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Supplement 3

Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development

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U.S. Nuclear Regulatory Commission

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ABSTRACT

This supplemental report is the fourth in a series that document and use methods developed by the Pacific Northwest Laboratory to calculate, for prioritization purposes, the risk, dose and cost impacts of implementing resolutions to reactor safety issues. The initial report in this series was published by Andrews et al. in 1983 as NUREG/CR-2800. This supplement consists of two parts describing separate research efforts: (1) an alternative human factors methodology approach and (2) a prioritization of the NRC's Human Factors Program Plan. The alternative human factors methodology approach may be used in specific future cases in which the methods identified in the initial report (NUREG/CR-2800) may not adequately assess the proper impact for resolution of new safety issues. The alternative methodology included in this supplement is entitled Methodology for Estimating the Public Risk Reduction Affected by Human Factors Improvement. The prioritization sect.on of this report is entitled Prioritization of the U.S. Nuclear Regulato'y Commission Human Factors Program Plan.

PREFACE

This report was prepared by the Pacific Northwest Laboratory (PNL) to communicate results of the Prioritization of Safety Issues (PSI) Project. An objective of the project is to develop a methodology for use in quantifying risk, dose and cost impacts of resolutions to reactor safety issues and to apply this methodology issues of interest to the NRC. Results of this project will be used by the NRC to support, in part, decisions on resource allocations to resolve specific issues. Prioritization decisions by the NRC are documented in NUREG-0933, A Prioritization of Generic Safety Issues.

This is the fourth in a series of reports from the PSI project. The first report, the initial NUREG/CR-2800, contains a description of the methodology and three example issue analyses. The second report (Supplement 1) contains results in 15 additional issues. The third (Supplement 2) contains results of analyses for 31 more issues.

This document (Supplement 3) consists of two parts. Each part describes the results of a research effort that was undertaken in the human factors area. The first part, entitled Estimating the Public Risk Reduction Affected by Human Factors Improvements, documents efforts to determine if currently used methods for assessing human-factors effects can be improved. The second part, entitled Prioritization of the U.S. Nuclear Regulatory Commission Human Factors Program, summarizes the results of risk and cost analyses conducted by the Pacific Northwest Laboratory in support of efforts for the Human Factors Program Plan.

The following listing identifies issues that were documented in the initial NUREG/CR-2800 report and in the two supplements previously published.

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B-56	Diesel Generator Reliability
I.A.2.2	Training and Qualifications of Operations Personnel

NUREG/CR-2800 (PNL-4297) - SUPPLEMENT 1	
23	Reactor Coolant Pump Seal Failures
B-6	Loads, Load Combinations, Stress Limits
B-10	Behavior of BWR Mark III Containments
B-26	Structural Integrity of Containment Penetrations
B-55	Improved Reliability of Target Rock Safety Relief Valves
B-58	Passive Mechanical Failures
C-8	Main Steam Line Leakage Control Systems

NUREG/CR-2800 (PNL-4297) - SUPPLEMENT 1 (CONTD)

I.A.2.7	Accreditation of Training Institutions
I.C.1(4)	Confirmatory Analysis of Selected Transients
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities
II.C.2	Continuation of Interim Reliability Evaluation Program
II.C.3	Systems Interaction
II.C.4	Reliability Engineering
III.D.3.1	Radiation Protection Plans
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NUREG/CR-2800 (PNL-4297) - SUPPLEMENT 2

15	Radiation Effects on Reactor Vessel Support Structures
A-18	Pipe Rupture Design Criteria
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage
C-11	Assessment of Failure and Reliability of Pumps and Valves
D-1	Advisability of a Seismic Scram--High Trip Level
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I.A.2.6(4)	Long-Term Upgrading of Training and Qualifications (Simulators)
I.A.2.6.(6)	Long-Term Upgrading of Training and Qualifications (Nuclear Power Fundamentals for Operator Training
I.A.3.3	Requirements or Operator Fitness
I.A.3.4	Licensing of Additional Operations Personnel
I.A.4.2	Long-Term Training Simulator Upgrade
I.B.1.1(5-7)	Management for Operations: Organization and Management of Long-Term Improvements

I.C.9	Long-Term Program Plan for Upgrading Procedures
I.D.3	Safety System Status Monitoring
I.D.4	Control Room Design Standard
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II.D.2	Research on Relief and Safety Valve Test Requirements
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II.J.3.1/	Organization and Staffing to Oversee Design and Construction
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EXECUTIVE SUMMARY

Human factors issues have comprised a significant portion of ongoing nuclear power plant safety research at the U.S. Nuclear Regulatory Commission (NRC). Many of these issues were prioritized using methods described in NUREG-0933 (NRC 1983) and NUREG/CR-2800 (Andrews 1983) during FY 1982 (FY82). This document presents the results from two extensive efforts performed by the Pacific Northwest Laboratory (PNL) in the human-factors area. The first report that is presented documents the development of an alternative human factors methodology. This effort was undertaken in response to concerns about the methods and models used in FY82. The second report documents the results of calculations of the risk, dose and cost impacts of implementing the elements of the NRC's Human Factors Program Plan which supports prioritization of the program plan.

ALTERNATIVE HUMAN FACTORS METHODOLOGY

The development of an alternative human factors methodology involved investigation and documentation of four attributes of human factor analyses. These four attributes were: (1) the general guidelines used in FY82 by the individuals on the PNL decision making panel, (2) the impact of using alternate representative plants, (3) human factors modeling related to maintenance and plant availability, and (4) human factors data bases.

The first attribute was addressed by documenting the bases considered by the PNL panel in generating estimates for quantifying impacts of human factors issues. The bases or guidelines are grouped under the following four headings: Plant-Related Guidelines, Human Error Assumptions, Independence, and Cost Guidance.

The second attribute was investigated and documented by comparing Oconee (PWR) results with Calvert Cliffs (PWR) and Grand Gulf (BWR) results, all based on Reactor Safety Studies Methodology Applications Program (RSSMAP) information. It was concluded that the Oconee results can be considered as representative of all PWRs for prioritization purposes. It was also concluded that more representative BWR results can be obtained from actual BWR assessment results rather than from ratioed PWR results for maintenance affected parameters.

The third attribute of investigation involved development and documentation of a new maintenance model and a plant availability model. The new maintenance model involved a division of the maintenance outage into duration and frequency terms. Each of these terms was further divided into human and design contributions. Changes in the human contribution can result in changes in the maintenance outage term. A plant availability model was developed that can be used in calculating the economic benefit related to improved plant performance and reduced downtime for human factors issues. This model used the new maintenance model as a basis for correlating changes in maintenance outage unavailability due to human performance to changes in plant availability.

The final attribute, human factors data bases, was investigated and documented by examining information on available human factors data bases. Additional human factors data do not appear to be readily available beyond those

which are found in NUREG/CR-1278 (Swain and Guttman 1980). Furthermore, the data in NUREG/CR-1278 are not readily applicable to prioritization analyses because of the level of detail available from reactor studies used for the analyses and because of manpower constraints.

One primary concern identified from this effort is that a safety issue resolution needs to be defined more carefully to acknowledge overlaps with other issues.

HUMAN FACTORS PROGRAM PLAN

The efforts made in support of prioritizing the NRC's Human Factors Program Plan involved assessment of the first six of the following seven elements identified in the July 1983 version of NUREG-0985 (NRC 1983):

1) Staffing and Qualification
2) Training
3) Licensing Examinations
4) Procedures
5) Management and Organization
6) Man-Machine Interface
7) Human Reliability

The Human Reliability Data element was considered a licensing issue, to be worked on regardless of safety priority, and thus was not assessed with the other six plan elements. An important guideline that was established by the NRC staff was that the plan elements were to be evaluated without considering the contribution or effect of maintenance. The maintenance issue has been assessed as a separate issue and involved more than just human factor related aspects.

Industry total public risk reductions for implementing the whole HFPP totaled 2.0E+05 man-rem. Elements that exceed 50,000 man-rem include Management and Organization and Training. All of the remaining elements exceeded 5000 man-rem per element.

Costs for implementing the whole HFPP are expected to total $7400M over the remaining lives of 134 plants. Costs for individual elements range from $660M to $2700M. Issues costing less than $1000M include Management and Organization, Man-Machine Interface, and Licensing Examinations. Possible cost savings were not included in these totals.

A ranking of the HFPP elements based on total public risk reductions indicates that the top two items are Management and Organization and Training. The bottom two items are Staffing and Qualifications and Licensing Examination. There is roughly a tie for the middle two items (Procedures, Man-Machine Interface). On an absolute scale as defined in NUREG 0933, the HFPP ranks as a high priority issue based on total risk reduction. On an incremental element basis, Management and Organization and Training would be ranked as high priority elements, with the remaining elements ranked in the low and medium categories.

If cost savings estimated for improvements to plant availability are included in the cost estimates for the HFPP, net cost savings for the entire program would result. However, these estimates are speculative and not easily attributed to specific HFPP elements. For the purposes of this analysis, it was concluded that these cost savings should be considered to the extent that they

provide incentive to improve overall human performance. To obtain the benefits, specific activities need to be defined under each of the HFPP elements with the objective being to improve availability. The element descriptions used for this analysis do not reflect this objective.

In summary, the HFPP has significant risk importance but could incur substantial costs during implementation. By considering overlaps in both the costs and benefits of the individual elements, significant economies of effort could be achieved in both the Industry and NRC costs without foregoing all of the available risk reduction.

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INTRODUCTION

This supplement documents both an alternative method developed by and a prioritization performed by the Pacific Northwest Laboratory (PNL) to provide the NRC Office of Nuclear Reactor Regulation (NRR) with information to use in prioritizing new safety issues related to nuclear power plants. Although the alternative method is not currently being used in the development of prioritization information, it is intended that this method may be utilized as an alternative approach in specific future cases in which the methods identified in the initial NUREG/CR-2800 report are deemed inadequate to adequately assess the proper impact for resolution of these new safety issues. The development of this alternative method and performance of this prioritization represents a significant effort by research staff at PNL.

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

It is felt that the approach used for issue analysis provides adequate information to the NRC for their use in prioritizing issues. It may not be adequate for making decisions or taking regulatory action for specific issues; however, this level of analysis can provide useful perspective in guiding future work.

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

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

PART 1

ESTIMATING THE PUBLIC RISK REDUCTION AFFECTED BY HUMAN FACTOR IMPROVEMENTS

SUMMARY AND CONCLUSIONS

Human factors issues have comprised a significant portion of ongoing nuclear power plant safety research at the U.S. Nuclear Regulatory Commission (NRC). These issues were prioritized using methods described in NUREG-0933 (NRC 1983) and NUREG/CR-2800 (Andrews 1983) during fiscal year 1982 (FY82). Quantification of safety benefits of human factors issues was based primarily on expert judgment; whereas, such judgment was only one aspect used in quantifying other safety issues. In response to concerns about the methods and models used in FY82, four attributes of the human factors analyses were investigated and documented further. These four attributes were: (1) the general guidelines used by individuals on the Pacific Northwest Laboratory (PNL) decision-making panel, (2) the impact of using alternative representative plants, (3) human factors modeling related to maintenance and plant availability, and (4) human factors data bases.

The first attribute was addressed by documenting the bases considered by the PNL panel in generating estimates for quantifying impacts of human-factors issues. The bases or guidelines are grouped under four headings: Plant-Related Guidelines, Human Error Assumptions, Independence, and Cost Guidance.

The second attribute was investigated and documented by comparing Oconee (PWR) results with Calvert Cliffs (PWR) and Grand Gulf (BWR) results, all based on the Reactor Safety Study Methodology Applications Program (RSSMAP). A comparison between the use of Oconee versus Calvert Cliffs as a representative plant shows reasonable correlation. The difference in percentage changes between the Oconee and Calvert Cliffs core-melt frequency and public risk reductions is at most 6 percent. It was concluded that the Oconee results can be considered as representative of all PWRs for prioritization purposes. BWR results obtained directly from Grand Gulf parameters were compared to previous BWR results obtained from ratioing Oconee results. This comparison indicates that more representative BWR results can be obtained from actual BWR assessment results rather than from ratioed PWR results for maintenance affected parameters. The difference in percentage changes between the Oconee ratioed and Grand Gulf actual results is at least 17 percent and at most 62 percent.

The third attribute of investigation involved development and documentation of a new maintenance model and a plant availability model. A more accurate maintenance model was developed, but its use in reassessing FY82 maintenance issues is not expected to significantly change the issue results. This maintenance model involved a division of the maintenance outage into duration and frequency terms. Each of these terms was further divided into human and design contributions. Changes in the human contribution can result in changes in the maintenance outage term. A plant availability model was developed that can be used for calculating the economic benefit related to improved plant performance and reduced downtime for human factors issues. This model used the new maintenance model as a basis for correlating changes in maintenance outage unavailability caused by human performance to changes in plant availability.

The final attribute, human factors data bases, was investigated and documented by examining information on available human factors data bases. Additional human factors data do not appear to be readily available beyond those

which are found in NUREG/CR-1278 (Swain and Guttman 1980). Furthermore, the data in NUREG/CR-1278 are not readily applicable to prioritization analyses because the level of detail of the reactor studies used for the analyses and manpower constraints.

In conclusion, the methods applied to prioritization of human factors safety issues in FY82 need little revision. Additional parameters can be considered to enhance the comparablity of these issues with hardware-type issues. BWR impacts can be considered separately to refine results. Available data do not warrant development of a more structured approach. Of primary concern for future issues is the definition of the safety issue resolution to acknowledge overlaps with other issues. Minor revisions to existing analyses may be needed to make them comparable to those analyzed in the future using more refined methods. This need will be determined by the significance of the difference in results obtained when more refined methods are used.

1.0 INTRODUCTION

The Prioritization of Safety Issues (PSI) project is being conducted by the Pacific Northwest Laboratory (PNL) for the Safety Program Evaluation Branch (SPEB) of NRC. The goal of this project is to provide risk and cost information to support NRC priority decisions on research and development for safety issues. During FY82, a number of human factors related reactor safety issues were analyzed as part of the PSI project. The following issues were analyzed:

- I.A.2.2, Training and Qualifications of Operating Personnel
- I.A.2.6(1,2,3,5), Long-Term Upgrading of Training and Qualifications
- I.A.2.6(4), Long-Term Upgrading of Training and Qualifications (Operator Workshops)
- I.A.2.7, Accreditation of Training Institutions
- I.A.3.3, Requirements for Operator Fitness
- I.A.3.4, Licensing of Additional Operations Personnel
- I.A.4.2, Long-Term Training Simulator Upgrade
- I.C.9, Long-Term Program for Upgrading of Procedures
- I.D.3, Safety System Status Monitoring
- I.D.4, Control Room Design Standard
- I.D.5(3,4,5), Improved Control Room Instrumentation Research

Because of time and funding limitations and the relative lack of specific and applicable data, these human factors issues were largely assessed by a panel of PNL experts using an informal Delphi technique. Estimates of reduction in human error were introduced into the dominant accident sequences of the Oconee Reactor Safety Study Methodology Applications Program (RSSMAP) study (Kolb et al. 1981) to calculate the risk reduction.

While this assessment technique was the best method available within the limitations of the FY82 effort, four areas of concern were identified. First, because the method employed individual judgments, substantial variability could occur if a different panel were used to assess future human factors issues. Second, the Oconee RSSMAP assessment was assumed to have generic applicability. Third, the human factors model itself was simplified. Finally, the need to determine the availability of human factors data was identified.

This study was performed to address those four areas. The concern about the possible variability of a panel's judgments is addressed by documenting the bases of the PNL panel's decisions. These bases are listed and explained in Section 2.0 for use on future issue analyses and reviews. Appendix B presents a list of the names of the PNL panel. The concern about the generic

applicability of the Oconee RSSMAP assessment was examined by comparing the Oconee RSSMAP results with two other reactor RSSMAP results. These comparisons are discussed in Section 3.0. Appendix A provides the risk parameters for Calvert Cliffs #2 (PWR). The human factors model was enhanced with an improved maintenance model and a plant availability model. The maintenance model and plant availability model are explained in Section 4.0. The availability of human factors data was investigated, and a discussion of human factors data bases is presented in Section 5.0.

2.0 BASES FOR PANEL DECISIONS ON HUMAN FACTORS ISSUES

A number of the safety issues examined by PNL as part of the PSI project were related to human factors. Because of time, funding and data limitations, these human factors issues were largely assessed by a panel of PNL experts using an informal Delphi technique. The panel consisted of four staff members, whose names are given in Appendix B. Also included in Appendix B are the names of those PNL risk analysis staff members who implemented the panel results to calculate risk reductions. In order to assist others who would seek to either carry on the assessment of human factors related issues or better understand the issues assessed by PNL, a set of interim assessment guidelines has been developed. These guidelines are intended to show some of the basic assumptions of the PNL panel and to provide bounds and general assistance in the treatment of human factors issues. If future analysts have access to better information, these bases can be easily modified.

A number of issues falling under the heading of "human factors" involve the collection of data. While these activities have value, they have no direct safety benefit. It would be inappropriate to compare them directly with other safety issues that would have a more immediate safety benefit. Therefore, such data-gathering tasks have been excluded from the PSI project with the rationale that no safety benefit is incurred in data collection. Issues that were excluded for this reason involve the following:

- I.E.3, Analysis and Dissemination of Operator Experience (Operator Safety Data Analysis)
- I.E.6, Analysis and Dissemination of Operator Experience (Reporting Requirements)
- I.E.7, Analysis and Dissemination of Operator Experience (Foreign Sources)
- I.E.8, Analysis and Dissemination of Operator Experience (Human Error Rate Analysis)

Only as the information is used is the benefit realized. If the potential use of the data can be associated with another issue, then reference should be made to the data-collection issue. Potential use of the data should be considered in the evaluation of the human factors risk reduction in the issue where the data will actually be used.

The PNL panel's assessment guidelines are listed and discussed in the following subsections. These guidelines are grouped under four headings: Plant-Related Guidelines, Human Error Assumptions, Independence, and Cost Guidance. Plant-Related Guidelines extend to two areas: status of existing nuclear power plants and plants that are applicable. The Human Error Assumptions concern four areas: affected staff, maintenance treatment, limitation on human error reduction, and complexity of control rooms.

2.1 PLANT-RELATED GUIDELINES

In general, the human factors operational capability at nuclear power plants is good. Based on the NRC's Systematic Assessment of Licensee Performance (SALP) reports, there is a distinct variation among facilities' management programs from a safety standpoint. Some have outstanding programs, some have average programs, and some have programs that are below average. This does not imply inadequate plant safety. Rather, it reflects the high standards that are applied to nuclear power plant safety. While the performance of average and even below-average facilities may be acceptable, it may be desirable to bring their performance up to the level of the outstanding facilities.

In general, the FY82 safety issue assessments did not explicitly account for variations in different facility performance. Rather, estimates for human error rate reduction or cost impact were intended to be representative of the entire industry. In these generic estimates, the individual facility variations were intuitively balanced. There were two exceptions to this general practice. For Issue I.A.2.2, Training and Qualifications, which was an example given in NUREG/CR-2800 (Andrews et al. 1983), plants were explicitly divided into three categories. Each category was independently assessed and a mathematical average calculated. For Issue I.A.2.7, Accreditation of Training Institutions, a number of plants were considered exempt because their training practice would be expected to closely approach or meet accreditation standards.

It was generally assumed that human factors issues would affect all plants, including both those currently operating and those planned. The only exceptions were cases where the issue resolution specifically excluded a class or classes of plants, or where the resolution was currently being effectively implemented for a class or classes of plants.

2.2 HUMAN ERROR ASSUMPTIONS

Issues involving human factors can affect any and all members of the operating staff, including managers, engineers, operators, technicians, and maintenance personnel. However, the Oconee RSSMAP study has no terms that directly treat any operational staff except operators. Since the Oconee RSSMAP study was, for the most part, the risk estimation model, operator error rates had to be the principal avenue for estimating human factors risk reduction. Therefore, for issues that affected personnel other than operators and/or maintenance staff, the effect on plant safety was represented by including affected personnel estimated error reduction in the representative operator and/or maintenance staff error reduction. For example, licensing non-operator operating personnel was assumed to have both direct and indirect effects on operators. Licensing of managers and engineers was assumed to result in better performance of their jobs. This would enhance plant safety by also helping operators perform their functions better. The effect on plant safety was estimated as a change in operator and/or maintenance staff error rate.

The Oconee RSSMAP allows an indirect treatment of maintenance errors. Estimates of component unavailability in the Oconee RSSMAP included downtime due to maintenance. The PNL panel was aware that maintenance outages consist of

human- and non-human-related factors contributing to downtime. But a conservative, simplistic approach was taken by applying the fractional improvement estimated for maintenance staff performance to the entire maintenance outage term. Due to the limitations of the FY82 effort, this was the best treatment available. Efforts to improve the treatment of maintenance errors are discussed in Section 4.0.

Intuitively, there is a limit to human error rate reduction. Perfection is not a human characteristic. A bound on the limiting reduction can be useful in estimating the effects of a single safety issue or any number of issues in combination. The PNL panel was asked to develop an estimate of the limiting value of the total human error reduction for any issue. The consensus was a value on the order of 50 percent; i.e., the current error rate could potentially be halved. Reductions beyond this were judged unlikely. It is quite important to emphasize that the 50 percent value is a subjective judgment. Stating the value does, however, demonstrate the range of improvements that the panel felt were reasonably achievable.

It is widely felt that nuclear power plant control rooms are very complex and not optimally designed from a human factors viewpoint. The PNL panel assumed that the addition of any new control room equipment without a concurrent control room redesign would have limited value. The potential benefit of new equipment would tend to be cancelled by its addition to the complexity of the control room. For example, evaluation of Issue I.D.3, Safety System Status Monitoring, resulted in an estimated reduction in human error of 2 percent, which was dependent on an assumed concurrent control room redesign (Issue I.D.1).

The area of diagnostics is closely related to the complexity of control rooms. A procedural, training, or hardware change that would provide operators with enhanced diagnostic capabilities was given extra consideration when human error reductions were estimated. While a hardware change that enhances diagnostic capabilities could be considered as adding new equipment in the control room, thus requiring control room redesign to be most effective, this enhancement was assumed to result in an independent human error reduction. Diagnostic equipment was viewed as becoming a primary source of operator information rather than just more monitoring equipment. For example, in Issue I.D.5(3,4,5), Improved Control Room Instrumentation Research, part of the resolution was to add an advanced diagnostic system such as the Safety Parameter Display System (SPDS). It was recognized that the effectiveness of the advanced diagnostic system in aiding control room operators would be greater if implemented with a complete control room redesign. However, while the resolution did not involve control room redesign, it still was assumed to result in an estimated human error reduction of 2 percent.

2.3 INDEPENDENCE

Most safety issues assessed by PNL during FY82 were assumed to be independent of other issue resolutions. Each issue was examined with the question, "What effects will the resolution of this single issue have, based upon the current, relevant conditions?" Many improvements have been made in the

human factors area since the development of the Three-Mile Island (TMI) Action Plan. The PNL panel felt that, in some cases, the improvements mitigated the need for the regulatory change indicated by a safety issue. Clearly the assessment of each new safety issue must independently consider this mitigating effect, but its overall importance should be noted. The benefit to be derived from a safety issue resolution should be its change from the current conditions. The interplay and overlap of other issues, whether related to human or other factors, were generally not considered. In some instances, overlaps were identified, but they did not enter into the numerical estimates and thus contributed to double counting of the benefits and costs for some issues.

If an accounting of interdependence is to be required in future analyses, several approaches are possible. First, with a known group of human-factors issues, the interdependence of the issue resolutions could be evaluated. Given one or more projected sequences for implementation of resolutions to selected human-factors issues, the PNL expert panel could estimate the incremental benefit (in terms of human error rate reductions) attainable for each resolution. The panel would assume some maximum overall benefit attainable by implementation of all selected issue resolutions. Each projected implementation sequence would then be limited to this maximum. Second, a simpler (and perhaps less accurate) technique would employ Boolean or logic as a mathematical tool to adjust for double counting from issues with overlapping resolutions. Again, some maximum overall benefit would be assumed. The percent reductions in human error rates could then be combined as a Boolean or equation to calculate the total reduction. For example, assume that the human error rate reductions associated with issues A and B are 20 percent and 30 percent, respectively, of the attainable overall maximum. The total reduction from both would become

$$0.20 + 0.30 - (0.20)(0.30) = 0.44,$$

or 44 percent of the maximum using the Boolean "or" combination. If the total maximum reduction attainable was 50 percent, then these two issues would result in 0.44(0.50) = 0.22, or 22 percent, reduction in the human error rate. The use of the Boolean "or" combinational technique ensures that, no matter how many issues are credited, the total reduction is limited to the maximum attainable. In each of these techniques, care must be taken to ensure that variations in the affected risk equation parameters and affected plants from issue to issue are accounted for consistently.

2.4 COST GUIDANCE

Development of cost estimates requires familiarity with the specific programs. Therefore, it is difficult to provide generic guidance. A few observations can be made, however. First, NRC costs for development and implementation vary widely for human factors issues. However, for many human factors issues, the NRC operational cost is estimated to be relatively small. For these issues, the NRC operational effort consists only of the Office of Inspection and Enforcement review to ensure that the training programs, procedures, or other changes are in effect. It is estimated that only a few person-weeks per plant-year (py) are required.

Human factors issues do not generally require significant investments in hardware. One major exception is the purchase of plant-specific simulators at an industry cost of $1 billion, as considered in Issue I.A.4.2, Long-Term

Training Simulator Upgrade. Many other resolutions of human factors issues are manpower intensive. At the assumed $2270 per person-week, such labor-intensive programs can quickly become expensive. For example, Issues I.A.3.3, Requirements for Operator Fitness, and I.A.2.6(1,2,3,5), Long-Term Upgrading of Training and Qualification, involve industry costs of $1.2 billion and $2.1 billion, respectively. Over 70 percent of these industry costs are labor costs.

Cost estimates for the FY82 issues did not involve detailed job-task analysis to establish incremental staffing levels. However, estimates of additional staffing for issue resolutions were generally made on a conservative basis. Most issues did not include any costs related to plant availability, except for Issue I.D.5(3,4,5), Improved Control Room Instrumentation Research. This issue included cost savings resulting from reduced unscheduled outages, which were, in turn, related to reduced transient shutdowns.

3.0 COMPARISONS OF ADDITIONAL PROBABILISTIC RISK ASSESSMENTS (PRAs)

During the FY82 effort on the PSI project, a number of human-factors related issues dealing with maintenance outage and operator error were analyzed. Due to the time and funding limitations, one nuclear plant (Oconee 3 PWR) was chosen as the representative light water reactor for most issues. The Oconee RSSMAP assessment (Kolb et al. 1981) was the basis for the FY82 analyses. Reductions in maintenance outage unavailability or operator error probability were estimated and then propagated through the Oconee RSSMAP dominant accident sequences to develop an estimation of risk reduction. For most issues where risk reductions were to be applied to BWRs, the results of the Oconee (PWR) analysis were ratioed by dividing the Grand Gulf 1 (BWR) total public risk by the Oconee total public risk.

