of about 1 chance in 80 of the operating staff failing to recover Low Pressure Service Water from another source before failure of all HPI pumps. That probability is marked on the scale to the right.

Please put a slash across the scale to indicate the probability with which you think this operator error would occur if NOPs and AOPs were improved as described in How Operating Procedures Could Be Improved.

Probability	Chance of Occurrence
1.0	1 chance in 1
.5	1 chance in 2
.2	1 chance in 5
.1	1 chance in 10
.05	1 chance in 20
.02	1 chance in 50
.01	1 chance in 100
.005	1 chance in 200
.002	1 chance in 500
.001	1 chance in 1,000
.0005	1 chance in 2,000
.0002	1 chance in 5,000
.0001	1 chance in 10,000
.00005	1 chance in 20,000
.00002	1 chance in 50,000
.00001	1 chance in 100,000
.000005	1 chance in 200,000
.000002	1 chance in 500,000
.000001	1 chance in 1,000,000
.0000005	1 chance in 2,000,000
.0000002	1 chance in 5,000,000
.0000001	1 chance in 10,000,000

PLEASE CONSIDER THE FOLLOWING INFORMATION

Operator Action #6

Operator Action Description:
Failure of the operating staff to recover the LPSW to HPI pumps given a failure of a manual valve in the common discharge header.

Situation:
A manual valve in the Low Pressure Service Water discharge header closes, resulting in common mode closing of the LPSW discharge cooling path from major loads in the LPSW system. The LPSW pumps continue to operate, and some parts of the system receive cooling as normal. HPI high temperatures will be reached (if operating) in 2 to 5 minutes and the RCP alarm setpoints will be reached in 10 to 15 minutes. The fault would require diagnosis, and recovery actions for the fault would be difficult. The operability of the HPI pumps is of primary concern. The operator could buy time for recovery by cycling the three HPI pumps to prevent overheating. Long-term recovery would involve the recovery of HPI-cooling by opening a discharge path (breaking or opening a drain in the 3-inch LPSW discharge line from the HPI pumps would suffice).

Current event frequency:
Research shows that in this situation there is currently a probability of about <u>1 chance in 9</u> of the operating staff failing to recover LPSW to HPI pumps given a failure of a manual valve in the common discharge header. That probability is marked on the scale to the right.

Probability	Chance of Occurrence
1.0	1 chance in 1
.5	1 chance in 2
.2	1 chance in 5
.1	1 chance in 10
.05	1 chance in 20
.02	1 chance in 50
.01	1 chance in 100
.005	1 chance in 200
.002	1 chance in 500
.001	1 chance in 1,000
.0005	1 chance in 2,000
.0002	1 chance in 5,000
.0001	1 chance in 10,000
.00005	1 chance in 20,000
.00002	1 chance in 50,000
.00001	1 chance in 100,000
.000005	1 chance in 200,000
.000002	1 chance in 500,000
.000001	1 chance in 1,000,000
.0000005	1 chance in 2,000,000
.0000002	1 chance in 5,000,000
.0000001	1 chance in 10,000,000

Please put a slash across the scale to indicate the probability with which you think this operator error would occur if NOPs and AOPs were improved as described in How Operating Procedures Could Be Improved.

ATTACHMENT

PLEASE CONSIDER THE FOLLOWING TRANSIENT-INITIATING EVENT

REACTOR/TURBINE TRIP

Base Freq. (X times per year)	Base Freq. (1 time per Y years)
1.0	1 in 1 year
.5	1 in 2 years
.2	1 in 5 years
.1	1 in 10 years
.05	1 in 20 years
.02	1 in 50 years
.01	1 in 100 years
.005	1 in 200 years
.002	1 in 500 years
.001	1 in 1,000 years
.0005	1 in 2,000 years
.0002	1 in 5,000 years
.0001	1 in 10,000 years
.00005	1 in 20,000 years
.00002	1 in 50,000 years
.00001	1 in 100,000 years
.000005	1 in 200,000 years
.000002	1 in 500,000 years
.000001	1 in 1,000,000 years
.0000005	1 in 2,000,000 years
.0000002	1 in 5,000,000 years
.0000001	1 in 10,000,000 years

Event description:
An event resulting in a reactor trip, but not significantly degrading the operability of equipment needed to respond to the event. This transient occurs when a turbine or reactor trip occurs, or if turbine problems occur which in effect decrease the steam flow to the turbine, causing a rapid change in the amount of energy removed from the primary system. This event may be caused by hardware failure or by human error.

Current event frequency:
Research shows that this transient can occur at the reference plant about .28 times per year due to procedure-related operator errors. This is equivalent to once every 3.5 years. The mark on the scale to the left indicates this frequency.

Please put a slash across the frequency scale to indicate the frequency with which you think this transient would occur due to procedure-related operator errors if NOPs and AOPs were improved as described in How Operating Procedures Could Be Improved.