The questions of the generic applicability of the Oconee 3 RSSMAP study and whether ratioing the PWR risk reduction results reasonably represents BWR results are examined in this section. The analysis was done by examining two other RSSMAP PRAs. The Calvert Cliffs RSSMAP study (Hatch et al. 1982) was examined as a comparison of PWR results. The Grand Gulf RSSMAP study (Hatch et al. 1981) was selected for comparing ratio-derived BWR results with actual BWR-assessed results. The following sections present the results of the considerations of these two nuclear plants (Calvert Cliffs and Grand Gulf) as related to maintenance outage, Section 3.1, and operator error, Section 3.2.

3.1 MAINTENANCE OUTAGE

"Maintenance outage," as defined for the FY82 analyses, is the unavailability of a component due to maintenance operations while the reactor is at power. For purposes of analysis, the maintenance outage was assumed to consist of unavoidable downtime due to component cooldown and repair procedures and avoidable downtime due to inefficiencies and/or errors attributable to the maintenance staff. However, when postulating the effects of issue resolutions upon maintenance outages, the PNL panel estimates of the reduction in human error were conservatively applied to the entire maintenance outage. This same approach was followed in analyzing the Calvert Cliffs and Grand Gulf reactors. The results of the maintenance outage reduction analyses for Calvert Cliffs and Grand Gulf are summarized in Table 3.1 and shown graphically in Figures 3.1 and 3.2.

Table 3.1 shows the changes in core-melt frequency and public risk resulting from three different percentage reductions (10%, 30%, 50%) in unavailability due to maintenance outages. These reductions are sensitivity case examples. The change is expressed as an absolute (per py, core-melt frequency or man-rem/py, public risk) or percentage change from the base case. The base-case core-melt frequency and base-case public risk are given for each of the three reactors considered (Oconee-B&W PWR; Calvert Cliffs-CE PWR; Grand Gulf-GE BWR). The absolute changes in core-melt frequency and public risk are shown as a basis for the calculation of the percentage change in base-case numbers. The base-case and absolute changes in core-melt frequency and public risk for the three reactors show little correlation. The percentage changes in core-melt frequency and public risk were used as a normalized basis for comparison.

TABLE 3.1 Change in Core-Melt Frequency and Public Risk Due to Reduction in Maintenance Outage (MO)

Reactor	Base Case Core-Melt Frequency per py	Change in Core-Melt Frequency for Given MO Reduction					
		10%	30%	50%	10%	30%	50%
		Frequency Change per py			% Change of Base Case		
Oconee	8.2E-05	3.8E-06	1.1E-05	1.6E-05	4.6	13	20
Calvert Cliffs	2.0E-03	1.2E-05	1.6E-04	3.E-04	0.60	8.0	15
Grand Gulf	3.7E-05	4.0E-06	1.2E-05	1.7E-05	11	32	46

Reactor Case	Base Case Public Risk, man-rem/py	Change in Public Risk for Given MO Reduction					
		10%	30%	50%	10%	30%	50%
		Public Risk Change, man-rem/py			% Change of Base		
Oconee	2.1E+02	9.7	2.6E+01	3.9E+01	4.6	12	19
Calvert Cliffs	7.6E+03	4.3E+01	5.8E+02	1.1E+03	0.57	7.6	14
Grand Gulf	2.5E+02	2.7E+01	8.0E+01	1.2E+02	11	32	48

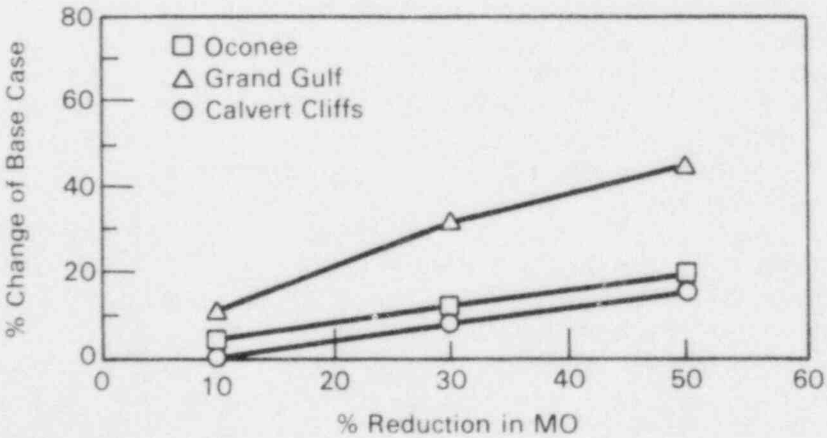


FIGURE 3.1 Percentage Change in Core Melt Frequency with Reduction in Maintenance Outage (MO)

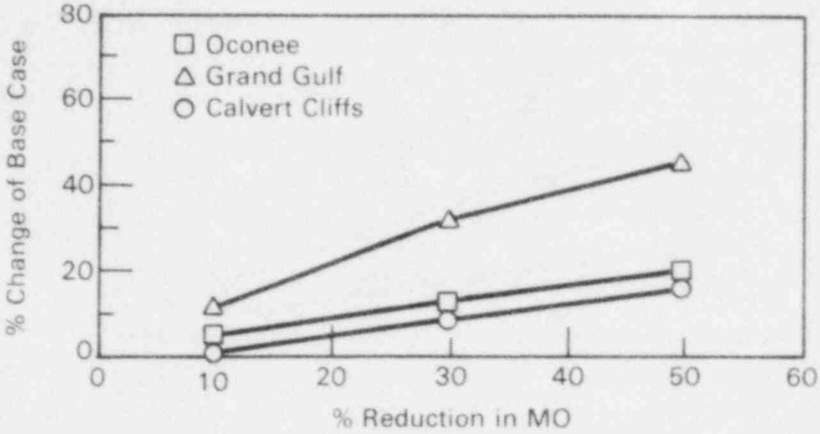


FIGURE 3.2 Percentage Change in Public Risk with Reduction in Maintenance Outage (MO)

Since representative dose factors from NUREG/CR-2800 (Andrews et al. 1983) were used, the percentage changes in base-case core-melt frequency and base-case public risk show close correlation--within 1 percent. The percentage changes in base-case core-melt frequency and public risk between Oconee and Calvert Cliffs vary with the percentage change in maintenance outage. Oconee shows a larger percentage change than Calvert Cliffs for a 10 percent reduction in maintenance outage, a slightly lower percentage change for a 30 percent reduction, and a lesser change (roughly 0.6 times that of Calvert Cliffs) for a 50 percent reduction. Overall the difference between the Oconee and Calvert Cliffs results is at most 5 percent. This indicates that either PWR would have given approximately the same relative measure of change in core-melt frequency or public risk if used consistently for a number of issues.

For Grand Gulf (BWR), the percentage change in core-melt frequency and public risk from the base case is roughly seven times that of Oconee (PWR). Overall the difference between the Oconee and Grand Gulf results is at least 6 percent and at most 62 percent. Maintenance would appear to be more important in BWRs than PWRs, although additional PRAs should be evaluated before drawing a definite conclusion. All of the human factors issues analyzed in NUREG-0933 (NRC 1983) were evaluated, and no changes in priority ranking resulted from using the actual Grand Gulf maintenance results. The numerical difference in public risk reduction between the ratioed PWR technique and the application of the actual Grand Gulf results is illustrated in an example at the end of Section 3.2.

3.2 OPERATOR ERROR

"Operator error" is defined in this report as an error made by the operations personnel during all operations not designated as maintenance or testing. These operator errors may occur inside or outside the control room. The results of the analyses for Calvert Cliffs and Grand Gulf relative to operator error reduction are summarized in Table 3.2 and shown graphically in Figures 3.3 and 3.4.

TABLE 3.2 Changes in Core-Melt Frequency and Public Risk Due to Reduction in Operator Error (OE)

Reactor	Base-Case Core-Melt Frequency per py	Change in Core-Melt Frequency for Given OE Reduction: 10% Frequency Change per py	30%	50%	10% % Change of Base Case	30%	50%
Oconee	8.2E-05	3.7E-06	1.1E-05	1.9E-05	4.5	13	23
Calvert Cliffs	2.0E-03	6.1E-05	1.8E-04	3.6E-04	3.1	9.0	18
Grand Gulf	3.7E-05	2.7E-06	4.0E-06	5.8E-06	7.3	11	16

Reactor	Base-Case Public Risk, man-rem/py	Change in Public Risk for Given OE Reduction: 10% Public Risk Change, man-rem/py	30%	50%	10% % Change of Base Case	30%	50%
Oconee	2.1E+02	9.1	2.8E+01	4.7E+01	4.3	13	22
Calvert Cliffs	7.6E+03	2.3E+02	6.8E+02	1.3E+03	3.0	8.9	17
Grand Gulf	2.5E+02	1.9E+01	2.7E+01	3.9E+01	7.6	11	16

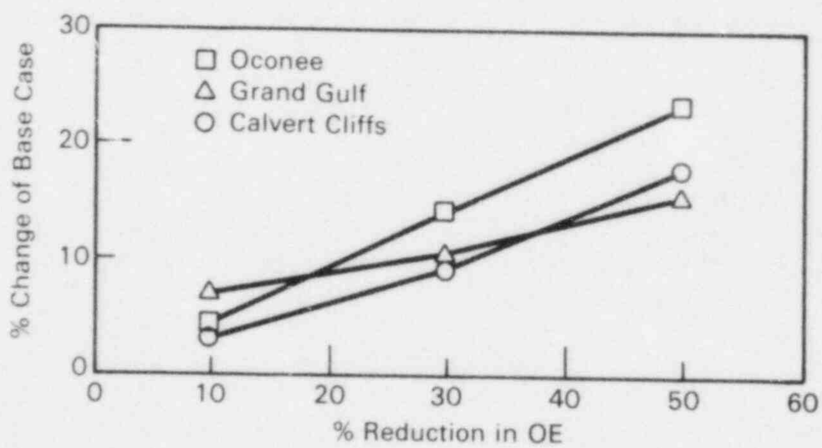


FIGURE 3.3 Percentage Change in Core-Melt Frequency with Reduction in Operator Error (OE)

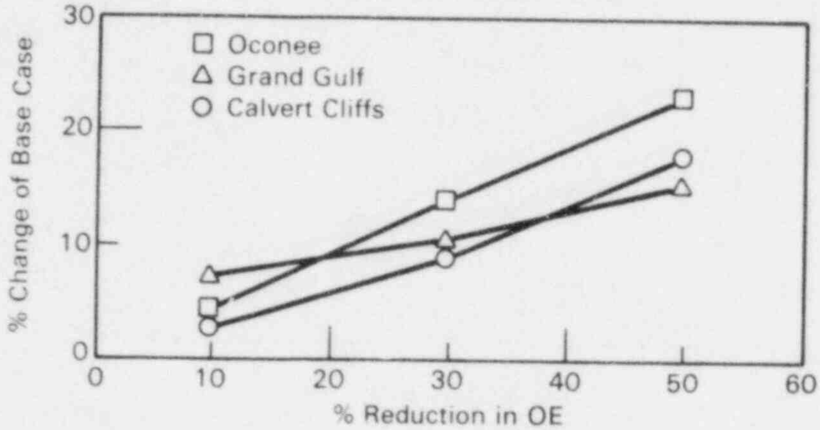


FIGURE 3.4 Percentage Change in Public Risk with Reduction in Operator Error (OE)

Table 3.2 shows the changes in core-melt frequency and public risk resulting from three different percentage reductions (10%, 30%, 50%) in operator error probabilities. These reductions are sensitivity case examples. Similar to the comparison for maintenance outage, the change in core-melt frequency and public risk is expressed either as an absolute change (per py, core-melt frequency or man-rem/py, public risk) or as a percentage change from the base case. The base-case core-melt frequency and base-case public risk are given for each of the three reactors considered (Oconee-B&W PWR; Calvert Cliffs-CE PWR; Grand Gulf-GE BWR). The absolute changes in core-melt frequency and public risk are shown as a basis for the calculation of the percentage change in base-case numbers. The base-case and absolute changes in core-melt frequency and public risk for the three reactors show little correlation. The percentage changes in core-melt frequency and public risk were used as a normalized basis for comparison.

Since representative dose factors from NUREG/CR-2800 (Andrews et al. 1983) were used, the percentage changes in base-case core-melt frequency and base-case public risk show close correlation--within 1 percent. The percentage change of base-case core-melt frequency or public risk for Calvert Cliffs is within 20 to 40 percent of that for Oconee. Overall the difference between the Oconee and Calvert Cliffs results is at most 6 percent. Therefore, either PWR would have given approximately the same relative measure of change in core-melt frequency or public risk if used consistently for a number of issues.

For comparison between Oconee (PWR) and Grand Gulf (BWR), the percentage change of base-case core-melt frequency or public risk for Grand Gulf is found to lie within ± 25% to 65% of Oconee. Grand Gulf shows a larger percentage change of base-case than Oconee at 10 percent operator error reduction and a

smaller percentage change at 50 percent reduction. Overall, the difference between the Oconee and Grand Gulf results is at most 8 percent. The operator-only human factors issues [I.A.3.3; I.D.3; I.A.2.6(4); I.A.2.7; I.A.4.2; I.A.2.6(1,2,3,5)] were evaluated, and no changes in priority ranking resulted from using the actual Grand Gulf operator results. Issue I.C.9, involving both operator and maintenance parameters, was evaluated to illustrate the numerical difference in public risk reduction between the ratioed PWR technique and application of the actual Grand Gulf results. With the ratioed PWR technique, the public risk reduction for I.C.9 is 2.1E+05 man-rem, and with the actual Grand Gulf results, an upper-bound reduction is 3.0E+05 man-rem.

3.3 RECOMMENDATIONS

PNL recommends that for future issues involving maintenance outage or operator error reductions Oconee calculations should be used for PWR results and Grand Gulf calculations should be used for BWR results. References can be made to Figures 3.1 through 3.4 for preliminary estimates of percentage changes in base-case core-melt frequency or public risk.

4.0 MAINTENANCE AND AVAILABILITY MODELS

In the analysis of safety issues performed by PNL for the NRC (NUREG/CR-2800)(Andrews et al. 1983), it has been assumed that a reduction in error rates for plant operators or improved performance during maintenance outages can lower the occurrence rate of accidents leading to core melt. Variables associated with these two factors in representative plant risk equations were identified and estimated changes made, depending on the specific safety issue being examined and its proposed resolution.

Two general areas were identified where potential improvements to that analysis could be made. The first concerned a better understanding of what constituted the "maintenance outage" term. The second dealt with the goal to quantify the economic benefits of increased plant availability associated with reduced human error. A potential approach to quantifying these parameters more effectively is presented in this section. As of this report, these approaches have not been used to reassess previously analyzed human factors issues.

4.1 MAINTENANCE OUTAGE

The purpose of this task is to further quantify the factors which constitute the maintenance outage term. The specific goal is to quantify that portion of the term directly related to human performance.

The risk equations for the Oconee PWR (Kolb et al. 1981) and the Grand Gulf BWR (Hatch et al. 1981) used essentially the same assumptions and data as the original Reactor Safety Study (WASH-1400)(NRC 1975). Plant-specific information on maintenance frequency and duration was used when available.

Maintenance outage terms as used in the risk equations were originally defined in Section 5 of Appendix III of WASH-1400. It was noted that certain testing and maintenance acts cause the effective removal of a component from a system, rendering it unavailable for some period of time. This unavailability is then a function of the duration of time required for the act of maintenance or testing, and the frequency of the act. Unavailability can be expressed as

$$Q = [f(\text{avg acts/month}) \times t(\text{avg hrs/act})]/720 \text{ hrs/month}$$

where q is the expression of unavailability, f is the frequency and t is the duration time (downtime) for the repair.

Four major classes of components were considered in WASH-1400: pumps, valves, diesels, and instruments. It was recognized that the testing frequency would vary for components, ranging from every month for safety-related equipment to yearly for other components. A lognormal distribution was assumed, giving a mean interval of 4.6 months between test or outage, or a frequency of 0.22 outages per month.

An examination of the data indicated that the outage duration for components also fit the lognormal distribution assumption. The mean duration times for outages by equipment class are shown in Table 4.1, along with the calculated unavailabilities.

TABLE 4.1. Component Unavailabilities from WASH-1400

Component	Unavailability	Outage Frequency (per/mo)	Act Duration (hr)
Pumps (24 hr)	2E-03	0.22	7
Pumps (72 hr)	6E-03	0.22	19
Valves	2E-03	0.22	7
Diesels	6E-03	0.22	21
Instruments	2E-03	0.22	6

It was pointed out that the plants' technical specifications in many cases restrict the time during which a component can be down for maintenance while the plant remains in operation. Certain pumps are limited to 24 hr, while others have a 72-hr repair period. A maximum bound can then be defined for the unavailability, since plant shutdown or placement in hot shutdown may be required beyond that point.

4.1.1 Constituents of Maintenance Outage

The question remains as to what factors contribute to the frequency and duration of maintenance outages. WASH-1400 mentions several factors, including the component involved, the complexity of the maintenance, the magnitude of the repair, and contingencies which arise. However, these factors are not quantified in WASH-1400. Note that some of these factors will be more closely related to equipment design. The scope of this study was to include only those factors of human performance or human interface in the maintenance operation which would be subject to improvement. The optimization of equipment design and the work environment for maintenance is an obvious program objective in many safety issues related to maintenance upgrade, but design modifications were specifically deleted here as being outside the scope of this study.

To see how the subject of human performance fits with other factors in this area, a current summary of ongoing research related to nuclear plant maintenance is given in Table 4.2 (Badalamente et al. 1982). Of the areas listed in Table 4.2, all deal with human performance to some degree, with the exception of Design for Maintainability and Inventory Control. Preventive Maintenance has some performance review associated with it, but it is also primarily machine oriented. To further illustrate design features desirable from a human engineering standpoint which are outside the scope of this issue, a summary of those attributes most often cited by maintenance personnel is given in Table 4.3 (Seminara et al. 1980). The Seminara EPRI study, entitled A Human Factors Review of Power Plant Maintainability, was an extensive review of difficulties found in operating power plants. (Note that Table 4.3 does include good manuals for procedures and instructions, which would be considered here to be human-performance related.) However, the conclusions and recommendations from the study are qualitative in nature, and no attempt was made to estimate the potential for improving performance in any of these areas.

TABLE 4.2. Summary of Approach to Improving NPP Maintenance

MAINTENANCE IMPROVEMENT OBJECTIVES	IMPROVEMENT STRATEGIES	PRIORITY	ELEMENT*	OTHER INVOLVEMENT
Design for Maintainability				
a. Assure design-induced error is minimized by application of human engineering design principles	• Develop a comprehensive NNP-MDG	1a	MMI	• EPRI (Maintainability Self-Improvement Guide) • EPRI (Anthropometric Data Base) • EPRI (Cool Suits)
	• Develop an interim retrofit MDG	1	MMI	
	• Amend 10CFR50, App. A&B	1	MMI	
b. Assure maintenance facilities are suitable	• Expand Reg. Guide 1.70.6	1a	MMI	
	• Revise Reg. Guide 1.64 (Develop new ANSI/ASME Std)	1a	MMI	
	• Evaluate maintainability issue under TAP Item IV.F	1a	MMI	
	• Develop comprehensive Reg. Guide on Maintenance	1a.	MMI	
	• Develop anthropometric data base	1a	MMI	
Maintenance Procedures and Technical Information				
a. Assure use of maintenance procedures by all maintenance specialties; and require application of human factors principles to preparation of procedures	• Develop Maintenance Procedures Guidelines	1	PT	• EPRI (JPA Study) • INPO (Procedures Study) • INPO (Writer's Guide)
	• Examine JPAs	2	PT	
	• Expand SRP to address vendor documentation	1a	PT	
	• Increase I&E emphasis on procedures	1	PT	
b. Assure availability of adequate vendor technical information				
c. Assure maintenance procedures, vendor manuals, and reference materials are technically accurate and up-to-date				

TABLE 4.2. (contd)

MAINTENANCE IMPROVEMENT OBJECTIVES	IMPROVEMENT STRATEGIES	PRIORITY	ELEMENT*	OTHER INVOLVEMENT
Selection and Training of Maintenance Personnel				
Assure maintenance personnel are adequately qualified to perform maintenance tasks reliably; minimizing errors which may degrade public health and safety	• Develop maintenance personnel selection and training guidelines	1	T	• INPO (Job and Task Analysis) • INPO (Qualification Guidelines • ISA (I&C Tech Guide)
	• Develop maintenance personnel qualification requirements for inclusion in Reg. Guide 1.8	2	SQ	
Staffing				
Assure maintenance and maintenance engineering/technical support functions are staffed with sufficient numbers of qualified people to accomplish required maintenance tasks safely	• Establish realistic rule on overtime	1	SQ	• INPO (Survey) • IBEW (Guide)
	• Develop guidelines for minimum around-the-clock manning	2	SQ	
	• Develop guidelines for manning maintenance functions	3	SQ	
Preventive Maintenance				
Assure realization of inherent safety and reliability levels of all safety-related equipment and restoration of safety and reliability when deterioration has occurred, by requiring and promoting adequate preventive maintenance programs/practices	• Develop guidelines for PM program	1a	MMI	• INPO (RCM) • EPRI (PM Model)
	• Reexamine ST/II requirements	2	PT	
	• Examine on-line dianostic monitoring	2	MMI	
Maintenance Control				
Assure maintenance is conducted in a way which minimizes probability of error due to inadequate job planning information and/or insufficient provisions for work authorization and control	• Increase I&E emphasis on work authorization and control practices	1	PT	
	• Review need for development of maintenance control guidelines	3	PT	

TABLE 4.2. (contd)

MAINTENANCE IMPROVEMENT OBJECTIVES	IMPROVEMENT STRATEGIES	PRIORITY	ELEMENT*	OTHER INVOLVEMENT
Outage Planning and Management				
Assure maintenance work performed during outages is accomplished safely and reliably	• Examine issue of outside personnel qualification and training under 4.2.3.3	2	SQ	• EPRI (Improvement Studies)
	• Develop guidelines for outage planning and management	3	PT	
Inventory Control				
Assure proper replacement components, spare parts, maintenance equipment and tools are available	• Monitor developments in inventory control practices	3	MMI	• INPO, EPRI, Owner's Groups
Management				
Assure maintenance programs which promote safe plant operations are planned for, implemented, allocated sufficient resources, supported and guided by utility and plant managers	• Address maintenance management issues in revision to NuReg-0731	3	OM	• INPO (Corporate Evaluations)
	• Examine Organization and Management Performance Standards		OM	

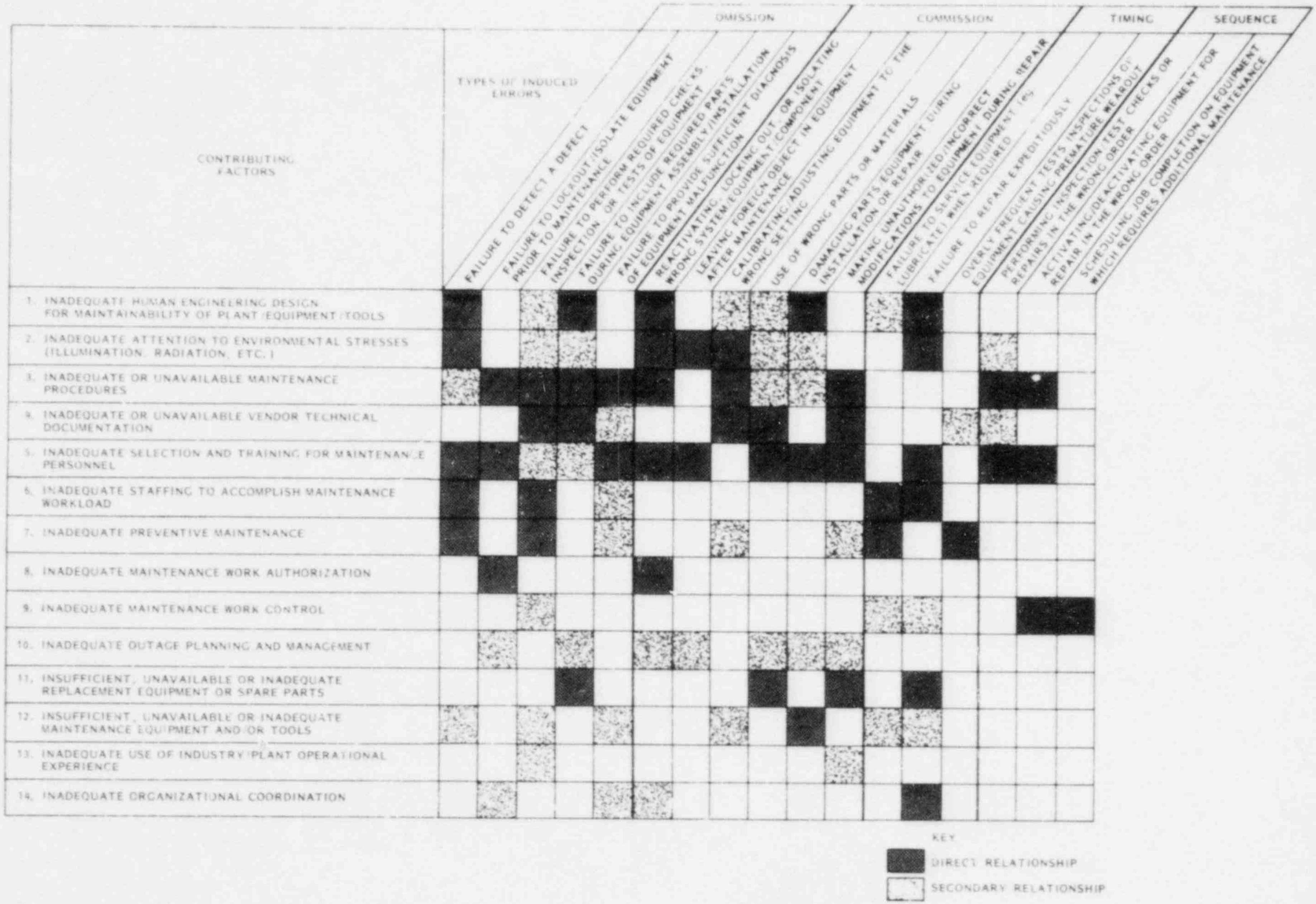
*Plan elements refer to the NRC Long-Range Human Factors Plan. The elements included in that plan were:

Organization & Management . . . OM
Staffing & Qualification . . . SQ
Training . . . . T
Examinations . . . . E
Man-Machine Interface . . . . . MMI
Procedures & Testing . . . . . PT

TABLE 4.3. Desirable Human Engineering Attributes of Well Designed Systems

	CATEGORY	FREQUENCY
A.	ACCESSIBILITY (good accessibility around the diesels, plant batteries are out in the open, turbine easy to get to, pulverizers well-spaced, good access to rod controls for repair, easy access to air compressors, good accessibility to MSIV, boiler feedpump out in the open and well-separated)	25
B.	EASE OF DISASSEMBLY, REMOVAL AND REPAIR (4KV breakers can be easily removed - isolated from other breakers, modular design of rod controls, ease of dismantling RCP motors, modules on roll-out rails, bolted instead of welded feedwater heaters)	24
C.	EASE OF SYSTEM MONITORING, TESTING AND TROUBLESHOOTING (good vibration and temperature monitoring on RCP motors, clear trouble indication, engineered safeguards easy to test, OFF-GAS system has everything up front - test points, switches, etc., built-in calibration system, control cabinet for boiler control easy to troubleshoot, good test jacks and easy to input signals)	17
D.	GOOD LIFTING AND MOVEMENT CAPABILITY (access for vehicles, built-in hoist always in-place, skid mounting, easy removal through roof, platforms very good, crane where you need it)	9
E.	HIGHLY RELIABLE EQUIPMENT (highly reliable engineered safeguards actuation system - settings don't drift, air compressor easy to operate and rarely breaks down, reliable relays, all components durable)	6
F.	EASE OF SERVICING AND INSPECTION (ease of oil changes, good access for preventive maintenance, easy to spot problems)	5
G.	GOOD MANUALS AND PRINTS (prints are readable, manuals very good, understandable procedures, detailed operating instructions)	5
H.	AVOIDANCE OF CONTAMINATED AREAS (equipment placed in an accessible location outside the "hot" areas)	3
I.	GOOD LAYDOWN AREA (excellent laydown area for turbine-generator)	3
J.	AVAILABILITY OF NECESSARY TOOLS (necessary tooling available, it's a complicated assembly but the necessary special tools were provided)	2
K.	MISCELLANEOUS (fail-safe design, mockups for training - should be used more often)	2
	TOTAL	101

TABLE 4.4. Induced Errors and Contributing Factors



The following areas from Table 4.2 are seen as particularly amenable to improvement in human performance:

- Maintenance Procedures and Technical Information
- Selection and Training of Maintenance Personnel
- Staffing
- Maintenance Control
- Outage Planning and Management
- Management

The most comprehensive review of this material to date is the study by Badalamente et al. (1983). This study, entitled Recommended Program for the Development of Maintenance Guidelines for Nuclear Power Plants, summarizes maintenance problems in a qualitative fashion. However, the above factors are ranked in order of estimated importance. This order is displayed in Table 4.2, and the basis for judgment is summarized in Table 4.4. In Table 4.4, types of errors induced in maintenance operations (i.e., errors of omission, commission, timing, or sequence) are correlated with contributing factors. The information in Table 4.4 indicates that the areas of most concern within the scope of this study fall under the categories of procedures and technical information, training, and staffing during the repair operation.

The information presented in Table 4.4 can be restructured to present more clearly the factors which dominate the error process. This has been done in Table 4.5 by dividing the contributing factors into design-oriented and human-performance-oriented factors. The number of entries checked for direct or secondary relationships which induce errors has then been summed.

TABLE 4.5. Ranking of Contributing Factors

	Number of Relationships Cited	
Design & Environmental Factors	Direct	Secondary
1. Human Factor Design	5	4
2. Environmental Stress	5	5
7. Preventive Maintenance	4	3
11. Spare Parts (Inventory Control)	4	0
12. Equipment & Tools	1	6
TOTAL	19	18
Human Performance Factors		
3. Maintenance Procedures	9	3
4. Technical Documentation	5	3
5. Training	11	2
6. Staffing	4	1
8. Work Authorizations	2	0
9. Work Control	2	3
10. Outage Planning & Management	0	7
13. Use of Industry Experience	0	2
14. Organizational Coordination	1	3
TOTAL	34	24

Table 4.5 indicates that human performance factors, particularly procedures, documentation, and training, play a substantial role in defining areas for

improvement in maintenance. However, the Badalamente study (1983) ranked design for maintenance first in importance. This area was consistently ranked as most important by maintenance personnel. The above information would thus indicate that human performance contributes less than 50 percent to the outage factor. A range of 25 percent to 50 percent is suggested as a working hypotheses.

4.1.2 LER Review

The above conclusion can be substantiated to some extent through a review of Licensee Event Reports (LERs) for nuclear power plants. A review of LERs by error categories very similar to those used above in Table 4.4 is summarized in Figure 4.1 (Brune and Weinstein 1980). Note that the categories of Follow-on (interface instructions following a repair), Communication, Comprehension, and Incomplete, Unrevised, and Inaccurate Procedures make up 36 percent of the errors reported. These are considered to be singularly human-performance-related factors. Non-compliance and Misalignment will also include some fraction of human-performance-related errors as was seen in the correlation of errors and contributing factors presented in Table 4.4. These two categories are then shared to some degree with the influence from design and environmental factors.

There are indications, therefore, that human-performance-related factors contribute at least 36 percent of the total to maintenance errors. The remaining 64 percent in Figure 4.1 will also contain some contribution from human performance, but design would be expected to play the dominant role. To estimate the upper bound for human factors, it will be assumed that 25 percent of the remaining categories, or 25% × 64% = 16%, can be attributed to human factors. This then gives an upper bound of approximately 50 percent, (i.e., 36% + 16% = 50%) for an estimated range of 25 percent to 50 percent.

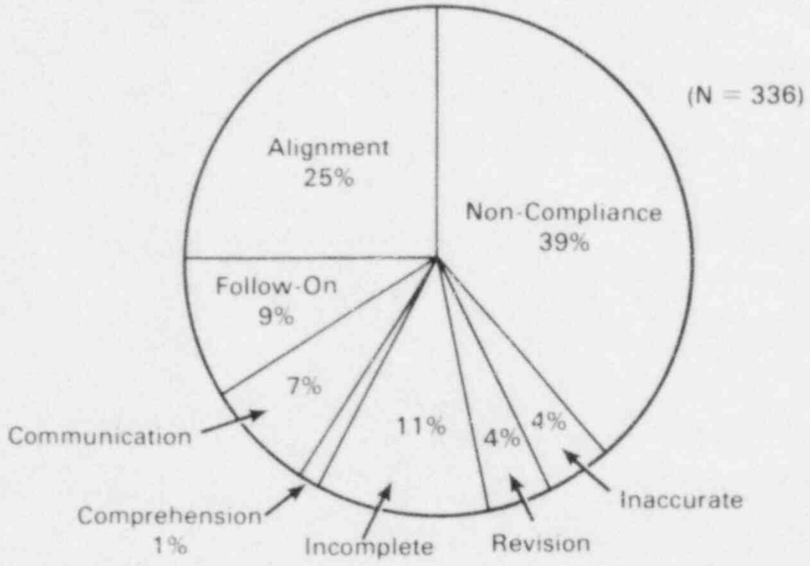
4.1.3 Proposed Model

The two data points above tend to confirm that the human-performance-related factors in maintenance represent 25 percent to 50 percent of the errors involved. It remains to be seen how these error rates affect outage duration and outage frequency. The LER review indicates that human errors will be more dominant during the actual outage, thus playing a more important role in determining the outage duration. The outage frequency is intuitively more dependent on design factors. An even division between human and design factors for outage duration is suggested as a working hypothesis, with a 25 percent to 75 percent division between the factors (human/design) for outage frequency.

The following new model is then proposed to examine the results of such a hypothesis. Factors which consist of the contributions from human performance, (H), and design, (D), the sum of which adds to one, will be associated with both the frequency, f, and duration, t, of the repairs in the equation for unavailability. This becomes

$$Q(\text{new}) = [(H1 + D1) \times t \times (H2 + D2) \times f]/720$$

where f is again in acts/month, and t is in hours/act.



Definitions of Performance
Deviation Categories Used to Classify LERs

Category	Definition
Non-Compliance	Unauthorized Action. A procedure was available but not used or followed. An action not in the procedure was performed or a step was performed out of the sequence specified in the procedure.
	Omission. A procedure was used but steps were not performed.
	Incorrect Action. A procedure was used but an error was committed in the performance of a step.
Misalignment	Valves, switches, jumpers, relays, fuses, breakers or solenoids were incorrectly positioned before, during or after a procedure.
Preparation of Inaccurate Procedures	Instructions in a procedure were incorrect.
Preparation of Incomplete Procedures	Instructions in a procedure were incomplete. The procedure did not contain necessary information.
Failure to Revise Interface Instructions	The procedure did not reflect changes in equipment or limits. An obsolete procedure was used.
Failure to Provide Interface Instructions	Personnel were not informed of necessary actions following completion of a procedure, e.g. a test following repair.
Communication	An extra-procedural communication, e.g. phone, intercom, was misunderstood.
Comprehension	The user misinterpreted the instructions in a procedure.

FIGURE 4.1. Distribution of LERs by Performance Deviation for the Maintenance Activity (Brune and Weinstein 1980)

The PNL expert panel did consider the existence of unavoidable (non-human-related) factors contributing to downtime during maintenance outages. But as a conservative approach, the fractional improvement estimated for maintenance staff performance was applied to the entire maintenance outage term. That is,

$Q(old) = (H \times t \times f)/720$, where H is the human factor.

The old model was thought to overestimate the reduction in Q by applying the percentage decrease in the human factor to the entire maintenance outage. However, combinations of H and D above are possible which bracket the old model. The cases listed in Table 4.6 are used to illustrate this point for reductions in H1 and H2 of 2.5, 5, 10, 20, and 40 percent (i.e., a reduction in human error), comparing the percentage difference between the reduction predicted for the old and new model. (There are some duplications in Table 4.6, these being Cases 2 and 4, 3 and 7, 6 and 8.) Cases 6 through 9 (hence Case 3) are not considered credible, given the examination of human error data which indicates a bound at the 50 percent contribution level.

TABLE 4.6. Potential Overestimation of Reduction in Maintenance Outage Unavailability in Old and New Maintenance Model for a Range of Human and Design Factor Combinations

					Percent Reduction in Human Factors (H)				
Case	H1	D1	H2	D2	2.5	5	10	20	40
					[100 x (Old - New)/Old]				
1	0.25	0.75	0.25	0.75	50.2	50.3	50.6	51.2	52.5
2	0.25	0.75	0.50	0.50	25.3	25.6	26.3	27.5	30.0
3	0.25	0.75	0.75	0.25	0.5	0.9	1.9	3.8	7.5
4	0.50	0.50	0.25	0.75	25.3	25.6	26.3	27.5	30.0
5	0.50	0.50	0.50	0.50	0.6	1.3	2.5	5.0	10.0
6	0.50	0.50	0.75	0.25	-24.1[(a)]	-23.1	-21.3	-17.5	-10.0
7	0.75	0.25	0.25	0.75	0.5	0.9	1.9	3.8	8.0
8	0.75	0.25	0.50	0.50	-24.1	-23.1	-21.3	-17.5	-10.0
9	0.75	0.25	0.75	0.25	-48.6	-47.2	-44.4	-38.8	-27.5

(a) minus sign indicates underestimate.

The results are shown graphically in Figures 4.2, 4.3 and 4.4. Again, the results are given as the percentage difference between the old and new models, i.e., [100 x (Old - New)/Old] for the various assumed reductions in the human performance factors. The results indicate a significant spread in the predicted overestimate of the old model, depending on the assumed split between human and design factors. The predicted overestimates range from approximately a 50 percent overestimate to a 50 percent underestimate for case 9, which, again, is not considered a viable example.

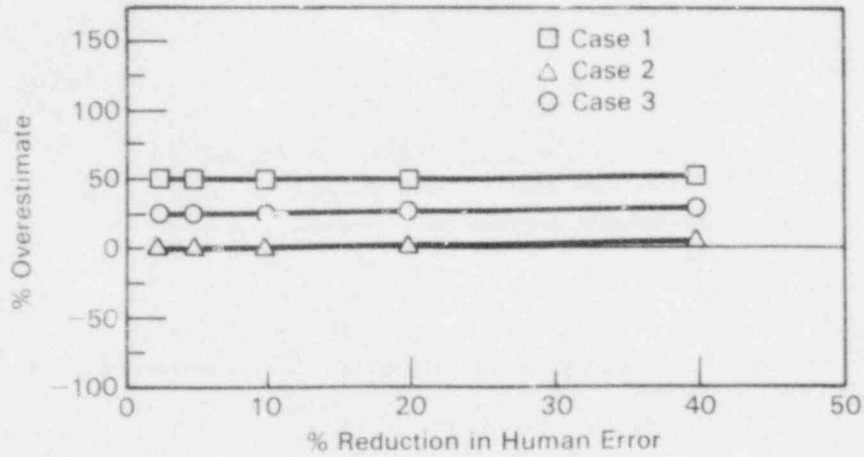


FIGURE 4.2 Old Maintenance Model Overestimation versus New Model (Cases 1,4,7)

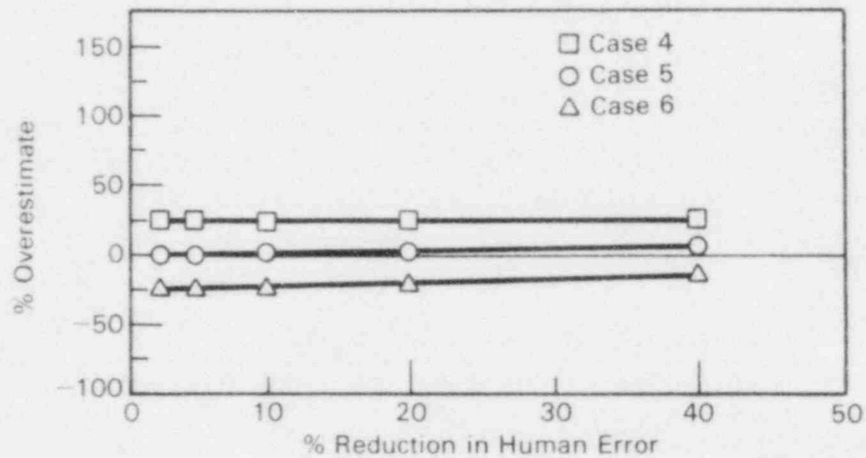


FIGURE 4.3 Old Maintenance Model Overestimation versus New Model (Cases 2,5,8)

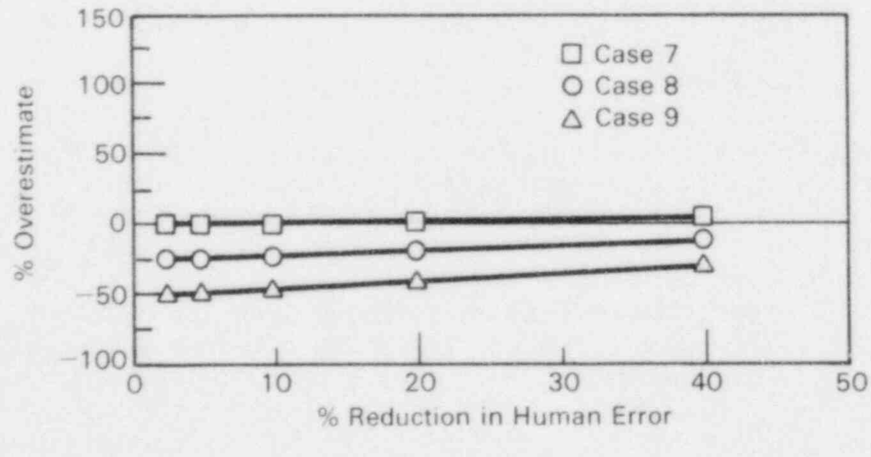


FIGURE 4.4 Old Maintenance Model Overestimation versus New Model (Cases 3,6,9)

The best engineering estimate at this time is that of Case 4. This case indicates an overestimate in unavailability, made by the old model, of approximately 25 percent. Case 5, which represents the upper bound on human performance factors assumed here, gives essentially the same results as the previous model.

Because the old maintenance model was a conservative model, the 25 percent overestimation of the old model versus the new model is within the range of uncertainty resulting from the PNL expert panel's estimates of maintenance performance improvements. Because the new maintenance model simply gives a more detailed breakdown of the constituents of the maintenance outage term, it is believed that use of the new maintenance model and new human performance improvements estimations would not change the results of the FY82 human factors issue analyses.

4.2 IMPACT OF REDUCTION IN HUMAN ERROR ON PLANT AVAILABILITY

In the PNL analysis, economic credit was taken for the industry cost savings due to accident avoidance after implementation of a safety issue resolution. This is simply the estimated cost of cleanup operations following a core-melt accident multiplied by the calculated reduction in accident frequency. For many safety issues, it is apparent that additional economic benefit would be derived from improved plant performance and reduced downtime. Replacement power costs can exceed several hundred thousand dollars per day, depending on the location of generation, they can quickly become substantial. It has been proposed that replacement power costs be consistently considered in future analyses of this nature.

A reduction in human error for plant operation and maintenance can logically be assumed to have some influence on plant availability in addition to the influence on core-melt frequency assumed in NUREG/CR-2800. In fact, the connection between human errors and day-to-day plant availability is easier to conceptualize than the tie to core-melt frequency. Errors in the daily operation and maintenance of the plant that lead to less serious results, such as plant shutdown, can be expected to occur much more frequently than accidents which lead to core damage or core melt.

The purpose of this particular task was to correlate a reduction in human error at nuclear power plants with increased plant availability and its associated economic benefits. The problem was examined to identify the data needed to make these correlations, and the available literature was studied to see if such data currently exist. When data were lacking, assumptions were made as necessary to provide the program with a working model. This information is presented below.

For FY82 analyses, the impact of reduction in human error has been confined to the plant risk equations and their associated reductions in core-melt frequency. These calculations have been applied specifically to equipment needing repair during plant operation. This creates a period of unavailability that could increase the potential for related accident sequences. No credit was given for the economic benefits associated with improved performance through the

reduction in outage duration and frequency. However, the techniques applied to improve human performance on equipment during plant operation could logically be assumed to apply to the duration of maintenance operations during forced outages. In addition, the reduction in repair frequency of this equipment could also have some impact on future failure rates and possible forced outages. Thus, there is some justification for estimating the economic benefits associated with reduced human error through reduced outage time.

Information on outages in general for nuclear plants is found in the NRC docuent entitled Operating Units Status Report for License Operating Reactors NUREG-0020, commonly known as the "Grey Book". A most recent examination of outages from this source was performed at INPO[a] for the 6-month period ending in January 1983. The results are given below in Table 4.7. The reason for the type of outage is correlated to the number of that type of outage that occurred per py and the average number of days the outage lasted.

TABLE 4.7. Nuclear Power Plant Outages (to January 1983)

NRC Code	Reason	Outages/py	Days/py
A	Equip. failure	6.5	26.5
B	Maintenance or testing	2.6	21.7
C	Refueling	2.3	53.7
D	NRC restrictions	3.0	5.1
E	Operator training	0.0	0.0
F	Administrative	0.05	0.4
G	Operational error	1.0	1.0
H	other	1.6	9.0
	Total	14.75	117.4

Exact definitions to the above categories could not be found in NUREG-0020. Note that the outages/py total approximately 17 in Table 4.7 but are corrected to 14.75 to correct for double counting of entries by the NRC. This correction is necessary because NRC records outages per month, and outages that last longer than one month may be counted more than once.

The potential outages that may be impacted are A and B, outages due to maintenance or testing. It is assumed that these are forced outages, as opposed to scheduled outages. Scheduled outages essentially consist only of refueling outages, where scheduled maintenance and repairs would be performed.

The information available at this time is insufficient for any direct correlation between a reduction in maintenance error and a reduction in forced outage time. However, the model proposed in Section 4.1 can be used as a working hypothesis. Again, this model represents both the duration, t, and frequency, f, of outage unavailability, Q, as the summed contribution of human performance and design factors:

$$Q = [(H1 + D1) \times t \times (H2 + D2) \times f]/720$$

where t is again in hours/act, and f is in acts/month. The best estimates available for these factors as discussed in Section 4.1 are recommended as H1=D1=0.5, H2=0.25, D2=0.75. Again, human performance is thought to play a more

(a) Personal Communication between W.E. Bickford, PNL and W.W. Little, INPO.

important role in reducing outage duration than in reducing outage frequency. The resulting percentage reduction in unavailability with percentage reduction in the human performance is presented in Table 4.8. It is further assumed that this percentage reduction in unavailability can be multiplied directly by the days per plant-year of forced outage, giving the improvement in days available. Note that this correlation is not expected to be 1 to 1. Even if maintenance operations are completed more rapidly, the time required for controlled shutdown and startup will most likely be dominant. The reduction in outage frequency, however, would have a direct relationship to improved availability. Thus, this model will again most likely be a conservative overestimate of the real improvement available from improved human performance.

TABLE 4.8. Estimated Improvement in Plant Availability with Reduction in Maintenance Human Error

Reduction in Human Error, %	Reduction in Plant Unavailability, %	Estimated Improvement in Availability, Days(based on 26.5+21.7 days/py)(a)
2.5	1.9	0.9 (0.6)
5.0	3.7	1.8 (1.2)
10.0	7.4	3.6 (2.4)
20.0	14.5	7.0 (4.7)
40.0	28.0	13.5 (9.0)
50.0	34.4	16.6 (11.1)
100.0 (limiting case)	62.5	30.1 (20.1)

(a) These values come from Table 4.7, Rows A and B.

Note that the model still has the problem of dealing with the cumulative benefit of numerous safety issues. A 100 percent reduction in human performance error will reduce the unavailability to (1-0.625), or 37.5 percent of its original value. Likewise, a 100 percent reduction in design performance will reduce unavailability to 12.5 percent of its original value, indicating a net reduction in unavailability of 150 percent. As a first estimate, the predicted days of reduction in outage time will be divided by 1.5 to account for this factor. The results of this adjustment are shown in parentheses in Table 4.8. The results are also displayed graphically in Figure 4.5.

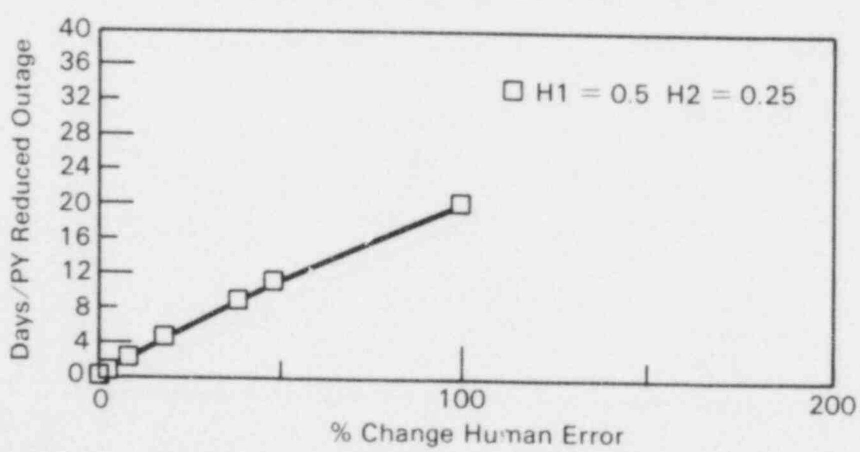


FIGURE 4.5 Reduction in Plant Outage Due to Reduction in Human Error

This model is recognized as being simplistic and lacks the necessary data to correlate improvements in plant performance with the subjective, qualitative types of studies that have been done to date on human factors in nuclear power plant maintenance. However, the available information suggests that the model presented here is sufficient as a working hypothesis. Further investigation is required to see whether additional information is available to draw the necessary connections, perhaps from individual utility records where improvement programs have been implemented and objectively evaluated.

5.0 HUMAN FACTORS DATA

The use of improved quantitative methods for determining potential risk reduction benefits which might be affected by human factors changes in nuclear power plants (NPPs) is dependent on the availability of useful data. Such data allow competent analysts to make quantitative or qualitative assessments of occurrences of human errors in NPPs that affect the availability or operational reliability of engineered safety systems and components.

The basic measure of human performance used in PRA methods is the human error probability (HEP). The HEP is the likelihood that when a given task is performed an error will occur. There are many ways to estimate the HEP; some are statistical and some are nonstatistical (i.e., judgemental).

The most useful information for human reliability analysis is actuarial data which consist of the number of errors of a given type that have occurred divided by the number of opportunities for that error to occur.

If a data-based estimate is not available, an estimate derived from information from similar tasks can be used. In these cases, similarity is judged in terms of the correspondence of behavioral variables. Two physically dissimilar items of equipment might be quite similar in terms of the human behavior involved in their operations, calibration, or maintenance. Therefore, an observed HEP for one item of equipment might be used as the estimate of the HEP for the same type of task on another item of equipment. Often, expert opinion is needed in the application of derived HEPs.

5.1 EXISTING HUMAN FACTORS DATA BANKS

For the past few years, the main source of data for quantifying the human contribution to NPP system performance has been NUREG/CR-1278 (Swain and Guttman 1980). Recently, with an increase in activity in the risk assessment area, there has developed a need to supplement and update the data in NUREG/CR-1278. Consequently, the NRC and Sandia National Laboratories (SNL) initiated a research project to develop a program for aggregating human reliability data relating to nuclear power plant operations. As part of that project, data bases other than NUREG/CR-1278 were reviewed to ascertain whether they could be used as sources of additional, updated data for inclusion in NUREG/CR-1278. The results of this data base review are reported in NUREG/CR-2744 (Topmiller, Eckel and Kozinsky 1983). In that study, the following 10 data bases, which represent all of the known data bases that deal with human performance data, were reviewed.

- American Institute for Research (AIR) Data Store
- Bunker-Ramo
- Aerojet General
- Technique for Establishing Personnel Performance Standards (TEPPS)
- Operational Recording and Data Systems (OPREDS)
- Air Force Inspection and Safety Center (AFISC)
- Aviation Safety Reporting System (ASRS)
- Nuclear Plant Reliability Data Systems (NPRDS)
- Safety Related Operator Action (SROA) Program
- IEEE Human Factors Working Group SC 5.5 Survey

The reviews of those data bases revealed very little additional data that could be used to improve the HEP values currently contained in NUREG/CR-1278. It was determined, however, that the AIR Data Store could be considered as a supplement to the data in the NUREG/CR-1278 tables dealing with displays (Tables 20-3 through 20-7) and controls (Table 20-13). The AIR data would be best applied at the detailed design level.

It was also observed that very little information was available in existing data banks that would assist human factors analysts in the scaling of appropriate performance-shaping factors (PSFs). The use of appropriately scaled PSFs is extremely important in arriving at specific HEP values that take into account significant contextual and environmental conditions affecting human performance. Topmiller, Eckel and Kozinsky (1983) also noted that the development of a method for integrating experimental and simulator-based data with data in NUREG/CR-1278 would be desirable. This concept will become increasingly important as more of the advanced man-machine interfaces (e.g., advanced cathodes ray tube displays and communication interfaces) are subjected to experimental treatment.

Based on the scarcity of available additional data for supplementing NUREG/CR-1278 and the increased interest in factoring HEP considerations into NPP PRAs, Topmiller, Eckel and Kozinsky concluded that a human reliability data bank specifically tailored for NPP PRA applications should be established.

The results of two recently completed studies in which data base concepts are described were reported by Comer et al. (1982) and Finlayson (1983). In their study of a new data bank concept, Comer et al. focused primarily on four requirements: human performance data collection, treatment, structuring (storage), and retrieval. Based on the premise that it is important for PRA applications that a Human Reliability Data Bank be established and working as soon as possible, four different data collection methods were proposed. Three of the data collection methods are extensions of existing data collection systems: the NPRDS, LERs, and Plant Incident Reports. The fourth data collection system would be a new system, the Nuclear Safety Reporting System (NSRS), which would be similar to the Aviation Safety Reporting System (ASRS). It will be established and used as another means of collecting data. In the proposed NSRS concept, two groups, each performing a separate function, would evaluate the field data and prepare them for a Human Reliability Analysis Group. The Human Reliability Analysis Group would evaluate data from sources other than operating power plants and classify all data for entry into the data bank. A Human Reliability Data Manual would also be developed to provide the data bank users with sufficient instructions and examples to use the data bank quickly and efficently. In addition, the concept of a Data Clearinghouse has been developed. The primary functions of this group would be to maintain the data bank, assist data bank users, and provide backup information to users who request it.