PLEASE CONSIDER THE FOLLOWING
TRANSIENT-INITIATING EVENT

LOSS OF MAIN FEEDWATER

Base Freq. (X times per year)	Base Freq. (1 time per Y years)
1.0	1 in 1 year
.5	1 in 2 years
.2	1 in 5 years
.1	1 in 10 years
.05	1 in 20 years
.02	1 in 50 years
.01	1 in 100 years
.005	1 in 200 years
.002	1 in 500 years
.001	1 in 1,000 years
.0005	1 in 2,000 years
.0002	1 in 5,000 years
.0001	1 in 10,000 years
.00005	1 in 20,000 years
.00002	1 in 50,000 years
.00001	1 in 100,000 years
.000005	1 in 200,000 years
.000002	1 in 500,000 years
.000001	1 in 1,000,000 years
.0000005	1 in 2,000,000 years
.0000002	1 in 5,000,000 years
.0[illegible]	1 in 10,000,000 years

Event description:
An interruption of main-feedwater flow from one or both trains of the system. This event may be caused by hardware failure or human error.

Current event frequency:
Research shows that this transient can occur at the reference plant about <u>.032 times per year due to procedure-related operator errors</u>. This is equivalent to once every 31 years. The mark on the scale to the left indicates this frequency.

Please put a slash across the frequency scale to indicate the frequency with which you think this transient would occur <u>due to procedure-related operator errors</u> if NOPs and AOPs were improved as described in How Operating Procedures Could Be Improved.

PLEASE CONSIDER THE FOLLOWING
TRANSIENT-INITIATING EVENT

LOSS OF CONDENSER VACUUM

Base Freq. (X times per year)	Base Freq. (1 time per Y years)
1.0	1 in 1 year
.5	1 in 2 years
.2	1 in 5 years
.1	1 in 10 years
.05	1 in 20 years
.02	1 in 50 years
.01	1 in 100 years
.005	1 in 200 years
.002	1 in 500 years
.001	1 in 1,000 years
.0005	1 in 2,000 years
.0002	1 in 5,000 years
.0001	1 in 10,000 years
.00005	1 in 20,000 years
.00002	1 in 50,000 years
.00001	1 in 100,000 years
.000005	1 in 200,000 years
.000002	1 in 500,000 years
.000001	1 in 1,000,000 years
.0000005	1 in 2,000,000 years
.0000002	1 in 5,000,000 years
.0000001	1 in 10,000,000 years

Event description:
A reduction of condenser vacuum to a level resulting in a feedwater-pump trip. This event may be caused by hardware failure or by human error.

Current event frequency:
Research shows that this transient can occur at the reference plant about <u>.01 times per year due to procedure-related operator errors</u>. This is equivalent to once every 100 years. The mark on the scale to the left indicates this frequency.

Please put a slash across the frequency scale to indicate the frequency with which you think this transient would occur <u>due to procedure-related operator errors</u> if NOPs and AOPs were improved as described in How Operating Procedures Could Be Improved.

PLEASE CONSIDER THE FOLLOWING TRANSIENT-INITIATING EVENT

SPURIOUS ENGINEERED SAFEGUARDS (ES) ACTUATION SIGNAL

Base Freq. (X times per year)	Base Freq. (1 time per Y years)
1.0	1 in 1 year
.5	1 in 2 years
.2	1 in 5 years
.1	1 in 10 years
.05	1 in 20 years
.02	1 in 50 years
.01	1 in 100 years
.005	1 in 200 years
.002	1 in 500 years
.001	1 in 1,000 years
.0005	1 in 2,000 years
.0002	1 in 5,000 years
.0001	1 in 10,000 years
.00005	1 in 20,000 years
.00002	1 in 50,000 years
.00001	1 in 100,000 years
.000005	1 in 200,000 years
.000002	1 in 500,000 years
.000001	1 in 1,000,000 years
.0000005	1 in 2,000,000 years
.0000002	1 in 5,000,000 years
.0000001	1 in 10,000,000 years

Event description:
The event is a spurious initiation of High Pressure Injection (HPI) flow. This event may be caused by hardware failure or by human error.

Current event frequency:
Research shows that this transient can occur at the reference plant about .0002 times per year due to procedure-related operator errors. This is equivalent to once every 5000 years. The mark on the scale to the left indicates this frequency.

Please put a slash across the frequency scale to indicate the frequency with which you think this transient would occur due to procedure-related operator errors if NOPs and AOPs were improved as described in How Operating Procedures Could Be Improved.