In his 1983 study, Finlayson developed a concept for a Nuclear Power Safety Reporting System (NPSRS). The purpose of the proposed NPSRS would be to provide a substantial, diverse source of data for assessing the influence of human factors data in the nuclear industry. Key elements of the NPSRS would include a third-party management organization and a NPSRS Advisory Committee.

The third-party management organization would provide a buffer between the support provided for the system by the NRC and reporters who submit incident descriptions to NPSRS. This third-party buffer would be essential to the success of the NPSRS in order to assure participants that if reports were submitted, the reporters would remain anonymous and that they would not incriminate themselves. The other key element of this proposed system is the NPSRS Advisory Committee. The Advisory Committee would be made up of a representative sample of the main participants in the system, both those providing input to the system and the users who receive output from the system. The Advisory Committee would be responsible for organizing and starting up the NPSRS and for monitoring subsequent operations. Data developed under the NPSRS would be fed into the Human Reliability Data Bank, which would be available for use by human factor system analysts, model developers for PRA events, utility and plant safety engineers, NRC regulatory staff, and others with an interest in the data.

5.2 CONCLUSIONS AND RECOMMENDATIONS

Based on the results of the review of existing human factors data banks by Topmiller, Eckel and Kozinsky, it does not appear that additional data are readily available to supplement NUREG/CR-1278. If current human factors data availability is limiting the refinement of PRA methods, attention may need to be focused on the development of a data bank specifically tailored for NPP PRA applications. However, the degree to which human factors data availability and quality affect the results of PRAs should be determined before efforts to develop a new data base are initiated. A survey of PRA analysts could be undertaken to determine if and to what degree they believe that additional human factors data would significantly affect the sensitivity of their analyses. If the results of the survey indicated that additional data might improve the quality of PRAs to some degree, a value analysis could be undertaken to determine the benefits to be attained as opposed to the costs for developing and implementing a system for collecting, storing and disseminating data specifically for use in NPP PRAs.

If these studies indicate that additional data will improve PRA analysis to an economically justifiable degree, development and implementation of a dedicated data base could proceed.

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APPENDIX A

RISK PARAMETERS FOR CALVERT CLIFFS 2 PWR

APPENDIX A

RISK PARAMETERS FOR CALVERT CLIFFS 2 PWR

The risk equation for Calvert Cliffs 2 (Combustion Engineering PWR with dry containment) has been summarized for the dominant accident sequences contributing to the seven PWR core-melt release categories as defined in WASH-1400 (NRC 1975). The Calvert Cliffs results have been extracted from its RSSMAP study, NUREG/CR-1659/3 (Hatch et al. 1982), and are provided here in Tables A.1 through A.4 with Addenda I and II. The information is presented so as to be compatible with the technique described in Section 3.0 of NUREG (Andrews et al. 1983) for estimating the risk reduction.

Table A.1 lists the dominant accident sequences for each PWR core-melt release category. The frequencies (reactor-year^{-1}) are given for each sequence listed in Table A.1. Also provided are the frequencies for the aggregates of non-dominant accident sequences per release category. Table A.2 defines the symbols used in Table A.1.

Table A.3 presents the dominant minimal cut sets for each dominant accident sequence listed in Table A.1. Also provided are the cut set frequencies, the containment failure modes for each sequence, the mode probabilities and corresponding release categories, and the frequencies of the sequences excluding the containment failure probabilities. Where appropriate, the contribution to the sequence frequency from the aggregate of non-dominant minimal cut sets is provided.

Table A.4 lists the elements of the dominant minimal cut sets given Table A.3. A brief description of each element is provided, with extended resolution into contributory failures were appropriate. The level of resolution is limited to that provided in the RSSMAP report. Probabilities are listed for each element. These can be viewed as unavailabilities unless otherwise specified. (Note that initiating event probabilities are occurrence rates in terms of reactor-year^{-1}).

Two elements in Table A.4 have somewhat detailed resolutions. For these, Addenda I and II have been provided. In some cases, additional detail can be found in the Calvert Cliffs RSSMAP report (Hatch et al. 1982). The analyst is referred to this for any further information that he may need.

Note that the Calvert Cliffs RSSMAP study was based upon auxiliary feedwater system (AFWS) design scheduled for upgrading at the end of 1982. This upgrade will affect some of the accident sequence frequencies given in the RSSMAP study. Estimates of these changes were made in the study. The analyst is referred there for further detail. These changes are not reflected in Tables A.1 through A.4 or the Addenda provided here.

Also note that some of the cut set frequencies in Table A.3 and some of the cut set element probabilities in Table A.4 differ slightly from the original values given in the RSSMAP study. These differences result from rounding the cut set element probabilities to two significant figures prior to their use in calculating the cut set frequencies. This is a simplification employed in this presentation to enhance the reproducibility of the results for use in subsequent

calculations. Note that none of these changes affect the total sequence frequencies (as given in the RSSMAP study) when measured to two significant figures.

TABLE A.1. Calvert Cliffs Dominant Accident Sequences and Frequencies (Reactor-Year^{-1})

Accident Sequence	PWR Release Category (based on WASH-1400) 1	2	3	4	5	6	7
T_2ML	α 9.0E-8		$(\gamma+\delta)$6.3E-4		β 6.3E-6		ε 2.7E-4
T_1ML	α 7.2E-8		$(\gamma+\delta)$5.0E-4		β 5.0E-6		ε 2.2E-4
T_3ML	α 1.2E-8		$(\gamma+\delta)$8.4E-5		β 3.4E-7		ε 3.6E-5
T_1MLOO'	α 1.0E-8	δ 8.0E-5	δ'2.0E-5	β 7.0E-7			
T_2KML	α 6.0E-9		$(\gamma+\delta)$4.2E-5		β 4.2E-7		ε 1.8E-5
T_1MQD	α 2.9E-9		$(\gamma+\delta)$2.0E-5		β 2.0E-7		ε 8.7E-6
T_1MQH	α 2.4E-9		$(\gamma+\delta)$1.7E-5		β 1.7E-7		ε 7.2E-6
T_1MQFH	α 1.2E-9	$(\gamma+\delta)$8.4E-6		β 8.4E-8		ε 3.6E-6	
T_2MQH	α 8.5E-10		$(\gamma+\delta)$6.0E-6		β 6.0E-8		ε 2.6E-6
Non-Dominant	1.3E-7	3E-5	--	3E-7	--	9E-6	1E-5
Total	3.3E-7	1.2E-4	1.3E-3	1.1E-6	1.3E-5	1.3E-5	5.7E-4

TABLE A.2. Symbols Used in Table A.1

Initiating Events

T_1 - Loss of Offsite Power Transient

T_2 - Loss of Power Conversion System Transient Caused by Other Than a Loss of Offsite Power

T_3 - Transients with the Power Conversion System Initially Available

System Failures

D - Emergency Coolant Injection System

F - Containment Spray Recirculation System

H - Emergency Coolant Recirculation System

K - Reactor Protection System

L - Auxiliary Feedwater System or Recovery of the Power Conversion System

M - Power Conversion System (Normal Operation)

Q - Reclosure of Pressurizer Safety/Relief Valves

O' - Containment Spray Injection System (transient event tree)

O - Containment Air Recirculation and Cooling System (transient event tree)

Containment Failure Modes

α - Vessel Steam Explosion

β - Containment Leakage

δ - Overpressure Due to Gas Generation

δ' - Overpressure Due to Gas Generation with Containment Failure Delayed Relative to Core Melt

γ - Overpressure Due to Hydrogen Burning

ε - Base Mat Melt Through

TABLE A.3. Dominant Minimal Cut Sets of Calvert Cliffs Dominant Accident Sequences

Accident Sequence	Sequence Frequency (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_2ML	9.0E-4	α	1E-4	1	T_2·M·CONST1·PCSNR	8.7E-4
(contribution from		$\gamma + \delta$	.7	3		
non-dominant minimal		β	.007	5		
cut sets = 3E-5)		ε	.3	7		
T_1ML	7.2E-4	α	1E-4	1	T_1·M·CONST1	5.8E-4
(contribution from		$\gamma + \delta$	.7	3	T_1·M·LOACRES·D21ST·D12ST	2.6E-5
non-dominant minimal		β	.007	5	T_1·M·LOACRES·D12ST·F2	1.8E-5
cut sets = 1.0E-4)		ε	.3	7		
T_3ML	1.2E-4	α	1E-4	1	T_3·M3·CONST1	1.2E-4
		$\gamma + \delta$	.7	3		
		β	.007	5		
		ε	.3	7		
T_1MLOO'	1.0E-4	α	1E-4	1	T_1·M·BATCM	8.0E-5
(contribution from		δ	.8	2	T_1·M·D21ST·D12ST·	
non-dominant minimal		δ'	.2	3	LOPNR8HR·DGNR8HR	5.4E-6
cut sets = 6E-6)		β	.007	4	T_1·M·D21ST·F2·	
					LOPNR8HR·DGNR8HR	3.8E-6
					T_1·M·LOACRES·LOPNRL·	
					D21ST·D12ST	2.6E-6
					T_1·M·LOACRES·LOPNRL·	
					D12ST·F2	1.8E-6
T_2KML	6.0E-5	α	1E-4	1	T_2·K·M·L	6.0E-5
		$\gamma + \delta$	.7	3		
		β	.007	5		
		ε	.3	7		

TABLE A.3. (contd)

Accident Sequence	Sequence Frequency (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_1MQD	2.9E-5	α	1E-4	1	$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ D21ST·D12ST·NREDF	4.1E-6
(contribution from		$\gamma + \delta$	.7	3	$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ D12ST·F2·NREDF	2.9E-6
non-dominant minimal		β	.007	5	$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ MT·D21ST·NREDF	1.5E-6
cut sets = 1.1E-5)		ϵ	.3	7	$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ HPCM·NRPORV	1.3E-6
					$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ M1·D12ST·NREDF	1.1E-6
					$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ MT·F2·NREDF	1.0E-6
					$T_1 \cdot M \cdot P_1 \cdot Q \cdot$ LT·D12ST·NREDF	9.8E-7
					$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ KT·D21ST·NREDF	9.8E-7
					$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ C·D12ST·NREDF	8.1E-7
					$T_1 \cdot M \cdot P_1 \cdot Q \cdot$ B·D21ST·NREDF	8.1E-7
					$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ N·D12ST·NREDF	7.9E-7
					$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ KT·F2·NREDF	6.8E-7
					$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ L1T1·D21ST·NREDF	4.3E-7
					$T_1 \cdot M \cdot P_1 \cdot Q \cdot$ F2·L1T1·NREDF	3.0E-7
T_1MQH	2.4E-5	α	1E-4	1	$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ R1·D21ST·NRLDF	5.8E-6
(contribution from		$\gamma + \delta$	.7	3	$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ R1·F2·NRLDF	4.0E-6
non-dominant minimal		β	.007	5	$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ N1·R1·NRLDF	4.0E-6
cut sets = 3E-6)		ϵ	.3	7	$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ N1·D12ST·NRLDF	1.4E-6
					$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ S1·D21ST·NRLDF	1.4E-6
					$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ S1·F2·NRLDF	1.0E-6
					$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ S1·N1·NRPORV	1.0E-6
					$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ R1·RASB1·NRPORV	8.0E-7
					$T_1 \cdot M \cdot P_1 \cdot Q \cdot$ R1·E2T1·NRLDF	5.6E-7
					$T_1 \cdot M \cdot \overline{P}_1 \cdot Q \cdot$ R1·G2T1·NRLDF	5.6E-7

TABLE A.3. (contd)

Accident Sequence	Sequence Frequency (ry-1)	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	CutSet Frequencies (ry-1)
T_1MQFH	1.2E-5	α	1E-4	1	$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot$ RASCM $\cdot$ NRPORV	1.6E-6
(contribution from		$\gamma + \delta$	.7	2	$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot$ R22 $\cdot$ D12ST $\cdot$ NRLDF	1.5E-6
non-dominant minimal		β	.007	4	$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot$ R21 $\cdot$ D21ST $\cdot$ NRLDF	1.2E-6
cut sets = 1E-6)		ε	.3	6	$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot$ R21 $\cdot$ R22 $\cdot$ NRPORV	8.3E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot W \cdot$ D21ST $\cdot$ NRLDF	7.5E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot V \cdot$ D21ST $\cdot$ NRLDF	7.5E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot$ F2 $\cdot$ R21 $\cdot$ NRLDF	8.0E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot W \cdot$ R22 $\cdot$ NRPORV	5.4E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot W \cdot$ F2 $\cdot$ NRLDF	5.2E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot V \cdot$ R21 $\cdot$ NRLDF	4.2E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot$ RASA1 $\cdot$ D21ST $\cdot$ NRLDF	2.9E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot$ RASB1 $\cdot$ D12ST $\cdot$ NRLDF	2.9E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot B \cdot$ R22 $\cdot$ NRLDF	2.9E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot W \cdot V \cdot$ NRPORV	2.7E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot C \cdot$ R21 $\cdot$ NRPORV	2.2E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot$ R22 $\cdot$ RASA1 $\cdot$ NRPORV	2.1E-7
					$T_1 \cdot M \cdot \bar{P}_1 \cdot Q \cdot$ F2 $\cdot$ RASA1 $\cdot$ NRPORV	2.0E-7

TABLE A.3. (contd)

Accident Sequence	Sequence Frequency (ry^{-1})	Cont. Fail. Modes	Mode Prob's	Rel. Cat's	Minimal Cut Sets	Cut Set Frequencies (ry^{-1})
T_2MQH	8.5E-6	α	1E-4	1	$T_2 \cdot M \cdot \bar{P}_{11} \cdot Q \cdot N1 \cdot R1 \cdot NRPORV$	4.2E-6
(contribution from		$\gamma + \delta$	.7	3	$T_2 \cdot M \cdot \bar{P}_{11} \cdot Q \cdot N1 \cdot S1 \cdot NRPORV$	1.1E-6
non-dominant minimal		β	.007	5	$T_2 \cdot M \cdot \bar{P}_{11} \cdot Q \cdot RASB1 \cdot R1 \cdot NRPORV$	8.4E-7
cut sets = 7E-7)		ε	.3	7	$T_2 \cdot M \cdot \bar{P}_{11} \cdot Q \cdot KT \cdot R22 \cdot NRPORV$	3.7E-7
					$T_2 \cdot M \cdot \bar{P}_{11} \cdot Q \cdot LT \cdot R21 \cdot NRPORV$	2.9E-7
					$T_2 \cdot M \cdot \bar{P}_{11} \cdot Q \cdot SIA2 \cdot R22 \cdot NRPORV$	2.2E-7
					$T_2 \cdot M \cdot \bar{P}_{11} \cdot Q \cdot RASB1 \cdot S1 \cdot NRPORV$	2.1E-7
					$T_2 \cdot M \cdot \bar{P}_{11} \cdot Q \cdot RASA1 \cdot N1 \cdot NRPORV$	2.0E-7
					$T_2 \cdot M \cdot \bar{P}_{11} \cdot Q \cdot KT \cdot V \cdot NRPORV$	1.9E-7
					$T_2 \cdot M \cdot \bar{P}_{11} \cdot Q \cdot LT \cdot W \cdot NRPORV$	1.9E-7

TABLE A.4. Elements of Dominant Minimal Cut Sets of Calvert Cliffs Dominant Accident Sequences

Symbol	Description	Probability
T_1	Transient initiating event caused by a loss of offsite power.	0.2/ry
T_2	Transient initiating event caused by a loss of feedwater. Offsite power is assumed to be available	3/ry
T_3	Transient initiating event with the power conversion system (PCS) initially available (e.g., turbine trip).	4/ry
M	Total interruption of the PCS	1
M3	Failure of the PCS to continue operation following reactor trip.	.01
D21ST	Unavailability of diesel generator #21 due to maintenance and start failures. It is needed to open one of two auxiliary feedwater system (AFWS) steam-admission motor-operated valves (MOVs) after a loss of offsite power. The failure probability of event D21ST is the sum of the following contributory modes: failure to start - .030 maintenance outage - .0064 .036	.036
D12ST	Unavailability of diesel generator #12 due to maintenance and start failures. It is needed to open one of two AFWS steam-admission MOVs after a loss of offsite power. The expansion of event D12ST into its contributory failures is analogous to event D21ST.	.036
BATCM	Common-cause failure of the DC power system due to miscalibration of the battery charger charging rate.	4.0E-4
K	Failure of the reactor protection system (RPS) due primarily to failure of the RPS trip circuit breakers to open. The expansion of event K into its contributory failures is somewhat complex. For more detail, see Addendum I.	2.0E-5

TABLE A.4. (contd)

Symbol	Description	Probability
B	Loss of flow path between the refueling water storage tank (RWST) and pumps for 1) trains #21 of the low pressure and containment spray injection/recirculation systems (LPIS/RS and CSIS/RS) and 2) trains #21 and 22 of the high pressure injection/recirculation system (HPIS/RS).	.0070
	Event B occurs if either of the following fails: 1. A normally-open (NO) MOV. Its failure probability is the sum of the following contributory modes: operator error - .001 plugged - .0001 maintenance outage - .0058 .0069 2. A check valve (CV). It has a hardware failure probability of .0001. The failure probability of event B is the sum of the above.	
C	Loss of flow path between the RWST and pumps in 1) trains #22 of the LPIS/RS and CSIS/RS and 2) train #23 of the HPIS/RS.	.0070
	The expansion of event C into its contributory failures is analogous to that for event B.	
KT	Loss of flow path through train #21 of the HPIS/RS.	.0085
	Event KT occurs if any of the following fails: 1. Either of two CVs. Each has a hardware failure probability of .0001. 2. Either of two NO manual valves (ManVs). The failure probability of each is the sum of the following contributory modes: plugged - .0001 operator error - .0001 .0002	

TABLE A.4. (contd)

Symbol	Description	Probability
	3. A pump. Its failure probability is the sum of the following contributory modes: hardware - .001 control circuitry - .0011 maintenance outage - .0058 .0079	
	The failure probability of event KT is the sum of the above: 2(.0001) + 2(.0002) + .0079 = .0085. The factors of two account for the contributions from two CVs and two NO ManVs.	
LT	Loss of flow path through train #23 of the HPIS/RS.	.0085
	The expansion of event LT into its contributory failures is analogous to that for event KT.	
MT	Loss of one of two common flow paths downstream from trains #21-23 in the HPIS/RS.	.013
	Event MT occurs if a normally-closed (NC) MOV fails. Its failure probability is the sum of the following contributory modes: hardware - .001 plugged - .0001 control circuitry - .0064 maintenance outage - .0058 .013	
N	Loss of one of two common flow paths downstream from trains #21-23 in the HPIS/RS.	.0069
	Event N occurs if an NO MOV fails. It has a failure probability of .0069 as shown in event B.	
HPCM	Diversion of HPIS flow through any one of eight interfacing lines of the cold leg accumulator system (CLAS) of the LPIS/RS, a common-cause failure	8E-4

TABLE A.4. (contd)

Symbol	Description	Probability
	Event HPCM occurs if any one of eight CVs fails. Each has a hardware failure probability of .0001.	
	The failure probability of event HPCM is the sum of that for each CV.	
V	Loss of the flow path from the containment sump to train #23 of the HPRS or train #22 of the LP/CSRS.	.013
	Event V occurs if either of the following fails: 1. A CV. It has a hardware failure probability of .0001. 2. An NC MOV. It has a failure probability of .013 as shown in event MT.	
	The failure probability of event V is the sum of the above	
W	Loss of the flow path from the containment sump to trains #21 and 22 of the HPRS or train #21 of the LP/CSRS.	.013
	The expansion of event W into its contributory failures is analogous to that for event V.	
SIA2	Failure of the safety injection actuation (SIA) logic channel A2 in the engineered safety features actuation system (ESFAS).	.0050
	The failure probability of event SIA2 is the sum of the following contributory modes: hardware (single or double failures) - .0029 test or maintenance outage - .0021 .0050	
RASA1	Failure of the recirculation actuation (RA) logic channel A1 in the ESFAS.	.0050
	The expansion of event RASA1 into its contributory failures in analogous to that for event SIA2.	

TABLE A.4. (contd)

Symbol	Description	Probability
RASB1	Failure of the RA logic channel B1 in the ESFAS.	.0050
	The expansion of event RASB1 into its contributory failures is analogous to that for event SIA2.	
RASCM	Common-cause failure of the RA logic channels in the ESFAS due to sensor/comparitor miscalibrations.	.0010
F2	Loss of flow path through the service water heat exchanger in train #22 of the salt water system (StWS).	.025
	Event F2 occurs if one of three control valves (ConVs) fails. Two have failure probabilities of .0058 due to maintenance outage. The failure probability of the third is the sum of the following contributory modes:	
	hardware - .0006 plugged - .0001 control circuitry - .0064 maintenance outage - .0058 .0129	
	The failure probability of event F2 is the sum of the above: 2(.0058) + .0129 = .025. The factor of two accounts for the contribution from the first two ConVs.	
M1	Loss of flow path through the pump in train #22 of the component cooling system (CCS).	.0098
	Event M1 occurs if any of the following fails:	
	1. A CV. It has a hardware failure probability of .0001.	
	2. Either of two NO ManVs. Each has a failure probability of .0002 as shown in event KT.	
	3. A pump. Its failure probability is the sum of the following contributory modes:	

Symbol	Description	Probability
	hardware - .0010 control circuitry - .0018 maintenance outage - .0058 failure to run for 24 hr (λ = 3.0E-5/hr) - .00072 .0093	
	The failure probability of event M1 is the sum of the above: .0001 + 2(.0002) + .0093 = .0098. The factor of two accounts for the contribution from two NO ManVs.	
N1	Loss of flow path through the component cooling heat exchanger in train #22 of the StWS.	.025
	Event N1 occurs if either of two ConVs fails. The failure probability of each is the sum of the following contributory modes: hardware - .0003 control circuitry - .0064 maintenance outage - .0058 .0125	
	The failure probability of event N1 is the sum of that for each ConV.	
R1	Loss of normal flow path through the component cooling heat exchanger in CCS loop #21 given failure of CCS loop #22.	0.10
	Event R1 occurs if either of the following fails: 1. Either of two NO ManVs. Each has a failure probability of .0002 as shown in event KT. 2. An NC ConV. It has a failure probability of 0.10 due to operator failure to open it given failure of CCS loop #22.	
	The failure probability of event R1 is the sum of the above: 2(.0002) + 0.10 = 0.10. The factor of two accounts for the contribution from two NO ManVs.	

TABLE A.4. (contd)

Symbol	Description	Probability
S1	Loss of flow path through the component cooling heat exchanger in train #21 of the StWS.	.025
	The expansion of event S1 into its contributory failures is analogous to that for event N1.	
R21	Failure of the emergency core cooling system (ECCS) pump room cooler #21.	.020
	The failure probability of event R21 is the sum of the following contributory modes:	
	fan failure - .00084 control circuitry - .0060 hardware - .013 .020	
R22	Failure of the ECCS pump room cooler #22.	.026
	The failure probability of event R22 is the sum of the following contributory modes:	
	fan failure - .00084 control circuitry - .0060 hardware - .019 .026	
E2T1	Failure of pump #22 to restart in the StWS after a loss of offsite power.	.0035
	The failure probability of event E2T1 is the sum of the following contributory modes:	
	hardware - .0010 control circuitry - .0018 failure to run for 24 hr (λ = 3.0E-5/hr) - .00072 .0035	
G2T1	Failure of pump #22 to restart in the service water system (SvWS) after a loss of offsite power.	.0035
	The expansion of event G2T1 into its contributory failures is analogous to that for event E2T1.	

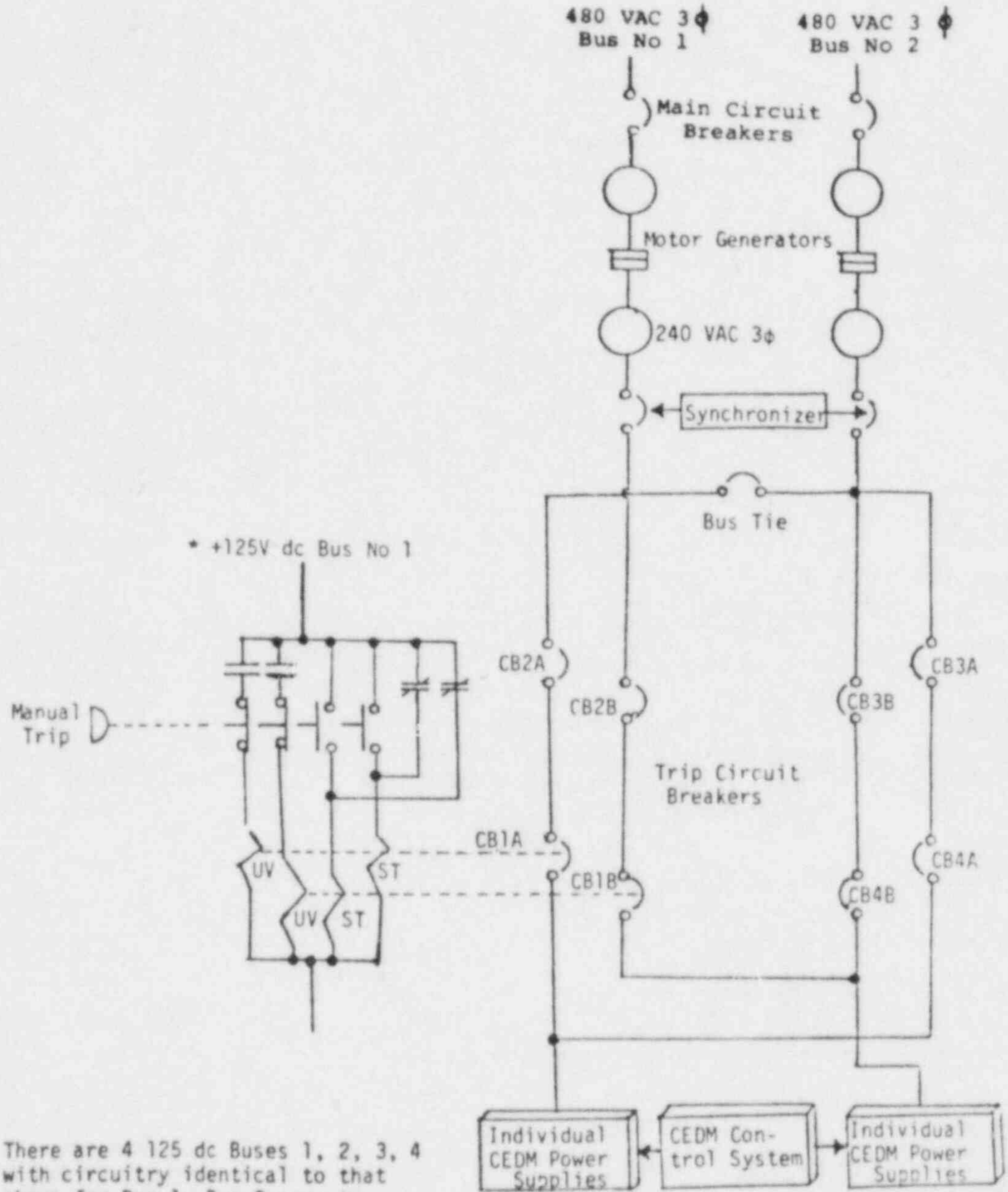
TABLE A.4. (contd)

Symbol	Description	Probability
L1T1	Failure to re-establish a flow path through the pump in CCS loop #21 after a loss of offsite power.	.0037
	Event L1T1 occurs if either of the following fails: 1. A pump (restart). It has a failure probability of .0035 as shown in event E2T1. 2. An NO ManV. It has a failure probability of .0002 as shown in event KT. The failure probability of event L1T1 is the sum of the above.	
CONST1	Failure of the AFWS due primarily to common-cause operator failure to manually initiate the AFWS or failure of both turbine pump trains.	.0029
	The expansion of event CONST1 into its contributory failures is somewhat complex. For more detail, see Addendum II.	
L	Failure of the AFWS given a loss of feedwater, failure of the RPS and total interruption of the PCS.	1
$\overline{P}_1$	Power-operated relief valves (PORVs) demanded open given a loss of offsite power or a transient with the PCS initially available.	1
$\overline{P}_{11}$	PORVs demanded open given a loss of feedwater.	.070
Q	Failure of a PORV to reclose given it opens.	.080
PCSNR	Failure to restore the PCS within ~30 minutes following a reactor trip caused by a PCS interruption	0.10
LOACRES	Failure to restore the AFWS given a total loss of AC power (LOAC). The AFWS is restored by opening manual valves in the AFWS steam-admission lines.	0.10

TABLE A.4. (contd)

Symbol	Description	Probability
LOPNRL	Failure to restore offsite power within 3 hours given LOAC. Restoration of offsite power would allow operation of containment systems capable of mitigating the accident.	0.10
LOPNR8HR	Failure to restore offsite power within 8 hours given LOAC and success of AFWS. Restoration of offsite power or a diesel generator is required to prevent AFWS failure via depletion of the station batteries.	.030
DGNR8HR	Failure to restore a diesel generator within 8 hours given LOAC and success of AFWS. It is assumed repair of only one diesel will be attempted within this time-frame.	0.70
NRPORV	Failure to recover a stuck-open PORV by closing its block valve given AC power is available. This includes the operator failing to close the valve or hardware failure of the valve.	0.10
NREDF	Failure to recover failed components within ~1 hour following a stuck-open PORV LOCA and failure of one or both diesels to prevent core-melt. Recovery actions consist of restoring offsite power so that high pressure injection can be utilized or so that the PORV block MOV can be closed.	0.20
NRLDF	Failure to recover failed components within ~3 hours following a stuck-open PORV LOCA and failure of a single diesel to prevent core-melt.	0.10

ADDENDUM I



*Note: There are 4 125 dc Buses 1, 2, 3, 4 with circuitry identical to that shown for Bus 1, Bus 2 goes to CB2A and CB2B, etc.