PLEASE CONSIDER THE FOLLOWING
TRANSIENT-INITIATING EVENT

LOSS OF LOW PRESSURE SERVICE WATER

Base Freq. (X times per year)	Base Freq. (1 time per Y years)
1.0	1 in 1 year
.5	1 in 2 years
.2	1 in 5 years
.1	1 in 10 years
.05	1 in 20 years
.02	1 in 50 years
.01	1 in 100 years
.005	1 in 200 years
.002	1 in 500 years
.001	1 in 1,000 years
.0005	1 in 2,000 years
.0002	1 in 5,000 years
.0001	1 in 10,000 years
.00005	1 in 20,000 years
.00002	1 in 50,000 years
.00001	1 in 100,000 years
.000005	1 in 200,000 years
.000002	1 in 500,000 years
.000001	1 in 1,000,000 years
.0000005	1 in 2,000,000 years
.0000002	1 in 5,000,000 years
.0000001	1 in 10,000,000 years

Event description:
This event is a failure of the Low Pressure Service Water system resulting in insufficient flow in the main headers or failure to supply service water to vital equipment. Three potential failure modes have been identified for the LPSW system: (1) a pipe break in the supply header, (2) an operating pump failure followed by failure of the standby pump, and (3) a failure of the main discharge header by blockage. This transient occurs when the service water system fails to perform its function either as a result of hardware failure or human error.

Current event frequency:
Research shows that this transient can occur at the reference plant about <u>.00008 times per year due to procedure-related operator errors</u>. This is equivalent to once every 12,500 years. The mark on the scale to the left indicates this frequency.

Please put a slash across the frequency scale to indicate the frequency with which you think this transient would occur <u>due to procedure-related operator errors</u> if NOPs and AOPs were improved as described in How Operating Procedures Could Be Improved.

APPENDIX D

NUCLEAR POWER PLANT CHARACTERISTICS

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NUCLEAR POWER PLANT CHARACTERISTICS

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 ($\bar{T}$) for groups of plants.

The calculation of the average remaining life of reactors affected by the resolution of a safety issue ($\bar{T}$) 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
3. sum remaining lives in existing plants by assuming a 40-year life and subtracting past service years
4. sum back fit and forward-fit life and divide by the total number of plants.

An estimate of number of plants (N) and average remaining life ($\bar{T}$) in each of the four reactor categories (backfit, forward-fit, BWR, PWR) was completed. The base year is 1989.

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.

The NRC has extended operating plant's licenses to 40 years. This is the maximum allowed by federal law and is figured from the date the Operating License (OL) is issued rather than the date of initial operation. The dates shown for the OL being issued in the attached tables are taken from the January 6, 1986 issue of Inside NRC. The net MWe and the start dates, shown for plants under construction, were taken from the February, 1989 issue of Nuclear News.

TABLE D.1. Type and Lifetimes of U.S. Nuclear Power Plants

Reactor Supplier	Type	No. of Units (N) Completed	Planned	Average Remaining Life (T) (years) Completed	Planned
Combustion Engineering	PWR	15	0	29.9	--
Babcock & Wilcox	PWR	8	2	25.4	40
Westinghouse	PWR	49	6	29.3	40
General Electric	BWR	37	1	28.1	40

	N	T (years)
All PWR	80	30.1
Backfit	72	29.0
Forward-fit	8	40
All BWR	38	28.4
Backfit	37	28.1
Forward-fit	1	40
All Plants	118	29.5
Backfit	109	28.7
Forward-fit	9	40

TABLE D.2. Combustion Engineering Plants

Name	Net MWe	Completed Date OL Issued	Backfit years from 1989
Calvert Cliffs 1	825	7/74	25
Calvert Cliffs 2	825	8/76	27
Maine Yankee	810	72	23
Millstone 2	863	8/75	26
Palisades	780	71	22
Fort Calhoun 1	478	5/73	24
Arkansas Nuc. I-2	858	7/78	29
St. Lucie 1	839	3/76	27
St. Lucie 2	839	4/83	34
Waterford 3	1075	12/84	35
Palo Verde 1	1221	12/84	35
Palo Verde 2	1221	86	37
Palo Verde 3	1221	87	38
San Onofre 2	1070	2/82	33
San Onofre 3	1080	11/82	33

TABLE D.3. Babcock & Wilcox Plants

Completed

Name	Net MWe	Date OL Issued	Backfit years from 1989
Three-Mile Island 1	808	4/74	25
Davis-Besse 1	860	4/77	28
Arkansas Nuc. I-1	836	5/74	25
Oconee 1	846	2/73	24
Oconee 2	846	10/73	24
Oconee 3	846	7/74	25
Crystal River 3	821	12/76	27
Rancho Seco	873	8/74	25