FIGURE I-1. Calvert Cliffs RPS Power Trip Circuit

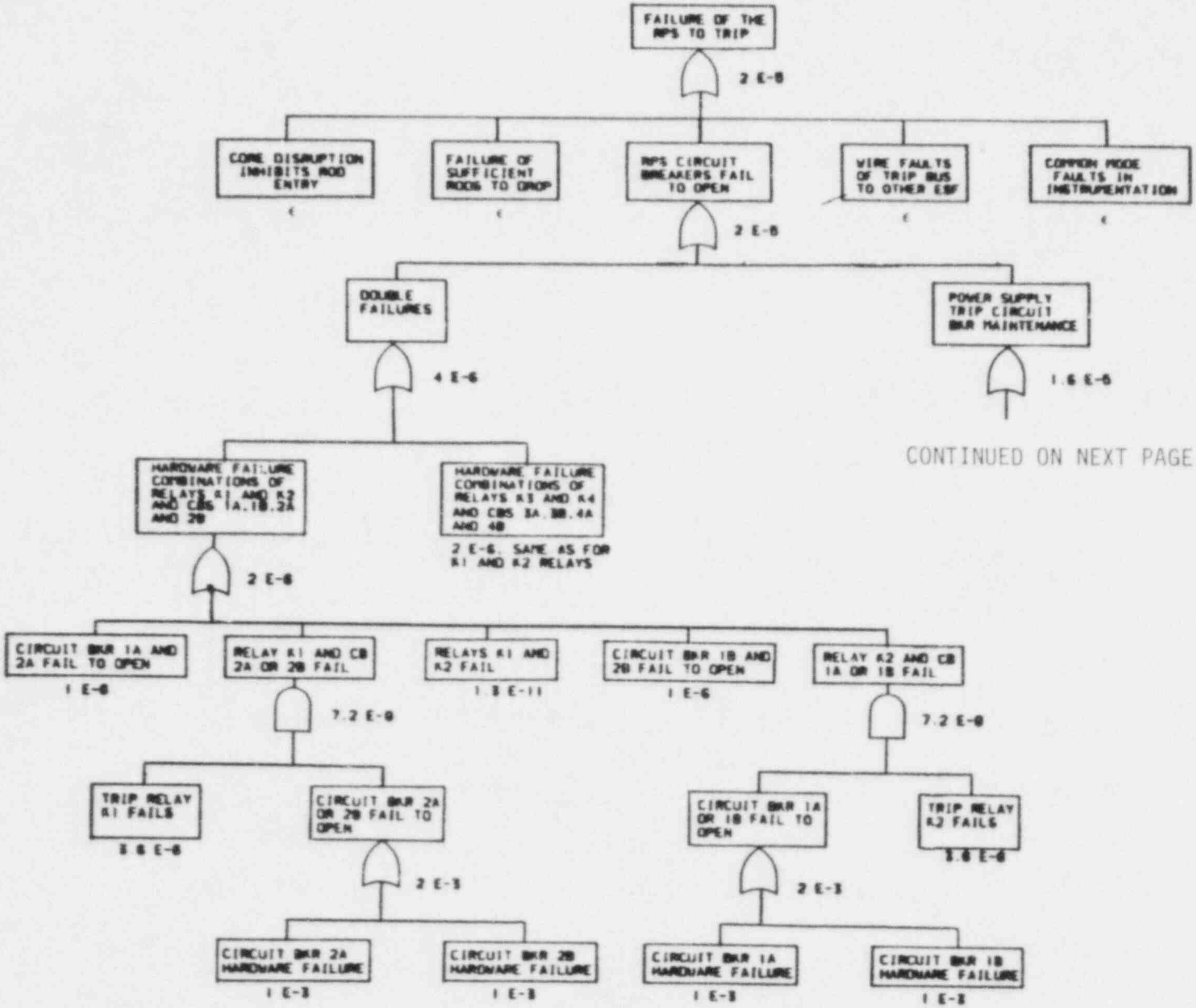


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FIGURE I-2. Calvert Cliffs Simplified RPS Fault Tree

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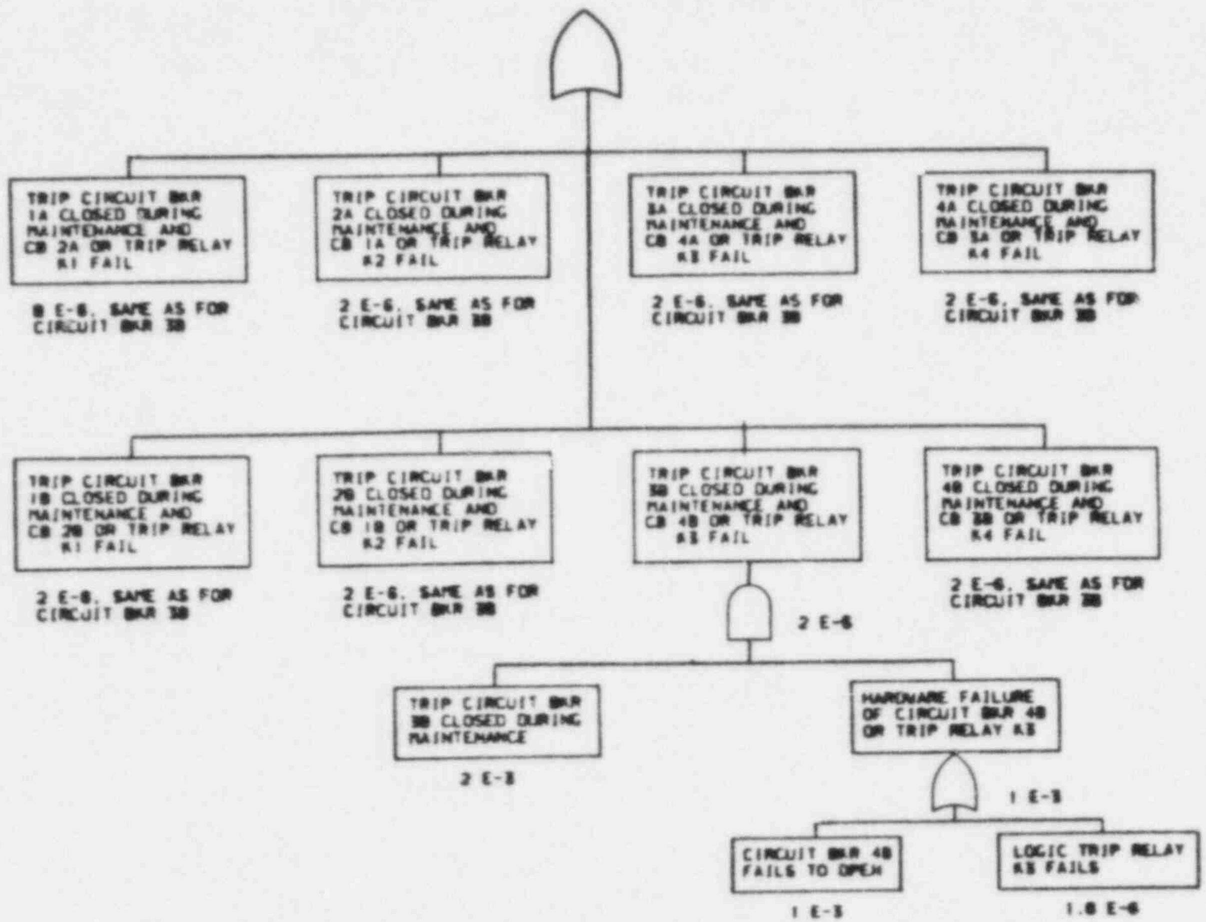

FIGURE I-2. FIGURE I-2. (contd)

TABLE II-1. Boolean Expansion of Term CONST1

The Boolean expansion of CONST1 is the sum of the following terms:

Terms	Probabilities
AFWSCM	1.0E-3
B1·C1	8.4E-4
A1	4.0E-4
D1·E1	1.7E-4
F1·G1	1.7E-4
F1·D1	1.7E-4
E1·G1	1.7E-4
$\sum$ = CONST1 =	2.9E-3

TABLE II-2. Boolean Terms Comprising CONST1

Boolean Term	Term Definition	Term Unavailability*
A1	C3 + C4	4.0×10^{-4}
B1	P1 + P4 + S6 + P3 + S5 + TP21	2.9×10^{-2}
C1	P2 + P6 + S8 + P5 + S7 + TP22	2.9×10^{-2}
D1	H1 + H5 + CV-4511	1.3×10^{-2}
E1	H2 + H6 + CV-4512	1.3×10^{-2}
F1	S3 + MOV-4071	1.3×10^{-2}
G1	S4 + MOV-4070	1.3×10^{-2}
AFWSCM	Operator fails to manually initiate the system	1.0×10^{-3}

*Double and triple maintenance contributions have been removed from these terms.

TABLE II-3. Component Failures Corresponding to Boolean Terms in CONST1

Component Description	Fault Identifiers	Failure Contributors	Q/Component
Check Valve	P3, P5 S3, S4, S5 S7, H5, H6	Hardware	1.0×10^{-4}
		Q Total	1.0×10^{-4}
Manual Valve (Normally Open)	C3, C4, P1 P2, P4, P6 S6, S8, H1 H3	Operator Error	1.0×10^{-4}
		Plugged	1.0×10^{-4}
		Q Total	2.0×10^{-4}
Turbine Pump	TP21 TP22	Hardware[1]	2.0×10^{-2}
		Control Circuit	1.8×10^{-3}
		Fails to operate 24 hrs. (3.0×10^{-5}/hr)	7.2×10^{-4}
		Maintenance	5.8×10^{-3}
		Q Total	2.8×10^{-2}
Control Valve	CV-4511 CV-4512	Hardware	3.0×10^{-4}
		Control Circuit	6.4×10^{-3}
		Plugged	1.0×10^{-4}
		Maintenance	5.8×10^{-3}
		Q Total	1.3×10^{-2}
Motor Operated Valve	MOV-4071 MOV-4070	Hardware	1.0×10^{-3}
		Plugged	1.0×10^{-4}
		Control Circuit	6.4×10^{-3}
		Maintenance	5.8×10^{-3}
		Q Total	1.3×10^{-2}

[1]Turbine pump data taken from "Boston Edison Co. Pilgrim Station Unit 2 Station Blackout and Emergency Feedwater System Reliability," Bechtel Power Corp., Sept. 1978.

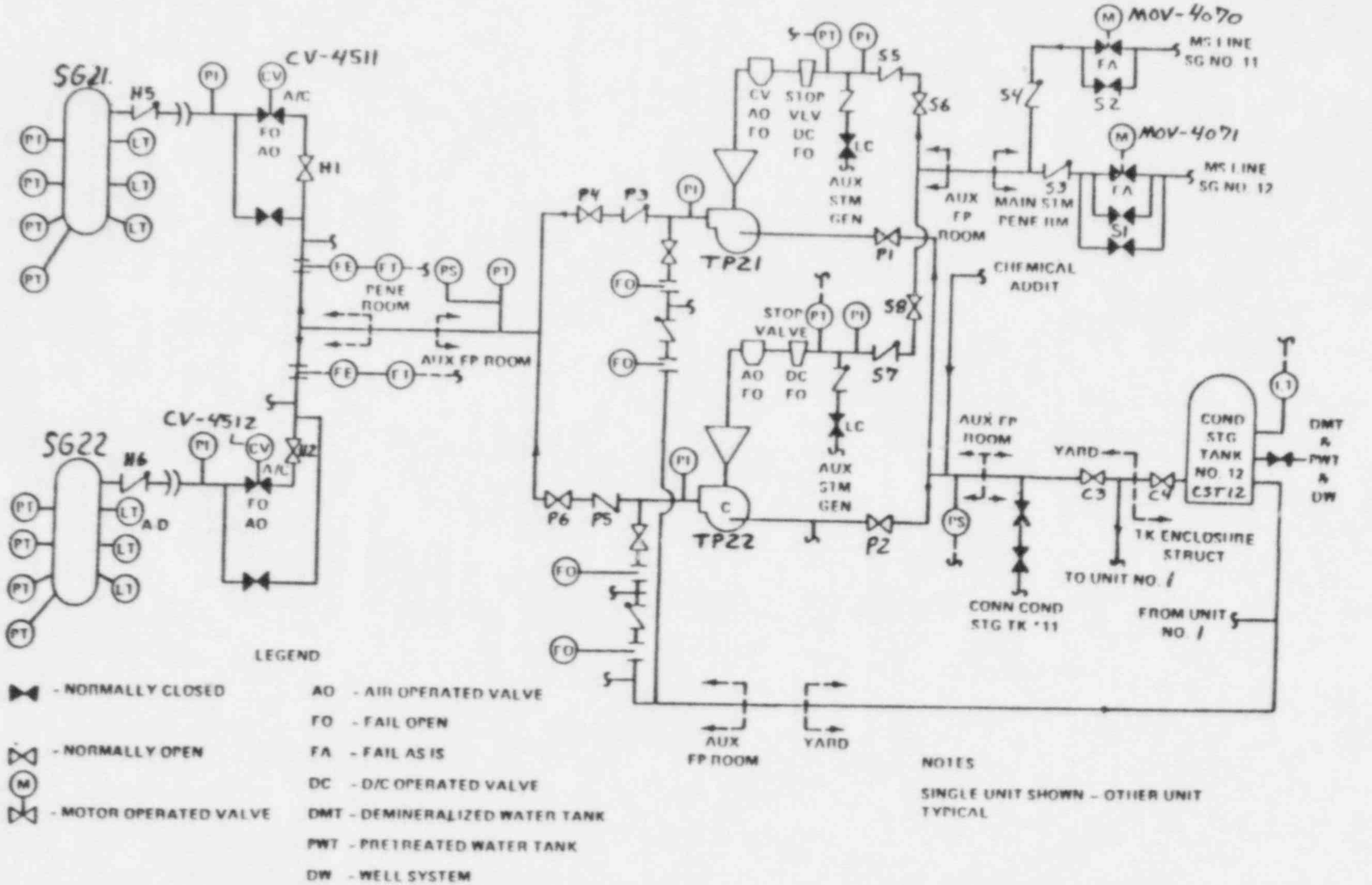

FIGURE II-1. Calvert Cliffs Auxiliary Feedwater System

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APPENDIX B

NAMES OF THE PNL EXPERT PANEL AND PNL RISK ANALYSTS

APPENDIX B

NAMES OF THE PNL EXPERT PANEL AND PNL RISK ANALYSTS

The PNL Expert Panel consisted of A. J. Boegel, M. J. Clausen, B. F. Gore, and S. W. Heaberlin. The PNL risk analysts involved in developing the human factors techniques used in the PSI project were W.E. Bickford, R.H.V. Gallucci, and T. B. Powers.

PART 2

PRIORITIZATION OF THE U.S. NUCLEAR REGULATORY COMMISSION HUMAN FACTORS PROGRAM PLAN

1.0 INTRODUCTION

This report summarizes the results of risk and cost analyses conducted by the Pacific Northwest Laboratory (PNL) for the Safety Program Evaluation Branch (SPEB) of the U.S. Nuclear Regulatory Commission (NRC) in support of the prioritization of the Human Factors Program Plan (HFPP). The main part of the report is divided into discussions of risk, cost, summary of results and conclusions.

The accident at Three Mile Island Unit 2 (TMI-2) identified the need to bring human factors considerations into the mainstream of NPP regulation and operation. A thorough understanding of functions, capabilities, and limitations of the personnel involved is needed to evaluate the safety of nuclear power plants (NPPs). NUREG-0660, The NRC Action Plan Developed as a Result of the TMI-2 Accident, described a number of tasks to be performed by the nuclear industry and the NRC. A significant number of these tasks were aimed at improving NPP safety through increased attention to the human element. Considerable progress has been made on many of these Action Plan items.

The purpose of the 1983 version of the NRC HFPP was to ensure that proper consideration is given to human factors in the design, operation, and maintenance of NPPs. This 1983 plan addressed NPPs and described (1) the technical assistance and research activities planned to provide the technical bases for the resolution of the remaining human factors related tasks described in NUREG-0660 and NUREG-0737, Clarification of TMI Action Plan Requirements, and (2) the additional human factors efforts identified during implementation of the Action Plan that should receive NRC attention. The plan is intended to represent a systematic and comprehensive approach for addressing human factors concerns important to NPP safety in the FY83 through FY85 time frame. The initial HFPP identifies seven elements perceived as addressing the human factors concerns in connection with NPP safety. These elements, which are described in Appendix A, are listed below:

1) Staffing and Qualification (SQ)
2) Training (T)
3) Licensing Examinations (LE)
4) Procedures (P)
5) Management and Organization (MO)
6) Man-Machine Interface (MMI)
7) Human Reliability (HR)

The NRC staff has recognized that there must be orders of priority among the plan elements. While all plan elements are important, some have more immediate application to improved plant safety. The seven plan elements have been qualitatively prioritized but to date they have not been quantitatively prioritized. The quantitative prioritization of these plan elements, based on risk and cost, is the purpose of this prioritization task.

Methods used in this analysis are described in NUREG/CR-2800. Risk impact was determined by using the results of an importance rating exercise to calculate risk changes based on two Reactor Safety Study Methodology Application Program (RSSMAP) risk assessments. Risk is discussed in Section 2.0. Costs were determined based on the costs associated with previously assessed safety issues and estimates for additional new activities. Cost is discussed more fully in Section 3.0. Results are summarized in Section 4.0. Conclusions are presented in Section 5.0.

The nature of both the recipients and resources involved in improving human performance result in significant overlaps in both costs and benefits. One major deviation from the approach taken in the analysis of a single issue, described in NUREG/CR-2800, is the consideration of the risk and cost overlaps. A discussion of these effects is provided in both Sections 2.0 and 3.0.

1.1 SCOPE

Certain guidelines were established by the NRC staff to define this work. First, the plan elements were to be evaluated without considering the contribution or effect of maintenance. The maintenance issue has been prioritized as a separate issue and involves more than just human factor related aspects. Second, since the Human Reliability Data element was considered to be a licensing issue, to be worked on regardless of safety priority, it was not prioritized with the other six plan elements. HFPP element issues were defined by the NRC as of January 1984, and are described in Appendix A.

2.0 ASSESSMENT OF RISK CHANGES

The assessment of risk changes associated with the implementation of the six selected human factors program plan elements involved composing a questionnaire that addressed the effects of the results of the plan elements on human performance improvement, selecting a panel of knowledgeable staff to evaluate the questionnaire, and using the questionnaire results to perform sensitivity studies on two risk assessments, Oconee and Grand Gulf RSSMAP assessments. Results include quantitative risk changes related to the various plan elements. This methodology and the results are discussed in the following subsections.

2.1 METHODOLOGY

The methodology used in quantifying the risk changes associated with the implementation of the human factors program plan elements involves the following five steps:

- composing the questionnaire
- selecting the staff panel
- evaluating the questionnaire
- compiling questionnaire results
- using risk assessments to calculate risk changes

2.1.1 Composing the Questionnaire

The first risk-related step taken in quantitatively prioritizing the six plan elements to be evaluated was to compose a questionnaire to be given to a panel of staff. The questionnaire that was finally evaluated by the whole panel is duplicated in Appendix B. The panel of staff was asked to assess the direct and indirect effects of the implementation of these plan elements on human performance improvement. The direct and indirect effects can be understood better by considering the illustration in Figure 2.1.

In Figure 2.1, the largest circle represents the total risk related to operation of an NPP. The second largest circle represents the part of the total risk that is related to human factors. Within the human factors risk circle is a third largest circle that represents the total human factors risk minus the human factors risk related to maintenance. Within this third largest circle are two smaller circles that overlap one another. Each of the overlapping circles has a crosshatched area. This crosshatched area, which includes the overlap, represents the direct effect of a human factors element on improvement of human performance. The direct effect is that part of the human factors element that can cause an improvement of human performance. The crosshatched area does not fill the entire circle because it is believed that human error in any human factors element area can not be completely eliminated. The overlapped portion of the crosshatched circles is representative of the indirect effect of other human factors elements that are interrelated.

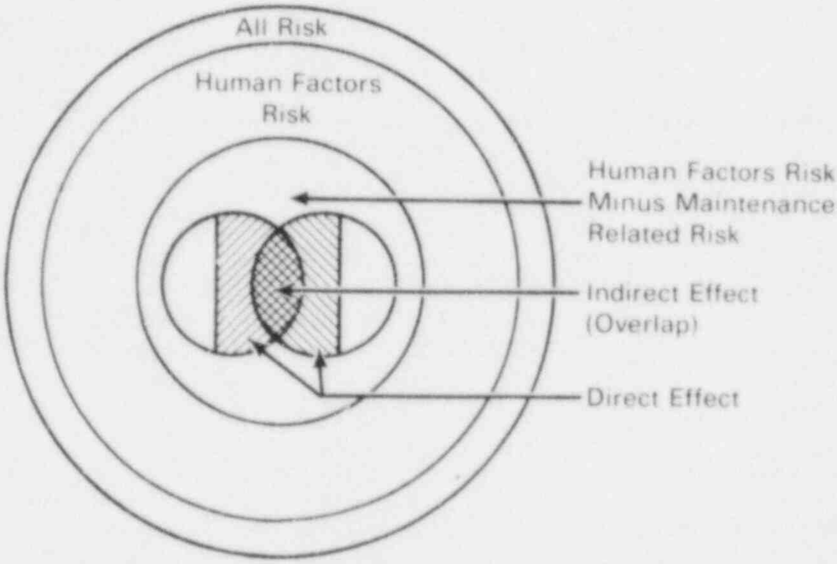

FIGURE 2.1 Illustration of the Levels of Human Factors Risk

2.1.2 Selecting Staff Panel

The staff selected to be on the panel consisted of personnel who either have been or are involved in one or more of NRC's human factors projects or who have background in nuclear reactor research and human factors research. The panel was composed of staff from the PNL and the Human Affairs Research Center (HARC). The staff selected have educational backgrounds in nuclear engineering, industrial engineering, physics, psychology, human factors engineering, business administration, and chemical engineering. They also have practical experience related to nuclear submarines, human factors design criteria for NPP control rooms, NPP operator license examination, NPP emergency preparedness, development of guidelines for writing emergency operating procedures, utility management and organization design, and control room design for NPP and liquefied natural gas plants. Names of those staff who were selected to be on the panel and who evaluated the questionnaire (in Appendix B) are given in Appendix C.

2.1.3 Evaluating Questionnaire

The panel staff were asked to (1) assess the direct effects of resolution of each of the six plan elements on improving human performance at NPPs, (2) assess how much reduction in human error would be expected if resolutions to each of the six plan elements were implemented and (3) assess the indirect effects or the extent to which any five element areas affected the sixth element area. Assessment of the indirect effects involved a seventh area titled "OTHER" (in the questionnaire) that was intended to provide an opportunity for expressing that either areas other than the five listed affected the sixth area or that there was a portion of the sixth area that was independent of all other indirect effects. This approach basically allowed each panel participant to form his own model of the interactions of the various elements on one another and ultimately on improvement in human performance.

2.1.4 Compiling Questionnaire Results

In attempting to model the overlaps that exist between the six human factors program plan elements, four approaches were assessed. Each of these approaches accounts for overlaps by compiling the questionnaire results in a different way. These approaches were not the only possible approaches, but were believed to cover the range of overlap affects. No one approach is believed to accurately model the real overlap that exists between the various plan elements. The first approach simply considered that there were no overlaps among elements. The second approach considered that all overlaps of other elements (on the first element) were removed from the first element and then all overlaps (of the first element) on the other elements were added to the remaining direct effect of the first element. The third approach considered overlap removal and addition based on an initial order of implementation. The fourth approach considered overlap removal and addition based on the most incrementally efficient order of implementation. The reevaluation occurred after each element was implemented. These four approaches were intended to consider the various ways that element overlaps can be accounted for and how these ways affect improvement of human performance assessments. The four approaches are discussed in more detail in the following paragraphs.

The first approach, Approach 1, averaged the direct effects data from the respondents to the questionnaire. Approach A does not consider the interactions or overlaps between the various plan elements. This approach will under or overestimate the effect of individual HFPP elements depending on the relative independence of the activities. For example, training benefits would be underestimated because of the ability to modify the course content to reflect results of other HFPP elements. This method also overestimates the effect of not implementing an element because any overlaps are assumed lost.

The second approach, Approach 2, is based on the assumption that the "OTHER" category, as described in the questionnaire on page B.4 (Appendix B), under indirect effects was a measure of the independence of the HFPP element. This model quantifies the portion of public risk reduction that would be lost if a single element were eliminated from the HFPP. All indirect benefits of the eliminated elements would show up under other elements. Approach 2 involved multiplying the direct effect of an element by its related "OTHER" category to determine the element's independent direct effect. Then the indirect effect of element 1 on the other five element areas was added. For example, consider the examples given in the questionnaire (see Appendix B). The direct effect of PROCEDURES on improvement of human performance is 25 points out of 100. The "OTHER" area related to PROCEDURES was 20 points out of 100. Thus PROCEDURES independent direct effect is calculated to be $25 \times (20/100) = 5$. Then the indirect effects of PROCEDURES on the other five element areas would be added to the PROCEDURES independent direct effect (5). Approach 2 determines each element's total independent effect on improving human performance but is limited by the fact that the indirect effect of element 1 on element 2 implicitly assumes that element 1 is implemented before element 2. This may not be the case.

The remaining two models postulate an order of implementation to calculate incremental HFPP element benefits. Even if the elements are not completed sequentially, these models provide insights on allocation of resources. These models are of the type that could be useful in the periodic review of funding allocations and reassessment of remaining benefits.

The third approach, Approach 3, assumes an order of implementation based on the direct effects ranking of the six element areas. The element that ranked highest is assumed to be implemented first. To calculate the value of the first element, indirect effects of the element on the other five elements were added to the direct effect of the first element. The second element implemented was valued by adding the indirect effects of the second element on the other four elements to the direct effect of the second element minus the indirect effect of the first element (already implemented) on the second. This process continues through the sixth element implemented. The value of the last element is the sum of direct effect of the sixth element minus all the indirect effects of other elements on the sixth element. Approach 3 models the effect of order of implementation of elements on the rating of the elements. This approach assumes an initial order of implementation, given by the direct effects ranking, that does not change during the process of calculating the rating of the elements. This approach also has the limitation that it assumes the direct effects of each element is independent of the other elements.

The fourth approach, Approach 4, is similar to Approach 3 but it assumes a reevaluation of the order of implementation after each element is implemented. The next element to be implemented is selected after subtracting the indirect effects of the implemented elements from the direct effect of unimplemented elements. For example, if the first element is implemented first based on the direct effects ranking, then the second element to be implemented is selected based on the ranking of direct effect of the second element minus the indirect effect of the first element on the second. This is then compared to the direct effect of the third element minus the indirect effect of the first element on the third element and so on. Approach 4 is similar to Approach 3 because it models the effect of order of implementation of elements on the rating of the elements. Approach 4 differs from Approach C because it models the potential for the order of implementation to change from the initial order when the remaining elements are reevaluated after each element is implemented. Approach D also has the limitation that it assumes the direct effects of each element is independent of the other elements.

2.1.5 Using Risk Assessments to Calculate Risk Changes

The questionnaire results that were used to calculate risk changes using the Oconee and Grand Gulf RSSMAP risk assessments were the overall reduction in human error that was judged to be acheiveable by implementing all six of the elements and the compiled ratings of the six elements according to either Approach 1, 2, 3, or 4. The overall reduction in human error value was applied to all operator error parameters in the Oconee and Grand Gulf risk assessments and the resulting risk change was calculated. For example, an overall

reduction in human error value of 10 percent would be applied to an operator error parameter, say Z, by reducing the value of Z by 10 percent. This risk change value is the the total risk change expected if all six plan elements were implemented. This risk change value is multiplied by the individual element ratings to yield individual risk changes for each element according to Approach 1, 2, 3, or 4. An assessment of the risk change from the entire HFPP is the sum of the elements.

2.2 RISK RESULTS

The risk results are presented in the following subsections according to either interim-ratings results or risk change results. The results are given according to each of the four approaches for compiling the questionnaire results. The average overall reduction in human error based on 11 questionnaire inputs was 59.7 percent which is rounded off to 60 percent for this analysis. Thus when all operator error terms in the Oconee and Grand Gulf risk assessments are reduced by 60 percent the resulting risk reduction is 56 man-rem/py for Oconee (PWR) and 43 man-rem/py for Grand Gulf (BWR). Thus 56 man-rem/py and 43 man-rem/py are the total risk reductions assumed if all six HFPP elements were implemented for a PWR and a BWR, respectively.

2.2.1 Interim Rating Results

The interim rating results based on the compilation of 11 questionnaire answers according to Appr aches A, B, C, and D are presented in Table 2.1. These rating results are given as percentage ratings. Thus each element's percentage
rating can be applied to the total risk reduction assumed above for a PWR or a BWR to yield a risk reduction per element.

TABLE 2.1 Interim Results Based on Compiled Questionnaire Answers

	RATINGS (%) BY APPROACH			
ELEMENT AREAS	1	2	3	4
Staffing and Qualifications	11.1	15.9	7.9	0.9
Training	22.3	21.7	32.8	41.6
Licensing Examinations	11.4	10.4	3.0	6.7
Procedures	18.6	16.1	17.1	9.0
Man-Machine Interfaces	17.7	10.9	3.4	14.9
Management and Organization	18.9	25.0	35.8	26.9
TOTAL	100.0	100.0	100.0	100.0

2.2.2 Risk Change Results

The risk change results based on multiplying the total risk reduction assumed for a PWR or a BWR by the interim rating results according to Approaches 1, 2, 3, and 4 are presented in Table 2.2. These risk change results are given as man-rem/py results.

TABLE 2.2 Risk Change Results Based on Interim Rating Results

	RISK REDUCTIONS (MAN-REM/PY) BY APPROACH AND REACTOR TYPE							
	1		2		3		4	
ELEMENT AREAS	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR
Staffing and Qualifications	6.2	4.8	8.9	6.8	4.4	3.4	0.5	0.4
Training	12.5	9.6	12.2	9.3	18.4	14.1	23.3	17.9
Licensing Examinations	6.4	4.9	5.8	4.5	1.7	1.3	3.8	2.9
Procedures	10.4	8.0	9.0	6.9	9.6	7.4	5.0	3.9
Man-Machine Interfaces	9.9	7.6	6.1	4.7	1.9	1.5	8.3	6.4
Management and Organization	10.6	8.1	14.1	10.8	20.0	15.4	15.1	11.6
TOTAL	56.0	43.0	56.1	43.0	56.0	43.1	56.0	43.1

Based on 90 PWRs with average lives of 28.8 years and 44 BWRs with average lives of 27.4 years, the total industry risk reduction by approach is given in Table 2.3.

TABLE 2.3 Total Industry Risk Change Results

ELEMENT AREAS	RISK REDUCTIONS (MAN-REM) BY APPROACH A	B	C	D
Staffing and Qualifications	2.2E+04	3.1E+04	1.6E+04	1.8E+03
Training	4.4E+04	4.3E+04	6.5E+04	8.2E+04
Licensing Examinations	2.2E+04	2.0E+04	6.0E+03	1.3E+04
Procedures	3.7E+04	3.2E+04	3.4E+04	1.8E+04
Man-Machine Interfaces	3.5E+04	2.1E+04	6.7E+03	2.9E+04
Management and Organization	3.7E+04	5.0E+04	7.0E+04	5.3E+04
TOTAL	2.0E+05	2.0E+05	2.0E+05	2.0E+05

3.0 HUMAN FACTORS PROGRAM PLAN COSTS

This section describes the industry and NRC resources and costs that were assumed to be used in the development, implementation and operation of resolutions to six portions of the HFPP. As in the risk analysis, maintenance has been excluded from this assessment. Each of the six areas are discussed in terms of the activities currently perceived as necessary for their resolution, the effect of these activities on cost, quantification of cost, and the overlap of each cost with other issues.

3.1 APPROACH

The costs associated with generic safety issues examined by PNL were used to bound the HFPP costs. The entire safety issue program was examined to find those with scope applicable to the human factors program. The scope of each safety issue was then examined and assigned on a percentage basis to specific HF Program elements. Issues impacting maintenance will be identified and a percentage of issue scope assigned to this factor. Any overlaps in scope were allocated in this fashion.

The safety issue costs for NRC and utility development, implementation, and operation were examined and assigned to HF Program elements with the same percentage weights determined for the scope as described above. Costs associated with maintenance were removed.

The issues and costs were then tabulated by program element. This approach will result in a total projected cost for all unresolved safety issues associated with each specific HF Program element.

Finally, the combined scope of all the safety issues were examined in light of the scope for the individual HF Program elements. If the research identified was adequate given the program plans, the costs estimated above served as a first estimate for that HF Program element. If the research proposed for the safety issues was less comprehensive than the HF Program element, additional costs to cover the remaining scope were estimated.

3.2 DEFINITION OF HF PROGRAM PLAN ELEMENTS AND SCOPE

In this section, the individual program elements are reviewed for the purpose of defining the scope. This will serve as the basis of comparison with scope statements from the generic safety issues. Detailed items to be covered by the HFPP as defined by the NRC are described for reference in Appendix A.

3.2.1 Staffing and Qualifications

Consideration was given under this element to (1) the numbers and functions of the staff needed to safely perform all required plant operations and technical support for each operational mode; (2) the minimum qualifications of plant personnel, in terms of education, skill, knowledge, training, experience, and fitness for duty; and (3) appropriate limits and conditions for shift work, including overtime, shift duration, and shift rotation.

Areas of Cost Impact

This element affects both industry and NRC costs related to staffing levels and higher costs for some existing positions due to increased qualifications. Other costs will be incurred in the development of criteria and models, performing assessments of the current needs for improvement and implementation of changes. This area is believed to have some potential for improvement in plant availability through the mitigation of plant transients prior to scram. Improvements in worker efficiency are also possible through the rigorous use of Job Task Analysis. However, it is anticipated that this item will incur additional costs for industry and the NRC.

3.2.2 Training

The Nuclear Waste Policy Act of 1982 PL 97-425 (Section 306), directs the NRC to promulgate regulations and regulatory guidance for the training of nuclear power plant personnel. Areas to be addressed include simulator training requirements, operator requalification programs, team training, instructional requirements for training programs, and administration of examinations. The planned activities in this task and its end products support this act.

Areas of Cost Impact

Implementation of this issue will require the NRC to develop new regulations, evaluate the value of plant-specific simulators and provide a periodic review of training programs. Industry costs for implementation will include additional costs for training materials and services, a review of their current training programs and the addition or upgrading of plant specific simulators. Operational costs will include simulator operation and additional staff time for attendance of training courses. This could require the hiring of additional staff. Cost savings could result from the avoidance of some unplanned outages due to appropriate operator response to transients and reduced errors.

3.2.3. Licensing Examinations

Licensing exams for reactor operators and senior reactor operators will be used to measure the operator's ability to perform the necessary tasks and functions required to safely operate and control commercial nuclear power plants. Examinations and the examination process will be practiced and administered in a consistent manner by the various NRC examiners. The licensing examination is to be correlated to the improved training programs.

Areas of Cost Impact

Costs associated with this issue are primarily related to the NRC. Costs will be incurred during development of (1) new regulations, (2) a data base of test questions, (3) examination and validation strategies, (4) new examination procedures and (5) additional training for licensing examiners. Use of simulators as an examination tool will be defined. Industry costs will be affected in the development of new training courses. These courses are to be implemented under the Training issue.

3.2.4 Procedures

Procedures are to be improved by: (1) developing guidelines for preparing criteria and evaluating emergency operating, normal operating and the other procedures which affect plant safety, and (2) upgrading the procedures, training the operators in their use, and implementing the upgraded procedures.

Areas of Cost Impact

NRC will expend resources for the development of guidelines and criteria to aid in implementing new procedures, fund research for the development of procedure display and training techniques, and have a continuing program to assess the adequacy of operator training. Industry costs will accrue to provide technical guidance to the NRC in the development of procedure guidance, implementation of the results and operator time for additional training in off-normal events.

3.2.5 Management and Organization

This task will: (1) provide the guidance and requirements to define necessary management and organizational practices; (2) develop criteria and reliable evaluation objectives for staff use to assess the effectiveness of management and organizational functions; and (3) cooperate with INPO in the development of a training program for prospective plant managers and other appropriate utility managers.

Areas of Cost Impact

NRC costs will be effected through review and evaluation of the INPO efforts in this area. It is anticipated that ongoing NRC costs will be reduced by transferring some of the responsibility for performance appraisal to INPO. This will also reduce the utility costs over the current practice of NRC reviews due to more efficient reviews and sharing of utility experience. Remaining NRC reviews are also expected to change in scope. This will require the NRC to develop new assessment procedures and the industry to modify their approach to the NRC reviews. Reductions in unscheduled outages are expected from the implementation of this issue.

3.2.6 Man-Machine Interface

This task will develop (1) human factors engineering guidelines for correcting man-machine interface problems; and (2) regulatory guidance for integrating human factors engineering into new designs and into advanced technological improvements incorporated into existing designs. This activity will also provide for the preparation of evaluation tools for the next generation of nuclear power plants and for expected changes or upgrading of designed plants in the areas of data and information management and improved annunciator systems.

Areas of Cost Impact

NRC costs will cover the evaluation of additional guidance to cover design of local control centers and clarification of near-term control room changes covered under previous guidance. Industry will have costs associated with implementation of these guidelines through equipment modifications, additional training, and maintenance. The NRC will also incur costs associated with the evaluation of new technologies for application to existing and future plants.

3.3 CORRELATION OF SAFETY ISSUES SCOPE TO HF PROGRAM PLAN ELEMENTS

In this section, the generic safety issues identified by PNL as being associated with the HFPP will be identified. Their individual costs and overlaps with other program elements will be used as the method for bounding the scope and cost of the HFPP.

3.3.1 Identification of Generic Safety Issues

The safety issues identifed as being related to this program are shown in Table 3.1. These issues were examined to determine which program element applies. Estimates were then made as to the percentage of scope and cost that is related to each element of the current HFPP.

TABLE 3.1. Safety Issues Associated with Human Factors Program

I.A.1.4	Operating Personnel and Staffing: Long-Term Upgrades
I.A.2.2	Training and Qualifications of Operating Personnel
I.A.2.4	NRR Participation in Inspector Training
I.A.2.6(4)	Long-Term Upgrade of Training and Qualifications (Training Workshops)
I.A.2.6(6)	Nuclear Power Fundamentals for Operator Training
I.A.2.6(1-3,5)	Simulators
I.A.2.7	Accreditation of Training Institutions
I.A.3.2	Operator Licensing Program Changes
I.A.3.3	Requirements for Operator Fitness
I.A.3.4	Licensing of Additional Operations Personnel
I.A.4.2	Long-Term Training Simulator Upgrade
I.B.1.1(5-7)	Management for Operations: Organization and Management of Long-Term Improvements
I.C.9	Long-Term Program Plan for Upgrading Procedures
I.C.1(4)	Confirmatory Analysis of Selected Transients
I.D.3	Safety System Status Monitoring
I.D.4	Control Room Design Standard
I.D.5(3-5)	Control Room Design: Improved Control Room Instrumentation Research
II.J.3.1-2	Organization and Staffing to Oversee Design and Construction, Issue Regulatory Guide

3.3.2 Discussion of Safety Issue Scope and Costs

The scope and major cost contributors to each issue are discussed in this section.

3.3.2.1 Issue I.A.1.4 Operator and Staffing: Long-Term Upgrades

This issue is considered resolved with no further significant associated costs.

3.3.2.2 Issue I.A.2.2 Training and Qualifications of Operating Personnel

This issue deals with the periodic review of licensed and auxiliary operators, technicians, maintenance personnel, and supervisors. Based on the testing results, the justification for no need of further training is documented or the training and qualifications are upgraded as required.

The HFPP elements that apply and the degree to which they apply to this issue are Training (25 percent), Staffing and Qualifications (50 percent), and Licensing (25 percent).

The scope includes maintenance and technician testing, so it will be assumed that only 50 percent of the costs identified below are applicable to the HF Program.

Costs

NRC costs for development and implementation were established to be 0.55 man-yrs for each, or $55K each. Ongoing costs were established at 1 man-yr per year, or $100K/yr for a total of $2.8M after 28.3 years.

Industry implementation costs were estimated to be $335K/plant, which gives $45M for 134 plants. Ongoing costs were established at $160K/py, which gives $21.4M/yr and $607M after 28.3 years.

3.3.2.3 Issue I.A.2.4 NRR Participation in Inspector Training

This TMI action item would have IE inspector training augmented with input from NRR personnel familiar with operator licensing and human factors problems.

This issue involves the input of human factors research, but the inspection process is aimed at plant hardware and routine practices as well as emergency procedures. Thus, a large portion of the benefit from this issue will not be directed towards the operator. It will be assumed here that 30 percent of the scope/costs discussed below will be associated with the HF program.

This issue is assumed here to be 100 percent associated with the Training program element.

Costs

Industry costs for implementation of this issue are associated with actions of the IE inspector. These were estimated by PNL at $250K/reactor for the first year, for a total of $33.5M for 134 plants. Ongoing costs were estimated to be $100K/py, giving $13.4M/yr for a total of $380M for 134 plants over 28.3 years.

NRC development and implementation costs were estimated to be one man-yr, or $100K. This is assumed to be equally divided. Ongoing NRC costs were estimated to be 67 man-weeks per year, or $152K/yr. For 28.3 years this gives $4.3M.

3.3.2.4 Issue I.A.2.6(4) Long-Term Upgrading of Training and Qualifications: Training Workshops

The intent of this issue is to conduct seminar-type workshops to exchange information between the NRC and licensed utility staff. This would assist in improving operator performance.

If additional staff such as maintenance personnel are licensed, they may take advantage of the same workshop approach to share information. However, the original intent of this issue was directed at operators. For now, it will be assumed that 100 percent of the scope/costs discussed below apply to the HF program.

The issue deals entirely with upgrading the performance of existing staff, and presents no new qualifications or criteria for new hires. As a result, this issue is thought to be 100 percent related to the Training program element.

Costs

For costs, PNL estimated $1E+05 or 1 man-yr for issue development by the NRC, and an additional $2E+05 or 2 man-years for implementation. Annual costs were put at $3E+05. For the remaining 28.3 year lifetime assumed, this gave a total of $8.5E+06.

For industry, one man-month per plant, or $8300/plant was assumed for implementation. For 134 plants, this gave $1.1M. Ongoing costs were estimated to be put at one man-month per plant per year, or $1.1M per year. Again for 28.3 years this gave a total ongoing cost of $32M.

3.3.2.5 Issue I.A.2.6(6) Long-Term Upgrading of Training and Qualifications: Nuclear Power Fundamentals for Operator Training

This TMI action item calls for the NRR to establish definitive instructional requirements for inclusion in reactor operator training courses.

This issue was seen to have no safety benefit, and NUREG 0933 carries it as a drop item. Therefore, costs for this issue were not included for the HFPP.

3.3.2.6 Issue I.A.2.6(1,2,3,5) Long-Term Upgrading of Training and Qualifications: Simulators

This TMI action item calls for the upgrading of training through the use of simulators in training, requalification, and testing. This action is assumed to provide a major enhancement of training and requalification for reactor operators.

This issue deals with operators, so the scope/costs discussed below are assumed to apply 100 percent to the HF program.

This issue thus deals primarily with the training of existing operator staff at utilities. As the role of simulators becomes the industry requirement, it will likely play a significant role in defining licensing requirements. The requalification program will also likely define simulator time requirements for staff qualification.

The HFPP elements that apply and the degree to which they apply to this issue are Training (70 percent), Licensing (10 percent), Procedures (10 percent), Qualification (5 percent), and Man-Machine Interface (5 percent).

Costs

Utility costs were estimated by PNL to represent a major effort for implementation at an average of 12 man-years per plant, or \$160M. Ongoing costs were established at 3.46 man-years per plant year, or \$346K/py, plus \$160K/py in additional simulator time, giving a total of \$506K/py. For 134 reactors this yields \$67.8M/yr, and for 28.3 years, gives a total for ongoing costs of \$1920M.

For NRC, NUREG-0660 put the effort at 5.4 man-yrs and \$259K. Much of this is completed, but PNL estimated the remaining time and cost requirements at 4.5 man-yrs and \$100K divided between development and implementation. This gives \$0.550M, or approximately \$0.28M each.

For ongoing costs, it was assumed that the NRC would perform annual reviews and inspections plus conduct a portion of the requalification examinations. This effort was estimated to require 14.2 man-yrs per year, or \$1.4M/yr, for a total ongoing cost of \$40M

3.3.2.7 Issue I.A.2.7 Accreditation of Training Institutions

This TMI action plan seeks to ensure consistently high quality training by establishing a means and system for accreditation of institutions and programs that provide training for reactor operators.

Participation in such schools or programs will most likely be used for meeting NRC staffing experience requirements. Thus, formation of this system will require interaction with the Qualifications element. Interaction with the Licensing element will also be required to some degree. However, the primary

thrust of this issue is directed towards ensuring the technical merit of the training programs in question. The bulk of the work is thus expected to be directed towards the Training element.

This issue deals with operators, so the scope/costs discussed below are assumed to apply 100 percent to the HF Program.

The effect of this issue is thus estimated here to be split 80 percent to Training, and 10 percent each to Qualifications and Licensing.

Costs

The industry cost to implement the issue was estimated by PNL to be \$300K per reactor, for a total of \$36M for 121 reactors. The ongoing operation and maintenance costs were estimated to be \$100K per plant per year. This gives a total ongoing cost of \$12.1M/yr or \$340M after 28.3 years.

The cost for NRC development and implementation was established as \$250K each, for a total of \$500K. The annual ongoing cost was estimated to be one man-year, or \$100K, for a total of \$2.8M over 28.3 years.

3.3.2.8 Issue I.A.3.2 Operator Licensing Program Changes

This issue is considered to be resolved, with no significant future costs associated with it.

3.3.2.9 Issue I.A.3.3 Requirements for Operator Fitness

This issue would implement background searches and psychological assessments for operators before hiring, and behavioral observations after hiring. This would include screening for drug and alcohol abuse and criminal background.

This issue is 100 percent related to the HF program.

The Qualifications program elements will have the primary effect on this issue, but the Licensing element will also have some impact. It was assumed that the split for this issue was Qualifications (90 percent) and Licensing (10 percent).

Costs

The NRC cost for development and implementation was estimated to be 1.5 man-yrs, or \$150K each. Ongoing costs were established as 1 man-week/py, giving \$300K/yr and \$8.6M after 28.3 years.

Industry implementation costs were put at \$140K for existing plants and \$590K for new plants, giving \$47M. Ongoing costs were put at \$300K/py, giving \$40.2M/yr and a total of \$1140M over 28.3 years.

3.3.2.10 Issue I.A.3.4 Licensing of Additional Operations Personnel

This issue would seek licensing requirements for managers, engineers, technicians, etc. This is considered to require a major effort on the part of the NRC to develop and implement requirements and regulations. Implementation will require issuing a REG Guide. The industry cost for implementing the requirements is similar to NRC costs. The operational costs are also very high.

This issue represents a major effort to license personnel associated with plant operations and maintenance, which are outside the scope of this program. It will be assumed here that only 25 percent of the scope/costs discussed below apply here.

This issue impacts Licensing primarily, but also impacts Qualifications and to some extent Management and Organization by requiring line management to be licensed personnel.

It will be assumed here that the HFPP elements germane to this issue are Licensing (70 percent), Qualifications (20 percent), and Organization (10 percent).

Costs

NRC costs to develop this issue were estimated to be \$30M. Implementation was established as \$5M. No estimate of staff time required was made. Ongoing costs were estimated to be 50 man-yrs per year, plus \$7500/py travel and publication costs, giving \$6M/yr and a total ongoing cost of \$170M over 28.3 years.

Industry costs for implementation were established as \$35M. Ongoing costs were estimated to be \$50K/py, or \$6.7M/yr and \$190M over 28.3 years.

3.3.2.11 Issue I.A.4.2 Long Term Training Simulator Upgrade

This issue deals with the long term upgrade of simulators in terms of research to 1) improve the use of simulators, 2) develop guidance on the need for and nature of operator actions, and 3) gather data on operator performance.

This issue deals with operators, so 100 percent of the scope/costs discussed below are assumed to apply to the HF program.

This issue deals somewhat with training, but does not really affect current or proposed near-term simulator reqirements. It is likely that some work on future control room requirements will develop out of training experiences and recommendations. The PNL analysis of this issue assumed that this issue dealt primarily with upgrading the realism of simulators to make the simulators more site specific in nature. This issue is thought to be primarily directed towards a better definition of the role of the operator with respect to plant controls and the need for site-specific simulators. Thus, it impacts on the Man-Machine Interface program element. This work will have to be integrated to some extent with plant procedures if the need for site specific control-room simulators is to be defined.

Because of the long term aspects of the issue, it will be assumed that this contributes 25 percent to Man Machine Interfaces, 25 percent to Procedures, and 50 percent to Training.

Costs

Industry costs for implementation of this issue were assumed to require 62 new simulators spread over 134 reactors, giving a capital cost of $3.2M/reactor and 3.2 man-yrs/reactor for new training. This total is $470M. Ongoing utility costs were established as 1.4 man-yrs per reactor/year or $18.8M/yr for a total of $530M.

No developmental NRC costs were estimated. Implementation costs were estimated at one NRC staff-yr, and $1M contractor effort, plus 248 man-weeks for confirmatory site reviews. This yields 6.6 man-years plus $1M, or $1.7M implementation cost. Ongoing annual costs are estimated to be 124 man-weeks/yr for site review, or $0.28M/yr for 28.3 years giving a total ongoing cost of $8M.

3.3.2.12 Issue I.B.1.1(5-7) Management for Operations: Organization and Management of Long-Term Improvements

The original stated objective of this TMI action item was to improve plant groups responsible for radiation protection and operation. This included staffing size, education, experience, etc.

Because of the overlap of this issue in training and qualifications with other issues, the scope of I.B.1(5-7) was modified by PNL to consider long range improvements to utility management and organization during plant design, construction, startup, and operation. This issue still however has some recognized overlap with other program areas, primarily Staffing and Qualifications, and Training.

Approximately 60 percent of industry costs associated with this issue deal with maintenance. As a result, it will be assumed that only 40 percent of the scope/costs discussed below apply to the HF program.

It will be assumed here that this issue impacts the following HFPP elements: Management (60 percent), Procedures (15 percent), Staffing (10 percent), Training (10 percent), and Licensing (5 percent).

Costs

PNL estimates significant cost savings will result from this issue because of increased plant availability. A discussion of these costs is provided in Section 3.5. NUREG 0933 estimated a major NRC effort of 22 man-yrs to develop a regulatory position on management and organization, and 2 man-yrs to implement the new regulatory guides and 5 man-month/plant to review the initial management and organization plan for each of 134 plants. These totals are $2.2M and $5.8M, respectively.

One-half man-month per plant year was estimated for ongoing costs, giving $0.56M/yr and a total of $15.8M after 28.3 years.

Industry costs for implementation were estimated from NUREG 0933 to be 2.75 man-years/plant giving $275K/plant for the 134 plants giving $37M.