Under Construction

Name	Net MWe	Start Date
Bellefonte 1	1213	1994
Bellefonte 2	1213	1996

TABLE D.4. Westinghouse Plants

Completed

Name	Net MWe	Date OL Issued	Backfit years from 1989
Haddam Neck	565	6/67	18
Indian Point 2	864	10/71	22
Indian Point 3	965	12/75	26
Beaver Valley 1	810	1/76	27
Beaver Valley 2	833	87	38
Salem 1	1106	8/76	27
Salem 2	1106	4/80	31
Robert E. Ginna	470	9/69	20
Yankee	167	7/60	11
Zion 1	1040	4/73	24
Zion 2	1040	11/73	24
Donald C. Cook 1	1020	10/74	25
Donald C. Cook 2	1060	12/77	28
Prairie Island 1	503	8/73	24
Prairie Island 2	500	10/74	25
Point Beach 1	485	10/70	21
Point Beach 2	485	11/71	22
Kewaunee	503	12/73	24
Joseph M. Farley 1	813	6/77	28
Joseph M. Farley 2	823	10/80	31
Robinson 2	665	7/70	21
McGuire 1	1129	1/81	32
McGuire 2	1129	3/83	34
Turkey Point 3	666	7/72	23
Turkey Point 4	666	4/73	24
Sequoyah 1	1148	2/80	31
Sequoyah 2	1148	6/81	32
Surry 1	781	5/72	23
Surry 2	781	1/73	24
North Anna 1	915	11/77	28
North Anna 2	915	4/80	31
Trojan	1095	11/75	26
San Onofre 1	436	3/67	18
Millstone 3	1142	86	37
Seabrook 1	1150	89	40
Byron 1	1105	10/84	35
Byron 2	1105	87	38
Wolf Creek	1128	85	36
Callaway 1	1150	10/84	35
Catawba 1	1129	12/84	35
Catawba 2	1129	86	37
Vogtle 1	1079	87	38
Virgil C. Summer 1	885	8/82	33
Diablo Canyon 1	1073	11/84	35
Diablo Canyon 2	1087	86	37

TABLE D.4. (Contd)

Completed

Name	Net MWe	Date OL Issued	Backfit years from 1989
Shearon Harris 1	860	87	38
Braidwood 1	1120	88	39
Braidwood 2	1120	88	39
South Texas Project 1	1250	88	39

Under Construction

Name	Net MWe	Start Date
Vogtle 2	1079	7/89
Watts Bar 1	1177	92
Watts Bar 2	1177	----
South Texas Project 2	1250	6/89
Comanche Peak 1	1150	12/89
Comanche Peak 2	1150	92

TABLE D.5. General Electric Plants

Completed

Name	Net MWe	Date OL Issued	Backfit years from 1989
Pilgrim 1	670	6/72	23
Oyster Creek 1	620	4/69	20
Nine Mile Point 1	610	8/69	20
Nine Mile Point 2	1045	87	38
Millstone 1	654	10/70	21
Peach Bottom 2	1051	8/73	24
Peach Bottom 2	1035	7/74	25
James A. Fitzpatrick	778	10/74	25
Vermont Yankee	504	3/72	23
Dresden 2	772	12/69	20
Dresden 3	773	1/71	22
Quad-Cities 1	769	10/71	22
Quad-Cities 2	769	3/72	23
Big Rock Point	67	8/62	13
Duane Arnold	515	2/74	25
Cooper	764	1/74	25
Monticello	536	9/70	21
Brunswick 1	790	9/76	27
Brunswick 2	790	12/74	25
Edwin I. Hatch 1	756	8/74	25
Edwin I. Hatch 2	768	6/78	29
Browns Ferry 1	1065	6/73	24
Browns Ferry 2	1065	7/74	25
Browns Ferry 3	1065	7/76	27
Shoreham	809	89	40
Susquehanna 1	1032	7/82	33
Susquehanna 2	1038	3/84	35
Limerick 1	1055	10/84	35
Hope Creek 1	1067	86	37
Lasalle 1	1036	4/82	33
Lasalle 2	1036	12/83	34
Clinton 1	930	87	38
River Bend 1	936	86	37
WNP-2	1095	12/83	34
Grand Gulf 1	1142	6/82	33
Perry 1	1205	87	38
Fermi 2	1093	87	38

Under Construction

Name	Net MWe	Start Date
Limerick 2	1055	10/90

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11. ABSTRACT (200 words or less)

This report documents a value-impact assessment that was undertaken to assist the U.S. Nuclear Regulatory Commission (NRC) in determining whether it should implement regulatory action that would specify requirements for the preparation of acceptable normal and abnormal operating procedures by the NRC's licensee nuclear power plants. The following steps were used in this assessment: 1) the proposed NRC regulatory action was defined as the NRC requiring each U.S. nuclear plant to undertake a program to upgrade its normal and abnormal operating procedures, 2) the attributes effected by this action were identified, and 3) sensitivity analyses were performed to show how changes in important data would affect the expected changes in the attributes. These individual evaluations were then summarized and the value-impact results displayed.

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