For utility operating costs, NUREG 0933 estimates an average 7 man-years per plant including maintenance, or $700K/py would be required, giving $93.8M/yr and a total of $2655M after 28.3 years.

3.3.2.13 Issue I.C.9 Long Term Program Plan for Upgrading of Procedures

This TMI action item calls for a major effort to improve and coordinate plant procedures. This would include integrating and expanding on current efforts.

This issue is directed primarily at emergency procedures for plant operators, so 100 percent of the scope/cost discussion below is assumed to apply to the HF program.

This issue deals primarily with procedures, but will have obvious impacts on operator training and licensing requirements. It will be assumed that the program impact is divided with 15 percent applied to Training, 15 percent to Licensing, and 70 percent to Procedures.

Costs

The PNL estimate for utility implementation of this issue was estimated to be 5 man-yrs per reactor. At $500K per reactor, the total implementation cost for 134 reactors comes to $67M. Continuing costs were established as 1 man-yr/reactor-year, giving $13.4M/yr and a total of $379M after 28.3 years.

For NRC development and implementation, NUREG-0660 estimates 4.9 man-yrs of NRC staff time. PNL increased this effort to $1.5M with no direct estimate of staff requirements due to the likely use of contractors. The costs and staff time will be split here between development ($1.0M) and implementation ($500K), with the 4.9 man-yr NRC staff time estimate still used and split evenly between development and implementation. Ongoing NRC review of procedures was estimated to be 2.5 man-yrs per year, which comes to $7.1M after 28.3 years.

3.3.2.14 Issue I.C.1(4) Short-Term Accident Analysis and Procedures Revision: Confirmatory Analysis of Selected Transients

This TMI action item seeks to perform confirmatory analyses of selected transients by NRR, providing a basis for comparison with methods used by vendors. The object of this is to improve the quality of procedures, and to ensure that the operator and staff actions are correct.

This issue is assumed to apply 100 percent to the HF program in the scope/cost discussions below.

This issue is assumed to contribute primarily to the Procedures program area. However, results from this study will also provide input into Operator Training and Licensing requirements to some degree. It will be assumed here that this issue contributes 70 percent to Procedures, 15 percent to Training, and 15 percent to Licensing.

Costs

Development costs to the NRC were estimated by PNL to be 0.5 man-yrs per transient for 32 transients, which comes to 16 man-yrs and $1.6M. Computer time would raise this to approximately $1.8M. The split between NRC staff time and contractors is not given. Implementation costs were estimated to be 1.1 man-yrs, or $110K. Ongoing costs were established as 0.5 man-yrs per year. This gives $50K/yr and $1.4M after 28.3 years.

Industry costs for implementation were estimated to be 30 man-weeks per plant, or $9.1M for 134 plants. Ongoing costs were established as 7 man-weeks/py, or $2.1M/yr for 134 plants, and $60M after 28.3 years.

3.3.2.15 Issue I.D.3 Safety System Status Monitoring

This issue will study the need for utilities to implement an automatic status monitoring system. This issue is seen to go beyond Reg Guide 1.47, and would involve backfitting 71 operating plants.

This issue deals with operators, so the scope/costs discussed below are assumed to apply 100 percent to the HF program.

This issue is assumed to contribute 80 percent to Man-Machine Interface, 10 percent to Training, and 10 percent to Procedures, as it will play an important role in operator interactions and simulator training for accident analysis.

Costs

Industry implementation costs were estimated to be $761K/plant for 71 existing plants, or $54M.

Ongoing industry costs were established as 1 man-week/py, giving $0.16M/yr for 71 plants, and $4.3M after the average remaining life of 26.9 years.

NRC costs for development were estimated to be 0.5 man-yrs, or $50K. Implementation costs were established as 4 man-weeks/plant, or 284 man-weeks and $0.6M.

Ongoing operational costs were estimated to be 0.5 man-weeks/py, or $80K/yr for 71 plants and a total ongoing cost of $2.2M after 26.9 years.

3.3.2.16 Issue I.D.4 Control Room Design Standard

The purpose of this issue is to develop guidance on the design of control rooms to incorporate human factors. Control rooms would then be constructed in accordance with a regulatory guide.

This issue deals with operators, and so the scope/costs discussed below are assumed to apply 100 percent to the HF program.

Dealing primarily with design standards, this issue will be assumed to contribute 80 percent to Man Machine Interface, and 10 percent to Training and Procedures.

Cost

For plants completed after 1986 and before 1990, utility implementation of this issue was assumed to cost $100K per plant. For 10 plants, this is $1.0M. No additional ongoing costs were identified.

For the NRC, $300K was estimated for development, and 4 man-weeks/plant or $91K for implementation for the 10 plants. No additional operation and maintenance costs were estimated.

3.3.2.17 Issue I.D.5(3-5) Control Room Design: Improved Control Room Instrumentation Research

The objective of this issue is to develop new instrumentation and advanced diagnostics to improve the ability of the operator to prevent and cope with accidents.

This issue deals with operators, so the scope/costs discussed below are assumed to apply 100 percent to the HF program.

This issue still deals with Man-Machine Interface, but also introduces accident analysis and operator training to a small extent as with the previous issues. It will be assumed here that this issue is 80 percent Man-Machine Interface, and 10 percent Training and Procedures.

Costs

Industry implementation costs were estimated to be $2M/plant for 134 plants to cover the cost of advanced diagnostic and surveillance systems for a total of $268M.

Ongoing utility costs are established as 10 man-weeks/py, or $3M/yr and $86M after 28.3 years.

Note that the PNL analysis estimated a cost savings due to reduced down time of $184K per PWR plant year and $211K per BWR plant year for a total savings of $732M over 28.3 years.

Based on NRC costs for diagnostics, surveillance, and vessel liquid level systems in FY83, PNL estimated the NRC development costs at $1.98M.

Implementation costs for only the 71 operating plants were estimated to be 4 man-weeks per plant, or $645K.

No additional ongoing costs were estimated.

3.3.2.18 Issue II.J.3.1-2 Organization and Staffing to Oversee Design and Construction

This issue is addressed as part of Issue I.B.1.1 and, therefore, not considered separately here.

This issue is addressed as part of Issue I.B.1.1 and, therefore, not considered separately here.

3.3.3 Scope of Safety Issues by Dominant Human Factors Program Element

The generic safety issues presented above can now be categorized by their role in the Human Factors Program. Based on the scope discussions presented above, the issues are shown in Table 3.2.

The percentage HF figure again gives the overall estimate of scope overlap with the Human Factors Program. That fraction is then divided as shown with the estimates for scope overlap with the individual program elements.

The issues are listed under the HF program element with the largest common scope overlap.

TABLE 3.2. Allocation of Safety Issues Impact to Human Factors Program

Issue Title		% of Scope H	SQ	T	LE	P	MO	MMI
STAFFING AND QUALIFICATIONS (SQ)								
I.A.1.4	Operating Personnel and Staffing: Long Term Upgrades (Resolved)	-	0	0	0	0	0	0
I.A.2.2	Training and Qualifications of Operating Personnel	50	50	25	25	0	0	0
I.A.3.3	Requirements for Operator Fitness	100	90	0	10	0	0	0

TABLE 3.2. (Continued)

Issue Title		%H	% Scope SQ	T	LE	P	MO	MMI
TRAINING (T)								
I.A.2.6(4)	Long-Term Upgrade of Training and Qualifications (Training Workshops)	100	0	100	0	0	0	0
I.A.2.6(6)	Nuclear Power Fundamentals for Operator Training (Resolved)	-	0	0	0	0	0	0
I.A.2.6(1,2,3,5)	Simulators	100	5	70	10	10	0	5
I.A.2.7	Accreditation of Training Institutions	100	10	80	10	0	0	0
I.A.2.4	NRR Participation in Inspector Training	30	0	100	0	0	0	0
I.A.4.2	Long-Term Training Simulator Upgrade	100	0	50	0	25	0	25
LICENSING EXAMINATIONS (LE)								
I.A.3.4	Licensing of Additional Operations Personnel	25	20	0	70	0	10	0
I.A.3.2	Operator Licensing Program Changes (Resolved)	-	0	0	0	0	0	0
PROCEDURES (P)								
I.C.9	Long-Term Program Plan for Upgrading Procedures	100	0	15	15	70	0	0
I.C.1(4)	Confirmatory Analysis of Selected Transients	100	0	15	15	70	0	0
MANAGEMENT AND ORGANIZATION (MO)								
I.B.1.1(5-7)	Management for Operations: Organization and Management of Long-Term Improvements	40	10	10	5	15	60	0
II.J.3.1/II.J.3.2	Organization and Staffing to Oversee Design and Construction[a]	-	-	-	-	-	-	-
MAN-MACHINE INTERFACE (MMI)								
I.D.3	Safety System Status Monitoring	100	0	10	0	10	0	80
I.D.4	Control Room Design Standard	100	0	10	0	10	0	80
I.D.5(3-5)	Control Room Design: Improved Control Room Instrumentation Research	100	0	10	0	10	0	80

(a) Note the overlap discussion in Section 3.3.2.18.

3.3.4 Scope Adjustment

Finally, the scope of these combined safety issues was examined to see if they gave a comprehensive coverage of the stated goals of the individual Human Factors program elements. The possibility of duplication of effort among issues must also be considered.

As a first estimate, it was judged that the safety issues do in fact give a sufficient coverage of scope so that costs will be representative for the Staffing and Qualification, Training, Licensing Examination, and Man-Machine Interface program elements. Excessive overlap between issues was not readily apparent.

The Procedures element was covered by only two issues. However, its role in further development with training, and especially advanced control room design (MMI) was recognized. As a first estimate, the values in Table 3.2 will be used.

The Management and Organization element was judged to be lacking in coverage of scope as represented by the safety issues. Issue II.J.3.1 especially had no input to this program, however an issue could be drafted along similar lines to ensure consideration of human factors associated problems in management circles during stages of design, construction, and modification. An approach similar to I.A.3.3 which involves NRR human factors input to IE inspector training could also be applied to the training of engineers undergoing training to oversee design and construction. These engineers would then play a strong management role after construction, as with Issue II.J.3.1. This area has also been recently recognized as one where substantial committment and improvement on the part of utilities may be needed. As a result of this, it will be assumed that the costs identified by Issue II.J.3.1 will also be included here as representative of a similar type of program for increasing management awareness of human factors requirements in their organizational structure.

3.4 CORRELATION OF GENERIC SAFETY ISSUE COSTS TO HF PROGRAM PLAN

In this section, the correlation between generic safety issue and program element costs is tabulated.

3.4.1 Generic Safety Issue Costs

The raw data given earlier for safety issue cost is gathered in the following two tables for NRC and utility costs. These costs include development, implementation, and ongoing operational costs throughout the remainder of the affected plant lifetimes.

Note again that not all of these safety issues apply 100 percent in scope to the Human Factors Program. The costs were adjusted below to reflect this.

TABLE 3.3. Total NRC Generic Safety Issue Cost Requirements

Issue	Development		Implementation		Op & Maintenance	
	\$,M	man-yrs	\$,M	man-yrs	\$/yr,M	Total
I.A.1.4	0	0	0	0	0	0
I.A.2.2	0.055	0.55	0.055	0.55	0.1	2.8
I.A.2.4	0.05	0.5	0.05	0.5	0.15	4.3
I.A.2.6(4)	0.1	1	0.2	2	0.3	8.5
I.A.2.6(6)	0	0	0	0	0	0
I.A.2.6(1-3,5)	0.28	2.8	0.28	2.8	1.4	40
I.A.2.7	0.25	2.5	0.25	2.5	0.1	2.8
I.A.3.2	0	0	0	0	0	0
I.A.3.3	0.15	1.5	0.15	1.5	0.3	8.6
I.A.3.4	30.0	?	5.0	?	6.0	170
I.A.4.2	0	0	1.66	6.6	0.28	8.0
I.B.1.1(5-7)	2.2	22	5.8	5.8	0.56	15.8
I.C.1(4)	1.8	16	0.11	1.1	0.05	1.4
I.C.9	1.0	2.45	0.5	2.45	0.25	7.1
I.D.3	0.05	0.5	0.64	6.4	0.08	2.2
I.D.4	0.3	3.0	0.09	0.9	0	0
I.D.5(3-5)	1.98	?	0.64	6.4	0	0
TOTAL	38.22	52.8+?	15.43	91.7+?	9.57	271.50

(?) indicates NRC staff requirements not estimated

TABLE 3.4. Utility Generic Safety Issue Cost Requirements

Issue	Implementation $,M	Op & Maintenance $/yr,M	Total,M
I.A.1.4	0	0	0
I.A.2.2	45	21.4	607
I.A.2.4	33.5	13.4	380
I.A.2.6(4)	1.1	1.1	32
I.A.2.6(6)	0	0	0
I.A.2.6(1-3,5)	160	67.8	1920
I.A.2.7	36	12.1	340
I.A.3.2	0	0	0
I.A.3.3	47	40.2	1140
I.A.3.4	35	6.7	190
I.A.4.2	470	18.8	530
I.B.1.1(5-7)	37	93.8	2655
I.C.1(4)	9.1	2.1	60
I.C.9	67.0	13.4	379
I.D.3	54	0.16	4.3
I.D.4	1.0	0	0
I.D.5(3-5)	268	3	86
TOTAL	1263.7	294.0	8323.3

3.4.2 Estimated Cost of Human Factors Program

Combining the results of Table 3.2 giving the assignment of scope to the program, and Tables 3.3 and 3.4 which give estimated costs will allow a first rough estimate of the costs associated with the Human Factors Program Plan. This has been done for both NRC and Industry costs, as given below in Tables 3.5 and 3.6.

TABLE 3.5. NRC Human Factors Program Funding Requirements

Issue	Development $,M	Development man-yrs	Implementation $,M	Implementation man-yrs	Op & Maintenance $/yr,M	Op & Maintenance Total $,M
STAFFING AND QUALIFICATIONS						
I.A.2.2	0.0138	0.1375	0.0138	0.1375	0.025	0.7
I.A.3.3	0.135	1.35	0.135	1.35	0.27	7.74
I.A.2.6(1-3,5)	0.014	0.1·	0.014	0.14	0.07	2
I.A.2.7	0.025	0.25	0.025	0.25	0.01	0.28
I.A.3.4	1.5	?	0.25	0	0.3	8.5
I.B.1.1(5-7)	0.088	0.88	0.232	2.32	0.0224	0.632
TOTAL	1.776	2.758+?	0.670	4.198	0.697	19.85
NRC Budget						
FY83	2.3	2.6				
FY84	1.2	4.2				
TRAINING						
I.A.2.2	0.0069	0.0688	0.0069	0.0688	0.0125	0.35
I.A.2.6(4)	0.1	1.	0.2	2.	0.3	8.5
I.A.2.6(1-3,5)	0.196	1.96	0.196	1.96	0.98	28
I.A.2.7	0.2	2.	0.2	2.	0.08	2.24
I.A.2.4	0.015	0.15	0.015	0.15	0.045	1.29
I.C.1(4)	0.27	2.4	0.0165	0.165	0.0075	0.21
I.B.1.1(5-7)	0.088	0.88	0.232	2.32	0.0224	0.632
I.D.3	0.005	0.05	0.064	0.64	0.008	0.22
I.D.4	0.03	0.3	0.009	0.09	0	0
I.D.5(3-5)	0.198	?	0.064	0.64	0	0
I.A.4.2	0	0	0.83	3.3	0.14	4.
I.C.9	0.15	0.3675	0.075	0.3675	0.0375	1.065
TOTAL	1.259	9.176+?	1.908	13.70	1.633	46.51
NRC Budget						
FY83	1.45	4.3				
FY84	1.2	4.2				

TABLE 3.5 (Continued)

Issue	Development $,M	Development man-yrs	Implementation $,M	Implementation man-yrs	Op & Maintenance $/yr,M	Op & Maintenance Total $,M
LICENSING EXAMINATIONS						
I.A.2.2	0.0069	0.0688	0.0069	0.0688	0.0125	0.35
I.A.3.4	5.25	?	0.875	0	1.05	29.75
I.A.3.3	0.015	0.15	0.015	0.15	0.03	0.86
I.A.2.6(1-3,5)	0.028	0.28	0.028	0.28	0.14	4
I.A.2.7	0.025	0.25	0.025	0.25	0.01	0.28
I.A.4.2	0	0	0	0	0	0
I.C.9	0.15	0.3675	0.075	0.3675	0.0375	1.065
I.C.1(4)	0.27	2.4	0.0165	0.165	0.0075	0.21
I.B.1.1(5-7)	0.044	0.44	0.116	1.16	0.0112	0.316
TOTAL	5.789	3.956+?	1.157	2.441	1.299	36.83
NRC Budget						
FY83	0.358	1.5				
FY84	0.892	2.3				
PROCEDURES						
I.A.2.6(1,2,3,5)	0.028	0.28	0.028	0.28	0.14	4
I.B.1.1(5-7)	0.132	1.32	0.348	3.48	0.336	0.948
I.C.9	0.7	1.715	0.35	1.715	0.175	4.97
I.C.1(4	1.26	11.2	0.077	0.77	0.035	0.98
I.D.3	0.005	0.05	0.064	0.64	0.008	0.22
I.D.4	0.03	0.3	0.009	0.09	0	0
I.D.5(3-5)	0.198	0	0.064	0.64	0	0
I.A.4.2	0	0	0.415	1.65	0.07	2.0
TOTAL	2.353	14.87	1.355	9.265	0.462	13.12
NRC Budget						
FY83	0.752	2.4				
FY84	1.265	4.8				
MANAGEMENT AND ORGANIZATION						
I.B.1.1(5-7)	0.528	5.28	1.392	13.92	0.134	3.79
I.A.3.4	0.75	0	0.125	0	0.15	4.25
TOTAL	1.278	5.28	1.517	13.92	0.284	8.04

TABLE 3.5 (Continued)

Issue	Development $,M	Development man-yrs	Implementation $,M	Implementation man-yrs	Op & Maintenance $/yr,M	Op & Maintenance Total $,M
NRC Budget						
FY83	0.810	1.4				
FY84	0.24	1.0				
MAN-MACHINE INTERFACE						
I.D.3	0.04	0.4	0.512	5.12	0.064	1.76
I.D.4	0.24	2.4	0.072	0.72	0	0
I.D.5(3-5)	1.584	0	0.512	5.12	0	0
I.A.4.2	0	0	0.415	1.65	0.07	2
I.A.2.6(1,2,3,5)	0.014	0.14	0.014	0.14	0.07	2
TOTAL	1.878	2.94	1.525	12.75	0.204	5.76
NRC Budget						
FY83	3.6	7.5				
FY84	2.34	3.0				
TOTAL NRC FUNDING	14.335	38.98+?	8.133	56.28	4.579	130.1
TOTAL NRC BUDGET						
FY83	9.27	19.7				
FY84	7.117	19.5				

? indicates that NRC staffing requirements are not defined in PNL analysis of the safety issue.

TABLE 3.6. Industry Human Factors Program Funding Requirements

Issue	Implementation $,M	Op & Maintenance $/yr,M	Total $,M
STAFFING AND QUALIFICATIONS			
I.A.2.2	11.25	5.35	151.8
I.A.3.3	42.3	36.18	1026
I.A.2.6(1-3,5)	8.0	3.39	96
I.A.2.7	3.6	1.21	34
I.A.3.4	1.75	0.335	9.5
I.B.1.1(5-7)	1.48	3.752	106.20
TOTAL	68.38	50.22	1423.0
TRAINING			
I.A.2.6(4)	1.1	1.1	32.
I.A.2.6(1-3,5)	112.	47.46	1344.
I.A.2.7	28.8	9.68	272.
I.A.2.4	10.05	4.02	114.
I.A.2.2	5.625	2.675	75.88
I.C.1(4)	1.365	0.315	9.0
I.B.1.1(5-7)	1.48	3.752	106.2
I.D.3	5.4	0.016	0.43
I.D.4	0.10	0	0
I.D.5(3-5)	26.8	0.3	8.6
I.A.4.2	235.0	9.4	265.0
I.C.9	10.05	2.01	56.85
TOTAL	437.8	80.73	2284.0
LICENSING EXAMINATIONS			
I.A.2.2	5.625	2.675	75.88
I.A.3.4	6.125	1.173	33.25
I.A.3.3	4.7	4.02	114.
I.A.2.6(1-3,5)	16.0	6.78	192.
I.A.2.7	3.6	1.21	34.
I.A.4.2	0	0	0
I.B.1.1(5-7)	0.74	1.876	53.1
I.C.9	10.05	2.01	56.85
I.C.1(4)	1.365	0.315	9.0
TOTAL	48.21	20.06	568.1

TABLE 3.6. (Continued)

Issue	Implementation $,M	Op & Maintenance $/yr,M	Total $,M
PROCEDURES			
I.A.2.6(1,2,3,5)	16.0	6.78	192.0
I.B.1.1(5-7)	2.22	5.628	159.3
I.C.9	46.9	9.38	265.3
I.C.1(4)	6.37	1.47	42.0
I.D.3	5.4	0.016	0.43
I.D.4	0.10	0	0
I.D.5(3-5)	26.8	0.3	8.6
I.A.4.2	117.5	4.7	132.5
TOTAL	221.3	28.27	800.1
MANAGEMENT AND ORGANIZATION			
I.B.1.1(5-7)	8.88	22.51	637.2
I.A.3.4	0.875	0.1675	4.75
TOTAL	9.755	22.68	642.0
MAN-MACHINE INTERFACE			
I.D.3	43.2	0.128	3.44
I.D.4	0.8	0	0
I.D.5(3-5)	214.4	2.4	68.8
I.A.4.2	117.5	4.7	132.5
I.A.2.6(1,2,3,5)	8.0	3.39	96.0
TOTAL	383.9	10.62	300.7
TOTAL INDUSTRY COSTS	1169.3	212.6	6018.0

3.5 COST SAVINGS

Implementation of several of the safety issues examined for the NRC were postulated to result in cost savings due a reduction in down time and increased plant availability. Such factors will most likely also result from implementation of the Human Factors Program. However, the impact on plant availability could be expected to be more difficult to quantify the further the program moves away from hardware. Also, impact on plant availability was not estimated in a rigorous fashion for all issues. Because of this, the examples of cost savings from the analysis of safety issues are given here to illustrate the types of savings that may be available to offset utility costs for implementing the Human Factors Program.

Issue I.B.1.1(5-7) Management and Organization: Organization and Management of Long-Term Improvements

In this safety issue, the improved management and organizational practices were postulated to achieve a $7.5E5/py savings in staff costs, and a 4 percent improvement in availability. For the Human Factors Program, it has been assumed in Table 3.2 that 40 percent of this would be attributable to human factors, giving an assumed improvement of 1.6 percent.

For 134 LWRs with an average life remaining of 23.8 years and an availability of 65 percent, this gives a savings of:

(1.6%)(0.65)(365 plant days/py)($3.00E+05/pd)(134 plants)(23.8 yrs) +
($7.50E+05/py)(134 plants)(23.8 yrs) = $6.03E+09.

Issue I.D.5(3-5) Improved Control Room Instrumentation Research

This issue was estimated to reduce the number of unscheduled outages. This was assumed in Table 3.2 to apply entirely to the HFPP, so the cost savings could be as follows:

(approx. 0.72 days/py)($3.00E+05/day)(134 plants)(28.3 yrs) = $8.20E+09.

4.0 SUMMARY OF RESULTS

A summary of cost, risk, and priority scores for the six HFPP elements is shown in Table 4.1. Industry total public risk reductions for the HFPP total 2.0E+05 man-rem. Elements that exceed 50,000 man-rem include Management and Organization and Training. All remaining elements exceed 5000 man-rem per element.

Costs for the implementation of the whole HFPP are expected to total $7400M over the remaining lives of 134 plants. Costs for individual elements range from $660M to $2800M. Issues costing less than $1000M include management and organization, man-machine interface, and procedures. Cost savings discussed in the previous section were not included in these totals.

Priority scores for the HFPP elements range from 106 man-rem/$M to 1.2 man-rem/$M.

TABLE 4.1. Summary of Results

		HFPP Element						
		Staffing Qualifications	Training	Licensing Examination	Procedures	Man-Machine Interface	Management Organization	Total
Public Risk (man-rem)	A	2.2E+04	4.4E+04	2.2E+04	3.7E+04	3.5E+04	3.7E+04	2.0E+05
	B	3.1E+04	4.3E+04	2.0E+04	3.2E+04	2.1E+04	5.0E+04	2.0E+05
	C	1.6E+04	6.5E+04	6.0E+03	3.4E+04	6.7E+03	7.0E+04	2.0E+05
	D	1.8E+03	8.2E+04	1.3E+04	1.8E+04	2.9E+04	5.3E+04	2.0E+05
Cost								
Industry		1.5E+09	2.7E+09	6.2E+08	1.0E+09	6.8E+08	6.5E+08	7.2E+09
NRC		2.2E+07	5.0E+07	4.4E+07	1.7E+07	9.2E+06	1.1E+07	1.8E+08
Total		1.5E+09	2.8E+09	6.6E+08	1.0E+09	6.9E+08	6.6E+08	7.4E+09
Priority Score (man-rem/$M)	A	15	16	33	37	51	56	27
	B	21	15	30	32	30	76	27
	C	11	23	9.1	34	9.7	106	27
	D	1.2	29	20	18	42	80	27

5.0 CONCLUSIONS

The results of cost, risk and priority score calculations were reviewed for the purposes of drawing conclusions about the relative and absolute importance of the overall HFPP and individual elements. In general, it was concluded that consideration of element overlaps is important to the identification of the most effective way of reducing operator errors. Results of the models indicate a range of results for public risk reductions and priority scores. However, all the models indicated consistent results for each of these parameters.

A ranking of the HFPP elements based on total public risk reductions indicates that the top two items are Management and Organization and Training. The bottom two items are Staffing and Qualifications and Licensing Examination. There is roughly a tie for the middle two items (Procedures, Man-Machine interface). On an absolute risk scale as defined in NUREG 0933, the HFPP ranks as a high priority issue based on total risk reduction. On an incremental element basis, management and Organization and Training would be ranked as high priority elements, with the remaining elements ranked in the low and medium categories.

A ranking of the HFPP elements based on priority score indicates that Management and Organization is first, procedures a distant second and Man-Machine Interface is third. The remaining three elements are roughly tied. On an absolute priority score basis, the entire HFPP would rate a low priority. All other elements would fall into the low or drop categories.

If cost savings estimated for improvements to plant availability are included in the cost estimates for the HFPP, net cost savings for the entire program would result. However, as previously discussed, these estimates are speculative and not easily attributed to specific HFPP elements. For the purposes of this analysis, it was concluded that these cost savings should be considered to the extent that they provide incentive to overall human performance improvements. To obtain the benefits, specific activities need to be defined under each of the HFPP elements with the objective to improve availability. The element descriptions used for this analysis do not reflect this objective.

In summary, the HFPP has significant risk importance, but could incur substantial costs during implementation. By considering overlaps in both the costs and benefits of the individual elements, significant economies of effort could be achieved in both the Industry and NRC costs without foregoing all of the available risk reduction.

APPENDIX A

NRC HUMAN FACTORS PROGRAM PLAN ELEMENT ISSUE DESCRIPTION

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NRC HUMAN FACTORS PROGRAM PLAN ELEMENT ISSUE DESCRIPTION

ISSUE H.F. 01: HUMAN FACTORS PROGRAM PLAN

All the investigations which followed the Three Mile Island Unit 2 accident identified the need to incorporate human factors considerations into the regulations and guidance governing the design and operation of nuclear power plants. NUREG-0660 described a number of tasks to be performed by the nuclear industry and the NRC. A significant number of these tasks focused upon improving nuclear power plant safety through increased attention to the human element. Considerable progress has been made on many of these Action Plan items. The "Human Factors Program Plan", describes the human factors related work required to complete the human factors tasks described in NUREG-0660. In addition, it describes the additional human factors related efforts identified during the implementation of the Action Plan that requires NRC attention.

TASK H.F. 01.1.0: STAFFING AND QUALIFICATIONS

To assure that the staffing at nuclear power plants is adequate in both number and capability to provide for safe operation. To meet this goal, consideration will be given to (1) the numbers and functions of the staff needed to safely perform all required plant operations, maintenance and technical support for each operational mode; (2) the minimum qualifications of plant personnel, in terms of education, skill, knowledge, training, experience, and fitness for duty; and (3) appropriate limits and conditions for shift work, including overtime, shift duration, and shift rotation. The end products and planned activities of this element have been adjusted to support the requirements of Public Law 97-425.

ITEM H.F. 01.1.1: ESTABLISH STAFFING REQUIREMENTS

The NRC will determine the minimum appropriate shift crew staffing composition. This determination will be made from developed personnel projection and allocation models and from evaluations of job/task analysis data. Current staffing practice of both foreign and domestic utilities will be surveyed to evaluate current practices, regulations and current staffing levels, considering such variables as plant size, control room arrangement and configuration, and plant layout. A rule for inclusion in 10 CFR will be prepared regarding licensed operator staffing. A revision to Standard Review Plan which includes staffing will be developed.

The need for engineering expertise on shift will be decided. This decision will be based in part upon the functions and duties required using the results of job/task analysis and evaluation of the current shift technical advisor experience. Consideration will also be given on how best to incorporate his expertise into the plant crew compliment. A rule in 10 CFR or a policy statement on the inclusion of engineering expertise on shift will be formulated.

ITEM H.F. 01.1.2: PERSONNEL QUALIFICATION REQUIREMENTS

The minimum training, education, and experience requirements for shift operating crews will be determined. This is to be achieved by the implementation of SECY 82-162A, "An Integrated Plan on Shift Crew Qualifications", which will integrate industry and NRC efforts to establish appropriate crew qualification requirements. The relationship between education, training, and experience will be assessed and the trade-offs among these related factors determined. A rule for 10 CFR Part will be prepared on minimum crew qualifications.

The feasibility and value of licensing or certifying other nuclear power plant personnel will be evaluated. A rule on degree requirements for the operating staff will be prepared by inclusion in 10 CFR Part.

ITEM H.F. 01.1.3: GUIDANCE ON LIMITS AND CONDITIONS FOR SHIFT WORK

It is an accepted fact that shift work and the use of overtime can have an adverse effect upon operator performance. To determine the appropriate limits and conditions for shift work, activities are planned (1) to determine the effects of varying shift duration using nuclear power plant simulators; and (2) to survey and assess the experience of other industries with job requirements similar to the nuclear industry with regard to shift arrangements and rotation. This effort will allow the NRC to establish trade-offs among factors affecting shift work and overall safe performance requirements. The results are to be reported as a NUREG document and a specific research effort will be undertaken if shift rotation is found to be a serious human factor problem.

ITEM H.F. 01.1.4: MAINTENANCE STAFFING AND QUALIFICATION

The appropriate staffing and qualifications of maintenance personnel for nuclear power plant maintenance and repair will be determined. Problems areas associated with qualifications, staffing, training and overtime practices related to nuclear power plant maintenance personnel will be identified, documented, and evaluated. Selected maintenance jobs/tasks will be reviewed to identify their complexity and safety importance to determine the effects of certification of maintenance personnel on plant safety. Job/task analyses for maintenance activities will be performed and analyzed to develop a maintenance certification program. A rule for inclusion in 10 CFR Part will be prepared, if required, regarding the licensing/certification of other nuclear power plant personnel.

ITEM H.F. 01.1.5: FITNESS FOR DUTY

A rule, revising 10 CFR Part 73, relating to fitness for duty for personnel having access into nuclear power plants or involved with their operation will be prepared.

TASK H.F. 01.2.0: TRAINING

Description:

To provide assurance that personnel are able to meet job performance requirements, that training properly accounts for pertinent safety issues and that a mechanism exists for upgrading and assuring the quality of training programs. The Nuclear Waste Policy Act of 1982, Section 306 of PL 97-425, directs the NRC to promulgate regulations and regulatory guidance for the training of nuclear power plant personnel. Areas to be addressed include simulator training requirements, operator requalification programs, team training, instructional requirements for training programs, and the administration of examinations. The planned activities in this task and its end products are supportive of this act.

ITEM H.F. 01.2.1: DEVELOP TRAINING REGULATION AND GUIDANCE

NRR will perform the activities required to develop regulations and guidance for the training and qualifications of civilian nuclear power plant operators, supervisors, technicians, and other appropriate operating personnel. These regulations will direct the utilities to review their training programs to assure a systematic approach to training. The purpose of this effort is to focus the utility training program on the knowledge, skills and abilities required to operate the nuclear power plant safely and increase the licensed operators' ability to respond correctly to unexpected events. As appropriate, this program will recognize the relationship to INPO's accreditation activity and INPO's effort to develop a handbook on application of systematic approaches to training. This activity will result in a rule involving the training of nucler power plant personnel and revisions to 10 CFR Parts 50 and 55. Regulatory Guides will be prepared to accompany new training rules and a revision will be prepared to Regulatory Guide 1.8. A NUREG will be prepared to provide guidelines for team training.

b. The role of simulators, their requisite fidelity and type in training programs will be determined. The use of simulators in providing team training of the control room operating personnel will be determined. A regulatory guide on simulator certification will be prepared. A NUREG report defining the role of simulators in training is to be developed.

ITEM H.F. 01.2.2: TRAINING ASSESSMENT PROCEDURES

Criteria and procedures will be developed to enable the performance of consistent and objective reviews of nuclear power plant training programs. These criteria and objectives are to be based upon the systematic approach to training concept. A regulatory position is to be established regarding accreditation of training programs. Recognition of the accreditation efforts by INPO will be taken into account. This activity will also result in development of criteria to evaluate the qualifications for Training Instructor. Revisions to Chapter 13.2 of NUREG-0800, "Standard Review Plan", will be prepared. Inspection modules for assessing training programs will be prepared or modified for use by the Office of Inspection and Enforcement and the Regional Offices.

TASK H.F. 01.3.0: LICENSING EXAMINATIONS

It is the purpose of this task to ensure that the licensing examination for reactor operators and senior reactor operators is a valid measure of the operator's ability to perform the necessary tasks and functions required to safely operate and control commercial nuclear power plants. Secondly, that examinations and the examination process be practiced and administered in a consistent manner by the various NRC examiners. Finally, to correlate the licensing examination to the improved training programs. The intent is to perform these modifications to the examination process without unnecessary impact to current license candidates and training programs.

ITEM H.F. 01.3.1: EXAMINATION CONTENT

An identification of the reactor operator and senior reactor tasks and duties, and the required knowledge, skills and abilities necessary for safe performance will be determined using available generic job/tasks analyses. From this information a set of test questions will be developed or updated and stored in a computer data bank for use in test construction and examination validation. Additionally, test specifications will be developed for licensing examinations to provide examination plans which outline the necessary types of knowedge required for safe operator performance. An evaluation of the feasibility will be conducted for identifying or developing on the job performance measures which can be employed in the examination process to predict operator performance. Long term examination development/validation strategies will be developed based upon the results of current examination modifications and content validation.

ITEM H.F. 01.3.2: THE EXAMINATION PROCESS

To increase the efficiency, reliability, and validity of the licensing examination process, the DHFS will prepare new examination procedures. These new procedures will take into consideration the problems and issues associated with the current examination process from the examiner, candidate and utility perspectives. The examination process and practices of similar applicable agencies and organizations will be reviewed. The input from industry training staff and reactor operators regarding problems or issues underlying the current licensing examinations will be solicited. The result will be the identification of improvements to optimize the format and procedures relating to written, oral and simulator examinations. From this identification activity standardized examination practices and guidelines will be developed. The test examiners will also be trained on test development, administration and grading techniques to assure consistency and reliability. A revision to 10 CFR Part 50 will be prepared to reflect changes made in examinations and in the examination process. Revision to Regulatory Guide 1.149, "Nuclear Power Plant Simulators for Use in Operator Training", will also be accomplished.

TASK H.F. 01.4.0: PROCEDURES

This task is to provide assurance that plant procedures are adequate and can be used effectively. The objective is to provide procedures which will guide the operators in maintaining the plant in a safe state under all operating conditions, including the ability to control upset conditions without first having to diagnose the specific initiating event. This objective is to be met by: (1) developing guidelines for preparing, and criteria for evaluating emergency operating, normal operating and the other procedures which affect plant safety; and (2) upgrading the procedures, training the operators in their use, and implementing the upgraded procedures. Additionally, the adequacy of the operator training in off-normal events duing the initial test program will be reassessed.

ITEM H.F. 01.4.1: PROCEDURES, GUIDANCE AND EVALUATION CRITERIA

Guidance will be developed which will provide the necessary instructions to prepare improved emergency operating procedures, abnormal operating procedures, normal operating procedures, maintenance procedures, and procedures for emergency plan implementation, refueling, administration, safeguards, and security. Generic technical guidance is to be provided by industry; the NRC and industry will jointly coordinate the development of human factors guidelines. Research will be accomplished to develop methods and evaluate alternate techniques and formats for the display of procedures, e.g., computerized CRT presentation. Results of these activities will be published as NUREG reports.

Criteria to evaluate and audit emergency operating procedures by the regions will be prepared by NRR and IE. This criteria will be published as a revised inspection module. Similar criteria and inspection modules will be developed when the guidelines for the upgrading of other procedures are completed.

ITEM H.F. 01.4.2: INITIAL TEST PROGRAM TRAINING EFFECTIVENESS

Operator training for off-normal events is conducted during the initial test program. The need to reassess the adequacy of operator training during this test period will be determined.

TASK H.F. 01.5.0: MAN-MACHINE INTERFACE

The objective of this task is to ensure that the man-machine interface is adequate for the safe operation and maintenance of nuclear power plants. This objective will be attained by developing (1) human factors engineering guidelines for correcting man-machine interface problems; and (2) regulatory guidance for integrating human factors engineering into new designs and into advanced technological improvements incorporated into existing designs. This activity will also provide for the preparation of evaluation tools for (1) the next generation of nuclear power plants; and (2) for expected changes or upgrading to designed plants in the areas of data and information management and improved annunciator systems. In addition, these efforts will improve the staff's capability to evaluate reactor incidents involving man-machine interface errors.

ITEM H.F. 01.5.1: MAN-MACHINE INTERFACE GUIDANCE FOR EXISTING DESIGNS

The regulatory efforts to date dealing with the man-machine interfaces have been limited to the control room and the remote shutdown panel. Further guidance is necessary regarding: (1) maintenance; (2) local control stations and auxiliary operator interfaces. Additional guidance may also be required regarding improvements to the existing annunciator system.

a. This activity will establish the scope and direction which will be followed with respect to nuclear power plant maintenance activities. The other NRC human factors activities related to maintenance, such as procedures, qualifications and training, will be closely monitored. Maintenance improvement activities sponsored by the nuclear industry will be closely followed, for example, INPO's efforts on reliability centered maintenance work and EPRI's work in preventative maintenance and job performance aids. From these and other related efforts a regulatory position on the NRC's role in nuclear power plant maintenance will be developed. Additional criteria relating human factors in maintennce for design and operation will be prepared. A NUREG addressing design-for-maintainability for selected maintenance area will be written and published. A regulatory Guide outlining an integrated approach to the general issue of nuclear power plant maintenance will be prepared.

b. Information will be collected to determine if guidance on local control station design and auxiliary operator interfaces with these stations is required. To accomplish this subtask, job/task analyses of control room crew activities will be conducted to identify and describe communication and control links between the control room and the auxiliary control stations. In addition, the functions of the auxiliary personnel will be analyzed from the task analyses to estimate the potential impact of auxiliary personnel job errors on plant safety. NUREG reports will be published to report the findings discovered.

c. The information provided in NUREG-0700, "Guidelines for Control Room Design Reviews", provide the guidance, which if incorporated, should minimize the potential for human errors associated with these systems. However, some of these standards are difficult to apply except as long-term design changes. Guidance will be developed for near-term improvements which address the techniques for implementing the quality standards of NUREG-0700. An assessment of the impact of NUREG-0700 guidelines on operating control rooms will be performed to identify if revisions are needed to NUREG-0700.

ITEM H.F. 01.5.2: GUIDANCE FOR DESIGNS BASED ON ADVANCED TECHNOLOGIES

The existing human engineering guidelines for nuclear power plant control rooms primarily address the control, display and information concepts and technologies which are now being used in process control systems. While these guidelines are adequate for the current generation of nuclear power plants, they may not be sufficient for advanced and developing technologies which may be introduced into existing and future designs. This concern is addressed by the following activities.

a. Computers - A program plan will be developed to evaluate the safety significance and problems relating to the management of data and information in a nuclear power plant control room during abnormal events. Products may include the development of guidelines on control room information management during severe transients and accidents. These guidelines may be in the form of NUREG reports and Regulatory Guides.

b. Advanced Controls and Displays - Staff guidance pertinent to the man-machine interface involving new control and display techniques will be prepared. These guidance documents will require (1) the identification of new developing display and control technologies having a potential application in nuclear power plant control rooms; (2) development of evaluation methods and design criteria related to visual displays; and (3) establishing the criteria needed for regulatory assessment of advanced control room concepts. In addition the control and display requirements for crew response needs following a seismic event will be identified.

c. Function Allocation - An integrated program plan for the investigation of function allocation will be prepared. The plan will address (1) the identification of nuclear power plant functions involving the human; (2) whether the current function allocations permit the reliable performance of functions assigned to humans; (3) the need to reallocate functions between the human and the machine; (4) which functions should be reallocated; and (5) the identification of those design changes which enhance function performance. In addition, the plan will address the feasibility and desirability of applying cognitive workload measurement techniques to a selected list of operator functions.

d. Advanced Annunciator Systems - Improved annunciator systems are expected to become available which will utilize advanced technologies. Guidelines for the utilization and evaluation of these longer-term annunciator improvements will be developed. These guidelines will be based upon evaluations of results from advanced concept activities being performed by governmental and commercially sponsored research activities.

e. Safety Status Indication - Based upon the results of a current project investigating means for monitoring and verifying operations, tests, and maintenance activities, the staff will make determinations concerning: (1) the comparative adequacy of status monitoring in plants that do not have automatic monitoring systems; (2) the adequacy of operational systems designed to be in conformance with Regulatory Guide 1.47; and (3) the development of long-term improvement guidance addressing the feasibility and value/impact of instrumentation backfits.

TASK H.F. 01.6.0: MANAGEMENT AND ORGANIZATION

It is the objective of this task to ensure that the utility management and organization will be adequate to provide for safe nuclear power plant operation. This objective will be accomplished by: (1) providing the guidance and requirements which will define the necessary management and organizational

practices; (2) develop the necessary criteria and reliable evaluation objectives for staff use to assess the effectiveness of management and organizational functions; and (3) cooperate with INPO in the development of a training program for prospective plant managers and other appropriate utility managers. These efforts are to heighten management awareness and sensitivity for safe plant operation. Portions of this activity will be a joint effort between INPO and the NRC.

ITEM H.F. 01.6.1: MANAGEMENT ORGANIZATION AND GUIDANCE AND REGULATORY POSITION

INPO has developed general guidance and criteria for improving the quality of nuclear power plant management and is proceeding with efforts to evaluate utility corporate office structure. In recognition of INPO's corporate office evaluations, NRC has reduced the number of Performance Appraisal Team inspections. In order to accept INPO's corporate office evaluations, as an alternate to an expanded NRC effort in management, NRC needs to review and monitor INPO's corporate office program. To provide an evaluation and make recommendations to INPO the following will be accomplished: (1) evaluate the need for a regulatory position on Management and Organization; (2) develop the basis for the acceptance of INPO's criteia, including the functions necessary for safe operation; (3) prepare a detailed evaluation of INPO Corporate Evaluation criteria; (4) prepare recommendations for INPO on Management and Organization; and (5) develop a regulatory position.

ITEM H.F. 01.6.2: ASSESSMENT PROCEDURES

Assessment procedures will be developed to increase the reliability and consistency of the operating license management and organization reviews. The development of these procedures will require: (1) the development of criteria and procedures for NRC reviewers; (2) the preparation of management and organization guidelines for use by the applicants for operating licenses; (3) the revision of the Standard Review Plan; (4) the development and provision of training for the technical reviewers; (5) the preparation of a workbook for NRC use during site visits and use in the preparation of the SER; and (6) the review of applicable inspection modules to determine if revision to the applicable modules is necessary.

APPENDIX B

<u>QUESTIONNAIRE RELATED TO HUMAN FACTORS PROGRAM PLAN</u>

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QUESTIONNAIRE RELATED TO HUMAN FACTORS PROGRAM PLAN

The purpose of this questionnaire is to elicit your judgments about the impacts of human factors research on the improvement of human performance at nuclear power plants. The NRC Human Factors Program Plan identifies six areas of research. These are:

- Staffing and Qualification
- Training
- Licensing Examinations
- Procedures
- Management and Organization
- Man-Machine Interface

[NOTE: Maintenance is not included as an area. In addition, for purposes of this questionnaire, Training does not include maintenance training, Procedures does not include maintenance procedures, etc.]

Your task will be to determine the direct and indirect effects of these areas on improving human performance.

First, please read the brief definitions of the six human factors research areas that are presented below. Then you will be asked to divide 100 points among the six areas according to your assessment of the relative impact of research in each area on improving human performance.

DEFINITIONS

Staffing and Qualification is defined here as the number and minimum qualifications of Nuclear Power Plant (NPP) personnel that are adequate for safe operation and support of an NPP. This area involves understanding the functions of NPP staff needed to safely perform all required plant operations and technical support activities for each operational mode. It also involves determining the minimum education, skill, knowledge, training, and fitness for duty necessary to fulfill the functions required.

Training is defined here as instruction and experience given to staff personnel to enable them to meet job performance requirements. This instruction and experience involves preparation to handle pertinent safety issues.

Licensing Examinations are defined here as three part tests consisting of a written exam, an oral exam, and a simulator exam for reactor operators (RO) and senior reactor operators (SRO). The written exam tests the academic knowledge of systems, fundamentals, and procedures. The oral exam tests the integrated knowledge of fundamentals. The simulator exam tests abilities to safely respond to various reactor operation occurrences.

Procedures are defined here as the sequential steps of plant operation necessary to maintain plant safety fuctions. These procedures should indicate how to control upset conditions without first having to diagnose the specific initiating events.

Management and Organization is defined here as the functions and characteristics of utility management and organization that must be effective in influencing the safe operation of an NPP. These functions include operations, security, technical support, and safety review. The characteristics include communications and administrative controls.

Man-Machine Interface is defined here as boundary between human and equipment. Equipment can be either the indication of the status of a piece of equipment or the actuating devices to affect a piece of equipment or the piece of equipment itself. Man-Machine Interface can involve the design of equipment to promote safer performance of the man-machine system.

Part I: Direct Effects

Your first task is to divide 100 points among the six research areas according to your assessment of the relative impact of research in each area on improving human performance at nuclear power plants.

For example, if you believe that only research in Procedures and Training will directly improve human performance and that research on Training is three times more important for improving human performance than research on Procedures, then you would allot 75 points to Training, 25 points to Procedures, and 0 points to each of the other areas as shown below.

Direct Effects Example

Research Areas	Points Allotment
STAFFING & QUALIFICATION	0
TRAINING	75
LICENSING EXAMINATIONS	0
PROCEDURES	25
MANAGEMENT & ORGANIZATION	0
MAN-MACHINE INTERFACE	0

Total = 100 (Sum = 100)

Now please distribute the 100 points among the six areas according to your assessment of the relative impact of research in each area on improving human performance at nuclear power plants.

Research Areas	Points Allotment
STAFFING & QUALIFICATION	___
TRAINING	___
LICENSING EXAMINATIONS	___
PROCEDURES	___
MANAGEMENT & ORGANIZATION	___
MAN-MACHINE INTERFACE	___
Total =	(Sum = 100)

Consider the six areas that have been discussed. If research was performed and the results implemented in <u>all</u> of these areas at nuclear power plants, how much reduction in human error would you expect to occur? [Remember, a 100% reduction means a complete elimination of errors.]

___%

<u>Part II:</u> Indirect Effects

Some of the research areas can also indirectly improve human performance. For example, Management and Organization can affect whether people use Procedures and the use of Procedures improves human performance; changes in the Licensing Examinations will affect Training, which improves human performance. Below we ask you to consider each research area individually and to judge the extent to which the other five areas affect that area. Again, you are asked to distribute 100 points among the areas.

Consider the example below. In this example, we are looking at the effects of S&Q, T, LE, M&O, and MMI on PROCEDURES. Because there may be cases where areas other than the five listed have an effect on PROCEDURES, an "Other" category is included.

If you believe that the five research areas have no affect on PROCEDURES, then you should assign all of the 100 points to the "Other" category.

In the example below, however, S&Q and LE were believed to have no impact on PROCEDURES. MMI was believed to have twice the impact on PROCEDURES compared to T and M&O. In addition, there were believed to be "other" effects on PROCEDURES that were equal to the effects of T and M&O. Thus, the person arrived at the point distribution shown below.

Indirect Effects Example

Effect of Research Areas on Improving PROCEDURES

Research Areas	Points Allotment
STAFFING & QUALIFICATION	0
TRAINING	20
LICENSING EXAMINATIONS	0
MANAGEMENT & ORGANIZATION	20
MAN-MACHINE INTERFACE	40
OTHER	20
	Total = 100 (Sum = 100)

STAFFING AND QUALIFICATIONS

Effect of Research Areas on Improving STAFFING AND QUALIFICATIONS

Research Areas	Points Allotment
TRAINING	___
LICENSING EXAMINATIONS	___
PROCEDURES	___
MANAGEMENT & ORGANIZATION	___
MAN-MACHINE INTERFACE	___
OTHER	___
Total =	(Sum = 100)

For each research area above that received more than "0" points, please describe the indirect effects that you had in mind.

TRAINING____________________

LICENSING EXAMINATION____________________

PROCEDURES____________________

MANAGEMENT AND ORGANIZATION____________________

MAN-MACHINE INTERFACE____________________

OTHER____________________

TRAINING

Effect of Research Areas on Improving TRAINING

Research Areas	Points Allotment
STAFFING & QUALIFICATION	___
LICENSING EXAMINATIONS	___
PROCEDURES	___
MANAGEMENT & ORGANIZATION	___
MAN-MACHINE INTERFACE	___
OTHER	___
Total =	(Sum = 100)

For each research area above that received more than "0" points, please describe the indirect effects that you had in mind.

STAFFING AND QUALIFICATION ____________________

LICENSING EXAMINATION ____________________

PROCEDURES ____________________

MANAGEMENT AND ORGANIZATION ____________________

MAN-MACHINE INTERFACE ____________________

OTHER ____________________

LICENSING EXAMINATIONS

Effect of Research Areas on Improving LICENSING EXAMINATIONS

Research Areas	Points Allotment
STAFFING & QUALIFICATION	___
TRAINING	___
PROCEDURES	___
MANAGEMENT & ORGANIZATION	___
MAN-MACHINE INTERFACE	___
OTHER	___
Total =	(Sum = 100)

For each research area above that received more than "0" points, please describe the indirect effects that you had in mind.

STAFFING AND QUALIFICATION____
____

TRAINING____
____

PROCEDURES____
____

MANAGEMENT AND ORGANIZATION____
____

MAN-MACHINE INTERFACE____
____

OTHER____
____

PROCEDURES

Effect of Research Areas on Improving PROCEDURES

Research Areas	Points Allotment
STAFFING & QUALIFICATION	___
TRAINING	___
LICENSING EXAMINATIONS	___
MANAGEMENT & ORGANIZATION	___
MAN-MACHINE INTERFACE	___
OTHER	___
Total =	(Sum = 100)

For each research area above that received more than "0" points, please describe the indirect effects that you had in mind.

STAFFING AND QUALIFICATION____

____

TRAINING____

____

LICENSING EXAMINATIONS____

____

MANAGEMENT AND ORGANIZATION____

____

MAN-MACHINE INTERFACE____

____

OTHER____

____

MANAGEMENT AND ORGANIZATION

Effect of Research Areas on Improving MANAGEMENT AND ORGANIZATION

Research Areas	Points Allotment
STAFFING & QUALIFICATION	___
TRAINING	___
LICENSING EXAMINATIONS	___
PROCEDURES	___
MAN-MACHINE INTERFACE	___
OTHER	___
Total =	(Sum = 100)

For each research area above that received more than "0" points, please describe the indirect effects that you had in mind.

STAFFING AND QUALIFICATION______________________________________

____

TRAINING____

____

LICENSING EXAMINATIONS____

____

PROCEDURES____

____

MAN-MACHINE INTERFACE____

____

OTHER____

____

MAN-MACHINE INTERFACE

Effect of Research Areas on Improving MAN-MACHINE INTERFACE

Research Areas	Points Allotment
STAFFING & QUALIFICATION	___
TRAINING	___
LICENSING EXAMINATIONS	___
PROCEDURES	___
MANAGEMENT AND ORGANIZATION	___
OTHER	___
Total =	(Sum = 100)

For each research area above that received more than "0" points, please describe the indirect effects that you had in mind.

STAFFING AND QUALIFICATION____

TRAINING____

LICENSING EXAMINATIONS____

PROCEDURES____

MANAGEMENT AND ORGANIZATION____

OTHER____

APPENDIX C

NAMES OF THE HUMAN FACTORS STAFF PANEL

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NAMES OF THE HUMAN FACTORS STAFF PANEL

The following personnel were included on the Human Factors Staff Panel.

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This supplemental report is the fourth in a series that document and use methods developed by the Pacific Northwest Laboratory to calculate, for prioritization purposes, the risk, dose and cost impacts of implementing resolutions to reactor safety issues. The initial report in this series was published by Andrews et al. in 1983 as NUREG/CR-2800. This supplement consists of two parts describing separate research efforts: (1) an alternative human factors methodology approach and (2) a prioritization of the NRC's Human Factors Program Plan. The alternative human factors methodology approach may be used in specific future cases in which the methods identified in the initial report (NUREG/CR-2800) may not adequately assess the proper impact for resolution of new safety issues. The alternative methodology included in this supplement is entitled <u>Methodology for Estimating the Public Risk Reduction Affected by Human Factors Improvement.</u> The prioritization section of this report is entitled <u>Prioritization of the U.S. Nuclear Regulatory Commission Human Factors Program Plan.</u>

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